ENCLOSURE 1

Examination Report No.: 50-206/OL-92-01

Facility: San Onofre Nuclear Generating Station Units 1

Docket Nos.: 50-206

Examinations administered at San Onofre Nuclear Generating Station, San Clemente, California and CAE Electronics Ltd., Montreal, Canada.

Chief Examiner: G. Johnston, Operator Licensing Examiner

Accompanying Personnel: L. Sherfey, Battelle Contractor
J. Nickolaus, Battelle Contractor

Approved: Lewis F. Miller, Jr., Chief
Reactor Safety Branch

6/26/92
Date Signed

Summary:

Examinations on April 27 - May 9, 1992 (Report No. 50-206-OL-92-01)

Examinations were administered to 14 Reactor Operator (RO) Candidates and 7 Senior Reactor Operator (SRO) Candidates. 9 RO candidates passed the written and operating portions of the examinations. 6 SRO candidates passed the written and operating examinations.

Safety Significant Issues:

No safety significant issues were identified.

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REPORT DETAILS

1. Examiners

G. Johnston, RV (Chief Examiner)
L. Sherfey, Battelle Contractor
J. Nickolaus, Battelle Contractor

2. Persons Attending the Exit Meeting

NRC:

G. Johnston, Chief Examiner
L. Sherfey, Battelle Contractor
J. Nickolaus, Battelle Contractor
C. Caldwell, Senior Resident Inspector
D. Solorio, Resident Inspector
C. Townsend, Resident Inspector

Southern California Edison:

H. Morgan, Vice President, Site Manager
R. Krieger, Station Manager
R. Waldo, Station Operations Manager
J. Reeder, Nuclear Training Manager
R. Sandstrom, Operations Training Supervisor
J. Vandenbroek, Compliance Supervisor
D. Brevig, Onsite Nuclear Licensing Supervisor
J. Jamerson, Onsite Nuclear Licensing Lead Engineer
J. Sutton, Onsite Nuclear Licensing Engineer
R. Clement, Nuclear Training Instructor
R. Plappert, Compliance Supervisor
A. Schramm, Unit 1 Plant Superintendent
D. Powers, Shift Technical Advisor
G. Moore, Unit 1 Assistant Plant Superintendent
H. Shutter Onsite Nuclear Licensing Lead Engineer
D. Axline, Onsite Nuclear Licensing Engineer
M. Kirby, Simulator Project Engineer
R. Kratz, Nuclear Training Instructor

3. Written Examination

The written examinations were administered on April 27, 1992. Following the administration of the written examinations the facility training staff was given copies of the examinations as administered. The facility provided comments on the examinations. Twenty nine comments were received for the Reactor Operator examination and twelve for the Senior Reactor Operator examination. An extensive regional review of the comments was conducted. Eleven comments were in common between the two examinations. The review of the comments resulted in eleven deletions of questions and fifteen changes to the answer key to the Reactor Operator examination. Two questions were deleted from the Senior Reactor Operator examination and nine changes were made to the answer key.

The modified examination was reviewed to determine whether or not it remained valid. The NRC concluded that the examination was valid. Specifically it met the criteria for validity as described in Examiner Standard ES-403. The resolution of the facility comments is attached as an enclosure to this report.

All seven Senior Reactor Operator candidates passed the written examination. Nine Reactor Operator candidates passed the written examination and five Reactor Operator candidates failed the written examination.

4. Operating Examination

Job Performance Measures/Walkthrough Examination

The Job Performance Measures (JPM) Examinations were conducted over two weeks from May 1 to May 9, 1992. All Reactor Operator candidates passed the JPM portion of the operating examinations. One Senior Reactor Operator candidate failed the JPM portion of the operating examinations and six Senior Reactor Operator candidates passed the examinations. The Chief Examiner noted that some candidates experienced some problems with responding to a loss of Component Cooling Water to the Charging Pumps. One candidate failed the JPM, and another utilized a procedure different than anticipated in the JPM. One other candidate experienced considerable difficulty in identifying the applicable procedure and then following the actions required. The examiners concluded that the operators were generically weak in responding to this scenario.

The candidates collectively had some difficulty with administrative questions on Section A of the operating examination. The examiners noted that the candidates relied heavily on facility references and often experienced confusion about facility requirements. One examination failure occurred, due to weak performance on Section A of the operating examination.

Simulator Examinations

The simulator used for this examination had been recently turned over to the licensee from the manufacturer. The machine had been available only since February for the purposes of training. During the interval prior to the examinations the licensee had made numerous changes to the Emergency Operating Instructions (EOI). These changes occurred frequently and complicated the preparation of simulator scenarios that reflected the EOI procedures the candidates were being trained to use. The reason for these changes was that the validation of the Ace had not been done with a plant specific simulator that could have provided operating characteristics. Consequently, the Ace reflected operating characteristics more suited to larger Westinghouse designs rather than the San Onofre Unit 1 design. The Chief Examiner reviewed the changes to the Ace and found them to be appropriate.

The simulator examinations were conducted on April 28 - 30, 1992. All

of the candidates passed the simulator examinations. During the simulator examinations the examiners noted some concerns regarding command and control of crew activities. Two of the SRO instant candidates experienced some problems coordinating and directing the activities of the crews. This was not apparent with the SRO upgrade candidates. This indicates that training efforts in this area may have to be improved. The RO candidates communicated well and listened well to the directions of the SRO candidates.

The candidates also experienced some problems implementing the immediate actions of the Ace. The most apparent example of this was during the verification actions of SO1-1.0-10, "Reactor Trip or Safety Injection." Several crews only fortuitously noticed that an incomplete Containment Isolation had occurred. The examiners did not observe a conscious effort on the part of the candidates to verify this parameter. Nevertheless, all of the candidates passed this portion of the operating examination.

5. Reference Material

The examiners experienced problems in preparing the examinations from the reference material provided. No control logic diagrams were provided. The licensee indicated that these were not available. The reference material did not include elementary diagrams that could have substituted for the control logic diagrams. Because of these deficiencies the examiners had to rely on the system descriptions which were provided. The facility system descriptions were not completely up to date. During the resolution of the facility comments on the written examinations, it also became apparent that some reference material was not provided. That new material could have averted some of the errors that occurred during the preparation of the RO written examination.

6. Exit Meeting

The NRC representatives met with the persons identified in Paragraph 2 on May 8, 1992. The Chief Examiner summarized the preliminary results of the examinations to date. He also stated that the results would await the final grading of the written examinations by the NRC examiners.

As Given

Nuclear Regulatory Commission
Operator Licensing
Examination

RO Test
and
Answer Key

This document is removed from
Official Use Only category on
date of examination.

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
REACTOR OPERATOR LICENSE
REGION 5

CANDIDATE'S NAME: _____

FACILITY: San Onofre 1

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 92/04/27

INSTRUCTIONS TO CANDIDATE:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	CANDIDATE'S SCORE	%	
100.00		___ %	TOTALS
	FINAL GRADE		

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

_____
Candidate's Signature

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

MULTIPLE CHOICE

001	a	b	c	d	___
002	a	b	c	d	___
003	a	b	c	d	___
004	a	b	c	d	___
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006	a	b	c	d	___
007	a	b	c	d	___
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043	a	b	c	d	___
044	a	b	c	d	___
045	a	b	c	d	___

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

046	a	b	c	d	___
047	a	b	c	d	___
048	a	b	c	d	___
049	a	b	c	d	___
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087	a	b	c	d	___
088	a	b	c	d	___
089	a	b	c	d	___
090	a	b	c	d	___
091	a	b	c	d	___

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

092	a	b	c	d	___
093	a	b	c	d	___
094	a	b	c	d	___
095	a	b	c	d	___
096	a	b	c	d	___
097	a	b	c	d	___
098	a	b	c	d	___
099	a	b	c	d	___
100	a	b	c	d	___

(********** END OF EXAMINATION **********)

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done after you complete the examination.

3. Restroom trips are to be limited and only one applicant at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

4. Use black ink or dark pencil ONLY to facilitate legible reproductions.

5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet and each answer sheet.

6. Mark your answers on the answer sheet provided. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

7. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.

8. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

9. The point value for each question is indicated in parentheses after the question.

10. Show all calculations, methods, or assumptions used to obtain an answer to any short answer questions.

11. Partial credit may be given except on multiple choice questions. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

12. Proportional grading will be applied. Any additional wrong information that is provided may count against you. For example, if a question is worth one point and asks for four responses, each of which is worth 0.25 points, and you give five responses, each of your responses will be worth 0.20 points. If one of your five responses is incorrect, 0.20 will be deducted and your total credit for that question will be 0.80 instead of 1.00 even though you got the four correct answers.

13. If the intent of a question is unclear, ask questions of the examiner only.

14. When turning in your examination, assemble the completed examination with examination questions, examination aids and answer sheets. In addition, turn in all scrap paper.

15. Ensure all information you wish to have evaluated as part of your answer is on your answer sheet. Scrap paper will be disposed of immediately following the examination.

16. To pass the examination, you must achieve a grade of 80% or greater.

17. There is a time limit of four (4) hours for completion of the examination.

18. When you are done and have turned in your examination, leave the examination area (EXAMINER WILL DEFINE THE AREA). If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION: 001 (1.00)

WHICH ONE (1) of the following is the ALTERNATE AC power supply for the Rod Control System?

a. MCC #1

b. MCC #2

c. MCC #3

d. MCC #4

QUESTION: 002 (1.00)

Given the following:

- Turbine Runback from Underfrequency has reduced power to 60% (Overshoot on turbine controls)
- Rod Control System in manual for duration of runback
- 10.5 degree F Tavg-Tref mismatch

WHICH ONE (1) of the following rod speeds would be observed if the Rod Control Selector switch was taken to the AUTO position? (Figure 5 of SD-SO1-400, "Rod Control System" is attached)

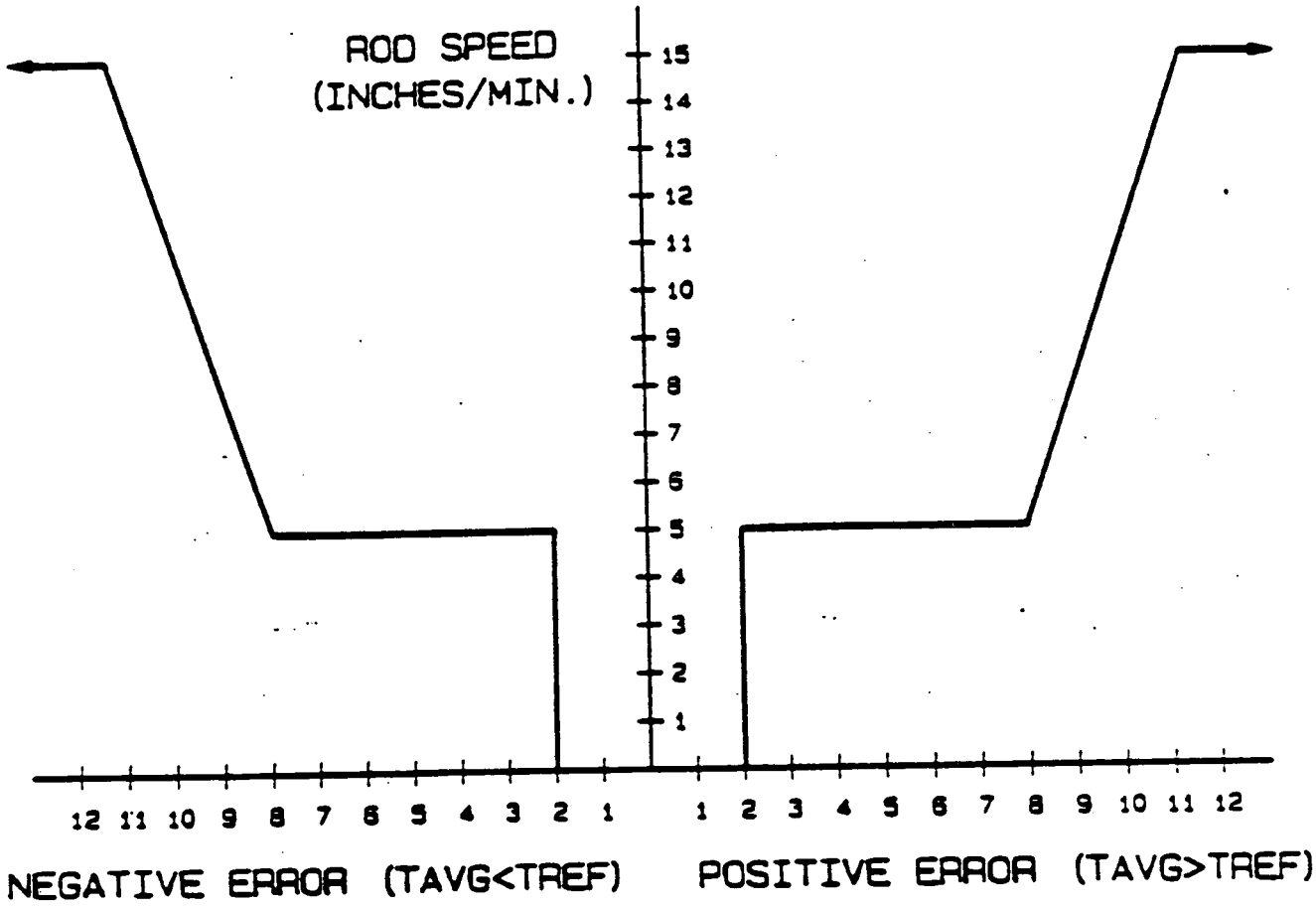
a. 8.75 inches/min

b. 9.25 inches/min

c. 11.25 inches/min

d. 13.0 inches/min

FIGURE 5: ROD CONTROL PROGRAM



QUESTION: 003 (1.00)

Given the following:

- Power level 80%
- Pressurizer level is at program setpoint.
- Tavg recorder pens all indicate approximately 549 degrees F.
- Tavg/Tref are matched on recorder TR-405.
- Rods start stepping in at high speed.
- No turbine runback is in progress.

WHICH ONE (1) of the following would cause these indications?

a. TM-407 - Avg Tave module to Rod Control loss of power.

b. TE-401A, Loop A hot leg temperature failed high.

c. TM-405 - Avg Tave summing computer failed low.

d. PT-415, 1st Stage pressure failed high.

QUESTION: 004 (1.00)

WHICH ONE (1) of the following constitutes an "operable low temperature overpressure protection system", for low temperature RCP starts?

a. One PORV with a lift setting of less than or equal to 360 psig.

b. Two PORV's with a lift setting of less than or equal to 400 psig

c. RHR relief valve RV-206 aligned to the RCS with a lift setting of less than or equal to 515 psig

d. A Reactor Coolant System vent of greater than or equal to 1.75 square inches.

QUESTION: 005 (1.00)

Given the following:

- Plant is in Mode 6.
- RCP "B" motor is uncoupled from the pump.
- Loop "B" is full.
- Maintenance is working on the RCP "B" pump.

WHICH ONE (1) of the following prevents leakage of reactor coolant up the RCP shaft?

a. Pump shaft mates with the top of the thermal barrier assembly.

b. Seal Leakoff collects any RCS leakage up the shaft and directs it back to the VCT.

c. Seal injection is maintained.

d. Nozzle dam installation.

QUESTION: 006 (1.00)

Given the following:

Make-up to the RCS has increased to 95 gpm, and ONLY the following alarms are received:

"RC PUMP A NO. 1 SEAL LOW DELTA P"
"RC PUMP A NO. 1 SEAL LEAKOFF FLOW OFF NORMAL"
"RC PUMP A SEAL WATER HI FLOW"
"RC PUMP A NO. 2 SEAL HI FLOW"

WHICH ONE (1) of the following has occurred to the A RCP?

a. #1 seal has failed.

b. #1 and #2 seals have failed.

c. All the seals have failed.

d. Seal injection has failed.

QUESTION: 007 (1.00)

WHICH ONE (1) of the following actions will move Control Bank II in MANUAL utilizing the In-Hold-Out switch? Assume the Reactor is at 92% power with rod control in automatic.

a. Place the Group Selector Switch in the "2" position, and pull it out.

b. Place the Overlap Cutout Switch in the "Bank 2" position, and the Group Selector Switch in the "Manual" position.

c. Place the Overlap Cutout Switch in "Override" and the Group Selector Switch in the "Bank 2" position.

d. Place the Overlap Cutout Switch in the "Overlap" position, and pull it out.

QUESTION: 008 (1.00)

WHICH ONE (1) of the following indications are printed on YR-404 "Rod Position Recorder" with the Rod Position Recorder Selector Switch in the Control Bank 2 position?

a. - Individual rod positions of the 17 rods in CBI by LVDT signal.
   - Control Bank 1 and 2 positions from the Digital Indication System.

b. - Control Bank 1 and 2 bank positions from the Digital Indication System.
   - Calculated Shutdown Margin Bank 1 and 2 LO alarm setpoints from the Reactor Protection and Control System.

c. - Individual rod positions of the 17 rods in CBII from the individual digital detectors in containment.
   - Calculated Shutdown Margin Bank 1 and 2 LO alarm setpoints from the Reactor Protection and Control System.

d. - Individual rod positions of the 17 rods in CBI AND CBII by LVDT signal.
   - Control Bank 1 and 2 positions from the Digital Indication System.

QUESTION: 009 (1.00)

Given the following:

- Plant has been shutdown for 6 days.
- RCS temperature is 145 degrees F.
- RCS pressure is 340 psig.
- Pressurizer level is 62%.
- S/G's have been in wet layup for 2 days.
- Tag on Loop "A" RCP control switch has 190 degrees F written on it.
- SRO Ops Supervisor directs the RO to start the Loop "A" RCP.

WHICH ONE (1) of the following allows the start of the Loop "A" RCP.

a. Pressure is adequate for D/P across the #1 Seal.

b. Secondary temperature can be determined from steam pressure.

c. Adequate time for temperature gradients to dissipate has passed.

d. RCP shutoff temperature is within allowable value.

QUESTION: 010 (1.00)

Given the following conditions:

- RCS pressure is 350 psig
- South Charging Pump is operable
- North Charging Pump is selected to be made inoperable, normal ACB has been racked out and tagged.

WHICH ONE (1) of the following assures that the North Charging Pump will NOT start?

a. Transfer switch A4S1-2 DSD OPEN and transfer switch A4S1-1 NORMAL CLOSED and tagged.

b. Transfer switch A4S1-1 NORMAL CLOSED and tagged and transfer switch A4S1-2 DSD CLOSED and tagged.

c. Transfer switch A4S1-2 DSD CLOSED and tagged and transfer switch A4S1-1 NORMAL OPEN.

d. Transfer switch A4S1-2 DSD OPEN and transfer switch A4S1-1 NORMAL OPEN and tagged.

QUESTION: 011 (1.00)

WHICH ONE (1) of the following will automatically occur when VCT level decreases to less than 20%?

a. MOV-1100 B & E will close and MOV-1100 C & D will open.

b. MOV-1100 C & D will close and MOV-1100 B & E will open.

c. MOV-1100 B & D will close and MOV-1100 C & E will open.

d. MOV-1100 C & E will close and MOV-1100 B & D will open.

QUESTION: 012 (1.00)

WHICH ONE (1) of the following occurs at "DIESEL GENERATOR BREAKER CLOSED +2 (TWO) SECONDS" after receipt of a SIS/LOP signal?

a. TCV-601 A & B, CCW to RHR Heat exchangers close.

b. Sequencer output to Load Group "B".

c. Feedwater pump recirc to condenser closes.

d. CCW pumps start.

QUESTION: 013 (1.00)

WHICH ONE (1) of the following represents the MINIMUM shutoff head for the Main Feed Pumps when they are operating in the Safety Injection mode?

a. 1170 psig

b. 1200 psig

c. 1230 psig

d. 1260 psig

QUESTION: 014 (1.00)

Given the following conditions:

- Containment pressure 1.6 psig
- SI has been reset

WHICH ONE (1) of the following is the MINIMUM action that will allow CV-525, (Letdown isolation valve) to open.

a. CV-525 (Letdown isolation valve), valve control switch turned to OPEN.

b. Reset CIS, CV-525 (Letdown isolation valve), control switch turned to OPEN.

c. Depress associated CIS Override and take CV-525 (Letdown isolation valve), control switch to OPEN.

d. Reset CIS, depress associated CIS Override and take CV-525 (Letdown isolation valve), control switch to OPEN.

QUESTION: 015 (1.00)

WHICH ONE (1) of the following describes the response of the CVCS system following SIS actuation?

a. Charging flow control valve FCV1112 opens, Letdown isolates, and Seal Injection valve position does not change.

b. Charging isolates, Letdown isolates, and FCV-1115 D,E, &F (Seal Injection valves) position does not change.

c. Charging flow control valve FCV1112 opens, Letdown valve positions does not change, and Seal Injection valve position does not change.

d. Charging isolates, Letdown isolates, and Seal Injection isolates.

QUESTION: 016 (1.00)

Given the following conditions:

The unit tripped from 92% power. During the trip transient the following conditions are experienced:
- 4KV bus 1C bus voltage dropped to 3900 volts.
- Primary plant pressure dropped to 1752 psig.
- Containment pressure increased to 1.6 psig.

WHICH ONE (1) of the following ESF actuations will occur?

a. CSAS

b. LOP

c. SIS

d. SIS/LOP

QUESTION: 017 (1.00)

Given the following:

- Reactor power is 92%
- The reactor has been operating at this power level for the past 2 weeks.
- A 6% flux tilt has been present for the past three hours.
- Reactor Engineering has determined that the tilt is due to rod misalignment and that a power reduction is prudent.
- The nuclear overpower trip setpoint is currently set at 108%.

WHICH ONE (1) of the following actions must be performed per SO1-2.1-14, "Nuclear Flux Tilt"? (See copy of Pg 4-5 of SO1-2.1-14, "Nuclear Flux Tilt" attached).

a. Reactor power must be reduced to 92% and the Nuclear Overpower trip setpoint must be reduced to 107%

b. Reactor power must be reduced to 88% and the Nuclear Overpower trip setpoint must be reduced to 102%

c. Reactor power must be reduced to 85% and the Nuclear Overpower trip setpoint must be reduced to 93%

d. Reactor power must be reduced to 50% and the Nuclear Overpower trip setpoint must be reduced to 55%

NUCLEAR FLUX TILT

OPERATOR ACTIONS

STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

PLANT CONDITIONS	REQUIRED ACTIONS
Reactor Power is greater than 50% and Flux Tilt is greater than 5%	a. Immediately notify Reactor Engineering. b. STOP power ascension, if in progress. c. REDUCE the Nuclear Overpower Trip setpoint 1% on each channel for each 1% Flux Tilt in excess of 5% (Tech. Spec. Table 2.1). d. Perform the following at least once per hour until Flux Tilt is reduced to less than 2%, or Reactor Power is reduced to less than 50%: 1) RECORD percent power, detector currents, and flux tilt on Attachment 1. e. If the Flux Tilt is due to rod misalignment, then reduce Reactor Power per SO1-3-8, Power Operations, at least 1.5% from rated thermal power for each 1% Flux Tilt within 30 minutes. 1) If appropriate, refer to SO1-2.3-1, Control Rod System Malfunctions. f. If Flux Tilt is not reduced to less than 2% within 2 hours after exceeding 2%, then perform the following: 1) REDUCE Reactor Power per SO1-3-8, Power Operations, to less than 50% within the following 2 hours, and 2) REDUCE Nuclear Overpower Trip setpoints to less than 55% within the following 4 hours. g. GO TO STEP 7.
Reactor Power less than 50% and Flux Tilt is greater than 4% and less than or equal to 5%	a. Immediately notify Reactor Engineering. b. STOP power ascension, if in progress. c. GO TO STEP 7.
Reactor Power less than 50% and Flux Tilt is greater than 5%	a. Immediately notify Reactor Engineering. b. STOP power ascension, if in progress. c. REDUCE the Nuclear Overpower Trip setpoint 1% on each channel for each 1% Flux Tilt in excess of 5% (Tech. Spec. Table 2.1). d. GO TO STEP 7.

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NUCLEAR FLUX TILT

OPERATOR ACTIONS

STEP ACTION/EXPECTED RESPONSE RESPONSE NOT OBTAINED

PLANT CONDITIONS	REQUIRED ACTIONS
Reactor Power greater than 50% and Flux Tilt greater than 4% and less than or equal to 5%.	a. Immediately notify Reactor Engineering. b. STOP power ascension, if in progress. c. Perform the following at least once per hour until Flux Tilt is reduced to less than 2%, or Reactor Power is reduced to less than 50%: 1) RECORD percent power, detector currents, and flux tilt on Attachment 1. d. Within 2 hours, either: 1) REDUCE Flux Tilt to less than 2%, or 2) REDUCE power per SO1-3-8, Power Operations, at least 1.5% from Rated Thermal Power for each 1% Flux Tilt, and REDUCE the Nuclear Overpower Trip setpoint 1.5% on each channel for each 1% Flux Tilt. e. If Flux Tilt is not reduced to less than 2% within 24 hours of exceeding 2%, then perform the following: 1) REDUCE Reactor Power per SO1-3-8, Power Operations, to less than 50% within the following 2 hours, and 2) REDUCE the Nuclear Overpower Trip setpoints to less than or equal to 55% within the following 4 hours. f. GO TO STEP 7.

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QUESTION: 018 (1.00)

Given the following:

- Reactor power is 8%.
- Plant startup is in progress.
- N-1203 - Intermediate Range channel 1 has failed.
- I&C is troubleshooting the drawer.
- I&C technician removes the control power fuses from the N-1203 instrument drawer.

WHICH ONE (1) of the following occurs when the fuses are pulled?

a. N-1203 bistables de-energize, reactor trips.

b. N-1203 indication lost at J console and remote shutdown panel, no trip occurs.

c. Instrument power supplies bistables, no trip occurs.

d. Source range re-energizes, reactor trips.

QUESTION: 019 (1.00)

WHICH ONE (1) of the following will cause immediate actuation of both trains of Containment Spray?

a. Containment pressure 7 psig on 1/3 containment pressure detectors, SI signal present.

b. Containment pressure 7 psig on 2/3 containment pressure detectors, SI and LOP signal are present.

c. CSAS Train A and B pump pushbuttons depressed.

d. CSAS Train A and B pump and valve pushbuttons depressed.

QUESTION: 020 (1.00)

WHICH ONE (1) of the following is the reason for maintaining a Nitrogen blanket on the Containment Spray Hydrazine Storage Tank?

a. Ensures proper metering of hydrazine flow into the Containment Spray system.

b. Minimize noncondensibles in containment following spray actuation.

c. Minimize hydrazine degradation from oxygen absorption during long term storage.

d. Provide overpressure to ensure that entire contents of tank are available following spray actuation.

QUESTION: 021 (1.00)

Given the following:

The plant in mode 1, with Unit load at 9%.

WHICH ONE (1) of the following is the MINIMUM acceptable number of running main feed pumps and condensate pumps?

	# of running MFP	# of running Cond. pumps
a.	1	1
b.	2	2
c.	1	2
d.	2	4

QUESTION: 022 (1.00)

Given the following:

- Reactor is at 11% power during Reactor Physics testing.
- Auxiliary feedwater is in service.
- Wide range S/G level has decreased to 254" in all 3 Steam Generators.
- The RO is told to establish MAXIMUM AFW flow to the Steam Generators.

WHICH ONE (1) of the following is the MAXIMUM AFW flow rate the RO may establish?

a. 25 gpm per S/G

b. 125 gpm per S/G

c. 300 gpm total flow

d. 450 gpm total flow

QUESTION: 023 (1.00)

Given the following:

- Reactor at 60% power.
- RO accidentally manually actuates AFW.

WHICH ONE (1) of the following describes the results of the manual AFW actuation?

a. Train B starts, delivers flow. Main Turbine does not trip

b. Train B starts, delivers flow, Main Turbine trips.

c. Train A starts, delivers flow, Main Turbine does not trip.

d. Train A starts, delivers flow, Main Turbine trips.

QUESTION: 024 (1.00)

WHICH ONE (1) of the following will result in flow from an automatic actuation of Train "A" Auxiliary Feedwater?

a. 2 of 3 S/G levels less than 5% on Train "A", AND no flow on Train B Aux Feed.

b. 2 of 2 level transmitters less than 5% on 1 of 3 S/G's, AND no flow on Train B Aux Feed.

c. 2 of 3 S/G low level, AND low pressure on Train B Aux Feed.

d. 2 of 3 low level on 1 of 3 S/G, AND low pressure on Train B Aux Feed.

QUESTION: 025 (1.00)

WHICH ONE (1) of the following will allow CV-110 (liquid release control valve) to open or remain open?

a. RE-1218 in alarm and Circulating water and Monitor tank pumps running and "FAIL/RESET" depressed on RE-1218

b. RE-1218 in alarm and Circulating water pump breaker racked to "TEST" and Monitor tank pump running and Salt Water cooling pump running.

c. RE-1218 NOT in alarm, RWS Switch and Purge Mode Switch in "On" and Monitor tank and Circulating water pumps running.

d. RE-1218 NOT in alarm, RWS Switch and Purge Mode Switch in "Off" and Monitor tank and Circulating water pumps running.

QUESTION: 026 (1.00)

WHICH ONE (1) of the following, if present in a sample from the Decontamination Drain Tank, would allow the operator to pump the tank to the Holdup tank?

a. Gross activity in excess of MDA.

b. 35 ppm Oil

c. Colored water

d. Foam (phosphates)

QUESTION: 027 (1.00)

WHICH ONE (1) of the following radiation monitors has automatic actions associated with its alarm?

a. R-1215 - Air Ejector Radiation Monitor

b. R-1212 - Containment Gas Radiation Monitor

c. R-1220 - Stack Iodine Radiation Monitor

d. R-1257 - Main Steam Line Radiation Monitor

QUESTION: 028 (1.00)

WHICH ONE (1) of the following describes the point of attachment of the pressurizer surge and spray lines?

	SURGE LINE	SPRAY LINES
a.	Loop A cold leg	Loop A and C hot legs.
b.	Loop A hot leg	Loop A and B cold legs
c.	Loop B cold leg	Loop A and C hot legs
d.	Loop B hot leg	Loop A and B cold legs

QUESTION: 029 (1.00)

WHICH ONE (1) of the following will trip the Safety Injection pumps during an SI?

a. Time delay undervoltage OR Loss of DC power

b. Time delay overcurrent OR Loss of DC power

c. Time Delay overcurrent OR Low-low RWST level

d. Loss of DC power OR Low-low RWST level

QUESTION: 030 (1.00)

WHICH ONE (1) of the following meets the MINIMUM criteria for enabling the Overpressure Mitigation System?

	PRESSURIZER LEVEL	RCS PRESSURE	RCS TEMPERATURE
a.	60%	400 psig	365 degrees F
b.	50%	465 psig	360 degrees F
c.	40%	410 psig	350 degrees F
d.	30%	430 psig	320 degrees F

QUESTION: 031 (1.00)

WHICH ONE (1) of the following is the reason for post accident injection of hydrazine into the Containment Spray system?

a. Containment oxygen removal.

b. Containment iodine removal.

c. Containment pH control.

d. Containment spray piping integrity.

QUESTION: 032 (1.00)

WHICH ONE (1) of the following reflects the response of LC-430F, (Pressurizer level controller) as power increases from 15% to 85%?

a. Output decreases as power increases.

b. Output increases until 50% power.

c. Output increases to 85% power.

d. Maintains 35% output continuously.

QUESTION: 033 (1.00)

Given the following:

- 92% power operations.
- Pressurizer level controls are in AUTO-CASCADE
- PZR pressure controls are in AUTOMATIC.
- LC430F fails HIGH.
- NO operator actions are taken.

WHICH ONE (1) of the following is the response of the pressurizer control/reactor protection systems to this failure?

a. Backup heaters energize, actual level decreases, letdown isolates, heaters deenergize, plant trips on LOW pressure.

b. Backup heaters energize, actual level increases, pressure increases, sprays actuate, plant trips on HIGH level.

c. Charging flow decreases, level decreases, letdown isolates, heaters deenergize, plant trips on HIGH level.

d. Charging flow increases, actual level increases, backup heaters energize, plant trips on HIGH pressure.

QUESTION: 034 (1.00)

WHICH ONE (1) of the following permissives prevents MANUAL and AUTOMATIC rod withdrawal?

a. P-1 - Overpower Rod Stop

b. P-2 - Low Power Cutout

c. P-3 - Rod Drop Rod Stop

d. P-5 - Shutdown Margin Alarm

QUESTION: 035 (1.00)

Given the following:

- Reactor power is 100%.
- VLPT setpoint display is inoperable
- VLPT setpoint formula is 26.15(.894 Delta T + Tavg) -14341

WHICH ONE (1) of the following is the current calculated VLPT trip setpoint?

a. 1156 psi

b. 1180 psi

c. 1203 psi

d. 1872 psi

QUESTION: 036 (1.00)

WHICH ONE (1) of the following provides the power level input to the shutdown margin computer?

a. Auctioneered high Power Range NI's.

b. Auctioneered high Delta T.

c. Auctioneered high Tavg.

d. Auctioneered high Tref.

QUESTION: 037 (1.00)

Given the following:

-Turbine tripped
-Primary pressure at 2020 psig
-T ave at 550 degrees
-Normal post-trip steam generator levels and pressures

WHICH ONE (1) of the following signals does the Steam Generator Water Level Control switching chassis direct to the regulating valve controller?

a. Feed flow controller output

b. 5% open signal

c. Full closed signal

d. Steam flow/feed flow error

QUESTION: 038 (1.00)

Given the following:

-An accident has occurred which required containment spray.
-The RWST is depleted, and recirculation flow is established.

WHICH ONE (1) of the following is the required position for the spray limiter orifice isolation valves?

	CV-517	CV-518
a.	open	open
b.	closed	open
c.	open	closed
d.	closed	closed

QUESTION: 039 (1.00)

Given the following:

- Spent fuel pit temperature is 135 degrees F and increasing.
- The loss of cooling has been traced to failure of the Spent Fuel heat exchanger.
- The heat exchanger cannot be repaired in a timely manner.

WHICH ONE (1) of the following is the preferred alternate source of Spent Fuel Pool Cooling?

a. CCW HX E-20-A (TOP)

b. CCW HX E-20-B (BOTTOM)

c. Recirculation HX CRS-E-11

d. Fire Protection System via fire hoses

QUESTION: 040 (1.00)

WHICH ONE (1) of the following conditions would prevent the Emergency Diesel Generator from starting automatically on a LOP?

a. Jacket water temperature of 210 degrees F.

b. Turbo oil pressure of 15 psig

c. Starting air pressure 145 psig

d. Lube oil temperature 90 degrees F.

QUESTION: 041 (1.00)

WHICH ONE (1) of the following is added to the Emergency Diesel Generator cooling water as a corrosion inhibitor?

a. Calgon PGH

b. Nalco 39M Nitrite

c. Potassium Chromate

d. Molybdates

QUESTION: 042 (1.00)

WHICH ONE (1) of the following statements describes how the Emergency Diesel Generator controls are reset to assure the correct voltage will be obtained upon engine start?

a. The operator nulls the Manual Voltage Adjust to match the setting of the Auto Voltage Control after engine shutdown.

b. The voltage control setting is fixed and requires no auto or manual adjustment.

c. The Auto Voltage Adjust will reposition to the correct startup voltage for up to 25 seconds after engine shutdown.

d. The Auto Voltage Adjust adjusts to 4160 volts at all times automatically.

QUESTION: 043 (1.00)

WHICH ONE (1) of the following statements describes when the Emergency Diesel Generator Standby Lube Oil Pump should be run?

a. Continuously when the diesel is shutdown to maintain engine pre-lube and temperature.

b. Started prior to engine start and run for 5 minutes to pre-lube the diesel.

c. Started prior to engine start to pre-lube the engine, but must be shutdown within 5 minutes after engine start.

d. Started prior to engine shutdown and allowed to run for approximately 1 minute following engine shutdown.

QUESTION: 044 (1.00)

WHICH ONE (1) of the following will cause an alarm to actuate in the State Office of Emergency services?

a. Wide Range Gas Monitor (R-1254) is at its high level alarm setpoint for greater than 10 minutes and the Auto Alert System Keyswitch in the AUTO position.

b. Wide Range Gas Monitor (R-1254) is at its high level alarm setpoint for greater than 10 minutes and the Auto Alert System Keyswitch in the OPERATE position.

c. Wide Range Gas Monitor (R-1254) is at its high level alarm setpoint for greater than 15 minutes and the Auto Alert System Keyswitch in the AUTO position.

d. Wide Range Gas Monitor (R-1254) is at its high level alarm setpoint for greater than 15 minutes and the Auto Alert System Keyswitch in the OPERATE position.

QUESTION: 045 (1.00)

WHICH ONE (1) of the following is the required method to align backup nitrogen to the PORV's and their block valves during a Dedicated Safe Shutdown condition?

a. Depressing the Contaiment Isolation OVERRIDE PB for CV-532, Containment Isolation valve for Nitrogen to the PORV's.

b. Depressing the Contaiment Isolation OPEN PB for CV-532, Containment Isolation valve for Nitrogen to the PORV's.

c. Locally OPENING the manual bypass around CV-532, Containment Isolation valve for Nitrogen to the PORV's.

d. Locally OPENING CV-532, Containment Isolation valve for Nitrogen to the PORV's.

QUESTION: 046 (1.00)

WHICH ONE (1) of the following is considered to be a "critical station function" requiring service air?

a. Cryogenic treatment room usage

b. TPCW Heat exchanger tube cleaning

c. Auxiliary saltwater cooling pump priming

d. Transfer pool upender testing

QUESTION: 047 (1.00)

Given the following:

-Reactor Coolant System is vented.
-Reactor Coolant System at 110 degrees F.
-Pressurizer level is at 50%.

WHICH ONE (1) of the following plant conditions will satisify the applicable LCO for RHR?

a. One RHR train is OPERABLE and in operation, with all steam generators at 180 inches.

b. One RHR train is OPERABLE and in operation, with 2 steam generators at 280 inches.

c. Two reactor coolant pumps are in operation.

d. Two RHR trains are OPERABLE and one is in operation.

QUESTION: 048 (1.00)

Given the following conditions:

- Plant is in Mode 4
- North Saltwater pump is running.
- CCW HX Outlet Valve, MOV-720B, is OPEN.
- The Reactor Operator takes CCW HX Outlet Valve, MOV-720B control switch to CLOSE.

WHICH ONE (1) of the following describes the response of MOV-720B to the CLOSE signal?

a. Valve does not respond to the CLOSE signal, it will remain full OPEN.

b. Valve moves in the CLOSED direction only while switch is held in the "CLOSE" position.

c. Valve will CLOSE and remain CLOSED.

d. Valve will CLOSE and then REOPEN.

QUESTION: 049 (1.00)

WHICH ONE (1) of the following describes the operation of the Main Turbine Overspeed Protection (OPC) Controller?

a. Actuates when turbine load is greater than 40% as sensed by 1st stage pressure and Reactor power is greater than 10% as sensed by Delta T.

b. Actuates when turbine load is greater than 50% as sensed by LP turbine inlet pressure and generator output is less than 20% as sensed by power relay.

c. Actuates when turbine load is greater than 50% as sensed by LP turbine inlet pressure and Reactor power is less than 20% as sensed by Power Range NI's.

d. Actuates when turbine load is greater than 40% as sensed by 1st stage pressure and generator output is greater than 10% as sensed by power relay.

QUESTION: 050 (1.00)

WHICH ONE (1) of the following is the pressure at which the Turbine Generator Emergency DC bearing oil pump will start?

a. 10 psig

b. 9 psig

c. 8 psig

d. 7 psig

QUESTION: 051 (1.00)

WHICH ONE (1) of the following turbine load limits is applicable when stop valve testing is in progress?

a. 210 Mwe

b. 250 MWe

c. 275 MWe

d. 300 MWe

QUESTION: 052 (1.00)

WHICH ONE (1) of the following would allow restart of a TPCW pump following a SIS/LOP?

a. TPCW pump handswitch taken to "STOP", then "AFTER STOP".

b. Dispatch a PEO to locally reset the breaker.

c. Depress the TPCW pump's OVERRIDE pushbutton.

d. Reset the Safety Injection Sequencer.

QUESTION: 053 (1.00)

WHICH ONE (1) of the following will cause an auto start of the standby Salt Water Cooling Pump which is in "PULL FOR AUTO"?

a. Overload trip of running Salt Water Cooling Pump.

b. Start of associated train Emergency Diesel Generator.

c. Start of associated train CCW Pump.

d. Low flow of 2500 gpm on running Salt Water Cooling Pump.

QUESTION: 054 (1.00)

Given the following:

- Reactor is at 92% steady state operation.
- SO1-2.3-1 "Control Rod System Malfunctions" in use.
- Operators are determining the operability of the RPI system.

WHICH ONE (1) of the following is the MINIMUM criteria requiring plant shutdown due to RPI INOPERABILITY per the Technical Specifications?

a. THREE (3) rod position indicators per bank inoperable.

b. TWO (2) rod position indicators per bank NOT capable of determining rod position within +/- 21 steps in a single bank.

c. LVDT current checks on more than one bank +/- 20 % of calibrated values.

d. ONE (1) step counter per bank NOT capable of determining rod position within +/- 21 steps in a single bank.

QUESTION: 055 (1.00)

WHICH ONE (1) of the following is the IMMEDIATE criteria for initiation of RCP/Reactor trip when responding to a loss of RCP motor cooling?

a. RO determination of severity.

b. RCP Oil bearing temperature 202 degrees F.

c. Containment area temperature 150 degrees F.

d. SRO determination of severity.

QUESTION: 056 (1.00)

Given the following:

- Reactor power 40%
- Loop "A" RCP trips
- All control systems are in automatic.

WHICH ONE (1) of the following is the IMMEDIATE expected plant response?

a. Reactor trips

b. Steam dumps open

c. Rods drive in

d. "A" S/G level decreases

QUESTION: 057 (1.00)

WHICH ONE (1) of the following is the MINIMUM required amount of emergency boration following a reactor trip with THREE (3) control rods NOT fully inserted?

a. 9% BAST level

b. 12% BAST level

c. 18% BAST level

d. 24% BAST level

QUESTION: 058 (1.00)

Given the following:

- The plant has been at 150 degrees F for 2 weeks.
- A Source Range High Flux Level alarm is received in the control room.
- The source range count rate is verified to be steadily increasing.

WHICH ONE (1) of the following actions should IMMEDIATELY be taken?

a. Immediately sample the RCS boron concentration, calculate the shutdown margin, and emergency borate as required.

b. Emergency borate 25% of the BAST, and terminate any dilution that maybe in progress.

c. Log the source range readings every 15 minutes.

d. Initiate action per SO1-1.1-2 "Potential Loss of Core Shutdown".

QUESTION: 059 (1.00)

WHICH ONE (1) of the following actions should be taken to verify that the DC RCP Thermal Barrier Pump is RUNNING during a loss of all AC power event?

a. Directly observe the pump to verify the shaft is turning and the suction and discharge pressures are normal.

b. Observe that the indicating light on the DC RCP Thermal Barrier Pump breaker in No. 2 DC Switchgear Room is lit.

c. Observe the Reactor Plant Annunciator #1 Panel on the west vertical board, "EMERGENCY DC RCP THERMAL BARRIER PUMP RUN" is lit.

d. Observe that the CCW discharge header pressure is being maintained greater than 20 psig on the main control board pressure gauge.

QUESTION: 060 (1.00)

WHICH ONE (1) of the following is the required operator action during response to an ATWS, if the operator cannot verify at least one charging pump running?

a. Start boric acid transfer pump in FAST speed.

b. Start the test pump.

c. Ensure CV-334, emergency boration valve, OPEN.

d. Manually initiate safety injection.

QUESTION: 061 (1.00)

Given the following:

- Steam line break inside containment has just occurred.
- SO1-1.0-30, "Loss of Secondary Coolant" is in use.
- AFW Pump G-10W RUNNING and G-10 in STANDBY.
- Core Exit Temperature is 395 degrees F.
- Containment pressure 6 psig.
- S/G "C" is faulted and dry.
- S/G's A/B are intact with NR level at 40%.

WHICH ONE (1) of the following is the APPROXIMATE allowable AFW flow to the non-faulted S/G's under these conditions?

a. 25 gpm

b. 50 gpm

c. 300 gpm

d. 450 gpm

QUESTION: 062 (1.00)

WHICH ONE (1) of the following parameters would allow the operator to distinguish between a feed line break and a steam line break?

a. Containment humidity and pressure

b. Steam line pressure

c. S/G level

d. Loop Delta T

QUESTION: 063 (1.00)

Given the following:

- Condenser backpressure is 5.6" Hg Abs.
- Unit load is 290 Mwe.

WHICH ONE (1) of the following is the required operator response to the above conditions? (Use Attachment 1 of SO1-2.4-3)

a. Reduce unit load

b. Manually start vacuum pump to adjust backpressure

c. Immediately trip the turbine

d. Immediately trip the reactor then the turbine.

QUESTION: 064 (1.00)

WHICH ONE (1) of the following will AUTOMATICALLY occur on a loss of the Utility Bus?

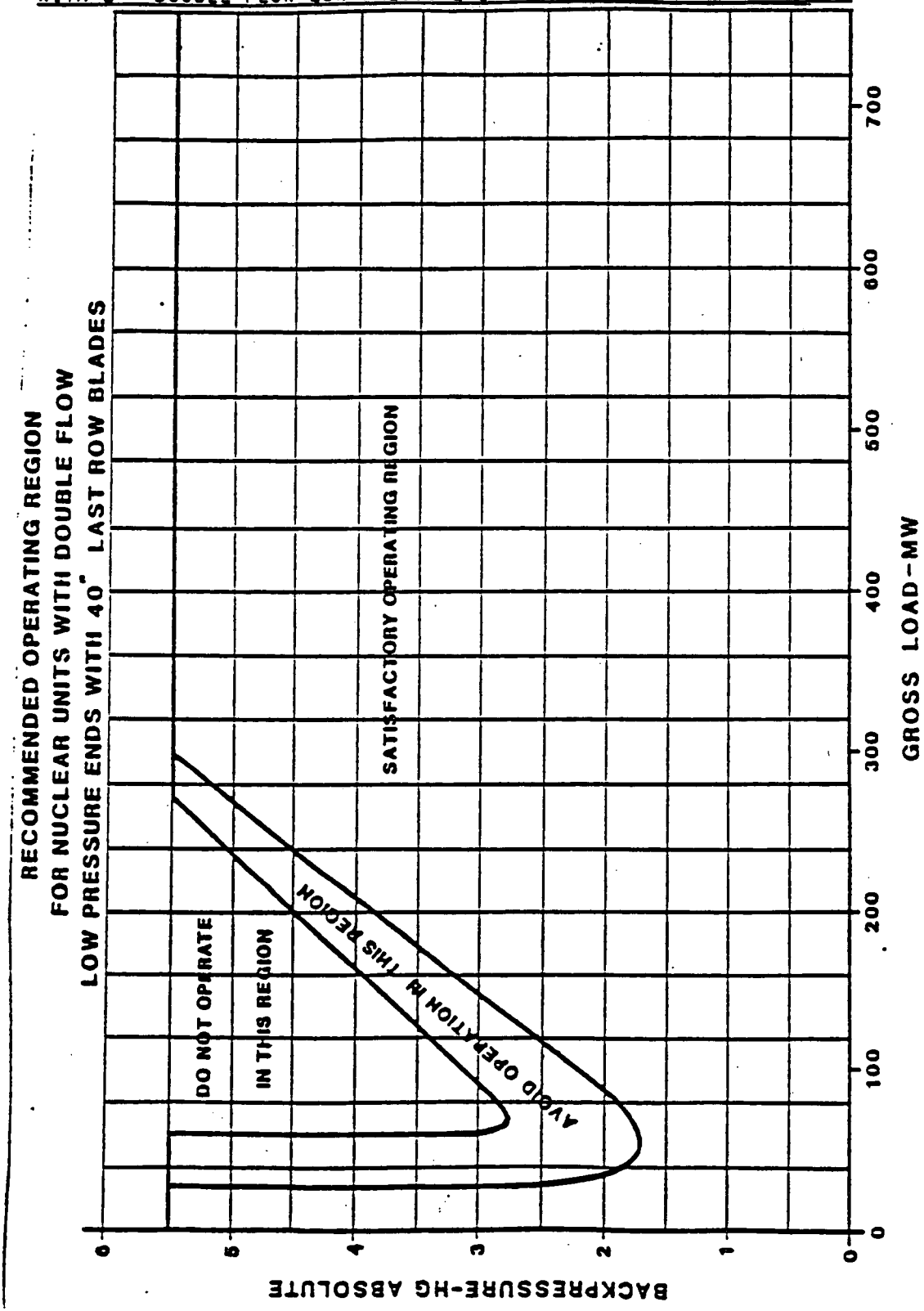
a. Pressurizer heaters trip OFF.

b. Charging and Letdown ISOLATE.

c. AFW Train "A" ACTUATES.

d. AFW Train "B" ACTUATES.

RECOMMENDED OPERATIONAL REGIONS FOR NUCLEAR UNITS
WITH 2 - DOUBLE FLOW LOW PRESSURE ENDS WITH 40" LAST ROW BLADES



QUESTION: 065 (1.00)

WHICH ONE (1) of the following plant conditions verifies adequate natural circulation cooling?

a. RCS subcooling 34 F.

b. RCS hot leg temperature trending with saturation temperature for Main Steam pressure.

c. Steam generator levels at or approaching 50% narrow range.

d. Core exit thermocouples stable and trending with RCS cold leg temperature.

QUESTION: 066 (1.00)

Given the following:

-Reactor power is 57%.
-The bus potential indicating light is OFF for Regulated Bus 3.

WHICH ONE (1) of the following automatic actions will occur?

a. FCV-458, Steam Generator C Feedwater Regulator Valve, fails OPEN.

b. Pressurizer heaters trip OFF when in AUTO.

c. PCV-430C and PCV-430H, Pressurizer Spray Valves, fail CLOSED.

d. FCV-1112, Charging Flow Control Valve, fails CLOSED.

QUESTION: 067 (1.00)

Given the following:

- Decision has been made to evacuate the control room.
- SO1-2.5-4, "Shutdown from Outside the Control Room" is in use.
- Reactor and turbine are tripped.

WHICH ONE (1) of the following are the remaining Immediate Actions performed by the control operator for SO1-2.5-4 "Shutdown From Outside The Control Room" prior to exiting the control room?

a. Isolate main feedwater, obtain radio prior to exit of control room.

b. Initiate aux feedwater.

c. Obtain radio prior to exit of control room.

d. Initiate emergency boration.

QUESTION: 068 (1.00)

Given the following:

- R-1218 Liquid Radwaste Effluent Monitor, is in a High Alarm condition.

WHICH ONE (1) of the following automatic actions will occur as a result of the high alarm?

a. R-1218 Sample Pump STOPS.

b. SV-1218-2 Purge valve CLOSES.

c. Yard Sump Pump TRIPS.

d. RWL-349, Sample Pump Suction valve CLOSES.

QUESTION: 069 (1.00)

Given the following:

- A remote manual containment isolation valves actuator is disconnected and inoperable.

WHICH ONE (1) of the following meets the MINIMUM required conditions to satisfy containment integrity per Tech Spec 3.6.2, Containment Integrity?

a. Upstream isolation valve closed and locked.

b. Upstream and downstream isolation valves closed and locked.

c. Qualified operator at the valve with a radio and appropriate valve operator tool.

d. Qualified mechanic at the valve with a radio and appropriate valve operator tool.

QUESTION: 070 (1.00)

WHICH ONE (1) of the following initiates core recovery by condensation, while responding to an inadequate core cooling event?

a. Dumping steam to atmosphere

b. Re-initiation of Safety Injection

c. Starting RCP's

d. Initiating Auxiliary Spray

QUESTION: 071 (1.00)

Given the following:

- The plant is at 25% power.
- 2 control rods have dropped to the bottom.

WHICH ONE (1) of the following actions should be taken?

a. Trip the Reactor and go to SO1-1.0-10, "Reactor Trip or Safety Injection".

b. Place rod control in manual, and use SO1-2.3-1 "Control Rod Malfunction" to recover the dropped rods.

c. Notify the on-shift STA and Reactor Engineering to make an evaluation for continued operation with rods on the bottom, and implement the action determined by the evaluation within 3 hours from the time the rods dropped.

d. Recover the dropped rods IF the rod recovery can be completed within 1 hour from the time the rods were dropped.

QUESTION: 072 (1.00)

WHICH ONE (1) of the following is the reason that a dropped rod must be recovered within THREE (3) hours ?

a. Xenon transient initiated by dropped rod at BOL could shutdown the reactor if not recovered within THREE hours.

b. Axial Flux Difference will be out of band if not recovered within THREE hours.

c. Violation of minimum Technical Specifications shutdown margin requirements will result if not recovered in THREE hours.

d. Fuel damage could result if not recovered in THREE hours.

QUESTION: 073 (1.00)

WHICH ONE (1) of the following is the MINIMUM required hydrazine flow rate if Containment Spray is required while performing SO1-1.0-10, "Reactor Trip or Safety Injection"?

a. 0.1 gpm

b. 0.4 gpm

c. 0.7 gpm

d. 1.0 gpm

QUESTION: 074 (1.00)

WHICH ONE (1) of the following would indicate that the PRT rupture disk had blown following a pressurizer PORV failing OPEN?

a. Pressurizer level decreasing

b. Relief line temperatures increasing

c. PRT temperature decreasing

d. PRT level low

QUESTION: 075 (1.00)

WHICH ONE (1) of the following is the criteria for verification of a Reactor trip per Step 1 of SO1-1.0-10, "Reactor Trip or Safety Injection"?

a. Both reactor trip breakers open, both turbine stop valves closed, and 4 KV buses 1C and/or 2C energized.

b. Both or either reactor trip breakers open, rod bottom indicator lights on, and neutron flux decreasing.

c. Both reactor trip breakers open, rods on the bottom, and neutron flux decreasing.

d. Both or either reactor trip breakers open, no more than ONE rod NOT fully inserted, and neutron flux decreasing.

QUESTION: 076 (1.00)

WHICH ONE (1) of the following parameters discriminates between a vapor space LOCA and a non-vapor space LOCA?

a. Pressurizer pressure

b. Pressurizer level

c. PRT pressure

d. Hot leg temperature

QUESTION: 077 (1.00)

Given the following:

- A Large break LOCA has occurred.
- Recirculation cooling mode has been in effect for 20 hours.
- Both trains of recirculation flow have been lost.
- RWST level is 5%.
- Operators are attempting to establish cold leg injection from an alternate suction supply while refilling RWST.

WHICH ONE (1) of the following is the PRIMARY alternate suction source for cold leg injection while refilling the RWST?

a. Unit 2/3 RWST

b. Unit 1 Radwaste Holdup Tank

c. Unit 1 Spent Fuel Pool

d. Unit 1 BAST and Primary Makeup Tank

QUESTION: 078 (1.00)

Given the following:

- The plant has been shutdown for 4 days and is at 140 F with RHR in service.
- A total loss of CCW occurs, with recovery of CCW projected to take 1 hour.
- The RCS is intact and not in midloop, with all systems available EXCEPT CCW.

WHICH ONE (1) of the following actions should be taken first?

a. Implement decay heat removal via primary system feed and bleed to radwaste using normal charging and letdown via RHR.

b. Implement decay heat removal via primary system feed and bleed to radwaste via PRT and RCS drain tank.

c. Implement decay heat removal via the primary system feed and bleed to containment.

d. Implement decay heat removal via the steam generators and AFW.

QUESTION: 079 (1.00)

Given the following:

- Mode 6, refueling complete.
- Lowering cavity level for head reinstallation.
- Loss of RHR occurs.
- Maintenance foreman wants to delay the containment closure for removal of equipment.

WHICH ONE (1) of the following core exit temperatures is the LIMIT when containment closure MUST be complete?

a. 140 degrees F.

b. 170 degrees F.

c. 200 degrees F.

d. 212 degrees F.

QUESTION: 080 (1.00)

WHICH ONE (1) of the following is required to be performed prior to placing the DTT (Diverse Turbine Trip) system in bypass during abnormal operations?

a. Perform Turbine trip test per SO1-12.9-3 "Offline Turbine Trip Test".

b. Direct PEO under SRO supervision to hold Main Turbine Overspeed Test lever in the "TEST" position while the DTT is removed from service.

c. A determination is made of the required compensatory action for lack of automatic response .

d. DTT System Trip test is completed.

QUESTION: 081 (1.00)

Given the following:

-The plant is in Mode 3.
-An Intermediate Range NI fails.

WHICH ONE (1) of the following actions should be taken to assure the associated Source Range Channel remains operable?

a. Pull the instrument fuses on the affected Intermediate Range Channel.

b. Place the HV Manual ON/OFF switch on the associated Source Range Channel in the "ON" position.

c. Place the Test Mode Selector Switch for the affected Intermediate Range Channel in the "FIXED" position.

d. Place the Operation Selector Switch for the affected Intermediate Range Channel in the "CPS LOCAL" position.

QUESTION: 082 (1.00)

Given the following:

- Power is 10 E-5%.
- Permissive annunciator #4, "S/U RATE TRIPS ACTIVE" is energized.
- N-1204, Intermediate Range level and SUR indications peg high.

WHICH ONE (1) of the following describes the response of the Nuclear Instrumentation/Reactor Protection system to this failure?

a. N-1204 High SUR trip bistable trips, no reactor trip occurs.

b. N-1204 High SUR trip bistable trips, reactor trip occurs.

c. N-1202 Source Range High volts de-energize, reactor trip occurs.

d. N-1201 Source Range High volts de-energize, no reactor trip occurs.

QUESTION: 083 (1.00)

WHICH ONE (1) of the following is the immmediate action required when steam generator tube leakage is identified?

a. Monitor leak rate to assure it does not exceed 10 gpm.

b. Monitor RCS pressure and pressurizer level to determine if a Reactor trip and Safety Injection is required.

c. Lower the affected steam generator level to assure the generator will not overfill during recovery.

d. Actuate or verify actuated S/G blowdown isolation.

QUESTION: 084 (1.00)

WHICH ONE (1) of the following is a reason for starting the RCP's in SO1-1.40, "Steam Generator Tube Rupture"?

a. Restore RCS inventory

b. Maintenance of subcooling

c. Minimize primary to secondary leak rate.

d. Eliminate reactor vessel head voiding

QUESTION: 085 (1.00)

Given the following conditions:

- Preparations are being made to release 2 liquid radioactive sources concurrently.
- R-1218 is in service.
- Release permits have been prepared for each source indicating that the sources can be released concurrently.

WHICH ONE (1) of the following conditions must be met prior to the start of the concurrent discharge?

a. Plant Manager approval, total number of MPC's of all sources released calculated to be less than ONE (1) prior to dilution.

b. Chemistry Department approval, total number of MPC's of all sources released calculated to be less than ONE (1) prior to dilution.

c. Plant Manager approval, total number of MPC's of all sources released calculated to be less than ONE (1) after dilution.

d. Chemistry Department approval, total number of MPC's of all sources released calculated to be less than ONE (1) after dilution.

QUESTION: 086 (1.00)

WHICH ONE (1) of the following is the reason for the difference in severity between a feedline break inside containment, upstream of the inside containment check valve, and a steam line break inside containment?

a. The feedline break will cause a less severe cooldown transient, and release less energy to the containment than a steamline break, due to the enthalpy difference between the feedwater and the steam.

b. The feedline break will cause the loss of feed to one steam generator, but no steam generator will blowdown and minimal RCS cooldown will occur.

c. The feedline break will cause one steam generator to blow down out the break, and steam the other generators dry. The energy lost, and cooldown associated with these occurrences is less severe than that caused by a steam line break.

d. The feedline break will result in an immediate loss of steam generator level, resulting in degraded heat transfer in that steam generator and a loss of heat sink.

QUESTION: 087 (1.00)

Given the following:

- Plant is in MODE 6.
- Refueling in progress.
- RHR in service.
- Letdown purification is in progress.
- A spent fuel assembly is dropped from the upender into the refueling cavity.
- A steady stream of gas bubbles is rising to the surface of the refueling cavity from the dropped assembly.

WHICH ONE (1) of the following is the reason that letdown is removed from service while responding to the above refueling accident?

a. Minimizes inventory losses.

b. Preclude the spread of contamination.

c. Protect the Letdown Ion Exchanger resin from exhaustion.

d. Ensure that containment closure is achieved.

QUESTION: 088 (1.00)

WHICH ONE (1) of the following is the preferred order of power sources for recovery from a loss of all AC power if the Main XFMR is not available?

a. Aux XFMR C, Emergency Diesel Generator, DSD

b. Aux XFMR C, Emergency Diesel Generator, SDG&E 12 Kv

c. Emergency Diesel Generator, Aux XFMR C, DSD

d. Emergency Diesel Generator, Aux XFMR C, SDG&E 12 Kv

QUESTION: 089 (1.00)

Given the following:

- Reactor trip and SIS/LOP has occurred.
- SO1-1.0-60 "Loss of All AC Power" is in progress.
- SI block does not function at Step 3.
- RNO directs the operator to de-energize the sequencer.

WHICH ONE (1) of the following is the reason for de-energizing the sequencer?

a. Locks SI signal into sequencer memory and allows SI reset.

b. Removes LOP signal from sequencer memory and allows SI reset.

c. Ensures all SI relays have actuated and allows SI reset.

d. Eliminates initiation signals from being generated and blocks SI.

QUESTION: 090 (1.00)

WHICH ONE (1) of the following represents the MAXIMUM allowable number of MOV motor starts that an operator can attempt in one minute ?

a. 3

b. 4

c. 5

d. 6

QUESTION: 091 (1.00)

WHICH ONE (1) of the following is considered to be a Dedicated Safe Shutdown Zone?

a. Reactor Aux Building Lower Level

b. Reactor Aux Building Upper Level

c. Turbine Building Ground Floor

d. Main Transformer Area

QUESTION: 092 (1.00)

WHICH ONE (1) of the following are types of valves that should NOT be operated with a leverage device?

a. Hydraulically operated POV and Kerotest

b. Pneumatically operated POV and Manual gate valves less than 2.5 inches.

c. Motor operated valves and Manual gate valves less than 2.5 inches.

d. Diaphragm valves and Manual gate valves in "Important to Safety" systems.

QUESTION: 093 (1.00)

WHICH ONE (1) of the following is a reason why Control Room Emergency Air Treatment System Fan A-33 is NOT started following a toxic gas hazard notification?

a. Butane/Propane could cause ignition of the filter.

b. Starting the system will draw more toxic fumes into the control room.

c. The TSC HVAC system, when placed in the filter mode, provides the control room with better protection against toxic gases.

d. Starting the system will make the filters inoperable for radiocative particulate.

QUESTION: 094 (1.00)

WHICH ONE (1) of the following are phone systems in the plant which allow communication with other SCE facilities and the SDG&E load dispatcher?

a. Edison Decision Circuit, PAX phone

b. Ringdown circuit, PAX Phone

c. Edison Decision Circuit, Emergency Notification System

d. Ringdown Circuit, Emergency Notification System

QUESTION: 095 (1.00)

WHICH ONE (1) of the following persons tracks the data from the completed Oil Monitoring Data Sheets?

a. In Service Test Coordinator

b. Shift Technical Advisor

c. Shift Superintendent

d. Chemistry Supervisor

QUESTION: 096 (1.00)

WHICH ONE (1) of the following is the MAXIMUM time that the Control Operator may be relieved for a "Short Term Absence"?

a. 5 minutes

b. 10 minutes

c. 15 minutes

d. 20 minutes

QUESTION: 097 (1.00)

WHICH ONE (1) of the following is the color of ink required to be used when logging the release of a clearance from the Energy Control Center?

a. Blue

b. Black

c. Red

d. Green

QUESTION: 098 (1.00)

WHICH ONE (1) of the following individuals, by title, can authorize operation of caution tagged equipment?

a. Control Operator

b. Equipment Control Supervisor

c. Maintenance Supervisor in charge of work

d. Nuclear Plant Equipment Operator

QUESTION: 099 (1.00)

WHICH ONE (1) of the following represents the MAXIMUM number of hours in a 7 day (168 hour) period that a licensed operator may work at SONGS 1?

a. 64 hours

b. 72 hours

c. 78 hours

d. 84 hours

QUESTION: 100 (1.00)

WHICH ONE (1) of the following represents the MAXIMUM allowable whole body dose for a female Control Room Operator while pregnant?

a. 200 mrem

b. 300 mrem

c. 400 mrem

d. 500 mrem

(********** END OF EXAMINATION **********)

ANSWER: 072 (1.00)

d. [+1.0]

REFERENCE:

1. L.P. 1AI720, Obj. 1.1.2, Pg 25
2. SO1-2.3-1 Control Rod System Malfunctions Rev 2
3. SO1-4-35, Control Rod Drive System, Precaution Pg 12
4. BOTH RO AND SRO (TV)

[3.5/3.8]

000003K103 ..(KA's)

ANSWER: 073 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.0-10, Reactor Trip or Safety Injection, Step 12 RNO
2. L.P.- 1XA208, Obj. 2.5, Pg 13
3. BOTH RO AND SRO (TV)

[4.2/4.1]

000007G010 ..(KA's)

ANSWER: 074 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-2.1-3, alarms
2. L.P. 1AI704, Obj. 1.1.2, Pg 7
3. BOTH RO AND SRO (TV)

[3.8/3.8]

000008A108 ..(KA's)

ANSWER: 075 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.0-10 "Reactor Trip or Safety Injection", REV 9, EFF 3/6/91, STEP 1
2. BOTH RO AND SRO (JB)

(4.3/4.5)

000007A206 ..(KA's)

ANSWER: 076 (1.00)

b. [+1.0]

REFERENCE:

1. L.P. 1TA703, Obj. 3.2
2. BOTH RO AND SRO (JB)

(3.4/3.7)

000008A212 ..(KA's)

ANSWER: 077 (1.00)

c [+1.0]

REFERENCE:

1. SO1-1.0-25
2. L.P.- 1EI711, Obj. 1.1.2.5, Pg 59
3. BOTH RO AND SRO (TV)

[4.5/4.5]

000011G011 ..(KA's)

ANSWER: 078 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-2.1-9 "Loss of RHR"
2. BOTH RO AND SRO (JB)

(3.9/4.3)

000025K101 ..(KA's)

ANSWER: 079 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-2.1-9, Loss of Residual Heat Removal System, Step 1 note
2. L.P. 1AI710, Obj. 1.1.2, Pg 5
3. BOTH RO AND SRO (TV)

[3.6/3.5]

000025A112 ..(KA's)

ANSWER: 080 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-4-34, Precaution 4.6, Pg 54
2. L.P.- 1XT204, Obj. 5.0, Pg 39
3. BOTH RO AND SRO (TV)

[4.2/4.3]

000029K306 ..(KA's)

ANSWER: 081 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-4-34 "Reactor Plant Instrumentation Operation, REV 2, EFF 10/30/91, STEP 6.8.1
2. L.P. 1XC205, Obj.. 1.2.2, 1.5.2
3. BOTH RO AND SRO (JB)

(3.1/3.4)

000032A101 ..(KA's)

ANSWER: 082 (1.00)

d. [+1.0]

REFERENCE:

1. SD-SO1-380 REV 2
2. LP-1XC205 Rev 4 Obj. 1.5.2, Pg 51
3. RO AND SRO (TV)

[3.9/4.2]

000033A207 ..(KA's)

ANSWER: 083 (1.00)

b. [+1.0]

REFERENCE:

1. L.P. 1AI750, Obj. 1.1.2
2. SO1-2.1-17, "Steam Generator Tube Leakage"
3. BOTH RO AND SRO (JB)

(3.7/3.9)

000037G010 ..(KA's)

ANSWER: 084 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-1.40.1, Steam Generator Tube Rupture Background Document, Pg 24
2. L.P. 1EI715, Obj. 1.1.3, Pg 5
3. BOTH RO AND SRO (TV)

[4.2/4.5]

000038K306 ..(KA's)

ANSWER: 085 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-5-16 "Liquid Radioactive Waste Releases", Precaution 4.8
2. L.P.- 1XR204, Obj. 5.5, Pg 53
3. BOTH RO AND SRO (TV)

[2.9/3.9]

000059A202 ..(KA's)

ANSWER: 086 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.0-30
2. SO1-14-47, EOI Users guide, Pg 19
3. L.P.- 1EI701, Obj. 1.7, Pg 9
4. RO ONLY (TV)
[4.5/4.7]

000040K106 ..(KA's)

ANSWER: 087 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-2.1-16, Step 4.1.3
2. RO ONLY (TV)

[3.7/4.1]

000036K303 ..(KA's)

ANSWER: 088 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-1.0-60/60.1 Loss of All AC Power/Background Document
2. BOTH RO AND SRO (TV)

[4.3/4.6]

000055K302 ..(KA's)

ANSWER: 089 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-1.0-60.1 Loss of All AC Power Background Document Pg 9
2. 1XC207, Obj. 4.2, Pg 26
3. RO ONLY (TV)

[4.3/4.6]

000055K302 ..(KA's)

ANSWER: 090 (1.00)

c. [+1.0]

REFERENCE:

1. SO123-0-23.1 - Valve Operation Pg 3 of 24 Precaution 4.7
2.. RO ONLY (TV)

[3.6/3.7]

194001K107 ..(KA's)

ANSWER: 091 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-2.7-1, "APPENDIX R FIRE ZONES", Pg 3 of 7, (rev date 2/7/92).
2. Lesson Plan 1AI741 Rev 2 Obj. 1.1.4., Pg 3
3. RO ONLY (TV)

[3.5/4.2]

194001K116 ..(KA's)

ANSWER: 092 (1.00)

a. [+1.0]

REFERENCE:

1. SO123-0-23.1, "Valve Operation" Pg 2
2. Lesson Plan 1AP112, Obj. 1.1.16 Pg 41
3. RO ONLY (TV)

[3.6/3.7]

194001K101 ..(KA's)

ANSWER: 093 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-2.2-3, "Toxic gas", Pg 2 Rev date 10/10/91.
2. LP 1AI747, Obj. 1.1.2.2, Pg 2
3. RO AND SRO (TV)

[3.4/3.5]

194001K111 ..(KA's)

ANSWER: 094 (1.00)

b. [+1.0]

REFERENCE:

1. SD-S01-480, "Communications System" Pg 3.
2. LP 1XM205, "Communications", Obj. 2.2 Pg 4-8
3. RO ONLY (TV)

[3.0/3.2]

194001A104 ..(KA's)

ANSWER: 095 (1.00)

a. [+1.0]

REFERENCE:

1. SO123-0-9, "Operator Rounds and Inspections", Pg 4, Rev Date 5/30/90.
2. RO ONLY (TV)

[3.4/3.4]

194001A106 ..(KA's)

ANSWER: 096 (1.00)

c. [+1.0]

REFERENCE:

1. SO123-0-10, "Operations Shift Relief", Pg 6 of 99, Rev Date 7/30/91
2. RO ONLY (TV)

[2.8/4.1]

194001A112 ..(KA's)

ANSWER: 097 (1.00)

a. [+1.0]

REFERENCE:

1. SO123-0-11, "NARRATIVE LOGS" - Pg 9 of 14, Sect. 6.2.2.5.23, Rev date 12/19/91
2. RO ONLY (TV)

[3.4/3.4]

194001A106 ..(KA's)

ANSWER: 098 (1.00)

a. [+1.0]

REFERENCE:

1. SO123-0-29, "Use of Information Tags" Pg 3 of 11 6.1.1
2. BOTH RO AND SRO (TV)

[3.7/4.1]

194001K102 ..(KA's)

ANSWER: 099 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-14-23 "Assignment and Approval of Operations Overtime", Rev 2, Pg 3 of 15,
2. Unit 1 Technical Specifications Section 6.2.2.f
3. RO ONLY (TV)

[2.7/3.9]

194001A109 ..(KA's)

ANSWER: 100 (1.00)

d. [+1.0]

REFERENCE:

1. SO123-VII-4 - Personnel Monitoring Program, Rev 7, Pg 6
2. RO ONLY (TV)

[2.8/3.4]

194001K103 ..(KA's)

(********** END OF EXAMINATION **********)

RO Exam PWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
001	1.00	8000012
002	1.00	8000013
003	1.00	8000014
004	1.00	8000016
005	1.00	8000034
006	1.00	8000077
007	1.00	8000078
008	1.00	8000080
009	1.00	8000015
010	1.00	8000017
011	1.00	8000018
012	1.00	8000019
013	1.00	8000020
014	1.00	8000029
015	1.00	8000075
016	1.00	8000079
017	1.00	8000021
018	1.00	8000064
019	1.00	8000027
020	1.00	8000028
021	1.00	8000076
022	1.00	8000025
023	1.00	8000026
024	1.00	8000082
025	1.00	8000031
026	1.00	8000032
027	1.00	8000023
028	1.00	8000033
029	1.00	8000083
030	1.00	8000036
031	1.00	8000037
032	1.00	8000035
033	1.00	8000038
034	1.00	8000039
035	1.00	8000040
036	1.00	8000030
037	1.00	8000084
038	1.00	8000081
039	1.00	8000045
040	1.00	8000043
041	1.00	8000044
042	1.00	8000085
043	1.00	8000086
044	1.00	8000024
045	1.00	8000041
046	1.00	8000042
047	1.00	8000087
048	1.00	8000046
049	1.00	8000047

RO Exam PWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
050	1.00	8000048
051	1.00	8000050
052	1.00	8000049
053	1.00	8000088
054	1.00	8000052
055	1.00	8000056
056	1.00	8000059
057	1.00	8000051
058	1.00	8000090
059	1.00	8000091
060	1.00	8000092
061	1.00	8000057
062	1.00	8000062
063	1.00	8000053
064	1.00	8000061
065	1.00	8000093
066	1.00	8000094
067	1.00	8000054
068	1.00	8000095
069	1.00	8000055
070	1.00	8000058
071	1.00	8000089
072	1.00	8000070
073	1.00	8000067
074	1.00	8000069
075	1.00	8000096
076	1.00	8000097
077	1.00	8000063
078	1.00	8000098
079	1.00	8000071
080	1.00	8000065
081	1.00	8000099
082	1.00	8000022
083	1.00	8000100
084	1.00	8000068
085	1.00	8000066
086	1.00	8000073
087	1.00	8000072
088	1.00	8000060
089	1.00	8000074
090	1.00	8000001
091	1.00	8000002
092	1.00	8000003
093	1.00	8000004
094	1.00	8000005
095	1.00	8000006
096	1.00	8000007
097	1.00	8000008
098	1.00	8000009

R O Exam PWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
099	1.00	8000010
100	1.00	8000011

	100.00	

	100.00	

R O Exam PWR Reactor

Organized by KA Group

PLANT WIDE GENERICS

QUESTION	VALUE	KA
094	1.00	194001A104
097	1.00	194001A106
095	1.00	194001A106
099	1.00	194001A109
096	1.00	194001A112
092	1.00	194001K101
010	1.00	194001K102
098	1.00	194001K102
100	1.00	194001K103
090	1.00	194001K107
093	1.00	194001K111
091	1.00	194001K116

PWG Total	12.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
001	1.00	001000K202
008	1.00	001000K401
007	1.00	001000K402
002	1.00	001000K402
004	1.00	003000A107
006	1.00	003000A201
009	1.00	003000G001
005	1.00	003000K103
011	1.00	004010K404
012	1.00	013000A401
014	1.00	013000A402
016	1.00	013000K101
015	1.00	013000K104
017	1.00	015000K101
018	1.00	015000K301
038	1.00	022000A404
021	1.00	056010K412
022	1.00	061000A101
024	1.00	061000K402
023	1.00	061000K407
025	1.00	068000A302
026	1.00	068000K107
027	1.00	072000K402

PS-I Total	23.00	

R O Exam PWR Reactor

Organized by KA Group

PLANT SYSTEMS

Group II

QUESTION	VALUE	KA
028	1.00	002000K109
003	1.00	002020K509
013	1.00	006000K601
029	1.00	006030A402
030	1.00	010000K403
033	1.00	011000A101
032	1.00	011000K101
034	1.00	012000K610
035	1.00	012000K611
036	1.00	014000K101
019	1.00	026000A301
020	1.00	026000G004
031	1.00	026000K402
039	1.00	033000K303
037	1.00	035010K101
042	1.00	064000A401
043	1.00	064000A406
041	1.00	064000K102
040	1.00	064000K401
044	1.00	073000K101

PS-II Total	20.00	

Group III

QUESTION	VALUE	KA
047	1.00	005000G005
048	1.00	008000A401
050	1.00	045000G007
049	1.00	045000K413
051	1.00	045010A301
052	1.00	076000K105
053	1.00	076000K402
046	1.00	078000K402

PS-III Total	8.00	

PS Total	51.00	

EMERGENCY PLANT EVOLUTIONS

Group I

R O Exam PWR Reactor

Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
054	1.00	000005A105
057	1.00	000005A203
055	1.00	000015A210
058	1.00	000024G010
059	1.00	000026G006
061	1.00	000040A110
062	1.00	000040A201
086	1.00	000040K106
063	1.00	000051A202
088	1.00	000055K302
089	1.00	000055K302
066	1.00	000057A219
064	1.00	000057A219
067	1.00	000068G010
069	1.00	000069A201
070	1.00	000074K311

EPE-I Total	16.00	

Group II

QUESTION	VALUE	KA
071	1.00	000003G010
072	1.00	000003K103
075	1.00	000007A206
073	1.00	000007G010
074	1.00	000008A108
076	1.00	000008A212
077	1.00	000011G011
079	1.00	000025A112
078	1.00	000025K101
080	1.00	000029K306
060	1.00	000029K312
081	1.00	000032A101
082	1.00	000033A207
083	1.00	000037G010
084	1.00	000038K306
085	1.00	000059A202
068	1.00	000059A205

EPE-II Total	17.00	

Group III

R O Exam PWR Reactor

Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group III

QUESTION	VALUE	KA
087	1.00	000036K303
065	1.00	000056K101
045	1.00	000065A207
EPE-III Total	3.00	
EPE Total	36.00	
Test Total	100.00	

ANSWER: 001 (1.00)

a. [+1.0]

REFERENCE:

1. SD-SO1-400 Rod Control System Pg 28 of 52
2. LP 1XI203 Obj.2.4 Pg 25
3. Electrical Dwg 5146828-34
4. RO ONLY (TV)

[3.6/3.7]

001000K202 ..(KA's)

ANSWER: 002 (1.00)

c. [+1.0]

REFERENCE:

1. SD-SO1-400 Rod Control System Pg 6 of 52, Fig 5
2. LP 1XI203 Obj. 1.3.2, Pg 26
3. RO AND SRO (TV)

[3.8/3.8]

001000K402 ..(KA's)

ANSWER: 003 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-13-3, "Annunciator Response", Pg 13
2. L.P. 1XC206, Obj. 2.2, Dwg RPS-1-7
3. BOTH RO AND SRO (TV)

[3.6/3.9]

002020K509 ..(KA's)

ANSWER: 004 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-4-3, RCP Operation, Pg 11 of 31 Rev date 3/14/91
2. LP-1XA203 Rev 2 Obj. 1.3.2, Pg 13
3. RO ONLY (TV)

[3.4/3.4]

003000A107 ..(KA's)

ANSWER: 005 (1.00)

a. [+1.0]

REFERENCE:

1. L.P. 1XA203 Obj. 1.2.4, Pg 17
2. RO ONLY (TV)

[3.3/3.6]

003000K103 ..(KA's)

ANSWER: 006 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 1XA203, Obj. 7.5, page 14, para 6.2.2.1.2
2. SONGS Exam Bank, Review Section 2, 2 of 15
3. System Description SD-S01-300 Reactor Coolant Pump System, page 27, step 3.2.2
4. SO1-13-4 "Reactor Plant #1 Annunciator", Windows 32, 52, 72
5. BOTH RO AND SRO (JB)

(3.5/3.9)

003000A201 ..(KA's)

ANSWER: 007 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1XI203 OBJ. 3.4
2. SO1-12.3-24 "Monthly Control Rod Exercise", Rev 3, Page 10, Step 2.8
3. SONGS Exam Bank 2191 2 of 35
4. BOTH RO AND SRO (JB)

(3.8/3.8)

001000K402 ..(KA's)

ANSWER: 008 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1XI204, OBJ 3.3
2. SD-SO1-400, "Rod Control System" Page 18, Step 2.2.3.2
3. BOTH RO AND SRO (JB)

(3.5/3.8)

001000K401 ..(KA's)

ANSWER: 009 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-4-3, "Reactor Coolant Pump Operation", ATT 5.
2. L.P. 1XA203, Obj. 1.6.6, Pg 28
3. RO ONLY (TV)

[3.7/3.8]

003000G001 ..(KA's)

ANSWER: 010 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-4-6, Charging and Letdown System, Pg 7 of 64, Rev date 8/30/91
2. L.P.- 1XA206, Obj. 7.0, Pg 51
3. SD-SO1-310 Pg 12
4. RO ONLY (TV)

[3.7/4.1]

194001K102 ..(KA's)

ANSWER: 011 (1.00)

d. [+1.0]

REFERENCE:

1. LP-1AI703 Rev 2, Obj. 1.1.2, Pg 9
2. SO1-2.1-2 - Rev 5, Pg 2 of 26
3. RO ONLY (TV)

[3.1/3.4]

004010K404 ..(KA's)

ANSWER: 012 (1.00)

c. [+1.0]

REFERENCE:

1. LP-1XC207 - Obj. 3.8.3, Pg 20
2. SO1-4-46, "Safety Injection System Operation"
3. Drawings 5149178 thru 5149182
4. RO ONLY (TV)

[2.8/3.2]

013000A401 ..(KA's)

ANSWER: 013 (1.00)

a. [+1.0]

REFERENCE:

1. SD-SO1-580 Safety Injection, Recirculation,and Containment Spray Systems Rev 2 Pg 13.
2. L.P.- 1XA207, Obj. 1.6.2, Pg 48
3. RO ONLY (TV)

[3.6/3.9]

006000K601 ..(KA's)

ANSWER: 014 (1.00)

d. [+1.0]

REFERENCE:

1. L.P.-1XA200, Obj. 3.2, Pg 18
2. SD-SO1-630, "Containment and Containment Isolation Systems" Pg 35.
3. BOTH RO AND SRO (TV)

[4.5/4.8]

013000A402 ..(KA's)

ANSWER: 015 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1XC207, OBJ 5.0
2. SD-S01-630 Containment and Containment Isolation Systems
3. Dwg 5150874, 5150875
4. BOTH RO AND SRO EXAMS (JB)

(3.9/4.4)

013000K104 ..(KA's)

ANSWER: 016 (1.00)

c. [+1.0]

REFERENCE:

1. L.P. 1XC207, Obj. 3.4
2. Technical Specifications Table 3.5.5-2, Figure 3.5.1.1
3. BOTH RO AND SRO (JB)

(4.2/4.4)

013000K101 ..(KA's)

ANSWER: 017 (1.00)

d. [+1.0]

REFERENCE:

1. Technical Specifications, Section 2.1
2. L.P. - 1AI715 - Obj 1.1.4, Pg 5
3. RO ONLY (TV)

[4.1/4.2]

015000K101 ..(KA's)

ANSWER: 018 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-4-34, Reactor Plant Instrumentation Operation, Precaution 4.9, Pg 30 of 97
2. L.P.- 1XC205, Obj. 1.5.2, Pg 51
3. RO ONLY (TV)

[3.9/4.3]

015000K301 ..(KA's)

ANSWER: 019 (1.00)

d. [+1.0]

REFERENCE:

1. SD-SO1-580, Pg 76 of 90
2. L.P. 1XA208, Obj. 3.4, Pg 17
3. RO ONLY (TV)

[4.3/4.5]

026000A301 ..(KA's)

ANSWER: 020 (1.00)

c. [+1.0]

REFERENCE:

1. SD-SO1-580 Safety Injection, Recirculation, and Containment Spray, Pg. 72 of 90
2. L.P.-1XA208, Obj. 6.1, Pg 11
3. RO ONLY (TV)

[3.6/3.9]

026000G004 ..(KA's)

ANSWER: 021 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-7-4 "Condensate System", Rev 4, Eff.9/8/90, Step 6.0 Caution
2. SO1-3-2 "Plant Startup from Hot Standby to Minimum Load" REV 6, EFF 11/18/91, STEP 6.15 AND 6.27
3. BOTH RO AND SRO EXAMS (JDB)

(2.2/2.6)

056010K412 ..(KA's)

ANSWER: 022 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-7-3 Auxiliary Feedwater System, Precaution 4.10
2. RO ONLY (TV)

[3.9/4.2]

061000A101 ..(KA's)

ANSWER: 023 (1.00)

a. [+1.0]

REFERENCE:

1. SD-SO1-620 Pg 6 of 37
2. SO1-7-3, Precaution 4.17
3. RO ONLY (TV)

[3.1/3.3]

061000K407 ..(KA's)

ANSWER: 024 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 89RS02, Review Question 3
2. SD-S01-620 Auxillary Feed, Page B-2
3. BOTH RO AND SRO (JB)

(4.5/4.6)

061000K402 ..(KA's)

ANSWER: 025 (1.00)

a. [+1.0]

REFERENCE:

1. LP-1XR201, Obj. 3.4, Pg 20
2. SO1-5-16, "Liquid Radwaste Releases", Pg 57, Step 2.22 Caution.
3. RO ONLY (TV)

[3.6/3.6]

068000A302 ..(KA's)

ANSWER: 026 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-5-14, Liquid Radioactive Waste Receiving and Storage Operations ATT 14
2. L.P.- 1XR205, Obj. 2.1, 6.1, Pg 9
3. RO AND SRO (TV)

[2.7/2.9]

068000K107 ..(KA's)

ANSWER: 027 (1.00)

b. [+1.0]

REFERENCE:

1. SD-SO1-540, Rev 1, Pg 3 of 18
2. L.P. 1XR201, Obj. 3.3, Pg 9
3. RO ONLY (TV)

[3.2/3.4]

072000K402 ..(KA's)

ANSWER: 028 (1.00)

d. [+1.0]

REFERENCE:

1. 1XA202 Obj. 4.0, Pg 13
2. SD-SO1-280, "Reactor Coolant System", Pg 5-6
3. RO ONLY (TV)

[4.1/4.1]

002000K109 ..(KA's)

ANSWER: 029 (1.00)

d. [+1.0]

REFERENCE:

1. Lesson Plan 1XA207, OBJ 1.3.1.6 Page 18
2. Drawing 5102063
3. BOTH RO AND SRO (JB)

(4.4/4.4)

006030A402 ..(KA's)

ANSWER: 030 (1.00)

a. [+1.0]

REFERENCE:

1. 1XI202 - Obj.2.4, Pg 56/58
2. SO1-3-5 - Plant Shutdown from Hot Standby to Cold Shutdown, Step 6.18
3. SO1-1.0-30 - Loss of Secondary Coolant, Caution prior to Step 31, Pg 43.
4. RO ONLY (TV)

[3.8/4.1]

010000K403 ..(KA's)

ANSWER: 031 (1.00)

b. [+1.0]

REFERENCE:

1. L.P.-1XA208, Obj. 1.1
2. Tech Spec 3.3.4, Basis
3. RO AND SRO (JDB)

[3.1/3.6]

026000K402 ..(KA's)

ANSWER: 032 (1.00)

b. [+1.0]

REFERENCE:

1. L.P. - 1XI202 Obj. 3.3 Pg 48
2. SO1-3-3, "Plant Operation from Minimum Load to Full Power", Pg 17
3. RO ONLY (TV)

[3.6/3.9]

011000K101 ..(KA's)

ANSWER: 033 (1.00)

c. [+1.0]

REFERENCE:

1. L.P. 1XI207, Obj 1.2.f, Pg 19
2. SO1-2.3-4, "Abnormal Pressurizer Level", Pg 8
3. RO ONLY (TV)

[3.5/3.6]

011000A101 ..(KA's)

ANSWER: 034 (1.00)

a. [+1.0]

REFERENCE:

1. L.P. 1XC204, Obj. 1.4.2, Pg 3-8
2. SO1-2.3-1, "Control Rod System Malfunctions", Pg 11
3. RO AND SRO (TV)

[3.3/3.5]

012000K610 ..(KA's)

ANSWER: 035 (1.00)

b. [+1.0]

REFERENCE:

1. L.P. 1XC204 Obj. 1.2.1, Pg 21
2. SO1-13-7, "Reactor Plant Partial Trip Matrix", Window 32
3. RO ONLY (TV)

[2.9/2.9]

012000K611 ..(KA's)

ANSWER: 036 (1.00)

b. [+1.0]

REFERENCE:

1. 1XA202, Obj. 5.2,6.4, Pg 14-15
2. SD-SO1-390, "Primary Process Instrumentation Systems", Pg 35
3. RO ONLY (TV)

[3.2/3.6]

014000K101 ..(KA's)

ANSWER: 037 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 1XI206, Obj. 2.1, Page iii
2. SD-SO1-260 Feedwater Control Systems, Page 18, Step 2.3.2
3. BOTH RO AND SRO (JB)

(4.2/4.5)

035010K101 ..(KA's)

ANSWER: 038 (1.00)

d. [+1.0]

REFERENCE:

1. Lesson Plan 1XA208, OBJ 3.3
2. SO1-1.0-23.1 "Background Document for Loss of Recircuation", Step 3
3. BOTH RO AND SRO (JB)

(3.1/3.2)

022000A404 ..(KA's)

ANSWER: 039 (1.00)

b. [+1.0]

REFERENCE:

1. L.P. 1AI714, Obj. 1.1.3.2, Pg 3
2. SO1-2.1-13, "Loss of Spent Fuel Pit Cooling", ATT 1
3. RO ONLY (TV)

[3.0/3.3]

033000K303 ..(KA's)

ANSWER: 040 (1.00)

c. [+1.0]

REFERENCE:

1. L.P.-1CF704, Obj.A.1, Pg 14
2. SO1-10-2, "Diesel Generator Starting Air System", Section A, Precaution 4.1
3. BOTH RO AND SRO (TV)

[3.8/4.1]

064000K401 ..(KA's)

ANSWER: 041 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-10-4, Diesel Generator Cooling Water System Objective 1.2, Pg 2
2. SD-SO1-600, Diesel Generator Systems Pg 40
3. RO ONLY (TV)

[3.1/3.6]

064000K102 ..(KA's)

ANSWER: 042 (1.00)

c. [+1.0]

REFERENCE:

1. System Description SD-S01-600, Rev. 2, Page 76
2. S01-10-1 "Diesel Generator Operations", Rev. 3, Eff. 12/31/91, Step 6.0 Caution
3. BOTH RO AND SRO (JB)

(4.0/4.3)

064000A401 ..(KA's)

ANSWER: 043 (1.00)

d. [+1.0]

REFERENCE:

1. SYSTEM DESCRIPTION SD-S01-600, REV 2, STEP 3.1, Pg 91
2. S01-10-1, "Diesel Generator Operations", REV 3, EFF 12/31/91, STEP 6.19.3.3
3. BOTH RO AND SRO (JB)

(3.9/3.9)

064000A406 ..(KA's)

ANSWER: 044 (1.00)

d. [+1.0]

REFERENCE:

1. L.P. - 1XR201, Obj. 2.3, Pg 55
2. SO1-2.2-1, "High Activity Operational Monitoring System", Step 2, Pg 7.
3. BOTH RO AND SRO (TV)

[3.6/3.9]

073000K101 ..(KA's)

ANSWER: 045 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-12.8-18, "PORV and BLock Valve Backup Nitrogen Supply Test"
2. Plant Drawings 5178404, 5178405
3. L.P.- 1XQ206, Obj. 1.3.2, Pg 12
4. SONGS 1 Question Bank
5. RO ONLY (TV)

[2.8/3.2]

000065A207 ..(KA's)

ANSWER: 046 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-7-1, Instrument and Service Air System Operation, Precaution 4.3
2. L.P. 1AI726 Obj. 1.1.3.2, Pg 6
3. RO ONLY (TV)

[3.2/3.5]

078000K402 ..(KA's)

ANSWER: 047 (1.00)

d. [+1.0]

REFERENCE:

1. SONGS Technical Specifications, 3.1.2.H.1 (Action Statement is: "IMMEDIATELY INITIATE CORRECTIVE ACTION...")
2. S01-4-9 "RESIDUAL HEAT REMOVAL SYSTEM OPERATION", REV 5, EFF 4/30/91, STEP 4.4
3. BOTH RO AND SRO (JB)

(3.2/3.8)

005000G005 ..(KA's)

ANSWER: 048 (1.00)

d. [+1.0]

REFERENCE:

1. L.P. 1XB201, Obj. 3.3, Pg 9
2. SO1-4-19, "CCW System Operations", Section 6.2
3. BOTH RO AND SRO (TV)

[3.3/3.1]

008000A401 ..(KA's)

ANSWER: 049 (1.00)

b. [+1.0]

REFERENCE:

1. SD-SO1-270 Turbine Control System Pg 25 of 45
2. L.P. 1XT204, Obj. 3.9, Pg 35
3. RO ONLY (TV)

[2.6/2.8], [3.4/3.6]

045000K413 ..(KA's)

ANSWER: 050 (1.00)

c. [+1.0]

REFERENCE:

1. L.P. 1XT204, Obj. 6.1, Pg 11
2. SO1-6-3, "Main Turbine Oil System Operation", Section A, Pg 5
3. RO ONLY (TV)

[2.6/2.8]

045000G007 ..(KA's)

ANSWER: 051 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-3-8 Precaution 4.3
2. L.P. XT204, Obj. 5.1, Pg 46
3. RO ONLY (TV)

[3.5/3.6]

045010A301 ..(KA's)

ANSWER: 052 (1.00)

d. [+1.0]

REFERENCE:

1. L.P. 1CF701, Obj. 1.2.1, Pg 53
2. SONGS 1 Question bank
4. RO ONLY (TV)

[3.8/4.0]

076000K105 ..(KA's)

ANSWER: 053 (1.00)

a. [+1.0]

REFERENCE:

1. System Description SD-S01-340, REV 1, Pg 4, STEP 2.2.1
2. BOTH RO AND SRO (JB)

(2.9/3.2)

076000K402 ..(KA's)

ANSWER: 054 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-2.3-1, RPI failure step 7
2. L.P. 1XI204, Obj. 5.1, Pg 17
3. Technical Specifications 3.5.4, Action C.3
4. BOTH RO AND SRO (TV)

[3.4/3.4]

000005A105 ..(KA's)

ANSWER: 055 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-2.1-7, Step 7
2. L.P. IA1709, Obj. 1.1.1
3. RO ONLY (TV)

[3.7/3.7]

000015A210 ..(KA's)

ANSWER: 056 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-2.1-1
2. L.P. 1AI702, Obj. 1.1.2, Pg 2
3. RO ONLY (TV)

[4.1/4.2]

000017K307 ..(KA's)

ANSWER: 057 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.0-11, Reactor Trip Response, Step 5
2. L.P. 1AI713, Obj. 1.1.2, Pg 5
3. RO ONLY (TV)

[3.5/4.4]

000005A203 ..(KA's)

ANSWER: 058 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-2.1-12 "Emergency Boration", Rev. 2, 10/23/91, Pages 1-2.
2. BOTH RO AND SRO (JB)

(4.0/4.0)

000024G010 ..(KA's)

ANSWER: 059 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-1.0-60.1 "Loss of All Ac Background Document", Rev. 3, Eff. 8/30/91, Step 24A
2. BOTH RO AND SRO (JB)

(3.4/3.6)

000026G006 ..(KA's)

ANSWER: 060 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.1-1, "Response to Nuclear Power Generation/ATWS", Step 4 RNO
2. RO ONLY (TV)

(4.4/4.7)

000029K312 ..(KA's)

ANSWER: 061 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-1.0-30, Loss of Secondary Coolant, Caution
2. RO ONLY (TV)

[4.1/4.1]

000040A110 ..(KA's)

ANSWER: 062 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-1.0-30.1, Background Document for Loss of Secondary Coolant, Pg 4 of 90
2. RO ONLY (TV)

[4.2/4.7]

000040A201 ...(KA's)

ANSWER: 063 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-2.4-3, Loss of Condenser Vacuum ATT 1, (Required handout)
2. L.P.- 1AI727, Obj. 1.1.1, Pg 4
3. BOTH RO AND SRO (TV)

[3.9/4.1]

000051A202 ..(KA's)

ANSWER: 064 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-2.6.3, Loss of Vital or Utility Bus, Part H, Note 4.6, Pg 62
2. L.P.- 1AI736, Obj. 1.2, Pg 84
3. BOTH RO AND SRO (TV)

[4.0/4.3]

000057A219 ..(KA's)

ANSWER: 065 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-1.0-60.1 "Background Document For Loss of All AC", Rev. 3, Eff. 3/6/91, Step 27, Page 29
2. SO1-3-6 "Plant Operation With Natural Circulation", page 3, step 4.2
3. BOTH RO AND SRO (JB)
(3.7/4.2)

000056K101 ..(KA's)

ANSWER: 066 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-2.6-5, "Loss of Regulated Bus", REV 1, EFF 3/6/91, Pg 18, AUTOMATIC ACTION
2. BOTH RO AND SRO (JB)

(4.0/4.3)

000057A219 ..(KA's)

ANSWER: 067 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-2.5-4
2. L.P.-1AI733, Obj. 1.1, Pg 3
3. RO ONLY (TV)

[4.1/4.2]

000068G010 ..(KA's)

ANSWER: 068 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-5-13 "Radiation Monitoring System Operation", Rev. 1, Eff. 8/14/91, Page 21-23, step 6.9.5.2
2. BOTH RO AND SRO (JB)

(3.6/3.9)

000059A205 ..(KA's)

ANSWER: 069 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-2.1-6, Loss of Containment Integrity
2. L.P.- 1XA200 - Obj. 5.2, Pg 30
3. RO ONLY (TV)

[3.7/4.3]

000069A201 ..(KA's)

ANSWER: 070 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-1.2-1.1 Inadequate Core Cooling Background Document, Pg 6 of 24
2. L.P.- 1MD701 Obj. 3.5, Pg 30
3. RO ONLY (TV)

[4.0/4.4]

000074K311 ..(KA's)

ANSWER: 071 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-2.3-1 "Control Rod Malfunction", REV 2, EFF 3/10/91, STEP 1
2. BOTH RO AND SRO (JB)

(3.9/4.0)

000003G010 ..(KA's)

A N S W E R K E Y

MULTIPLE CHOICE

001	a
002	c
003	a
004	d
005	a
006	a
007	b
008	b
009	d
010	a
011	d
012	c
013	a
014	d
015	b
016	c
017	d
018	a
019	d
020	c
021	c
022	d
023	a
024	a
025	a
026	a
027	b
028	d
029	d
030	a
031	b
032	b
033	c
034	a
035	b
036	b
037	a
038	d
039	b
040	c
041	b
042	c
043	d
044	d
045	c

A N S W E R K E Y

046	c	069	c
047	d	070	a
048	d	071	a
049	b	072	d
050	c	073	b
051	c	074	c
052	d	075	b
053	a	076	b
054	b	077	c
055	d	078	d
056	d	079	c
057	b	080	c
058	b	081	b
059	c	082	d
060	b	083	b
061	a	084	d
062	d	085	b
063	a	086	b
064	a	087	b
065	c	088	c
066	a	089	d
067	c	090	c
068	a	091	c

A N S W E R K E Y

092	a
093	d
094	b
095	a
096	c
097	a
098	a
099	b
100	d

(********** END OF EXAMINATION **********)

AS GIVEN

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Examination

SRO Test
and
Answer Key

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SITE SPECIFIC EXAMINATION
SENIOR OPERATOR LICENSE
REGION 5

CANDIDATE'S NAME: _____

FACILITY: San Onofre 1

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 92/04/27

INSTRUCTIONS TO CANDIDATE:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	CANDIDATE'S SCORE	%	
100.00		___ %	TOTALS
	FINAL GRADE		

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

_____
Candidate's Signature

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

MULTIPLE CHOICE

001	a	b	c	d	___	023	a	b	c	d	___
002	a	b	c	d	___	024	a	b	c	d	___
003	a	b	c	d	___	025	a	b	c	d	___
004	a	b	c	d	___	026	a	b	c	d	___
005	a	b	c	d	___	027	a	b	c	d	___
006	a	b	c	d	___	028	a	b	c	d	___
007	a	b	c	d	___	029	a	b	c	d	___
008	a	b	c	d	___	030	a	b	c	d	___
009	a	b	c	d	___	031	a	b	c	d	___
010	a	b	c	d	___	032	a	b	c	d	___
011	a	b	c	d	___	033	a	b	c	d	___
012	a	b	c	d	___	034	a	b	c	d	___
013	a	b	c	d	___	035	a	b	c	d	___
014	a	b	c	d	___	036	a	b	c	d	___
015	a	b	c	d	___	037	a	b	c	d	___
016	a	b	c	d	___	038	a	b	c	d	___
017	a	b	c	d	___	039	a	b	c	d	___
018	a	b	c	d	___	040	a	b	c	d	___
019	a	b	c	d	___	041	a	b	c	d	___
020	a	b	c	d	___	042	a	b	c	d	___
021	a	b	c	d	___	043	a	b	c	d	___
022	a	b	c	d	___	044	a	b	c	d	___
						045	a	b	c	d	___

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

046	a	b	c	d	___	069	a	b	c	d	___
047	a	b	c	d	___	070	a	b	c	d	___
048	a	b	c	d	___	071	a	b	c	d	___
049	a	b	c	d	___	072	a	b	c	d	___
050	a	b	c	d	___	073	a	b	c	d	___
051	a	b	c	d	___	074	a	b	c	d	___
052	a	b	c	d	___	075	a	b	c	d	___
053	a	b	c	d	___	076	a	b	c	d	___
054	a	b	c	d	___	077	a	b	c	d	___
055	a	b	c	d	___	078	a	b	c	d	___
056	a	b	c	d	___	079	a	b	c	d	___
057	a	b	c	d	___	080	a	b	c	d	___
058	a	b	c	d	___	081	a	b	c	d	___
059	a	b	c	d	___	082	a	b	c	d	___
060	a	b	c	d	___	083	a	b	c	d	___
061	a	b	c	d	___	084	a	b	c	d	___
062	a	b	c	d	___	085	a	b	c	d	___
063	a	b	c	d	___	086	a	b	c	d	___
064	a	b	c	d	___	087	a	b	c	d	___
065	a	b	c	d	___	088	a	b	c	d	___
066	a	b	c	d	___	089	a	b	c	d	___
067	a	b	c	d	___	090	a	b	c	d	___
068	a	b	c	d	___	091	a	b	c	d	___

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

092	a	b	c	d	___
093	a	b	c	d	___
094	a	b	c	d	___
095	a	b	c	d	___
096	a	b	c	d	___
097	a	b	c	d	___
098	a	b	c	d	___
099	a	b	c	d	___
100	a	b	c	d	___

(********** END OF EXAMINATION **********)

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done after you complete the examination.

3. Restroom trips are to be limited and only one applicant at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

4. Use black ink or dark pencil ONLY to facilitate legible reproductions.

5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet and each answer sheet.

6. Mark your answers on the answer sheet provided. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

7. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.

8. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

9. The point value for each question is indicated in parentheses after the question.

10. Show all calculations, methods, or assumptions used to obtain an answer to any short answer questions.

11. Partial credit may be given except on multiple choice questions. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

12. Proportional grading will be applied. Any additional wrong information that is provided may count against you. For example, if a question is worth one point and asks for four responses, each of which is worth 0.25 points, and you give five responses, each of your responses will be worth 0.20 points. If one of your five responses is incorrect, 0.20 will be deducted and your total credit for that question will be 0.80 instead of 1.00 even though you got the four correct answers.

13. If the intent of a question is unclear, ask questions of the examiner only.

14. When turning in your examination, assemble the completed examination with examination questions, examination aids and answer sheets. In addition, turn in all scrap paper.

15. Ensure all information you wish to have evaluated as part of your answer is on your answer sheet. Scrap paper will be disposed of immediately following the examination.

16. To pass the examination, you must achieve a grade of 80% or greater.

17. There is a time limit of four (4) hours for completion of the examination.

18. When you are done and have turned in your examination, leave the examination area (EXAMINER WILL DEFINE THE AREA). If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION: 001 (1.00)

Given the following conditions:

Make-up to the RCS has increased to 95 gpm, and ONLY the following alarms are received:

"RC PUMP A NO. 1 SEAL LOW DELTA P"
"RC PUMP A NO. 1 SEAL LEAKOFF FLOW OFF NORMAL"
"RC PUMP A SEAL WATER HI FLOW"
"RC PUMP A NO. 2 SEAL HI FLOW"

WHICH ONE (1) of the following has occurred to the A RCP?

a. #1 seal has failed.

b. #1 and #2 seals have failed.

c. All the seals have failed.

d. Seal injection has failed.

QUESTION: 002 (1.00)

WHICH ONE (1) of the following are printed on YR-404 "Rod Position Recorder" with the Rod Position Recorder Selector Switch in the Control Bank 2 position?

a. -Individual rod positions of the 17 rods in CBI by LVDT signal.
-Control Bank 1 and 2 positions from the Digital Indication System.

b. -Control Bank 1 and 2 bank positions from the Digital Indication System.
-Calculated Shutdown Margin Bank 1 and 2 LO alarm setpoints from the Reactor Protection and Control System.

c. -Individual rod positions of the 17 rods in CBII from the individual digital detectors in containment.
-Calculated Shutdown Margin Bank 1 and 2 LO alarm setpoints from the Reactor Protection and Control System.

d. -Individual rod positions of the 17 rods in CBI AND CBII by LVDT signal.
-Control Bank 1 and 2 positions from the Digital Indication System.

QUESTION: 003 (1.00)

WHICH ONE (1) of the following actions will move Control Bank II in MANUAL utilizing the In-Hold-Out switch? Assume the Reactor is at 92% power with rod control in automatic.

a. Place the Group Selector Switch in the "2" position, and pull it out.

b. Place the Overlap Cutout Switch in the "Bank 2" position, and the Group Selector Switch in the "Manual" position.

c. Place the Overlap Cutout Switch in "Override" and the Group Selector Switch in the "Bank 2" position.

d. Place the Overlap Cutout Switch in the "Overlap" position, and pull it out.

QUESTION: 004 (1.00)

WHICH ONE (1) of the following describes the method to achieve a bumpless transfer from Manual to Auto - Cascade control on pressurizer level controller LC430F?

a. Transfer directly to Auto - Cascade.

b. Null by adjusting Auto setpoint, then transfer to Auto - Cascade.

c. Transfer to Auto - Manset, then to Auto - Cascade.

d. Null by adjusting manual control, then transfer to Auto - Cascade.

QUESTION: 005 (1.00)

Given the following conditions:

The unit tripped from 92% power. During the trip transient the following conditions are experienced:
- 4 KV bus 1C voltage dropped to 3900 volts.
- Primary plant pressure dropped to 1752 psig.
- Containment pressure increased to 1.6 psig.

WHICH ONE (1) of the following describes the ESF actuation response?

a. CSAS

b. LOP

c. SIS

d. SIS/LOP

QUESTION: 006 (1.00)

WHICH ONE (1) of the following describes the response of the CVCS system following SIS actuation?

a. Charging flow control valve FCV1112 opens, Letdown isolates, and Seal Injection valve position does not change.

b. Charging isolates, Letdown isolates, and Seal Injection valve position does not change.

c. Charging flow control valve FCV1112 opens, Letdown valve positions does not change, and FCV-1115 D, E, &F (Seal Injection valves) position does not change.

d. Charging isolates, Letdown isolates, and Seal Injection isolates.

QUESTION: 007 (1.00)

Given the following information:

-An accident has occurred which required containment spray.
-The RWST is depleted, and recirculation flow is established.

WHICH ONE (1) of the following is the required position for the spray limiter orifice isolation valves?

	CV-517	CV-518
a.	open	open
b.	closed	open
c.	open	closed
d.	closed	closed

QUESTION: 008 (1.00)

Given the following information:

The plant in mode 1, with Unit load at 9%.

WHICH ONE (1) of the following is the MINIMUM acceptable number of running main feed pumps and condensate pumps?

	# of running MFP	# of running Cond. pumps
a.	1	1
b.	2	2
c.	1	2
d.	2	4

QUESTION: 009 (1.00)

WHICH ONE (1) of the following will result in flow from an automatic actuation of Train A Auxiliary Feedwater?

a. 2 of 3 S/G level less than 5% on train A, AND no flow on Train B Aux Feed.

b. 2 of 2 level transmitters less than 5% on 1 of 3 S/G, AND no flow on Train B Aux Feed.

c. 2 of 3 S/G low level, AND low pressure on Train B Aux Feed.

d. 2 of 3 low level on 1 of 3 S/G, AND low pressure on Train B Aux Feed.

QUESTION: 010 (1.00)

WHICH ONE (1) of the following will trip the Safety Injection pumps during an SI?

a. Time delay undervoltage OR Loss of DC power

b. Time delay overcurrent OR Loss of DC power

c. Time Delay overcurrent OR Low-low RWST level

d. Loss of DC power OR Low-low RWST level

QUESTION: 011 (1.00)

Given the following conditions:

-Turbine tripped
-Primary pressure at 2020 psig
-T ave at 550 degrees
-Normal post-trip steam generator levels and pressures

WHICH ONE (1) of the following signals does the Steam Generator Water Level Control switching chassis direct to the regulating valve controller?

a. Feed flow controller output

b. 5% open signal

c. Full closed signal

d. Steam flow/feed flow error

QUESTION: 012 (1.00)

WHICH ONE (1) of the following statements describes how the Emergency Diesel Generator controls are reset to assure the correct voltage will be obtained upon engine start?

a. The operator nulls the Manual Voltage Adjust to match the setting of the Auto Voltage Control after engine shutdown.

b. The voltage control setting is fixed and requires no auto or manual adjustment.

c. The Auto Voltage Adjust will reposition to the correct startup voltage for up to 25 seconds after engine shutdown.

d. The Auto Voltage Adjust adjusts to 4160 volts at all times automatically.

QUESTION: 013 (1.00)

WHICH ONE (1) of the following statements describes when the Emergency Diesel Generator Standby Lube Oil Pump should be run?

a. Continuously when the diesel is shutdown to maintain engine pre-lube and temperature.

b. Started prior to engine start and run for 5 minutes to pre-lube the diesel.

c. Started prior to engine start to pre-lube the engine, but must be shutdown within 5 minutes after engine start.

d. Started prior to engine shutdown and allowed to run for approximately 1 minute following engine shutdown.

QUESTION: 014 (1.00)

Given the following information:

-Reactor Coolant System is vented.
-Reactor Coolant System at 110 degrees F.
-Pressurizer level is at 50%.

WHICH ONE (1) of the following plant conditions will satisfy the applicable LCO for RHR?

a. One RHR train is OPERABLE and in operation, with all steam generators at 180 inches.

b. One RHR train is OPERABLE and in operation, with 2 steam generators at 280 inches.

c. Two reactor coolant pumps are in operation.

d. Two RHR trains are OPERABLE and one is in operation.

QUESTION: 015 (1.00)

WHICH ONE (1) of the following will cause an auto start of the standby Salt Water Cooling Pump which is in "PULL FOR AUTO"?

a. Overload trip of running Salt Water Cooling Pump.

b. Start of associated train Emergency Diesel Generator.

c. Start of associated train CCW Pump.

d. Low flow of 2500 gpm on running Salt Water Cooling Pump.

QUESTION: 016 (1.00)

Given the following information:

-The plant is at 25% power.
-2 control rods have dropped to the bottom.

WHICH ONE (1) of the following actions should be taken?

a. Trip the Reactor and go to SO1-1.0-10, "Reactor Trip or Safety Injection".

b. Place rod control in manual, and use SO1-2.3-1 "Control Rod Malfunction" to recover the dropped rods.

c. Notify the on-shift STA and Reactor Engineering to make an evaluation for continued operation with rods on the bottom, and implement the action determined by the evaluation within 3 hours from the time the rods dropped.

d. Recover the dropped rods IF the rod recovery can be completed within 1 hour from the time the rods were dropped.

QUESTION: 017 (1.00)

Given the following information:

-The plant has been at 150 degrees F for 2 weeks.
-A Source Range High Flux Level alarm is received in the control room.
-The source range count rate is verified to be steadily increasing.

WHICH ONE (1) of the following actions should be immediately taken?

a. Immediately sample the RCS boron concentration, calculate the shutdown margin, and emergency borate as required.

b. Emergency borate 25% of the BAST, and terminate any dilution that may be in progress.

c. Log the source range readings every 15 minutes.

d. Initiate action per SO1-1.1-2 "Potential Loss of Core Shutdown".

QUESTION: 018 (1.00)

WHICH ONE (1) of the following actions should be taken to verify that the DC RCP Thermal Barrier Pump is RUNNING during a loss of all AC power event?

a. Directly observe the pump to verify the shaft is turning and the suction and discharge pressures are normal.

b. Observe that the indicating light on the DC RCP Thermal Barrier Pump breaker in No. 2 DC Switchgear Room is lit.

c. Observe the Reactor Plant Annunciator #1 Panel on the west vertical board, "EMERGENCY DC RCP THERMAL BARRIER PUMP RUN" is lit.

d. Observe that the CCW discharge header pressure is being maintained greater than 20 psig on the main control board pressure gauge.

QUESTION: 019 (1.00)

WHICH ONE (1) of the following is the reason for the difference in severity between a feedline break inside containment, upstream of the inside containment check valve, and a steam line break inside containment?

a. The feedwater break will cause a less severe cooldown transient and release less energy to the containment than a steamline break, due to the enthalpy difference between the feedwater and the steam.

b. The feedwater break will cause the loss of feed to one generator, but no steam generators will blowdown and minimal RCS cooldown will occur.

c. The feedwater break will cause one steam generator to blow down out the break, and steam the other generators dry. The energy lost and associated cooldown is less severe than that caused by steam line break.

d. The feedwater break will result in an immediate loss of steam generator level, resulting in degraded heat transfer in that generator and a loss of heat sink.

QUESTION: 020 (1.00)

WHICH ONE (1) of the following plant conditions verify adequate natural circulation cooling?

a. RCS subcooling 34 F.

b. RCS hot leg temperature trending with saturation temperature for Main Steam pressure.

c. Steam generator levels at or approaching 50% narrow range.

d. Core exit thermocouples stable and trending with RCS cold leg temperature..

QUESTION: 021 (1.00)

Given the following information:

-Reactor power is 57%.
-The bus potential indicating light is OFF for Regulated Bus 3.

WHICH ONE (1) of the following automatic actions will occur?

a. FCV-458, Steam Generator C Feedwater Regulator Valve, fails OPEN.

b. Pressurizer heaters trip OFF when in AUTO.

c. PCV-430C and PCV-430H, Pressurizer Spray Valves, fail CLOSED.

d. FCV-1112, Charging Flow Control Valve, fails CLOSED.

QUESTION: 022 (1.00)

Given the following information:

R-1218 Liquid Radwaste Effluent Monitor, is in a High Alarm condition.

WHICH ONE (1) of the following automatic actions will occur as a result of the high alarm?

a. R-1218 Sample Pump STOPS

b. SV-1218-2 Purge valve CLOSES

c. Yard Sump Pump TRIPS

d. RWL-349 Sample Pump Suction valve CLOSES

QUESTION: 023 (1.00)

WHICH ONE (1) of the following is the criteria for verification of a reactor trip per step 1 of SO1-1.0-10, "Reactor Trip or Safety Injection"?

a. Both reactor trip breakers open, both turbine stop valves closed, and 4 KV buses 1C and/or 2C energized.

b. Both or either reactor trip breakers open, rod bottom indicator lights on, and neutron flux decreasing.

c. 1 of 4 reactor trip or bypass breakers open, rods on the bottom, and neutron flux decreasing.

d. Both or either reactor trip breakers open, no more than ONE rod NOT fully inserted, and neutron flux decreasing.

QUESTION: 024 (1.00)

WHICH ONE (1) of the following parameters discriminates between a vapor space LOCA and a non-vapor space LOCA?

a. Pressurizer pressure

b. Pressurizer level

c. PRT pressure

d. Hot leg temperature

QUESTION: 025 (1.00)

Given the following information:

-The plant has been shutdown for 4 days and is at 140 F with RHR in service.

-A total loss of CCW occurs, with recovery of CCW projected to take 1 hour.

-The RCS is intact and not in midloop, with all systems available EXCEPT CCW.

WHICH ONE (1) of the following actions should be taken first?

a. Implement decay heat removal via primary system feed and bleed to radwaste using normal charging and letdown via RHR.

b. Implement decay heat removal via primary system feed and bleed to radwaste via PRT and RCS drain tank.

c. Implement decay heat removal via the primary system feed and bleed to containment.

d. Implement decay heat removal via the steam generators and AFW.

QUESTION: 026 (1.00)

Given the following information:

-The plant is in Mode 3.
-An Intermediate Range NI fails.

WHICH ONE (1) of the following actions should be taken to assure the associated Source Range Channel remains operable?

a. Pull the instrument fuses on the affected Intermediate Range Channel.

b. Place the HV Manual ON/OFF switch on the associated Source Range Channel in the "ON" position.

c. Place the Test Mode Selector Switch for the affected Intermediate Range Channel in the "FIXED" position.

d. Place the Operation Selector Switch for the affected Intermediate Range Channel in the "CPS LOCAL" position.

QUESTION: 027 (1.00)

WHICH ONE (1) of the following is the immediate action required when steam generator tube leakage is identified?

a. Monitor leak rate to assure it does not exceed 10 gpm.

b. Monitor RCS pressure and pressurizer level to determine if a Reactor trip and Safety Injection is required.

c. Lower the affected steam generator level to assure the generator will not overfill during recovery.

d. Actuate or verify actuated S/G blowdown isolation.

QUESTION: 028 (1.00)

WHICH ONE (1) of the following individuals, by title, can authorize operation of caution tagged equipment?

a. Control Operator

b. Equipment Control Supervisor

c. Maintenance Supervisor in charge of work

d. Nuclear Plant Equipment Operator

QUESTION: 029 (1.00)

Given the following:

- Power level 80%
- Pressurizer level is at program setpoint.
- Tavg recorder pens all indicate approximately 549 degrees F.
- Tavg/Tref are matched on recorder TR-405.
- Rods start stepping in at high speed.
- No turbine runback is in progress.

WHICH ONE (1) of the following would cause these indications?

a. TM-407 - Avg Tave module to Rod Control loss of power.

b. TE-401A, Loop A hot leg temperature failed high.

c. TM-405 - Avg Tave summing computer failed low.

d. PT-415, 1st Stage pressure failed high.

QUESTION: 030 (1.00)

WHICH ONE (1) of the following will cause an alarm to actuate in the State Office of Emergency services?

a. Wide Range Gas Monitor (R-1254) is at its high level alarm setpoint for greater than 10 minutes and the Auto Alert System Keyswitch in the AUTO position.

b. Wide Range Gas Monitor (R-1254) is at its high level alarm setpoint for greater than 10 minutes and the Auto Alert System Keyswitch in the OPERATE position.

c. Wide Range Gas Monitor (R-1254) is at its high level alarm setpoint for greater than 15 minutes and the Auto Alert System Keyswitch in the AUTO position.

d. Wide Range Gas Monitor (R-1254) is at its high level alarm setpoint for greater than 15 minutes and the Auto Alert System Keyswitch in the OPERATE position.

QUESTION: 031 (1.00)

WHICH ONE (1) of the following permissives prevents MANUAL and AUTOMATIC rod withdrawal?

a. P-1 - Overpower Rod Stop

b. P-2 - Low Power Cutout

c. P-3 - Rod Drop Rod Stop

d. P-5 - Shutdown Margin Alarm

QUESTION: 032 (1.00)

WHICH ONE (1) of the following conditions would prevent the Emergency Diesel Generator from starting automatically on a LOP?

a. Jacket water temperature of 210 degrees F.

b. Turbo oil pressure of 15 psig

c. Starting air pressure 145 psig

d. Lube oil temperature 90 degrees F.

QUESTION: 033 (1.00)

Given the following conditions:

- Plant is in Mode 4
- North Saltwater pump is running.
- CCW HX Outlet Valve, MOV-720B,is OPEN.
- The Reactor Operator takes CCW HX Outlet Valve, MOV-720B control switch to CLOSE.

WHICH ONE (1) of the following describes the response of MOV-720B to the CLOSE signal?

a. Valve does not respond to the CLOSE signal, it will remain full OPEN.

b. Valve moves in the CLOSED direction only while switch is held in the "CLOSE" position.

c. Valve will CLOSE and remain CLOSED.

d. Valve will CLOSE and then REOPEN.

QUESTION: 034 (1.00)

Given the following:

- Reactor is at 92% steady state operation.
- SO1-2.3-1 "Control Rod System Malfunctions" in use.
- Operators are determining the operability of the RPI system.

WHICH ONE (1) of the following is the MINIMUM criteria requiring plant shutdown due to RPI INOPERABILITY per Technical Specification?

a. THREE (3) rod position indicators per bank inoperable.

b. TWO (2) rod position indicators per bank NOT capable of determining rod position within +/- 21 steps in a single bank.

c. LVDT current checks on more than one bank +/- 20 % of calibrated values.

d. ONE (1) step counter per bank NOT capable of determining rod position within +/- 21 steps in a single bank.

QUESTION: 035 (1.00)

Given the following:

- Condenser backpressure is 5.6" Hg Abs.
- Unit load is 290 Mwe.

WHICH ONE (1) of the following is the required operator response to the above conditions? (Use Attachment 1 of SO1-2.4-3)

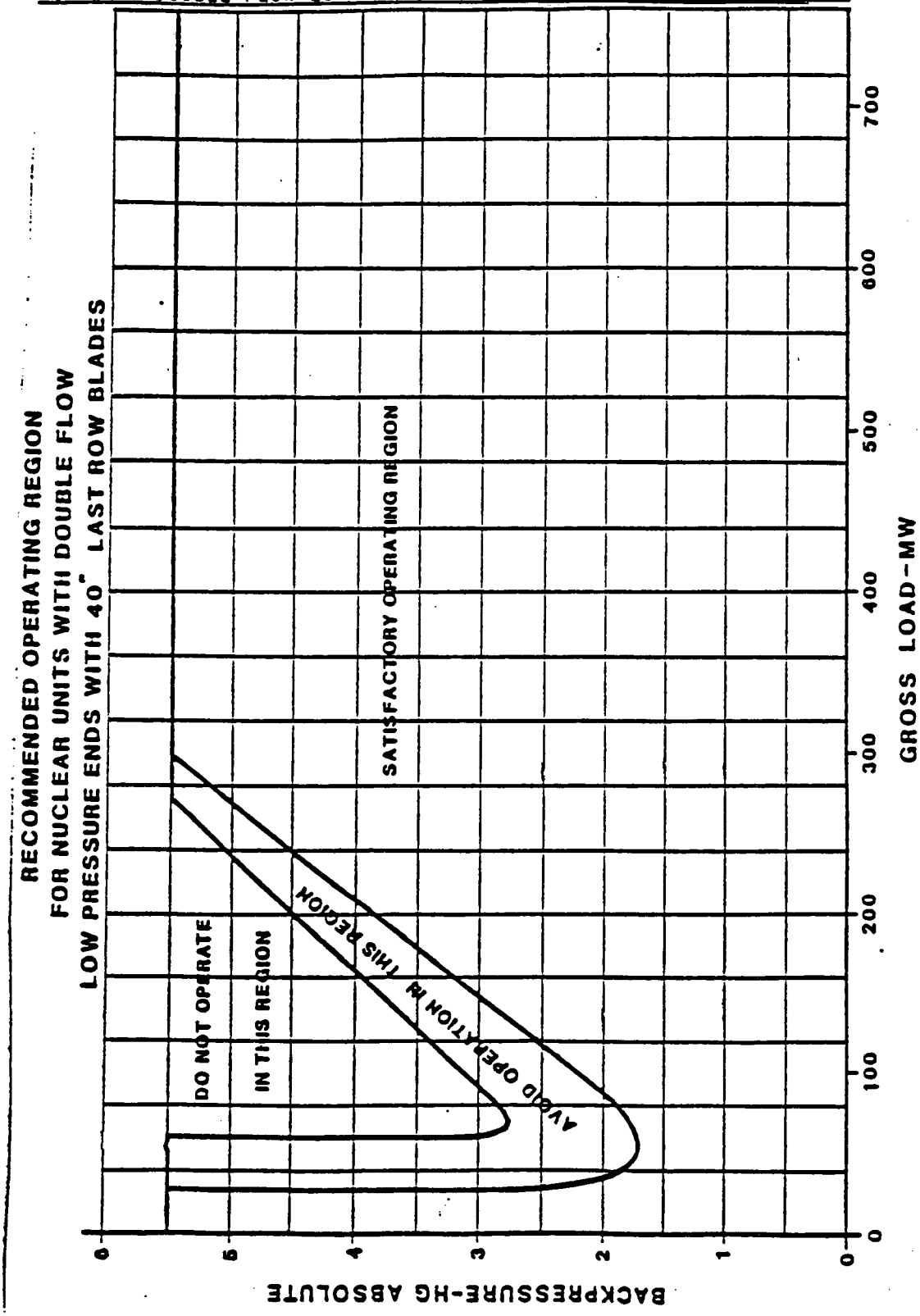
a. Reduce unit load

b. Manually start vacuum pump to adjust backpressure

c. Immediately trip the turbine

d. Immediately trip the reactor then the turbine.

RECOMMENDED OPERATIONAL REGIONS FOR NUCLEAR UNITS
WITH 2 - DOUBLE FLOW LOW PRESSURE ENDS WITH 40" LAST ROW BLADES



QUESTION: 036 (1.00)

WHICH ONE (1) of the following is the preferred order of power sources for recovery from a loss of all AC power if the Main XFMR is not available?

a. Aux XFMR C, Emergency Diesel Generator, DSD

b. Aux XFMR C, Emergency Diesel Generator, SDG&E 12 Kv

c. EDG, Aux XFMR C, DSD

d. EDG, Aux XFMR C, SDG&E 12 Kv

QUESTION: 037 (1.00)

WHICH ONE (1) of the following will AUTOMATICALLY occur on a loss of the Utility Bus?

a. Pressurizer heaters trip OFF.

b. Charging and Letdown ISOLATE.

c. AFW Train "A" ACTUATES.

d. AFW Train "B" ACTUATES.

QUESTION: 038 (1.00)

WHICH ONE (1) of the following is required to be performed prior to placing the DTT (Diverse Turbine Trip) system in bypass during abnormal operations?

a. Perform Turbine trip test per SO1-12.9-3 "Offline Turbine Trip Test".

b. Direct PEO under SRO supervision to hold Main Turbine Overspeed Test lever in the "TEST" position while the DTT is removed from service.

c. A determination is made of the required compensatory action for lack of automatic response .

d. DTT System Trip test is completed.

QUESTION: 039 (1.00)

Given the following conditions:

- Preparations are being made to release 2 liquid radioactive sources concurrently.
- R-1218 is in service.
- Release permits have been prepared for each source indicating that the sources can be released concurrently.

WHICH ONE (1) of the following conditions must be met prior to the start of the concurrent discharge?

a. Plant Manager approval, total number of MPC's of all sources released calculated to be less than ONE (1) prior to dilution.

b. Chemistry Department approval, total number of MPC's of all sources released calculated to be less than ONE (1) prior to dilution.

c. Plant Manager approval, total number of MPC's of all sources released calculated to be less than ONE (1) after dilution.

d. Chemistry Department approval, total number of MPC's of all sources released calculated to be less than ONE (1) after dilution.

QUESTION: 040 (1.00)

WHICH ONE (1) of the following is the MINIMUM required hydrazine flow rate if Containment Spray is required while performing SO1-1.0-10, "Reactor Trip or Safety Injection"?

a. 0.1 gpm

b. 0.4 gpm

c. 0.7 gpm

d. 1.0 gpm

QUESTION: 041 (1.00)

WHICH ONE (1) of the following is a reason for starting the RCP's in SO1-1.40, "Steam Generator Tube Rupture"?

a. Restore RCS inventory

b. Maintenance of subcooling

c. Minimize primary to secondary leak rate.

d. Eliminate reactor vessel head voiding

QUESTION: 042 (1.00)

WHICH ONE (1) of the following would indicate that the PRT rupture disk had blown following a PORV steam space accident?

a. Pressurizer level decreasing

b. Relief line temperatures increasing

c. PRT temperature decreasing

d. PRT level low

QUESTION: 043 (1.00)

WHICH ONE (1) of the following is the reason that a dropped rod must be recovered within THREE (3) hours?

a. Xenon transient initiated by dropped rod at BOL could shutdown the reactor if not recovered within THREE hours.

b. Axial Flux Difference will be out of band if not recovered within THREE hours.

c. Violation of minimum Technical Specifications shutdown margin requirements will result if not recovered in THREE hours.

d. Fuel damage could result if not recovered in THREE hours.

QUESTION: 044 (1.00)

Given the following:

- Mode 6, refueling complete.
- Lowering cavity level for head reinstallation.
- Loss of RHR occurs.
- Maintenance foreman wants to delay the containment closure for removal of equipment.

WHICH ONE (1) of the following core exit temperatures is the LIMIT when containment closure MUST be complete?

a. 140 degrees F.

b. 170 degrees F.

c. 200 degrees F.

d. 212 degrees F.

QUESTION: 045 (1.00)

WHICH ONE (1) of the following is a reason why Control Room Emergency Air Treatment System Fan A-33 is NOT started following a toxic gas hazard notification?

a. Butane/Propane could cause ignition of the filter.

b. Starting the system will draw more toxic fumes into the control room.

c. The TSC HVAC system, when placed in the filter mode, provides the control room with better protection against toxic gases.

d. Starting the system will make the filters inoperable for radiocative particulate.

QUESTION: 046 (1.00)

Given the following:

- Power is 10 E-5%.
- Permissive annunciator #4, "S/U RATE TRIPS ACTIVE" is energized.
- N-1204, Intermediate Range level and SUR indications peg high.

WHICH ONE (1) of the following describes the response of the Nuclear Instrumentation/Reactor Protection system to this failure?

a. N-1204 High SUR trip bistable trips, no reactor trip occurs.

b. N-1204 High SUR trip bistable trips, reactor trip occurs.

c. N-1202 Source Range High volts de-energize, reactor trip occurs.

d. N-1201 Source Range High volts de-energize, no reactor trip occurs.

QUESTION: 047 (1.00)

WHICH ONE (1) of the following would align backup nitrogen to the PORV's and their block valves during a Dedicated Safe Shutdown condition?

a. Depressing the Contaiment Isolation OVERRIDE PB for CV-532, Containment Isolation valve for Nitrogen to the PORV's.

b. Depressing the Contaiment Isolation OPEN PB for CV-532, Containment Isolation valve for Nitrogen to the PORV's.

c. Locally OPENING the manual bypass around CV-532, Containment Isolation valve for Nitrogen to the PORV's.

d. Locally OPENING CV-532, Containment Isolation valve for Nitrogen to the PORV's.

QUESTION: 048 (1.00)

Given the following:

- A Large break LOCA has occurred.
- Recirculation cooling mode has been in effect for 20 hours.
- Both trains of recirculation flow have been lost.
- RWST level is 5%.
- Operators are attempting to establish cold leg injection from an alternate suction supply while refilling RWST.

WHICH ONE (1) of the following is the PRIMARY alternate suction source for cold leg injection while refilling the RWST?

a. Unit 2/3 RWST

b. Unit 1 Radwaste Holdup Tank

c. Unit 1 Spent Fuel Pool

d. Unit 1 BAST and Primary Makeup Tank

QUESTION: 049 (1.00)

WHICH ONE (1) of the following represents the MAXIMUM allowable whole body dose in a calendar year for a Short Term Worker?

a. 200 mrem

b. 300 mrem

c. 400 mrem

d. 500 mrem

QUESTION: 050 (1.00)

Given the following:

- Turbine Runback from Under frequency has reduced power to 60% (Overshoot on turbine controls)
- Rod Control System in manual for duration of runback
- 10.5 degree F Tavg-Tref mismatch

WHICH ONE (1) of the following rod speeds would be observed if the Rod Control Selector switch was taken to the AUTO position? (Figure 5 of SD-SO1-400, "Rod Control System" is attached)

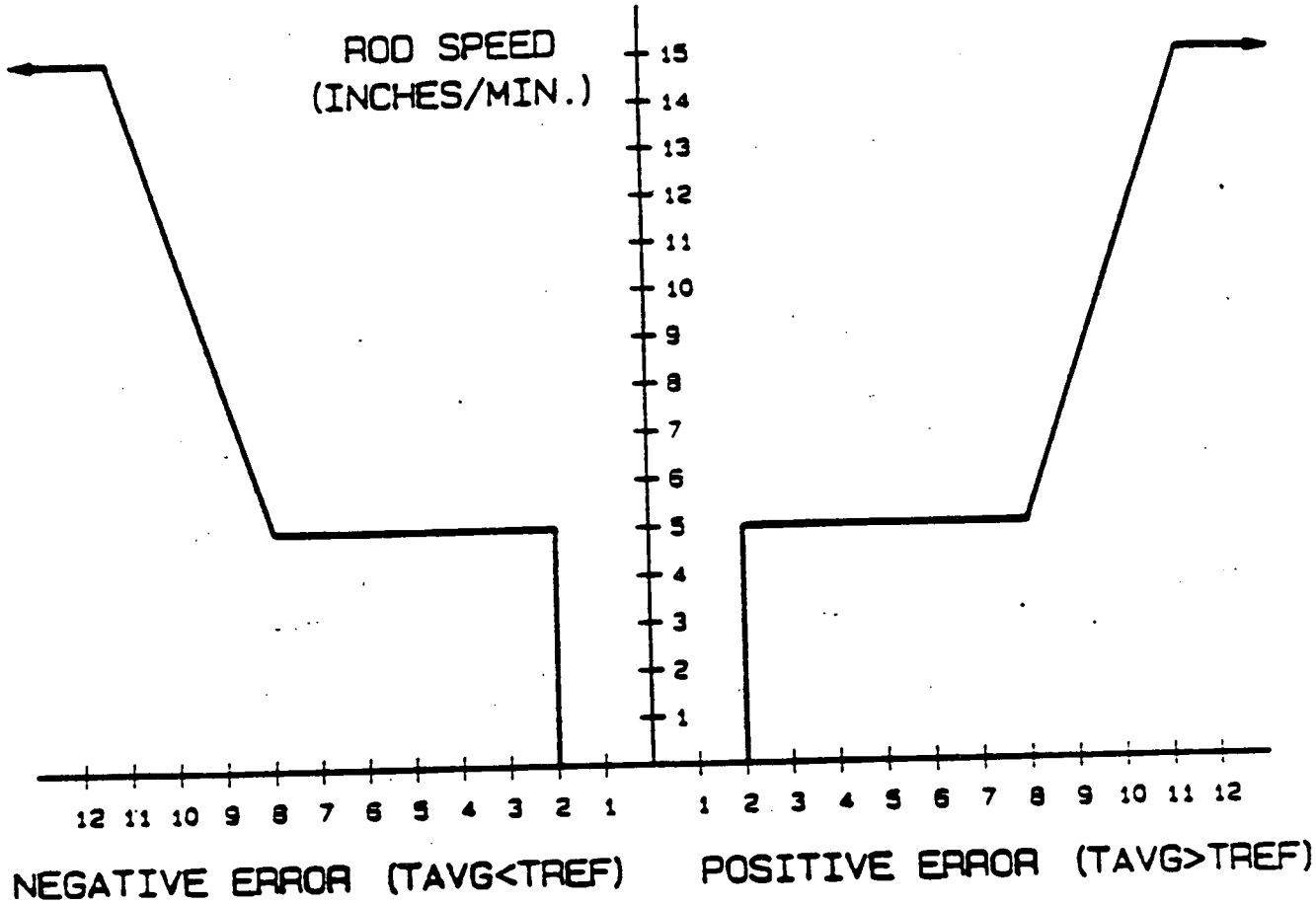
a. 8.75 inches/min

b. 9.25 inches/min

c. 11.25 inches/min

d. 13.0 inches/min

FIGURE 5: ROD CONTROL PROGRAM



QUESTION: 051 (1.00)

WHICH ONE (1) of the following, if present in a sample from the Decontamination Drain Tank, would allow the operator to pump the tank to the HUT tank?

a. Gross activity in excess of MDA.

b. 35 ppm Oil

c. Colored water

d. Foam

QUESTION: 052 (1.00)

WHICH ONE (1) of the following are the normal and backup AC power supplies to the rod control system?

a. Normal: MCC 1, Backup: MCC 2

b. Normal: MCC 3, Backup: MCC 4

c. Normal: MCC 4, Backup: MCC 3

d. Normal: MCC 2, Backup: MCC 1

QUESTION: 053 (1.00)

WHICH ONE (1) of the following Source Range Count rate and Audible Multiplier Switch position give the procedurally required audible beep rate?

	Actual SR Count Rate	Audible Multiplier Switch
a.	12 cps	100
b.	120 cps	100
c.	4 cps	10
d.	12 cps	10

QUESTION: 054 (1.00)

WHICH ONE (1) of the following is the LOWEST power level for a Power Range Nuclear Instrumentation high level Reactor trip, assuming a normal power distribution?

a. 101%

b. 104%

c. 110%

d. 114%

QUESTION: 055 (1.00)

Given the following information:

Following an accident the Core Exit Thermocouples indicate 680 degrees F with an RCS pressure of 350 psig.

WHICH ONE (1) of the following conclusions can be drawn from the above information?

a. Core geometry is lost

b. Core Exit Thermocouples have failed

c. Core uncovery is occurring

d. Core cooling mode is reflux

QUESTION: 056 (1.00)

WHICH ONE (1) of the following is the reason for the post-accident injection of hydrazine into the Containment Spray system?

a. Containment oxygen removal

b. Containment iodine removal

c. Containment ph control

d. Containment spray piping integrity

QUESTION: 057 (1.00)

WHICH ONE (1) of the following is a positive barrier between the RCS and the feedwater system?

a. Auto SIS actuation defeat

b. Control power removed from the MFP's

c. MOV 850 A, B, and C CLOSED and Operating power removed

d. De-energizing the sequencers

QUESTION: 058 (1.00)

WHICH ONE (1) of the following is the LOWEST steam generator level at which compensating measures are not required to avoid feed ring water hammer, when the plant is at 15% power?

a. 8%

b. 24%

c. 38%

d. 52%

QUESTION: 059 (1.00)

WHICH ONE (1) of the following describes the response of the Sphere Sump pumps to a Sphere Sump HI - HI level alarm?

a. East Sphere Sump Pump starts

b. West Sphere Sump Pump starts

c. The Sphere Sump Pump in standby starts

d. Both Sphere Sump Pumps trip

QUESTION: 060 (1.00)

EXCEEDING WHICH ONE (1) of the following RCS pressures is a Technical Specification one hour reportable event?

a. 2085 psig

b. 2485 psig

c. 2500 psig

d. 2735 psig

QUESTION: 061 (1.00)

WHICH ONE (1) of the following will immediately occur if PT 430 fails low when selected as a controlling channel?

a. Safety Injection alarm

b. Spray Valves open

c. Pressurizer Heaters in auto are energized

d. Variable Low Pressure Trip Channel 2 alarm

QUESTION: 062 (1.00)

WHICH ONE (1) of the following is lowest level at which a Pressurizer High Level Reactor trip could be expected to occur?

a. 55%

b. 65%

c. 85%

d. 95%

QUESTION: 063 (1.00)

WHICH ONE (1) of the following is the Reactor trip that protects the plant during a loss of load accident?

a. Pressurizer High Level Trip

b. Nuclear Overpower Trips

c. Variable Low Pressure Trip

d. Steam/Feedwater Flow Mismatch

QUESTION: 064 (1.00)

WHICH ONE (1) of the following statements is the required position for the steam dump valves when auto steam dump has initiated following a loss of MW signal, and with a current 4 degree T ave - T ref delta T?

	Atmospheric Dumps	Condenser Dumps
a.	Full open	Partially open.
b.	Full open	Full open.
c.	Partially open	Full open.
d.	Full closed	Full closed.

QUESTION: 065 (1.00)

Given the following information:

SO1-7-10 "Condenser Air Removal System" operating procedure, contains instructions to uncap and throttle open the East and West Air Ejector Second Stage Vents.

WHICH ONE (1) of the following is the reason for the above step?

a. To place the air ejectors into the hogging mode.

b. To minimize steam usage by the air ejectors.

c. To establish minimum flow for R-1215.

d. To establish a flowpath for condensate to escape from the air ejector.

QUESTION: 066 (1.00)

Given the following information:

-Train A Emergency Diesel Generator is inoperable.
-92% Power Operation.

WHICH ONE (1) of the following statements describes the operability of the other A train equipment?

a. All systems, equipment, components, or devices which normally receive emergency power from the train A Emergency Diesel Generator are also inoperable.

b. All systems, equipment, components, or devices which normally receive emergency power from the train A Emergency Diesel Generator are also inoperable, except those which are powered by an operable battery.

c. The operability of the remaining train A equipment is not impacted, but the train B equipment is required to be verified operable per Technical Specification 3.7.1.

d. The operability of the remaining train A equipment is not impacted, except for the ESF electrical bus that the Emergency Diesel Generator supports.

QUESTION: 067 (1.00)

WHICH ONE (1) of the following is the power supply for the turbine generator bearing emergency oil pump?

a. DC Bus #1

b. DC Bus #2

c. 850 C UPS

d. DSD 125 VDC System

QUESTION: 068 (1.00)

WHICH ONE (1) of the following assures the CCW pump will not air bind when flow is initiated through the recirculation heat exchanger?

a. The operators vent the system thoroughly prior to placing it in service.

b. The recirculation heat exchanger has automatic high point vents.

c. Continuous vents are installed on the CCW pumps to the surge tank.

d. The system is operated with 25 - 30 psig cover gas on the surge tank.

QUESTION: 069 (1.00)

WHICH ONE (1) of the following will cause an auto start of the containment spray hydrazine pumps?

a. Safety Injection actuation

b. Containment high radiation level on R 1212

c. Safety Injection AND containment pressure of 7 psig

d. Hydrazine tank low-low temperature

QUESTION: 070 (1.00)

WHICH ONE (1) of the following is the required plant condition when the RCS is open to containment atmosphere when the plant is in MODE 5?

a. Maintain containment integrity and greater than 5% delta k/k shutdown margin.

b. Maintain control rods fully inserted and greater than 5% delta k/k shutdown margin.

c. Maintain Sphere Purge in service and greater than 1% delta k/k shutdown margin.

d. Maintain containment integrity and do not allow positive reactivity changes.

QUESTION: 071 (1.00)

Given the following information:

-The Reactor is subcritical with the first control bank just off the bottom.
-Control rod withdrawal is in progress.
-The in-hold-out switch is returned to the hold position, and rods continue to step out.

WHICH ONE (1) of the following actions is required?

a. Monitor the position of the control rods, and trip the Reactor just prior to the estimated critical position if the rods do not stop by then.

b. Verify the rod control is in manual, check if rod motion is stopped, and if rod motion continues trip the Reactor.

c. Locally open the reactor trip breakers in the 4 KV room, or locally open DC supply breaker 72-141 to control rods in No. 1 DC room.

d. Transfer rod control to Auto, check if rod motion is stopped, and if rod motion continues trip the Reactor.

QUESTION: 072 (1.00)

Given the following information:

The plant is at 30% power, when it is noted that the RCP seal return temperature is 210 degrees F on an RCP.

WHICH ONE (1) of the following immediate actions is to be performed FIRST?

a. Trip the Reactor

b. Trip the affected RCP

c. Commence load reduction to less than 10% power and be at less than 10% within 1 hour.

d. Monitor and log seal return temperature once per shift. IF temperature continues to trend up, then commence rapid load reduction to less than 10% power.

QUESTION: 073 (1.00)

WHICH ONE (1) of the following is the basis for tripping the turbine during an ATWS?

a. To avoid motoring (reverse power) of the main generator.

b. To limit the RCS pressure increase in the event of a loss of feed ATWS.

c. To maintain steam generator inventory (level) high enough to avoid water hammer when Aux Feed is initiated.

d. To reduce reactor power (due to MTC) by increasing T ave.

QUESTION: 074 (1.00)

WHICH ONE (1) of the following condenser vacuum pressures will result in a turbine trip?

a. 1.6 to 2.0 psia increasing

b. 8 to 12 in Hg decreasing

c. 16 to 20 in Hg decreasing

d. 18 to 22 in Hg decreasing

QUESTION: 075 (1.00)

Given the following information:

-The plant is operating at 92% power and a loss of all AC occurs.
-SO1-1.0-60 "Loss of All AC Power" is followed, and due to extensive damage to switchgear power restoration appears to be several hours away.
-A RED PATH exists on Heat Sink.

WHICH ONE (1) of the following actions must be taken?

a. Transition to SO1-1.3-1 "Response to Loss of Secondary Heat Sink".

b. Stop the implementation of SO1-1.0-60 "Loss of All AC Power" and immediately direct that the steam driven AFW pump be restarted.

c. Continue with SO1-1.0-60 "Loss of All AC Power" and restore feed as directed by the procedure.

d. IMMEDIATELY depressurize the RCS by opening the primary PORV's, to the point where the accumulators will inject to provide core cooling.

QUESTION: 076 (1.00)

Given the following information:

-The plant is at 10% power.
-The "CONTAIN. INSIDE SEC SHIELD" and "CONTAIN. OUTSIDE SEC SHIELD" alarms are in alarm on the Fire Systems Annunciator Panel.
-The "RCP MOTOR AREA OR SPHERE AREA HIGH TEMP" alarm is in on the Sphere Heat and Vent Annunciator Panel.
-PORV Block Valve CV-530 spuriously closes.
-RCP C trips on overcurrent.

WHICH ONE (1) of the following actions should be taken?

a. Dispatch an Operator to verify the condition on R-9 at the Heat and Vent Panel.

b. Initiate a plant shutdown per SO1-2.7.2 "Plant Shutdown Using the Dedicated Shutdown System".

c. Initiate a plant shutdown using normal shutdown procedures (SO1-3-4 and SO1-3-5).

d. Initiate action per SO1-2.7-1 "Appendix R Fire Zone Response" and MANUALLY actuate containment spray for 2 minutes for fire suppression, then trip the plant if symptoms of the fire exist.

QUESTION: 077 (1.00)

WHICH ONE (1) of the following immediate actions must be taken in the event of a forced evacuation from the control room?

a. Verify open unit CB's 4012 and 6012.

b. Actuate Safety Injection.

c. Start the Aux Feed pumps.

d. Close the Primary PORV block valves, and trip the RCP's.

QUESTION: 078 (1.00)

WHICH ONE (1) of the following plant conditions satisfy CONTAINMENT INTEGRITY, assuming at least one door in each personnel airlock and the equipment hatch are properly closed?

a. All non-automatic and automatic containment isolation valves (or blind flanges) are closed.
All steam generator secondary manway covers are installed.

b. All non-automatic containment isolation valves (or blind flanges) are closed.
The actuation system for containment isolation is operable.

c. All non-automatic and automatic containment isolation valves (or blind flanges) are operable.
The actuation system for containment isolation is operable.

d. All non-automatic containment isolation valves (or blind flanges) are closed.
All automatic containment isolation valves are operable.

QUESTION: 079 (1.00)

WHICH ONE (1) of the following plant conditions exhibits inadequate core cooling?

a. 0% Pressurizer level, Containment pressure 4 psig, Core exit TC's 950-1000 degrees F.

b. 5% Pressurizer level, Wide range SG levels all less than 25%, RCS subcooling 25 degrees F.

c. 3 core exit TC's at 1300 degrees F, rest of core exit TC's 1000-1100 degrees F, RCS subcooling 15 degrees F.

d. All core exit TC's between 900-1000 degrees F, and 2 RCS hot leg RTD's 700 degrees F, rest of RCS hot leg RTD's at 600-650 degrees F.

QUESTION: 080 (1.00)

WHICH ONE (1) of the following is the bases for reducing T ave to less than 535 F following a shutdown required by a Dose Equivalent I-131 level of 90 micro curies per gram?"

a. Slows coolant/fuel reaction rate, immediately reducing the source term of the activity.

b. Prevents the release of activity following a steam generator tube rupture.

c. Minimize the temperature related degradation of the CVCS demineralizers while the RCS clean-up is in progress.

d. Accommodation of the iodine spiking phenomena which occurs due to the large change in THERMAL POWER caused by the unit shutdown.

QUESTION: 081 (1.00)

Given the following information:

A major plant transient has occurred:
-Pressurizer level and pressure are rapidly decreasing
-Reactor Trip and Safety Injection have occurred
-Containment pressure and temperature are increasing
-Steam generator levels decreased initially, then stabilized
-Steam header pressure stable

WHICH ONE (1) of the following conditions have occurred?

a. Steam space RCS break

b. Non steam space RCS break inside containment

c. Steam line break inside containment

d. Feed line break inside containment

QUESTION: 082 (1.00)

Given the following information:

Generic analysis has shown that if RCP's are left running during a small break LOCA and tripped later in the accident, more severe core uncovery and fuel damage could result due to the increased mass loss through the break while the RCP'S were running.

WHICH ONE (1) of the following statements describes the sensitivity of SONGS 1 to RCP operation during a small break LOCA?

a. SONGS 1 does not have a problem with inventory loss during a small break LOCA with RCP's running due to having high volume safety injection.

b. SONGS 1 is more impacted by this issue than the generic analysis would indicate, which is why the design change was installed to automatically trip the RCP's upon SI initiation.

c. SONGS 1 is closely represented by the generic analysis, and RCP restart during a small break LOCA must be only as a last ditch cooling effort.

d. SONGS 1 does not have a problem with inventory loss while running RCP's with a small break LOCA, due to EOP instructions which rapidly depressurize the RCS to get adequate SI flow to the core.

QUESTION: 083 (1.00)

Given the following information:

-The plant is in MODE 1.
-A PORV block valve was closed, and would not reopen.

WHICH ONE (1) of the following must be performed within 1 hour per Technical Specifications?

a. Restore the valve to OPERABLE status.

b. CLOSE and deactivate the block valve.

c. Initiate action to place the unit in a MODE that the specification does not apply.

d. Verify the other PORV block valve is OPERABLE.

QUESTION: 084 (1.00)

Given the following information:

-The plant is in Mode 3 with the Reactor trip breakers CLOSED.
-BOTH Source Range channels fail.

WHICH ONE (1) of the following actions is required for the conditions described?

a. Suspend all operations involving positive reactivity changes.

b. Restore channel to operable status within 48 hours or open Reactor trip breakers within the next hour.

c. Within 1 hour take action to place the unit in a MODE that the specification does not apply.

d. Verify shutdown margin requirements within 1 hour.

QUESTION: 085 (1.00)

Given the following information:

-Pressurizer Level Controller LC-430F output has failed high.
-Assume no operator action is performed, and all controls are in auto.

WHICH ONE (1) of the following will be the first auto action to occur?

a. FCV-1112 will go full open.

b. A PZR high level trip will occur.

c. Letdown isolation LCV-1112 closes.

d. PZR heaters energize at +4% program level.

QUESTION: 086 (1.00)

WHICH ONE (1) of the following is the basis for maintaining a MINIMUM of approximately 5% narrow range level in a ruptured steam generator?

a. To minimize radiological concerns.

b. To avoid overfilling the ruptured steam generator.

c. To provide adequate RCS cooling.

d. To avoid water hammer due to abnormal steam generator levels.

QUESTION: 087 (1.00)

WHICH ONE (1) of the following requires the CRS to evaluate sending two operators together (performer and checker) to complete a system alignment?

a. Alignments involving a potential personnel safety hazard.

b. Alignments that could affect plant efficiency.

c. Alignments removing "Important to Safety" equipment not required to be OPERABLE for the current mode.

d. Alignments which are complex, on non-safety related equipment.

QUESTION: 088 (1.00)

WHICH ONE (1) of the following methods should be used to VERIFY an ESF locked manual valve in the CLOSED position? The valve is accessible, with no personnel or radiological safety concerns.

a. Visual observation of the installed position indication device.

b. Unlock and turn the valve in the CLOSED direction using normal closing torque.

c. Verify by the previous system alignment.

d. Unlock and turn the valve slightly in the OPEN direction then in the CLOSED direction using normal closing torque.

QUESTION: 089 (1.00)

WHICH ONE (1) of the following individuals, by position, has the authority to release systems and equipment for isolation, maintenance, or testing?

a. Unit Assistant Control Operator

b. Unit Control Operator

c. SRO Operations Supervisor

d. Shift Superintendent

QUESTION: 090 (1.00)

WHICH ONE (1) of the following types of Work Authorizations must be used to allow a diver to enter the Circulating Water System to inspect the system piping?

a. In-Test

b. Approval

c. Clearance

d. Permission

QUESTION: 091 (1.00)

WHICH ONE (1) of the following is the MAXIMUM time for classification and declaration of an emergency after recognition of an abnormal plant condition?

a. As soon as possible.

b. Within 15 minutes.

c. Within 30 minutes.

d. Within 45 minutes.

QUESTION: 092 (1.00)

WHICH ONE (1) of the following requires a Temporary Facility Modification?

a. A Design Change package.

b. A non-conformance report with a non-permanent disposition of "Repair".

c. A Work Authorization that will not remain open for the duration of the modification.

d. An approved procedure which remains in use for the duration of the modification.

QUESTION: 093 (1.00)

WHICH ONE (1) of the following actions must be taken when a procedure step CANNOT be performed because a Work Authorization Record (WAR) is holding part of the system out of service for maintenance?

a. Place N/A in the procedure step sign-off/initial space, and explain in comments.

b. Initial the procedure step sign-off/initial space, and explain in comments.

c. Remove the step by a single line, initial, and date.

d. Place A/C in the procedure step sign-off/initial space, and put the WAR number in comments.

QUESTION: 094 (1.00)

WHICH ONE (1) of the following is the remaining annual Administrative Radiation Dose available to a 40 year old male Radiation Worker, who has a lifetime dose of 20 R and received 1250 mrem in the current year?

a. 750 mrem

b. 1000 mrem

c. 1250 mrem

d. 1750 mrem

QUESTION: 095 (1.00)

WHICH ONE (1) of the following (measured at 30 cm from the source) is the LOWEST AMOUNT which would require posting as a HIGH RADIATION AREA?

a. 25 mrem

b. 250 mrem

c. 2.5 R

d. 25 R

QUESTION: 096 (1.00)

Given the following conditions:

- One Saltwater Pump is inoperable due to a grounded motor. Maintenance projects the repairs will take about 1 week to complete.
- A plant shutdown has been initiated 30 hours following the failure of the Saltwater Cooling Water Pump.

WHICH ONE (1) of the following is the MAXIMUM time allowable to report to the NRC per 10CFR50.72?

a. Fifteen minutes

b. One hour

c. Four hours

d. Twenty four hours

QUESTION: 097 (1.00)

WHICH ONE (1) of the following is a NON-DELEGABLE responsibility of the Station Emergency Director following turnover of the Emergency Coordinator title to the Corporate Emergency Director?

a. Emergency Event Declaration/Classification

b. Exceeding 10CFR100 Exposure Limits

c. Notification to Offsite Agencies

d. Offsite Protective Action Recommendations

UESTION: 098 (1.00)

WHICH ONE (1) of the following would be a violation of the NRC Guidelines for Operator Overtime?

a. An ACO worked a 16 hour shift on his last night shift in and worked 16 hours on his second day off following the night shift.

b. A CO was working a 16 hour shift during a Unit startup and ended up staying 17 hours plus a long shift turnover.

c. A CRS worked a 12 hour shift 5 out of 7 days.

d. A SS worked 17 - 8 hour days in a row without a day off.

QUESTION: 099 (1.00)

WHICH ONE (1) of the following Radiation Exposure Permits is issued to cover a particular job, and expires upon completion of the job?

a. Special

b. Regular

c. Specific

d. Extended

QUESTION: 100 (1.00)

Given the following information:

The Security Leader reports that an unidentified individual was found tampering with a vital area door locking mechanism, apparently trying to force the door open. A security guard confronted the individual, who escaped into the protected area.

Using tab F1-F4 attached, WHICH ONE (1) of the following is the Emergency Classification for this event?

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

(********** END OF EXAMINATION **********)

SECURITY SAFEGUARDS CONTINGENCY

ATTACHMENT 2

UNUSUAL EVENT

TAB F1

1. For Modes 1 - 6:
The Shift Commander/Security Leader reports a security threat, attempted entry, or attempted sabotage. This includes any of the following Safeguards Contingency Plan conditions:

(a) A credible threat to attack the protected or vital area has been received.
(Contingency Threat Situation A1 (SECON Yellow))

(b) A credible threat to bomb the protected or vital area has been received.
(Contingency Threat Situation B1-A (SECON Yellow))

(c) A non-explosive sabotage attempt, to include tampering or unauthorized manipulation of vital area safety equipment or security-related safeguards equipment, for the purpose of sabotage or for gaining undetected or unauthorized protected or vital area entry, has been confirmed.
(Contingency Threat Situation S1 (SECON Orange))

(d) An attempt to introduce prohibited items (such as firearms, explosives or unauthorized incendiary devices) into the protected area, for the purpose of sabotage or disruption of plant activities has been discovered.
(Contingency Threat Situation M3-A (SECON Orange))

(e) Tampering with the protected area perimeter intrusion detection system (IDS) (e.g., E-fields, microwaves, and protected area entry card-readers) or the vital area IDS (e.g., vital area card-readers, portal locking hardware, or intrusion alarm mechanisms) for the purpose of gaining undetected or unauthorized access to the protected or vital area has been confirmed.
(Contingency Threat Situation I1 (SECON Yellow))

NOTE: If the condition described by the Shift Commander/Security Leader does not reasonably conform to one of the conditions above, declare an Unusual Event if the situation constitutes a security threat, attempted entry or attempted sabotage

SECURITY SAFEGUARDS CONTINGENCY

ATTACHMENT 2

UNUSUAL EVENT

TAB F1

(f) NRC notifies SONGS that a credible threat of attack by surface vehicle bomb(s) has been received, and that short-term contingency measures to protect against a surface vehicle bomb threat must be implemented within 12 hours.
(Contingency Threat Situation B1-B (SECON Orange))

NOTE: If the condition described by the Shift Commander/ Security Leader does not reasonably conform to one of the conditions above, declare an Unusual Event if the situation constitutes a security threat, attempted entry or attempted sabotage

SECURITY SAFEGUARDS CONTINGENCY

ATTACHMENT 2

ALERT

TAB F2

1. For Modes 1 - 6:
The Shift Commander/Security Leader reports an ongoing security compromise. This includes any of the following Safeguards Contingency Plan conditions:

(a) A bomb or unauthorized explosive device has been discovered in the protected or vital area.
(Contingency Threat Situation B2 (SECON Orange))

(b) An intruder (an unauthorized or unidentified person who purposely evades physical contact with security, or has escaped from security's custody) in the protected or vital area has been confirmed.
(Contingency Threat Situation I2 (SECON Orange))

(c) An unauthorized, forced entry through the protected perimeter barrier or a vital area barrier for the purpose of sabotage has been confirmed.
(Contingency Threat Situation I3 (SECON Orange))

(d) Discovery of prohibited items (such as firearms or unauthorized incendiary devices) inside the protected or vital area, for the purpose of sabotage, has been confirmed.
(Contingency Threat Situation M3-B (SECON Orange))

(e) An adversary force has assaulted the site, but has not attempted to or succeeded in penetrating the protected area perimeter. The adversary force has taken hostages and is barricaded in the owner controlled area.
(Contingency Threat Situation A4-A (SECON Red))

(f) A forceful attack on the protected or vital area by an adversary force is imminent. Adversary motives or extorsion demands may or may not be known. If not known, the assumption is made that radiological sabotage will be attempted.
(Contingency Threat Situation A2 (SECON Orange))

NOTE: If the condition described by the Shift Commander/Security Leader does not reasonably conform to one of the conditions above, declare an Alert if the situation constitutes an ongoing security compromise.

SECURITY SAFEGUARDS CONTINGENCY

ATTACHMENT 2

ALERT

TAB F2

(g) A bomb or unauthorized explosive device explodes within the protected area or vital area or a fire or explosion of suspicious origin, or other disruptive emergency occurs which evidence suggest/confirms willful intent to damage vital area safety equipment or security-related safeguards equipment.
(Contingency Threat Situation B3 (SECON Red))

(h) An act of sabotage has damaged or destroyed vital area safety equipment (may or may not result in a radiological release) or security safeguards equipment, but the security force maintains control of the protected area perimeter.
(Contingency Threat Situation S2-A (SECON Red))

(i) An adversary force has attacked the Site and has succeeded in penetrating the protected area perimeter. The adversary force has taken hostages and is barricaded within the protected area or vital area, but the security force maintains control of the protected area perimeter.
(Contingency Threat Situation A4-B (SECON Red))

A

NOTE: If the condition described by the Shift Commander/ Security Leader does not reasonably conform to one of the conditions above, declare an Alert if the situation constitutes an ongoing security compromise.

SECURITY SAFEGUARDS CONTINGENCY

ATTACHMENT 2

SITE AREA EMERGENCY

TAB F3

For Modes 1 - 6:

1. The Shift Commander/Security Leader reports the imminent loss of physical control of the protected area. This may include the following Safeguards Contingency Plan conditions:

   (a) A forceful assault on the protected area perimeter by an adversary force occurs, or has occurred and has succeeded in penetrating the protected area perimeter and is continuing in the protected or vital area.
   (Contingency Threat Situation A3 (SECON Red))

NOTE:	If the condition described by the Shift Commander/ Security Leader does not reasonably conform to one of the conditions above, declare a Site Area Emergency if the situation constitutes an imminent loss of physical control of the protected area.

SECURITY SAFEGUARDS CONTINGENCY

ATTACHMENT 2

GENERAL EMERGENCY

TAB F4

For Modes 1 - 6:

1. The Shift Commander/Security Leader reports the loss of physical control of the protected area. This may include either of the following Safeguards Contingency Plan conditions:

   (a) An adversary force has attacked the Site and has succeeded in penetrating the protected area perimeter. The adversary force has taken hostages and is barricaded within the protected area or vital area, and the security force has lost control of the protected area perimeter.
   (Contingency Threat Situation A4-B (SECON Red))

   (b) An act of sabotage has damaged or destroyed vital area safety equipment (may or may not result in a radiological release), and the security force has lost control of the protected area perimeter.
   (Contingency Threat Situation S2-B (SECON Red))

NOTE:	If the condition described by the Shift Commander/ Security Leader does not reasonably conform to one of the conditions above, declare a General Emergency if the situation constitutes a loss of physical control of the protected area.

R

A N S W E R K E Y

MULTIPLE CHOICE

001	a
002	b
003	b
004	d
005	c
006	b
007	d
008	c
009	a
010	d
011	a
012	c
013	d
014	d
015	a
016	a
017	b
018	c
019	b
020	c
021	a
022	a
023	b
024	b
025	d
026	b
027	b
028	a
029	a
030	d
031	a
032	c
033	d
034	b
035	a
036	c
037	a
038	c
039	b
040	b
041	d
042	c
043	d
044	c
045	d

A N S W E R K E Y

046	d	069	c
047	c	070	b
048	c	071	b
049	d	072	b
050	c	073	b
051	a	074	d
052	d	075	c
053	a	076	b
054	c	077	a
055	c	078	d
056	b	079	d
057	c	080	b
058	c	081	b
059	d	082	a
060	d	083	a
061	c	084	c
062	a	085	c
063	a	086	a
064	c	087	a
065	c	088	a
066	c	089	c
067	a	090	d
068	c	091	a

A N S W E R K E Y

092 c

093 d

094 c

095 b

096 b

097 a

098 b

099 b

100 b

(********** END OF EXAMINATION **********)

SRO Exam PWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
001	1.00	9000015
002	1.00	9000017
003	1.00	9000018
004	1.00	9000019
005	1.00	9000020
006	1.00	9000021
007	1.00	9000026
008	1.00	9000027
009	1.00	9000029
010	1.00	9000035
011	1.00	9000039
012	1.00	9000044
013	1.00	9000045
014	1.00	9000046
015	1.00	9000048
016	1.00	9000050
017	1.00	9000053
018	1.00	9000054
019	1.00	9000056
020	1.00	9000059
021	1.00	9000060
022	1.00	9000061
023	1.00	9000067
024	1.00	9000068
025	1.00	9000070
026	1.00	9000073
027	1.00	9000074
028	1.00	9000077
029	1.00	9000078
030	1.00	9000079
031	1.00	9000080
032	1.00	9000081
033	1.00	9000082
034	1.00	9000083
035	1.00	9000084
036	1.00	9000085
037	1.00	9000086
038	1.00	9000087
039	1.00	9000088
040	1.00	9000089
041	1.00	9000090
042	1.00	9000091
043	1.00	9000092
044	1.00	9000093
045	1.00	9000094
046	1.00	9000095
047	1.00	9000096
048	1.00	9000097
049	1.00	9000098

SRO Exam PWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
050	1.00	9000099
051	1.00	9000100
052	1.00	9000016
053	1.00	9000022
054	1.00	9000023
055	1.00	9000024
056	1.00	9000025
057	1.00	9000028
058	1.00	9000030
059	1.00	9000033
060	1.00	9000034
061	1.00	9000036
062	1.00	9000037
063	1.00	9000038
064	1.00	9000043
065	1.00	9000042
066	1.00	9000031
067	1.00	9000032
068	1.00	9000047
069	1.00	9000040
070	1.00	9000041
071	1.00	9000049
072	1.00	9000052
073	1.00	9000055
074	1.00	9000057
075	1.00	9000058
076	1.00	9000062
077	1.00	9000063
078	1.00	9000064
079	1.00	9000065
080	1.00	9000066
081	1.00	9000051
082	1.00	9000069
083	1.00	9000071
084	1.00	9000072
085	1.00	9000075
086	1.00	9000076
087	1.00	9000001
088	1.00	9000002
089	1.00	9000003
090	1.00	9000004
091	1.00	9000005
092	1.00	9000006
093	1.00	9000007
094	1.00	9000008
095	1.00	9000009
096	1.00	9000010
097	1.00	9000011
098	1.00	9000012

S R O E x a m P W R R e a c t o r

O r g a n i z e d b y Q u e s t i o n N u m b e r

QUESTION	VALUE	REFERENCE
099	1.00	9000013
100	1.00	9000014

	100.00	

	100.00	

S R O Exam PWR Reactor

Organized by KA Group

PLANT WIDE GENERICS

QUESTION	VALUE	KA
093	1.00	194001A102
098	1.00	194001A103
091	1.00	194001A116
097	1.00	194001A116
096	1.00	194001A116
100	1.00	194001A116
088	1.00	194001K101
089	1.00	194001K101
090	1.00	194001K102
087	1.00	194001K102
028	1.00	194001K102
092	1.00	194001K102
099	1.00	194001K103
094	1.00	194001K103
095	1.00	194001K103
049	1.00	194001K103
045	1.00	194001K111

PWG Total	17.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
052	1.00	001000K201
002	1.00	001000K401
003	1.00	001000K402
050	1.00	001000K402
001	1.00	003000A201
006	1.00	004000K115
005	1.00	013000K101
054	1.00	015000K105
053	1.00	015020A301
055	1.00	017020K503
007	1.00	022000A404
056	1.00	022000G004
008	1.00	056010K412
057	1.00	059000K105
009	1.00	061000K402
058	1.00	061000K505
067	1.00	063000K201
059	1.00	068000G007
051	1.00	068000K107

SRO Exam PWR Reactor

Organized by KA Group

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
PS-I Total	19.00	

Group II

QUESTION	VALUE	KA
060	1.00	002000G005
029	1.00	002020K509
010	1.00	006030A402
061	1.00	010000K302
062	1.00	011000K405
004	1.00	011000K604
031	1.00	012000K610
069	1.00	027000K101
011	1.00	035010K101
064	1.00	039000K402
065	1.00	055000K106
066	1.00	062000G005
012	1.00	064000A401
013	1.00	064000A406
032	1.00	064000K401
030	1.00	073000K101
070	1.00	103000G005

PS-II Total	17.00	

Group III

QUESTION	VALUE	KA
014	1.00	005000G005
033	1.00	008000A401
068	1.00	008000G007
015	1.00	076000K402

PS-III Total	4.00	

PS Total	40.00	

EMERGENCY PLANT EVOLUTIONS

Group I

S R O Exam PWR Reactor

Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
071	1.00	000001G010
016	1.00	000003G010
043	1.00	000003K103
034	1.00	000005A105
081	1.00	000011A213
048	1.00	000011G011
072	1.00	000015A210
017	1.00	000024G010
018	1.00	000026G006
038	1.00	000029K306
073	1.00	000029K312
019	1.00	000040K106
074	1.00	000051A202
035	1.00	000051A202
036	1.00	000055K302
021	1.00	000057A219
037	1.00	000057A219
039	1.00	000059A202
022	1.00	000059A205
076	1.00	000067A213
077	1.00	000068G010
078	1.00	000069A201
079	1.00	000074A207
080	1.00	000076G004

EPE-I Total	24.00	

Group II

QUESTION	VALUE	KA
023	1.00	000007A206
040	1.00	000007G010
042	1.00	000008A108
024	1.00	000008A212
063	1.00	000009G004
082	1.00	000009K323
044	1.00	000025A112
025	1.00	000025K101
083	1.00	000027G004
026	1.00	000032A101
084	1.00	000032G004
046	1.00	000033A207
027	1.00	000037G010
086	1.00	000038K306

S R O E x a m P W R R e a c t o r

O r g a n i z e d b y K A G r o u p

EMERGENCY PLANT EVOLUTIONS

Group II

QUESTION	VALUE	KA
041	1.00	000038K306
047	1.00	000065A207

EPE-II Total	16.00	

Group III

QUESTION	VALUE	KA
085	1.00	000028A201
020	1.00	000056K101
075	1.00	000056K302

EPE-III Total	3.00	

EPE Total	43.00	

Test Total	100.00	

ANSWER: 001 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 1XA203, Obj. 7.5, page 14, para 6.2.2.1.2
2. SONGS Exam Bank, Review Section 2, 2 of 15
3. System Description SD-S01-300 Reactor Coolant Pump System, page 27, step 3.2.2
4. SO1-13-4 "Reactor Plant #1 Annunciator", Windows 32, 52, 72
5. BOTH RO AND SRO (JB)

(3.5/3.9)

003000A201 ..(KA's)

ANSWER: 002 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1XI204, OBJ 3.3
2. SD-SO1-400, Page 18, Step 2.2.3.2
3. BOTH RO AND SRO (JB)

(3.5/3.8)

001000K401 ..(KA's)

ANSWER: 003 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1XI203 OBJ. 3.4
2. SO1-12.3-24 "Monthly Control Rod Exercise", Rev 3, Page 10, Step 2.8
3. SONGS Exam Bank 2191 2 of 35
4. BOTH RO AND SRO (JB)

(3.8/3.8)

001000K402 ..(KA's)

ANSWER: 004 (1.00)

d. [+1.0]

REFERENCE:

1. Lesson Plan 1XI202, Obj 3.2 and 3.3
2. SO1-4-34 "Reactor Plant Instrumentation Operation", Section B, Step 6.11
3. RO AND SRO (JDB)

(3.1/3.1)

011000K604 ..(KA's)

ANSWER: 005 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1XC207, OBJ 3.4
2. Technical Specifications Table 3.5.5-2, Figure 3.5.1.1
3. BOTH RO AND SRO (JB)

(4.2/4.4)

013000K101 ..(KA's)

ANSWER: 006 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1XC207, OBJ 5.0
2. SD-S01-630 Containment and Containment Isolation Systems
3. Dwg 5150874, 5150875
4. BOTH RO AND SRO EXAMS (JB)

(3.9/4.4) (3.9/4.3) (3.8/4.0)

004000K115 013000K111 013000K104 ..(KA's)

ANSWER: 007 (1.00)

d. [+1.0]

REFERENCE:

1. Lesson Plan 1XA208, OBJ 3.3
2. SO1-1.0-23.1 "Background Document for Loss of Recircuation", Step 3
3. BOTH RO AND SRO (JB)

(3.1/3.2)

022000A404 ..(KA's)

ANSWER: 008 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-7-4 "Condensate System", Rev. 4, Eff. 9/8/90, Step 6.0 Caution
2. SO1-3-2 "Plant Startup From Hot Standby To Minimum Load", Rev. 6, Eff. 11/18/91, Step 6.15 AND 6.27
3. BOTH RO AND SRO EXAMS (JDB)

(2.2/2.6)

056010K412 ..(KA's)

ANSWER: 009 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 89RS02, Review Question 3
2. SD-S01-620 Auxillary Feed, Page B-2
3. BOTH RO AND SRO (JB)

(4.5/4.6)

061000K402 ..(KA's)

ANSWER: 010 (1.00)

d. [+1.0]

REFERENCE:

1. Lesson Plan 1XA207, OBJ 1.3.1.6 Page 18
2. Drawing 5102063
3. BOTH RO AND SRO (JB)

(4.4/4.4)

006030A402 ..(KA's)

ANSWER: 011 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 1XI206, Obj. 2.1, Page iii
2. SD-SO1-260 Feedwater Control Systems, Page 18, Step 2.3.2
3. BOTH RO AND SRO (JB)

(4.2/4.5)

035010K101 ..(KA's)

ANSWER: 012 (1.00)

c. [+1.0]

REFERENCE:

1. System Description SD-S01-600, Rev. 2, Page 76
2. S01-10-1 "Diesel Generator Operations", Rev. 3, Eff. 12/31/91, Step 6.0 Caution
3. BOTH RO AND SRO (JB)

(4.0/4.3)

064000A401 ..(KA's)

ANSWER: 013 (1.00)

d. [+1.0]

REFERENCE:

1. System Description SD-S01-600, Rev. 2, Step 3.1, Page 91
2. S01-10-1, "Diesel Generator Operations", Rev. 3, Eff. 12/31/91, Step 6.19.3.3
3. BOTH RO AND SRO (JB)

(3.9/3.9)

064000A406 ..(KA's)

ANSWER: 014 (1.00)

d. [+1.0]

REFERENCE:

1. SONGS TECHNICAL SPECIFICATIONS, 3.1.2.H.1
2. S01-4-9 "Residual Heat Removal System Operation", Rev. 5, Eff. 4/30/91, Step 4.4
3. BOTH RO AND SRO (JB)

(3.2/3.8)

005000G005 ..(KA's)

ANSWER: 015 (1.00)

a. [+1.0]

REFERENCE:

1. System Description SD-S01-340, Rev. 1, Page 4, Step 2.2.1
2. BOTH RO AND SRO (JB)

(2.9/3.2)

076000K402 ..(KA's)

ANSWER: 016 (1.00)

a. [+1.0]

REFERENCE:

1. S01-2.3-1 "Control Rod MALFUNCTION", Rev. 2, Eff. 3/10/91, Step 1
2. BOTH RO AND SRO (JB)

(3.9/4.0)

000003G010 ..(KA's)

ANSWER: 017 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-2.1-12 "Emergency Boration", Rev. 2, Eff. 10/23/91, Pages 1 AND 2
2. BOTH RO AND SRO (JB)

(4.0/4.0)

000024G010 ..(KA's)

ANSWER: 018 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-1.0-60.1 "Loss of All Ac Background Document", Rev. 3, Eff. 8/30/91, Step 24A
2. BOTH RO AND SRO (JB)

(3.4/3.6)

000026G006 ..(KA's)

ANSWER: 019 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.0-30.1 "Background Document For The Loss of Secondary Coolant"
2. Both RO and SRO (JDB)

(3.7/3.8)

000040K106 ..(KA's)

ANSWER: 020 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-1.0-60.1 "Background Document For Loss of All AC", Rev. 3, Eff. 3/6/91, Step 27, Page 29
2. SO1-3-6 "Plant Operation With Natural Circulation", page 3, step 4.2
3. BOTH RO AND SRO (JB)

(3.7/4.2)

000056K101 ..(KA's)

ANSWER: 021 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-2.6-5, "Loss of Regulated Bus", Rev. 1, Eff. 3/6/91, Page 18, Automatic Action
2. BOTH RO AND SRO (JB)

(4.0/4.3)

000057A219 ..(KA's)

ANSWER: 022 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-5-13 "Radiation Monitoring System Operation", Rev. 1, Eff. 8/14/91, Page 21-23, step 6.9.5.2
2. BOTH RO AND SRO (JB)

(3.6/3.9)

000059A205 ..(KA's)

ANSWER: 023 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.0-10 "Reactor Trip Or Safety Injection", Rev. 9, Eff. 3/6/91, Step 1
2. BOTH RO AND SRO (JB)

(4.3/4.5)

000007A206 ..(KA's)

ANSWER: 024 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1TA703, OBJ 3.2
2. BOTH RO AND SRO (JB)

(3.4/3.7)

000008A212 ..(KA's)

ANSWER: 025 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-2.1-9 "Loss of RHR"
2. BOTH RO AND SRO (JB)

(3.9/4.3)

000025K101 ..(KA's)

ANSWER: 026 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-4-34 "Reactor Plant Instrumentation Operation", Rev. 2, Eff. 10/30/91, Step 6.8.1
2. Lesson Plan 1XC205, Obj. 1.2.2, 1.5.2
3. BOTH RO AND SRO (JB)

(3.1/3.4)

000032A101 ..(KA's)

ANSWER: 027 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1AI750, Obj 1.1.2
2. SO1-2.1-1.3 "SG Tube Leakage"
3. BOTH RO AND SRO (JB)

(3.7/3.9)

000037G010 ..(KA's)

ANSWER: 028 (1.00)

a. [+1.0]

REFERENCE:

1. SO123-0-29, "Use of Information Tags" Pg 3 of 11 6.1.1
2. BOTH RO AND SRO (TV)

[3.7/4.1]

194001K102 ..(KA's)

ANSWER: 029 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-13-3, "Annunciator Response", Pg 13
2. L.P. 1XC206, Obj. 2.2, Dwg RPS-1-7
3. BOTH RO AND SRO (TV)

[3.6/3.9]

002020K509 ..(KA's)

ANSWER: 030 (1.00)

d. [+1.0]

REFERENCE:

1. L.P. - 1XR201, Obj. 2.3, Pg 55
2. SO1-2.2-1, "High Activity Operational Monitoring System", Step 2, Pg 7.
3. BOTH RO AND SRO (TV)

[3.6/3.9]

073000K101 ..(KA's)

ANSWER: 031 (1.00)

a. [+1.0]

REFERENCE:

1. L.P. 1XC204, Obj. 1.4.2, Pg 3-8
2. SO1-2.3-1, "Control Rod System Malfunctions", Pg 11
3. RO AND SRO (TV)

[3.3/3.5]

012000K610 ..(KA's)

ANSWER: 032 (1.00)

c. [+1.0]

REFERENCE:

1. L.P.-1CF704, Obj.A.1, Pg 14
2. SO1-10-2, "Diesel Generator Starting Air System", Section A, Precaution 4.1
3. BOTH RO AND SRO (TV)

[3.8/4.1]

064000K401 ..(KA's)

ANSWER: 033 (1.00)

d. [+1.0]

REFERENCE:

1. L.P. 1XB201, Obj. 3.3, Pg 9
2. SO1-4-19, "CCW System Operations", Section 6.2
3. BOTH RO AND SRO (TV)

[3.3/3.1]

008000A401 ..(KA's)

ANSWER: 034 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-2.3-1, RPI failure step 7
2. L.P. 1XI204, Obj. 5.1, Pg 17
3. Technical Specifications 3.5.4, Action C.3
4. BOTH RO AND SRO (TV)

[3.4/3.4]

000005A105 ..(KA's)

ANSWER: 035 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-2.4-3, Loss of Condenser Vacuum ATT 1, (Required handout)
2. L.P.- 1AI727, Obj. 1.1.1, Pg 4
3. BOTH RO AND SRO (TV)

[3.9/4.1]

000051A202 ..(KA's)

ANSWER: 036 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-1.0-60/60.1 Loss of All AC Power/Background Document
2. BOTH RO AND SRO (TV)

[4.3/4.6]

000055K302 ..(KA's)

ANSWER: 037 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-2.6.3, Loss of Vital or Utility Bus, Part H, Note 4.6, Pg 62
2. L.P.- 1AI736, Obj. 1.2, Pg 84
3. BOTH RO AND SRO (TV)

[4.0/4.3]

000057A219 ..(KA's)

ANSWER: 038 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-4-34, Precaution 4.6, Pg 54
2. L.P.- 1XT204, Obj. 5.0, Pg 39
3. BOTH RO AND SRO (TV)

[4.2/4.3]

000029K306 ..(KA's)

ANSWER: 039 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-5-16 "Liquid Radioactive Waste Releases", Precaution 4.8
2. L.P.- 1XR204, Obj. 5.5, Pg 53
3. BOTH RO AND SRO (TV)

[2.9/3.9]

000059A202 ..(KA's)

ANSWER: 040 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.0-10, Reactor Trip or Safety Injection, Step 12 RNO
2. L.P.- 1XA208, Obj. 2.5, Pg 13
3. BOTH RO AND SRO (TV)

[4.2/4.1]

000007G010 ..(KA's)

ANSWER: 041 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-1.40.1, Steam Generator Tube Rupture Background Document, Pg 24
2. L.P. 1EI715, Obj. 1.1.3, Pg 5
3. BOTH RO AND SRO (TV)

[4.2/4.5]

000038K306 ..(KA's)

ANSWER: 042 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-2.1-3, alarms
2. L.P. 1AI704, Obj. 1.1.2, Pg 7
3. BOTH RO AND SRO (TV)

[3.8/3.8]

000008A108 ..(KA's)

ANSWER: 043 (1.00)

d. [+1.0]

REFERENCE:

1. L.P. 1AI720, Obj. 1.1.2, Pg 25
2. SO1-2.3-1 Control Rod System Malfunctions Rev. 2
3. SO1-4-35, Control Rod Drive System, Precaution Pg 12
4. BOTH RO AND SRO (TV)

[3.5/3.8]

000003K103 ..(KA's)

ANSWER: 044 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-2.1-9, Loss of Residual Heat Removal System, Step 1 note
2. L.P. 1AI710, Obj. 1.1.2, Pg 5
3. BOTH RO AND SRO (TV)

[3.6/3.5]

000025A112 ..(KA's)

ANSWER: 045 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-2.2-3, "Toxic gas", Pg 2 Rev date 10/10/91.
2. LP 1AI747, Obj. 1.1.2.2, Pg 2
3. RO AND SRO (TV)

[3.4/3.5]

194001K111 ..(KA's)

ANSWER: 046 (1.00)

d. [+1.0]

REFERENCE:

1. SD-SO1-380 REV 2
2. LP-1XC205 Rev 4 Obj. 1.5.2, Pg 51
3. RO AND SRO (TV)

[3.9/4.2]

000033A207 ..(KA's)

ANSWER: 047 (1.00)

c. [+1.0]

REFERENCE:

1. L.P.- 1XQ206, Obj. 1.3.2, Pg 12
2. SONGS 1 Question Bank
3. RO and SRO (TV)

[3.4/3.6]

000065A207 ..(KA's)

ANSWER: 048 (1.00)

c [+1.0]

REFERENCE:

1. SO1-1.0-25
2. L.P.- 1EI711, Obj. 1.1.2.5, Pg 59
3. BOTH RO AND SRO (TV)

[4.5/4.5]

000011G011 ..(KA's)

ANSWER: 049 (1.00)

d. [+1.0]

REFERENCE:

1. SO123-VII-4 - Personnel Monitoring Program, Rev 7, Pg 6
2. BOTH RO AND SRO (TV)

[2.8/3.4]

194001K103 ..(KA's)

ANSWER: 050 (1.00)

c. [+1.0]

REFERENCE:

1. SD-SO1-400 Rod Control System Pg 6 of 52, Fig 5
2. LP 1XI203 Obj. 1.3.2, Pg 26
3. RO AND SRO (TV)

[3.8/3.8]

001000K402 ..(KA's)

ANSWER: 051 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-5-14, Liquid Radioactive Waste Receiving and Storage Operations ATT 14
2. L.P.- 1XR205, Obj. 2.1, 6.1, Pg 9
3. BOTH RO AND SRO (TV)

[2.7/2.9]

068000K107 ..(KA's)

ANSWER: 052 (1.00)

d. [+1.0]

REFERENCE:

1. Lesson Plan 1XI203, Obj. 2.4
2. Dwg 5146828-34
3. SRO ONLY (JDB)

(3.5/3.6)

001000K201 ..(KA's)

ANSWER: 053 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-3-4 "Plant Shutdown From Full Power To Hot Standby", Eff. 10/17/91, Rev. 4, Step 6.20.7
2. SONGS Exam Bank 1B-258
3. SRO ONLY (JDB)

(2.6/2.7)

015020A301 ..(KA's)

ANSWER: 054 (1.00)

c. [+1.0]

REFERENCE:

1. TECHNICAL SPECIFICATIONS Table 2.1
2. SRO ONLY (JB)

(4.3/4.5)

015000K105 ..(KA's)

ANSWER: 055 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1MD705, OBJ 2.3
2. SO1-2.1-1.1 "Background Document for Inadequate Core Cooling", Page 5
3. SRO ONLY (JDB)

(3.7/4.1)

017020K503 ..(KA's)

ANSWER: 056 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1XA208, OBJ 1.1
2. TECHNICAL SPECIFICATIONS 3.3.4 Basis
3. SRO ONLY (JDB)

(3.1/3.3)

022000G004 ..(KA's)

ANSWER: 057 (1.00)

c. [+1.0]

REFERENCE:

1. Clarification Manual, No. 79 SIS Operability Requirements, 5/3/85
2. SRO ONLY (JDB)

(3.1/3.2)

059000K105 ..(KA's)

ANSWER: 058 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1XA201, Obj. 8.1, Precaution Following 8B Step
2. SO1-3-3 "Plant Operation from Minimum Load to Full Power", Page 7, Step 4.6.3
3. SRO ONLY (JDB)

(2.7/3.2)

061000K505 ..(KA's)

ANSWER: 059 (1.00)

d. [+1.0]

REFERENCE:

1. SYSTEM DESCRIPTION SD-S01 520, Page 6, 2.2.2
2. SRO ONLY (JDB)

(2.5/2.8)

068000G007 ..(KA's)

ANSWER: 060 (1.00)

d. [+1.0]

REFERENCE:

1. SONGS TECHNICAL SPECIFICATIONS, 6.7.1.a, 2.1
2. SRO ONLY (JDB)

(3.6/4.1)

002000G005 ..(KA's)

ANSWER: 061 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1XI207, OBJ 1.1, Page iii
2. SO1-2.3-3 "Abnormal Pressurizer Pressure", Page 3
3. SRO ONLY (JDB)

(4.0/4.1)
010000K302 ..(KA's)

ANSWER: 062 (1.00)

a. [+1.0]

REFERENCE:

1. SONGS TECHNICAL SPECIFICATIONS, Table 2.1
2. SRO ONLY (JDB)

(3.7/4.1)

011000K405 ..(KA's)

ANSWER: 063 (1.00)

a. [+1.0]

REFERENCE:

1. SONGS TECHNICAL SPECIFICATIONS, Page 2.1-2
2. SRO ONLY (JDB)

(2.8/3.9)

000009G004 ..(KA's)

ANSWER: 064 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1XC206, OBJ 2.4
2. Dwg N-1542 sh 102c, RLS 1017/83
3. SRO ONLY (JDB)

(3.1/3.2)

039000K402 ..(KA's)

ANSWER: 065 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1XR201, Obj.1.3
2. SO1-2.2-1 "High Activity Operational Radiation Monitoring System", Rev. 3, Eff. 12/19/91, Page 15, Step 6
3. SO1-7-10 "Condenser Air Removal System", Rev. 2, Eff. 7/17/91, Step 6.8
4. SRO ONLY (JDB)

(2.6/2.6)

055000K106 ..(KA's)

ANSWER: 066 (1.00)

c. [+1.0]

REFERENCE:

1. SONGS Unit 1 TECHNICAL SPECIFICATIONS, 3.7.1
2. SRO ONLY (JDB)

(3.1/3.8)

062000G005 ..(KA's)

ANSWER: 067 (1.00)

a. [+1.0]

REFERENCE:

1. System Description SD-S01-140, 125 VDC System, 2.1.1.1
2. S01-6-3 "Main Turbine Lube Oil Operation", Page 27
3. SRO ONLY (JDB)

(2.9/3.1)

063000K201 ..(KA's)

ANSWER: 068 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1XB201, Obj 9.1, Page v
2. S01-4-19 "CCW System Operation", Page 27
3. SRO ONLY (JDB)

(3.3/3.4)

008000G007 ..(KA's)

ANSWER: 069 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1XA208, OBJ 3.6, Page iv
2. SO1-4-46 "SI System Operations"
3. SRO ONLY (JDB)

(3.4/3.7)

027000K101 ..(KA's)

ANSWER: 070 (1.00)

b. [+1.0]

REFERENCE:

1. SONGS TECHNICAL SPECIFICATION 3.6.1
2. SRO ONLY (JDB)

(3.3/4.1)

103000G005 ..(KA's)

ANSWER: 071 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-2.3-1 "Control Rod Malfunctions", Rev. 2, Eff. 3/10/91, Page 10
2. SRO ONLY (JDB)

(3.9/4.0)

000001G010 ..(KA's)

ANSWER: 072 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-2.1-8 "Reactor Coolant Pump Trouble", Rev. 2, Eff. 11/16/91, Step 13
2. SRO ONLY (JDB)

(3.7/3.7)

000015A210 ..(KA's)

ANSWER: 073 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.1-1.1 "Background Document for Response to Nuclear Power Generation/ATWS"
2. Lesson Plan 1TA707, Obj 1.3, Section III.A.1.a
3. SRO ONLY (JB)

(4.4/4.7)

000029K312 ..(KA's)

ANSWER: 074 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-2.4-3 "Loss of Condenser Vacuum", Rev. 1, Eff. 12 1 89, Step 2 Automatic Actions
2. SRO ONLY (JDB)

(3.9/4.1)

000051A202 ..(KA's)

ANSWER: 075 (1.00)

c. [+1.0]

REFERENCE:

1. SO1-1.0-60.1 "Background Document For Loss of All AC Power", Rev. 3, EFF 3/6/91, Page 7 Note Explanation
2. SO1-1.0-60 "Loss of All AC Power", Conditional Information Page
3. Lesson Plan 1FG703, Obj. 1.1.2.1 (Red Path Entry Conditions)
4. SRO ONLY (JDB)

(4.4/4.7)

000056K302 ..(KA's)

ANSWER: 076 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-2.7-1 "Appendix R Fire Zone Response", Rev. 0, Eff. 2/7/92, Page 3, Step 4.3
2. Lesson Plan 1XB209, Obj. 1.5.1
3. SRO ONLY (JDB)

(3.3/4.4)

000067A213 ..(KA's)

ANSWER: 077 (1.00)

a. [+1.0]

REFERENCE:

1. SO1-2.5-4 "Shutdown From Outside The Control Room", Rev. 2, Eff. 10/19/90, Section 3.0
2. SRO ONLY (JDB)

(4.1/4.2)

000068G010 ..(KA's)

ANSWER: 078 (1.00)

d. [+1.0]

REFERENCE:

1. SONGS UNIT 1 TECHNICAL SPECIFICATIONS, Definitions, Page 1.0-2
2. SRO ONLY (JDB)

(3.7/4.3)

000069A201 ..(KA's)

ANSWER: 079 (1.00)

d. [+1.0]

REFERENCE:

1. SO1-1.2-1 "Response To Inadequate Core Cooling", Rev. 12, EFF 3/6/91, Step 19
2. Lesson Plan 1MD701, Obj. 3.4
3. SRO ONLY (JDB)

(4.1/4.7)

000074A207 ..(KA's)

ANSWER: 080 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-2.1-15 "High Activity In The Reactor Coolant System", Rev. 4, Eff 6/25/91, Step 1
2. Lesson Plan 1TS713, Obj. 1.1.7
3. SONGS TECHNICAL SPECIFICATIONS Basis Page 3.1-2
4. SRO ONLY (JDB)

(2.1/3.7)

000076G004 ..(KA's)

ANSWER: 081 (1.00)

b. [+1.0]

REFERENCE:

1. SO1-1.0-10.1 "Background Document for Reactor Trip or Safety Injection", Rev. 4, Eff. 9/15/91
2. SRO ONLY (JDB)

(3.9/3.9)

000011A213 ..(KA's)

ANSWER: 082 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 1TA703, Obj. 3.3
2. SO1-1.0-20.1 "Background Document for Loss of Reactor Coolant"
3. SRO ONLY (JDB)

(4.2/4.3)

000009K323 ..(KA's)

ANSWER: 083 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 1AI704, Obj. 1.4
2. SONGS Technical Specifications 3.1.5
3. SRO ONLY (JDB)

(3.1/3.6)

000027G004 ..(KA's)

ANSWER: 084 (1.00)

c. [+1.0]

REFERENCE:

1. S01-2.3-2 "Abnormal Nuclear Instrumentation Operation", Rev. 2, Eff. 2/23/91, Page 13
2. SRO ONLY (JDB)

(2.4/3.3)

000032G004 ..(KA's)

ANSWER: 085 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1XI207, Obj. 1.2.F
2. SONGS Exam Bank, 2161 - 57
3. SO1-4-34 "Reactor Plant Instrumentation Operation" Section B, Step 6.9
4. SRO ONLY (JDB)

(3.4/3.6)

000028A201 ..(KA's)

ANSWER: 086 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 1EI715, Obj. 1.1.3
2. SO1-1.0-40 "Steam Generator Tube Rupture", Rev. 8, Eff. 3/6/91, Step 1, and Background Document Step 1
3. SRO ONLY (JDB)

[4.2/4.5]

000038K306 ..(KA's)

ANSWER: 087 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 1AP112, Obj. 1.1.11
2. SO123-0-23 "Control of System Alignments" Rev. 0, Eff. 11/8/91, Step 6.4.7
3. Exam Bank 2316, Page 21 of 53
4. SRO ONLY (JDB)

(3.7/4.1)

194001K102 ..(KA's)

ANSWER: 088 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 1AP112, Obj. 1.1.16
2. SO123-0-23.1 "Valve Operation", Rev. 0, Eff. 10/29/91, Step 6.5.2.1.1
3. SONGS Exam Bank 2316 23 of 53
4. SRO ONLY (JDB)

(3.6/3.7)

194001K101 ..(KA's)

ANSWER: 089 (1.00)

c. [+1.0]

REFERENCE:

1. SO123-0-23 "Control of System Alignments", Rev. 0, Eff. 11/8/91, Step 4.2
2. SRO ONLY (JB)

(3.7/4.1) (3.6/3.7)

194001K101 194001K102 ..(KA's)

ANSWER: 090 (1.00)

d. [+1.0]

REFERENCE:

1. Lesson Plan 1AP112, Obj. 1.1.5
2. SO123-0-21 "Equipment Status Control", Rev. 1, Eff. 1/16/92, Step 6.13.9
3. SONGS Exam Bank 2316, 18 of 53
4. SRO ONLY (JDB)

(3.7/4.1)

194001K102 ..(KA's)

ANSWER: 091 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan ORP030, Obj. 1.2.4 AND Lesson Plan ORP194, Obj. 1.1.1
2. SONGS Exam Bank 2316 41 of 53
3. SRO ONLY (JDB)

(3.1/4.4)

194001A116 ..(KA's)

ANSWER: 092 (1.00)

c. [+1.0]

REFERENCE:

1. Lesson Plan 1AP112, Obj. 1.1.9
2. SO123-0-22 "Temporary Facility Modification Control", Rev. 0, 12/7/90, Step 4.1
3. SONGS Exam Bank 2324 21 of 48
4. SRO ONLY (JDB)

(3.7/4.1)

194001K102 ..(KA's)

ANSWER: 093 (1.00)

d. [+1.0]

REFERENCE:

1. SO123-0-20 "Use of Procedures" Rev. 0, Eff. 11/19/91, Step 6.4.3
2. Lesson Plan 1AP113, Obj. 1.1.3
3. SRO ONLY (JDB)

(4.1/3.9)

194001A102 ..(KA's)

ANSWER: 094 (1.00)

c. [+1.0]

REFERENCE:

1. SO123-VII-4 "Personnel Monitoring Program", Rev. 7, Eff. 7/16/91, Step 6.6.1
2. SRO ONLY (JDB)

(2.8/3.4)

194001K103 ..(KA's)

ANSWER: 095 (1.00)

b. [+1.0]

REFERENCE:

1. SO123-V11-7.4 "Health Physics Procedure", Rev. 16, Eff. 8/20/91, Step 6.1.3
2. SRO ONLY (JDB)

(2.8/3.4)

194001K103 ..(KA's)

ANSWER: 096 (1.00)

b. [+1.0]

REFERENCE:

1. SO123-0-14 "Notification and Reporting of Significant Events", Rev. 1, Eff. 11/27/91, Step 6.2.2
2. SRO ONLY (JDB)

(3.1/4.4)

194001A116 ..(KA's)

ANSWER: 097 (1.00)

a. [+1.0]

REFERENCE:

1. Lesson Plan 0RP030, Obj. 1.3.0
2. SO123-VIII-10 "Emergency Coordinator Duties", Rev. 5, Eff. 9/14/91, Step 4.1
3. SRO ONLY (JDB)

(3.1/4.4)

194001A116 ..(KA's)

ANSWER: 098 (1.00)

b. [+1.0]

REFERENCE:

1. Lesson Plan 1AP113, Obj. 1.1.15
2. SONGS Exam Bank 2317, Rev.IEW, 6 of 15
3 . SRO ONLY (JDB)

(2.5/3.4)

194001A103 ..(KA's)

ANSWER: 099 (1.00)

b. [+1.0]

REFERENCE:

1. SO123-VII-9.9 "Radiation Exposure Permit Program", Rev. 11, Eff. 11/27/91, Attachment 1 Step 1.1
2. SRO ONLY (JDB)

(2.8/3.4)

194001K103 ..(KA's)

ANSWER: 100 (1.00)

b. [+1.0]

REFERENCE:

1. SO123-VIII-1 "Recogniton and Classification of Emergencies" Rev. 9, Page 39, Step 1b
2. SRO ONLY (JDB)

(3.1/4.4)

194001A116 ..(KA's)

(********** END OF EXAMINATION **********)

Resolution of Facility Comments

Reactor Operator Examination

(Note: If a question was common to both RO and SRO examination the comment resolution applies to both unless an exception was made.)

Question 2

The answer to (d) is correct using the graph given to the candidates. Answer (c) is correct per the facility design and per the formula for the rod speed control program.

Resolution: Correct answers are (c) and (d). (SRO Question 50)

Question 3

Answer (a) could be correct if the initial conditions did not change at the moment of instrument failure. Answer (b) could be correct if initial conditions changed as a result of the instrument failure. The stem of the question can be read in two different ways, thus providing two different answers.

Resolution: The correct answers are (a) and (b). (SRO Question 29)

Question 4

The correct answer to the question is a combination of (b) and (c). There is no identifiable correct answer.

Resolution: The question was deleted based on no correct answer.

Question 5

Answer (a) is not correct based on the licensee statement that the maintenance seal (thermal barrier assembly) does not now "prevent leakage of reactor coolant up the RCP shaft." The conditions of "loops filled" in the stem would specifically exclude mid-loop conditions and reasonably specifies an RCS level which would wet the RCP seals. The thermal barrier assembly, according to the licensee, cannot be used in the manner specified in answer (a). Answers (b) and (c) are not compatible with plant conditions that would require a clearance being hung on seal injection and leakoff. Answer (d) cannot be used as a RCP maintenance seal.

Resolution: The question was deleted based on no correct answer.

Question 6

Number 1 seal failure would cause a slight increase in RCS makeup (leakage rate) due to increased flow past Number 2 seal, but it would not cause a 95 gpm increase. Normal RCS makeup is near zero (Technical Specification limit 6 gpm of unidentified leakage). A failed Number 1 seal would cause a small increase (on the order of a few cc/minute), but would still be enough to cause a "#2 SEAL HIGH FLOW" alarm. Simultaneous failure of Number 1 and 2 seals would allow a greatly increased seal flow up to about 100 gpm before the floating ring seals will begin sealing. The resultant 100 gpm will most likely be lost from the RCS (not going to the VCT) and therefore will contribute to a greatly increased RCS makeup. The answer in (a) could be viewed as a subset of (b) because it does not specify that only the Number 1 seal is affected. Therefore, (a) could be construed to be correct.

Resolution: Both (a) and (b) are accepted as an answer. (SRO Question 1)

Question 9

There is no correct answer because the conditions in the stem would not allow a RCP start according to referenced licensee procedures (SO1-4-3 attachments 4 and 5).

Resolution: The question was deleted based on no correct answer.

Question 12

There is no correct answer.

Resolution: The question was deleted based on no correct answer.

Question 13

Answer (a) is correct because it is the minimum pump shutoff head of the band specified in the referenced lesson plan and system description. The other answers are not correct because they are not the MINIMUM pressure within this band. The Emergency Operating Instruction value of 1040 psig was not a choice and included additional conservatism beyond just the pump design.

Resolution: Retain the question with no change to the answer key.

Question 14

Answer (a) is correct because it required only one action to open CV-525. The other three answers are not correct because they require more actions than answer (a).

Resolution: The correct answer is changed to (a).

Question 15

Answer (b) is correct because FCV-1115 D, E, and F are normally closed when they get SIS signals to close. Answers (a) and (c) are not correct because FCV-1112 does not get an open signal. Answer (d) is not correct because all seal injection valves (FCV-1115 A, B, and C) do not get closed signals.

Resolution: Retain the question with no change to the answer key. (SRO Question 6)

Question 18

The Source Range high voltage switch position was given in the stem and its position is pertinent to the question, allowance has to be made for the candidates to assume the NORMAL or OFF positions. Thus, if Source Range high voltage switch is in the OFF position, answer (a) is correct. If the Source Range high voltage switch is in the NORMAL position then answer (d) is correct. Answers (b) and (c) are incorrect because the reactor would trip in both cases.

Resolution: Retain the question with both (a) and (d) as correct answers.

Question 19

The high containment pressure condition occurring later than 21 seconds after a SIS/LOP is more plausible than the licensee postulated scenario which would make (b) correct. However, either case would make answer (b) correct.

Resolution: Retain the question with both (b) and (d) as correct answers.

Question 29

Answer (d) is not correct because Loss of DC power will not cause a SI pump trip according to the licensee comments. The answers require both conditions, independently, to trip the pump in order for the whole answer to be correct. This results from the use of the **OR** logic and the construction of the answers by pairs. Alternatively, if only one true condition of each pair is required to make a true answer, then without knowing any subject matter, a candidate would be able to select the only logically true answer (a), because answer (a) is the only answer with a trip condition that is not otherwise repeated in another answer. Answers (a), (b), and (c) are not correct per the licensee comments.

Resolution: Delete the question based on no correct answer. (SRO Question 10)

Question 30

Answer (a) is correct because the conditions of 60% Pressurizer level and 400 psig RCS pressure requires the Overpressure Mitigation System inservice per Technical Specification 3.20. The other answers do not preclude OMS being service, however the conditions presented in (a) are the only conditions that require OMS to be placed in service.

Resolution: Retain question with answer (a).

Question 32

Answer (c) appears to be the only correct response.

Resolution: Retain question and change key to (c).

Question 33

Answer (b) appears to be the only correct response.

Resolution: Retain question and change key to (b).

Question 35

Both answers b and d appear to be correct responses.

Resolution: Retain question with two correct answers (b) and (d).

Question 36

There are no auctioneered inputs to the shutdown margin monitor. There are no correct answers.

Resolution: Delete question based on no correct answer.

Question 50

More than two correct responses exist for this question.

Resolution: Delete question based on more than two correct responses to the question.

Question 55

Answers (a) and (d) are not "criteria" and are not "immediate" by any possible definition. Answers (b) and (c) are not correct. Answer (d) also contains an inconsistent title from the RCP Abnormal Operating Instruction. The question also requires exceptional knowledge of the AOI to answer the question.

Resolution: Delete the question based on no correct answer.

Question 61

Answers (a) and (b) are correct depending on the interpretation of whether the question is asking for the individual or total AFW flow.

Resolution: Retain question with correct answers (a) and (b).

Question 64

Answer (b) appears to be the only correct answer.

Resolution: Retain question and change key to (b). (SRO Question 37)

Question 65

Answer (c) is not correct because it alone does not "verify" natural circulation as "verification" is used in licensee procedure SO1-3-6 for natural circulation operation. The other answers are not correct for the same reason.

Resolution: Delete the question based on no correct answer. (SRO Question 20)

Question 69

Answers (a) and (c) meet the requirements for MINIMUM required Technical Specification conditions to satisfy containment integrity.

Resolution: Retain question with two correct answers (a) and (c).

Question 82

Answers (b) and (c) are correct.

Resolution: Retain question with two correct answers (b) and (c). (SRO Question 46)

Question 85

Answer (d) is the only correct response. The question however requires a level of knowledge beyond what could be expected under these circumstances for a RO candidate. The question is acceptable for the SRO candidate.

Resolution: RO examination - Delete question because of level of knowledge. SRO examination - Retain question with (d) as the correct answer. (SRO Question 39)

Question 86

The logic of answer (a) can be read two ways. First, the Feedwater break will cause a less severe cooldown and, independently, the energy release to containment is less with a Feedwater break specifically because of the enthalpy difference between Feedwater and Main Steam. However, due to sentence construction, an alternate reading of this answer could be that the Feedwater break will cause a less severe cooldown also due to the enthalpy difference between Feedwater and Main Steam. The first reading would make answer (a) correct, but the second reading makes answer (a) not correct because feedwater enthalpy would have no effect on RCS cooldown if Feedwater flow suddenly stopped or was dramatically reduced. Since answer (a) could reasonably be read as a technically correct response, it must be allowed.

Resolution: Retain question with correct answers (a) and (b). (SRO Question 19)

Question 88

Both answers (b) and (c) appear to be correct responses.

Resolution: Retain question with correct answers (b) and (c). (SRO Question 36)

Question 89

Answer (b) appears to be correct based on the reading of the background document. Resetting Safety Injection is the ultimate goal after blocking Safety Injection. Therefore, answer (b) is correct in that the reason for deenergizing the sequencer is to block and reset Safety Injection.

Resolution: Retain question with correct answers (b) and (d).

<u>Changes to RO examination based on post examination review.</u>

Question 51

During the post examination review this question was determined to have an inappropriate K/A designator (045010A301). The actual K/A designator was determined to be 045000G005 with an importance value of 2.4. Because no justification exists for increasing its importance rating NUREG BR 0122 page 2-21 requires Section Chief approval for use of a K/A importance value less than 2.5. No approval was granted for this question for use on the examination.

Resolution: Delete the question based on an inappropriate K/A importance value.

Senior Reactor Operator Examination

Question 85

Answers (a) and (b) both appear to be correct responses. FCV-1112 will go full open, and if no operator action is taken a pressurizer high level trip will eventually occur.

Resolution: Retain question with correct answers (a) and (b).

COMMENTS ON RO EXAMINATION
(Administered April 27, 1992)

1. **QUESTION** 002 (1.00)

Given the following:

- Turbine Runback from Underfrequency has reduced power to 60% (Overshoot on turbine controls)
- Rod Control System in manual for duration of runback
- 10.5 degree F Tavg-Tref mismatch

WHICH ONE (1) of the following rod speeds would be observed if the Rod Control Selector switch was taken to the AUTO position? (Figure 5 of SD-SO1-400, "Rod Control System" is attached)

a. 8.75 inches/min

b. 9.25 inches/min

c. 11.25 inches/min

d. 13.0 inches/min

ANSWER:

c

NRC REFERENCE: (TV)
1. SD-SO1-400, Rod Control System, Pg 6 of 52, Figure 5
2. L.P. 1XI203, Obj 1.3.2, Pg 26

SCE COMMENTS

FACILITY DISCUSSION: The graph removed from system description SD-SO1-400 which was attached to the test does not agree with the formula in system description SD-SO1-400. Choice "c" is correct using the formula, which is correct. Answer "d" is correct using the graph. The graphic artist when drawing the graph made a mistake and slipped the endpoint of the graph to the left one unit. The slope for Rod Speed rose from 5 in/min to 15 in/min from 8 to 11 degrees vs. the proper slope from 8 to 12 degrees.

REFERENCE MATERIAL: L.P. 1XI203, Rod Control System, Obj 1.3.2 , Pg 26, does not address this question clearly. Page 12, in lesson plan 1XI203 is the proper location to address

this question, it discusses the fact that the rod speed is variable between 5" to 15"/min and provides a handout/TP entitled Figure 1-7. This figure is similar to the graph referenced in the system description but is drawn correctly.

SD-SO1-400, Rod Control System, Pg. 6, properly discussed the correct Tavg program, this is the choice that the NRC chose as correct. The NRC examiner did not check the accuracy of SD-SO1-400, Figure 5, but included it on the examination and this is what led to their being two correct answers.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. L.P. 1XI203, Obj 1.3.2, Pgs 12 and 26
2. L.P. 1XI203, Student Handout Figure 1-7
2. SD-SO1-400, Rod Control System, Pg 6 and Figure 5

SCE REQUESTED RESOLUTION: Accept both answers c and d

2. **QUESTION** 003 (1.00)

Given the following:

- Power level 80%
- Pressurizer level is at program setpoint.
- Tavg recorder pens all indicate approximately 549 degrees F.
- Tavg/Tref are matched on recorder TR-405
- Rods start stepping in at high speed
- No turbine runback is in progress

WHICH ONE (1) of the following would cause these indications?

a. TM-407 - Avg Tave module to Rod Control loss of power.

b. TE-401A, Loop A hot leg temperature failed high.

c. TM-405 - Avg Tave summing computer failed low.

d. PT-415 - 1st Stage pressure failed high.

ANSWER:

a

NRC REFERENCE: (TV)
1. SO1-13-3, "Annunciator Response", Pg 13
2. L.P. 1XC206, Obj 2.2, Dwg RPS-1-7

SCE COMMENTS

FACILITY DISCUSSION: The sequence of the information presented in the initiating conditions resulted in confusion for the licensed operator candidates. The indications provided imply the rods start stepping in after all of the other information is provided. It does not state the rods are stepping in during the initial conditions but that they start stepping in. The setup of the question made it unclear as to whether the failure just occurred, with the conditions provided as the starting point or if the failure referenced was already present with the conditions provided.

Due to this confusion the students requested clarification as to which was the correct condition, were the conditions presented time 0, or after the failure. The response to the question was to consider the conditions time 0. In

addition, the students are not expected to know the failure mode of all components in a given circuit, this caused some to question what the result would be if TM-407 were to loose power.

Answer "a" is correct. If TM-407 loses power, it fails such that Tavg appears high, thus resulting in rods moving in.

Answer "b" is correct. If the conditions are assumed to be at time 0. At time 0, with all systems working properly, the power level provided does not make a difference. Pressurizer level will be on program as stated in the initial conditions, Tavg recorder pens would all indicate approximately 549 degrees F, and Tavg/Tref would be matched on the recorder TR-405. Turbine runback would be irrelevant at this time, and as stated in the conditions, rods would have just started stepping in at high speed after the failure initiated.

The initial condition of pressurizer level at program setpoint further led the candidates to believe that the conditions provided in the question marked time 0 vs. some time after the initiation of the failure. Pressurizer level would not be at program setpoint if the failure had occurred prior to the conditions given. Pressurizer level and pressure are normally seen prior to actual temperature changes on the recorders, if the conditions provided represented a time after time 0, the pressurizer level should have been less then program level, due to RCS temperature decrease, resulting in system shrinkage and an outsurge from the pressurizer.

This question was not reviewed during the examination pre-review and had not been previously validated. The SCE reviewer believes it was added to the examination after the pre-review.

REFERENCE MATERIAL: L.P. 1XC206, Objective 2.2, Dwg. RPS-1-7, has no application to this question. Objective 2.2 states " Identify the physical location of all indicators, recorders, and controls for the Reactor Control System."

It is not an expectation that an operator know the failure mode of each part of a circuit in the plant, only that he/she can respond to the symptoms as they occur. To know that loss of power to Tavg module TM-407 would result in Tavg appearing high is not an expectation of the lesson referenced, and is not covered by any of the objectives in the referenced lesson plan, 1XC206. Drawing RPS-1-7 does

not indicate/label TM-407 Tavg module and does not discuss or indicate how it will fail on a loss of power.

None of the references used by the NRC examiner were appropriate for this question, or could have provided the NRC examiner the information needed to correctly answer this question.

The reference material provided by SCE was adequate to allow addressal of this question and answer.

SCE REFERENCE:

To answer this question takes an integrated knowledge of the entire Reactor Control System and Nuclear Plant Fundamentals. There is no single set of references available that can be used to answer this question

1. 1542 sheet 102C

SCE REQUESTED RESOLUTION: Accept both a and b.

3. **QUESTION** 004 (1.00)

WHICH ONE (1) of the following constitutes an "operable low temperature overpressure protection system", for low temperature RCP starts?

a. One PORV with a lift setting of less than or equal to 360 psig.

b. Two PORV's with a lift setting of less than or equal to 400 psig.

c. RHR relief valve RV-206 aligned to the RCS with a lift setting of less than or equal to 515 psig.

d. A Reactor Coolant System vent of greater than or equal to 1.75 square inches.

ANSWER:

d

NRC REFERENCE: (TV)
1. SO1-4-3, RCP Operation, Pg 11 of 31, Rev date 3/14/91
2. LP-1XA203, Rev 2, Objective 1.3.2, Pg 13

SCE COMMENTS

FACILITY DISCUSSION: The inclusion of the statement "for low temperature RCP starts" makes answer "d" an incorrect choice. Had the question stated "which of the following constitutes an operable overpressure protection system?", and not mentioned a low temperature RCP start, then d would be a correct answer. The requirement for a 1.75 square inch vent for overpressure protection is not related to a RCP start. A 1.75 square inch vent would prohibit a RCP start.

A reference used by the NRC, SO1-4-3, Section B, Pg 11, Precaution 4.3, is a general precaution statement placed in all procedures that relate to low RCS pressure operation. The procedure precaution makes the following statement:

4.3 <u>When</u> the RCS Temperature is ≤ 360 °F, <u>then</u> at least one of the following overpressure protection systems shall be OPERABLE: (Tech. spec. 3.20.A, Reference 2.1.2)

4.3.1 Two Power Operated Relief Valves (PORVS) with a lift setting of ≤ 465 psig <u>and</u> RHR relief

valve RV-206 aligned to the RCS with a lift setting of ≤ 515 psig (Reference 2.4.6); or

4.3.2 A Reactor Coolant System vent(s) of ≥ 1.75 square inches.

The application of 4.3.2, will only be used when the RCS is depressurized. It can not be applied to the condition being addressed by this question, i.e. a low temperature RCP start.

The overpressure protection that would be in place when starting a RCP at low RCS pressure (≤ 400 psig), would be that referenced in SO1-4-3, Step 4.3.1, page 11. Two PORVs with a lift setting of ≤ 465 psig and RHR relief valve RV-206 aligned to the RCS with a lift setting of ≤ 515 psig.

To start a RCP per SO1-4-3, pg. 15, step 6.1.13, the RCS pressure must be 350 - 400 psig. Plant pressure can not be maintained at 350 - 400 psig with a 1.75 in or greater vent.

None of the other choices provided include all of the requirements for OMS and are also not related to the referenced low temperature RCP start.

REFERENCE MATERIAL: The NRC referenced LP-1XA203, Rev 2, Objective 1.3.2, Pg 13 as one of the source documents for their information. None of the answers provided as possible choices are supported by the referenced lesson plan. Objective 1.3.2 states "State the interlocks and administrative requirements to start on (sic) RCP." The referenced page does not have any discussion or implication about anything called a low temperature reactor coolant pump start. It has no discussion regarding any of the criteria provided in the question and its selection of answers. The LP simply states that the following is needed to start a RCP, lift pump on to upper thrust shoes, #1 seal $\Delta p > 275$ psi, #1 seal leakoff > .25 gpm, seal injection $\Delta P \approx 20$" WG, no reverse ΔT of ≥ 5°F only with $T_c > 520$°F, and no overcurrent relay.

The NRC examination did not properly apply the precaution referenced by SO1-4-3, RCP Operation, Pg 11 of 31, Rev date 3/14/91.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-4.3, pages 11, 12, 13, and 15.

SCE REQUESTED RESOLUTION: Delete question; no correct answers provided.

4. **QUESTION** 005 (1.00)

Given the following:

- Plant is in Mode 6
- RCP "B" motor is uncoupled from the pump.
- Loop "B" is full.
- Maintenance is working on the RCP "B" pump.

WHICH ONE (1) of the following prevents leakage of reactor coolant up the RCP shaft?

a. Pump shaft mates with the top of the thermal barrier assembly.

b. Seal Leakoff collects any RCS leakage up the shaft and directs it back to VCT.

c. Seal injection is maintained.

d. Nozzle dam installation.

ANSWER:

a

NRC REFERENCE: (TV)
1. L.P. 1XA203, Objective 1.2.4, Page 17

SCE COMMENTS

FACILITY DISCUSSION: System Description SD-SO1-300, page 21 discusses that the original design of the RCP incorporated a maintenance seal. If the system description is read further it states that work on the pump must be done at mid-loop. This is because the maintenance seal will no longer prevent leakage as stated in the question.

The current plant design and procedures do not allow the plant conditions stated in the question. In addition, the answer provided will not accomplish that which is stated, i.e. prevent leakage.

REFERENCE MATERIAL: The NRC reference to L.P. 1XA203, RCP and Seal Injection System, Objective 1.2.4, Page 17 does not agree with the question and the answer provided. The statement made in the L.P. is:

"Inside diameter of housing contains stellite overlayed seat to mate with similar seat on the outside of the shaft to act as a low pressure valve during seal section maintenance."

The objectives referenced as applicable to this question are 1.1.3 and 1.1.4, not 1.2.4 as referenced by the NRC Answer Key. Objective 1.2.4 states "Describe the functions for all major components of the RCP and Seal Injection system." The feature briefly mentioned in the L.P. is not considered a major component of the RCP and Seal Injection System. Objectives 1.1.3 and 1.1.4 reference discussions on how RCS leakage through the seals is controlled for various conditions. The discussion in the lesson plan does not state that the function of this seal design is to prevent leakage of reactor coolant up the RCP shaft during maintenance, it simply states that it acts as low pressure valve, not a low pressure cutoff valve.

Had the information assumed by the examination writer been cross checked with the system description SD-S01-300, and the plant operating procedures, it would have been known that we can not perform maintenance on the RCP pump or it's seals with the associated loop full, we must be at mid-loop.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SD-S01-300, page 21

SCE REQUESTED RESOLUTION: Delete question; no correct answer provided.

5. **QUESTION** 006 (1.00)

Given the following:

Make-up to the RCS has increased to 95 gpm, and ONLY the following alarms are received:

"RC PUMP A NO. 1 SEAL LOW DELTA P"
"RC PUMP A NO. 1 SEAL LEAKOFF FLOW OFF NORMAL"
"RC PUMP A SEAL WATER HI FLOW"
"RC PUMP A NO. 2 SEAL HI FLOW"

WHICH ONE (1) of the following has occurred to the A RCP?

a. #1 seal has failed.

b. #1 and #2 seals have failed.

c. All the seals have failed.

d. Seal injection has failed.

ANSWER:

a

NRC REFERENCE: (JB)

1. Lesson Plan 1XA203, Obj. 7.5, page 14, para. 6.2.2.1.2
2. SONGS Exam Bank, Review Section 2, 2 of 15
3. System Description SD-SO1-300, Reactor Coolant Pump System, Page 27, step 3.2.2
4. SO1-13-4, "Reactor Plant No. 1 Annunciator", Windows 32, 52, 72

SCE COMMENTS

FACILITY DISCUSSION: The conditions/symptoms provided to determine the problem would be indicative of both answers "a" and "b". The only difference between a number 1 seal failure and number 2 seal failure is the amount of flow past the number 2 seal. A small increase in flow would take place in a number 1 seal failure, or a large increase in flow which would occur for a number 1 and 2 seal failure.

Procedure SO1-2.1-8, Section C, RCP Seal Trouble, does not differentiate between a failure of the number 1 seal and a combined failure of the number 1 and 2 seals. The procedure simply treats the symptoms. Once number 1 seal has failed, it is hard to determine if number 2 seal has failed also.

The only means to identify if the number 2 seal has also failed is to have some idea of the amount of relative leakage past the number 2 seal. For either a number 1 seal failure only or a number 1 and 2 seal, the operator would expect to see the "RC PUMP A NO.2 SEAL HI FLOW" alarm, which is typically used to determine a number 2 seal problem. This alarm can only be accurately used when there is a problem only with the number 2 seal. If there is a problem with the number 1 seal, then the alarm can indicate either a failed number 1 seal or a failure of the number 1 and number 2 seals. Therefore, both answers "a" and "b" should be correct. The question should not have asked for this type of discrimination.

The NRC Referenced SONGS Exam Bank, Review Section 2, 2 of 15. A search of the Examination Bank indicated that this question appears to have come from examination number 2228, review section 2, and was question number 2. The NRC examiner changed the question by adding a 95 GPM makeup and a statement that alarm "RC PUMP A NO. 2 SEAL HI FLOW" was present. The SONGS Examination had only the following alarms: "RC Pump A No. 1 Seal Low ΔP"; "RC Pump A No. 1 Seal Leakoff Flow Off Normal"; and "RC Pump A Seal Water Hi Flow". The NRC's addition of the additional alarm created the situation where their were two correct answers.

REFERENCE MATERIAL: The NRC Examination reference L.P. 1XA203, Obj 7.5, page 14, para. 6.2.2.1.2 does not support the conclusion provided by the examination writer.

The NRC Examination references L.P. 1XA203, Obj. 7.5, page 14, para 6.2.2.1.2. There is no objective 7.5 in this L.P., an assumption is made that the examiner made a typographical error, and meant to reference 1.7.5, which states "State the symptoms of a failure of #1 seal for a single RCP. Describe the immediate operator actions and identify the applicable AOI." The referenced page 14, and paragraph 6.2.2.1.2 are not referenced in the lesson plan as supporting this objective. The LP only discusses that the purpose of the floating ring seal is to limit leakage to 100 GPM @ 100 psid, in the event #1 seal is lost. None of the other conditions described in the plant conditions are described by this part of the lesson plan. The only reference in the question that appears to touch on this topic is that RCS makeup has increased to 95 gpm. This flowrate was not the referenced number in the lesson plan and does not discriminate between a number 1 seal failure and a number 1 and 2 seal failure.

The initial conditions referenced that there is a "RC PUMP A NO.2 SEAL HI FLOW" alarm annunciated. The same L.P. referenced by the NRC examiner L.P. 1XA203, page 23, 6.2.4.1.11.5 states "Abnormal - High level may be indication of failure of #2 seal or both #1 and #2 seals". High level in the Vapor Seal Head Tank referenced above is indicated by a "RC PUMP A NO.2 SEAL HI FLOW" alarm. This section of the L.P. does reference the appropriate objective of 1.7.5. This section supports again that there is not enough information provided to discriminate between a #1 seal failure only or a #1 and #2 seal failure.

The NRC Referenced SONGS Exam Bank, Review Section 2, 2 of 15, was inappropriately modified. See previous discussion.

The reference material provided by SCE was adequate to allow addressal of this question and answer.

SCE REFERENCE:

1. SD-SO1-300, Reactor Coolant Pump System, page 27, step 3.2.2.
2. SO1-2.1-8 page 21, step 7.
3. SO1-2.1-8 background document, section G, page 4 of 6.
4. SONGS Exam Bank, Review Section 2, 2 of 15
5. L. P. 1XA203,, Pg 23.

SCE REQUESTED RESOLUTION: Accept "a" and "b" as correct answers.

6. **QUESTION** 009 (1.00)

Given the following:

- Plant has been shutdown for 6 days.
- RCS temperature is 145 degrees F.
- RCS pressure is 340 psig.
- Pressurizer level is 62%.
- S/G's have been in wet layup for 2 days.
- Tag on Loop "A" RCP control switch has 190 degrees F written on it.
- SRO Ops Supervisor directs the RO to start the Loop "A" RCP.

WHICH ONE (1) of the following allows the start of the Loop "A" RCP.

a. Pressure is adequate for D/P across the #1 Seal.

b. Secondary temperature can be determined from steam pressure.

c. Adequate time for temperature gradients to dissipate has passed.

d. RCP shutoff temperature is within allowable value.

ANSWER:

d

NRC REFERENCE: (TV)
1. SO1-4-3, Reactor Coolant Pump Operation, Attachment 5
2. L.P. 1XA203, Obj. 1.6.6, Pg 28

SCE COMMENTS

FACILITY DISCUSSION: If the procedure referenced by the NRC is used it specifically prohibits starting an RCP with the conditions given in the question. In addition, the initial conditions given are not achievable with San Onofre procedures and requirements. The plant must be solid to achieve a pressure of 340 psig at a temperature of 145 degrees F. The 62% level is misleading.

This is a very complex scenario that requires the use of procedural attachments and curves. An operator should not be required to memorize curves and attachments, only know

how to use them and the information contained in them. This would be a good open reference style question. It requires taking information using the procedures to perform calculations as directed and then taking the calculated information and conditions and applying this to a curve. There is no way that one could expect operators to have the curve memorized or the other details required to properly answer this question had it even provided a correct answer.

SO1-4-3, Page 11, Step 4.4 states: "When RCS Temperature is ≤ 360 °F, then, a RCP shall not be started unless requirements of Attachment 4 are satisfied." Based on Attachment 4, the stated conditions would not allow a RCP start, the conditions place the plant in the part of the attachment that is titled "RCP START FORBIDDEN". Per SO1-4-3, Page 15, a RCP could not be started for the following reasons:

a. Step 6.1.13 requires 350 - 400 psig, initial condition provided in the question was RCS pressure of 340 psig.

b. The Caution prior to Step 6.1.14 discusses the fact that if pressurizer level is less than 80% there are no temperature restrictions, it states:

   "A RCP shall not be started with RCS pressure ≤ 400 psig unless Pressurizer water level is < 80% **OR** the temperature difference between the secondary and primary systems is < 50°F."

   This OR situation is further discussed in SO1-4-3, RCP Operation, Page 12, steps 4.7, 4.7.1 and 4.7.2.

   This caution discriminates against the answer selected (d) as the correct answer on the answer key.

c. Step 6.1.14.2 would not allow a RCP per Attachment 4 discussed above.

REFERENCE MATERIAL: The NRC reference to L.P. 1XA203, Obj 1.6.6, Pg 28 does not support the answers provided by the NRC Examination.

Page 28 of L.P. 1XA203 does reference Obj. 1.6.6. The L.P. tells the instructor to utilize procedure SO1-4-3, Reactor Coolant Pump Operations, to discuss: 1. Precautions and limitations, 2. System Alignment, 3. Monitoring, 4. Seal bypass, 5. Abnormal conditions, and 6. Initial RCP starts. The lesson plan does not discuss any of the criteria addressed by the question.

From the above discussion it can be seen that if the NRC examination writer had properly read and interpreted SO1-4-3, the procedure he referenced, then the problem discussed would not have existed.

The reference material provided by SCE was adequate to allow addressal of this question and answer.

SCE REFERENCE:

1. SO1-4-3, Reactor Coolant Pump Operation, Pages 11, and 12; also pg 15, Steps 6.1.13, 6.1.14.2, Caution prior to step 6.1.14, and attachment 4.
2. Technical Specification 3.1.2.I.

SCE REQUESTED RESOLUTION: Delete question; no correct answer available.

7. **QUESTION** 012 (1.00)

WHICH ONE (1) of the following occurs at "DIESEL GENERATOR BREAKER CLOSED +2 (TWO) SECONDS" after receipt of a SIS/LOP signal?

a. TCV-601 A & B, CCW to RHR Heat exchangers close.

b. Sequencer output to Load Group "B".

c. Feedwater pump recirc to condenser closes.

d. CCW pumps start.

ANSWER:

c

NRC REFERENCE: (TV)
1. L.P. 1XC207, Obj 3.8.3, Pg 20
2. SO1-4-46, Safety Injection System Operation"
3. Drawings 5149178 through 5149182

SCE COMMENTS

FACILITY DISCUSSION: This question is requesting detailed circuitry knowledge that is unexpected of an operator without the use of references. It is true that allowed to use references, the operator would be able to go to the prints and determine which loads come off of which load groups for the sequencer. However, there are far too many loads to expect that the operator would have each of these load groups memorized.

To have memorized the individual actions that occur for each load group and the start times of each load group associated with the sequencer during a SIS, SIS/LOP, or a LOP is not an expected knowledge at San Onofre. To know what is going to occur to the plant and how each component is expected to operate, i.e should a pump start or stop, or should a valve open or close during these sequencer signals is the expected knowledge.

The wording for answer "c" states that "Feedwater pump recirc to condenser closes" at +2 seconds. The sequencer signal to the valve is generated at that time.

The Feedwater pump recirc to condenser valves are also operated based on the position of SI Valves HV853A and HV853B. HV853A and HV853B receive a signal to operate at time zero (0). As these valves move off of their closed seat the feedwater recirc valves (CV36 and 37) are operated. This contact is in parallel with the sequencer contacts, the sequencer contacts are a backup to the limit switches associated with SI Valves HV853A and HV853B.

As can be seen if all is working properly the feedwater pump recirc to condenser valves will start to close slightly after time zero, and not at DG Breaker Close +2 seconds.

REFERENCE MATERIAL: The objective referenced by the NRC examiner 3.8.3 states "Explain how the following components respond to SIS, LOP, or SISLOP: SISLOP 4KV and 480V Lockouts". The referenced objective is inappropriate, there is no expectation that the operator know what occurs at each given load block of the Sequencer. There is no objective that requires the operator to know this. The referenced portion of the lesson plan, does not reference itself to objective 3.8.3. All that is required is that the operator knows what happens to each component, i.e. end result for plant alignment upon the various sequencer initiation signals. L.P. 1XC207, does break each sequencer into the respective load groups by time just to make the flow of the lesson easier for operator understanding. It is easier to talk in blocks of information, building the entire picture and then summarizing at the end.

If the lesson plan is used in it's entirety, and the person using the lesson plan understands the systems at San Onofre, it would have informed the question writer that the wording of the question would not have provided the desired response.

Two pages prior to that referenced by the NRC Examiner, pg 18, the reason for the inaccuracy of the question wording is provided. In the content part of the lesson the response for time 0 seconds states Main Feedwater to S/G's isolates, and Feedwater pump's HVs reposition to SI position. From lessons that the students received earlier in their training program, they understand that when these valves reposition, whether initiated by an SI or manual operation of the valves, the recirc valves to the condenser reposition.

The lesson plan does state on the referenced page, pg 20, in the content section, that at time D.G. breaker closed +2 seconds, that the feedwater pump recirc to condenser closes. The Instructor activity section directly across from this

statement clarifies that this is only an initiation signal with the statement "Sequencer output to load group "C". The students would have already discussed the HV's closure before and be instructed that this is only another signal to close the recirc valves, and that they should already be closed.

Procedure SO1-4-46, is not titled "Safety Injection System Operation." The title of SO1-4-46, is "Sequencer Operation." This procedure does not support the NRC examination question. The procedure simply discusses the following: 1) De-energizing Sequencer Amber Lights, 2) Energizing Sequencer Amber Lights, 3) De-energizing the Sequencer, and 4) Energizing the sequencer.

Reference to Dwgs 5149178 through 5149182 are the elementaries associated with the Sequencer Load Blocks. They only support that a signal is provided to feedwater recirc valves at the time specified.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer. However, it would take knowledge of system interrelations not covered in the specific lesson plan referenced.

SCE REFERENCE:

1. L.P. 1XC207, Obj 3.8.3, Pgs 18 and 20
2. SO1-4-46, Safety Injection System Operation"
3. Drawings 5149178 through 5149182
4. 1543 sheet 1
5. 1542 sheet 33

SCE REQUESTED RESOLUTION: Delete question; no correct answer available.

8. **QUESTION** 013 (1.00)

WHICH ONE (1) of the following represents the MINIMUM shutoff head for the Main Feed Pumps when they are operating in the Safety Injection mode?

a. 1170 psig

b. 1200 psig

c. 1230 psig

d. 1260 psig

ANSWER:

a

NRC REFERENCE: (TV)
1. SD-SO1-580, Safety Injection, Recirculation, and Containment Spray Systems, Rev 2, Pg 13
2. L.P. 1XA207, Obj. 1.6.2, Pg. 48

SCE COMMENTS

FACILITY DISCUSSION: L.P. 1XA207, pg. 49 states that the shutoff head of the feedwater pumps is 1170 to 1200 psig. Both of these answers were available, and should be considered appropriate.

SD-SO1-580, pg 13 states that the SI shutoff head is 1170 to 1250 psig, the question gives three possible choices in this range a, b, and c.

A number more recently used based on our EOI revision process is 1040 psig. All of these EOIs were provided the NRC examination team. Technically, if one was to look at the absolute MINIMUM shutoff head for the Main Feedwater pumps it would be 1040 psig in the Safety Injection mode, as stated in SO1-1.2-1.1, pg 12, Step 9, and in SO1-1.0-10, SO1-1.0-12, and SO1-1.0-30. The value of 1040 psig was not given as a choice.

SO1-1.0-10 (pg 13), SO1-1.0-12 (Pg 1), SO1-1.0-30 (pg 7), and SO1-1.2-1.1 (pg 12) require RCS pressure be less than 1040 psig to ensure S.I. Flow.

The correct answer to this question is 1040 psig and is not available, therefore the question should be deleted.

L.P. 1XA207, indicates that shutoff head is 1170 to 1200 psig. If the question is not deleted based on the above information both choice "a" and "b" should be accepted.

REFERENCE MATERIAL: L.P. 1XA207, pg 49 should have been the actual reference used by the NRC Examiner. The lesson plan does not state a MINIMUM shutoff head for the feedwater pumps. Under the Operating Parameters section it states, shutoff head of the feedpumps is 1170 - 1200 psig. This should have told the examiner that both 1170 and 1200 should not have been choices for possible correct answers.

The NRC Examiner did have all of the EOIs and could have noted that the shutoff head in these procedures was lower than originally discussed in the S.I. Lesson plan.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-1.0-10, Reactor Trip or Safety Injection, pg 13
2. SO1-1.0-12, SI Termination, pg 1
3. SO1-1.0-30, Loss of Secondary Coolant, pg 7
4. SO1-1.2-1.1, Response to Inadequate Core Cooling, pg 12
5. SD-SO1-580, Safety Injection, Recirculation, and Containment Spray Systems, Rev 2, Pg 13
6. L.P. 1XA207, Obj. 1.6.2, Pg. 49

SCE REQUESTED RESOLUTION: Delete question, no correct answer provided. If determined that it should not be deleted then accept both answers "a" and "b".

9. **QUESTION** 014 (1.00)

Given the following:

- Containment pressure 1.6 psig
- SI has been reset

WHICH ONE (1) of the following is the MINIMUM action that will allow CV-525, (Letdown isolation valve) to open.

a. CV-525 (Letdown isolation valve), control switch turned to OPEN.

b. Reset CIS, CV-525 (Letdown isolation valve), control switch turned to OPEN.

c. Depress associated CIS Override and take CV-525 (Letdown isolation valve), control switch to OPEN.

d. Reset CIS, depress associated CIS Override and take CV-525 (Letdown isolation valve), control switch to OPEN.

ANSWER:

d

NRC REFERENCE: (TV)
1. L.P. 1XA200, Obj 3.2, Pg 18
2. SD-SO1-630, Containment and Containment Isolation Systems page 35.

SCE COMMENTS

FACILITY DISCUSSION: For all Containment Isolation Valves the answer provided as the correct choice "d" would be correct. Lesson Plan 1 XA200, and System Description SD-SO1-630, both discuss that certain valves do not operate as Containment Isolation System (CIS) valves, although these valves are physically located on the CIS panel.

CV-525, Letdown Isolation Valve does not receive a CIS signal on SIS, nor does it have an override feature. The valve does not require one since it does not isolate during SI.

REFERENCE MATERIAL: The NRC examination reference to L.P. 1XA200, objective 3.2, pg 18, would have been appropriate had CV-525 been a Containment Isolation Valve. Page 18, only discusses what the operator must do to operate the containment isolation valves.

L.P. 1XA200, Obj. 3.4 states "List which containment penetrations do not isolate upon a Containment Isolation Signal." L.P. 1XA200, page 26, specifically states that CV-525 and CV-526, Letdown Isolation Valves do not receive a Containment Isolation signal.

SD-SO1-630, Containment and Containment Isolation Systems, referenced by the NRC also discusses that CV-525 does not get an CIS. (See pg 21)

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. L.P. 1XA200
2. SD-SO1-630, Containment and Containment Isolation Systems, pg 21
3. 1542 sheet 91

SCE REQUESTED RESOLUTION: Change answer key to select choice "a" as the correct answer.

10. **QUESTION** 015 (1.00)

WHICH ONE (1) of the following describes the response of the CVCS system following SIS actuation?

a. Charging flow control valve FCV1112 opens, Letdown isolates, and Seal Injection valve position does not change.

b. Charging isolates, Letdown isolates, and FCV-1115 D, E, &F (Seal Injection valves) position does not change.

c. Charging flow control valve FCV1112 opens, Letdown valve positions does not change, and Seal Injection valve position does not change.

d. Charging isolates, Letdown isolates, and Seal Injection isolates,

ANSWER:

b

NRC REFERENCE: (JB)
1. L.P. 1XC207, Obj 5.0
2. SD-S01-630, Containment and Containment Isolation Systems
3. Dwg 5150874, 5150875

SCE COMMENTS

FACILITY DISCUSSION: The same question was on both the Reactor Operator and Senior Reactor Operator Examinations with only slightly different wording. The question on the SRO examination is Number 015. The wording on the RO versus SRO choice "b" is different.

Choice "b" on the SRO examination (Question 015) is "Charging isolates, Letdown isolates, and Seal Injection valve position does not change."

Choice "b" on the RO examination is "Charging isolates, Letdown isolates, and FCV-1115 D, E, and F (Seal Injection valves) position does not change."

Note that typically when discussing the Seal Injection Valves, the valves being referenced are FCV's 1115 A, B, and

C not FCV's 1115 D, E, and F. These latter valves are normally referenced as the Cold Leg Injection Valves.

Choice "d" on the RO examination states "Charging isolates, Letdown isolates, and Seal Injection isolates."

Choice "b" is a correct answer as worded on the RO examination because if one assumes the plant is in its normal condition valves FCV 1115 D, E, and F will be closed, and therefore will not change position. The answer key chose "b" as the correct answer based on this expected plant condition and is a correct answer.

Choice "d" is correct based on the statement in choice "b" stating that FCV's 1115 D, E, and F are considered the Seal Injection valves, in parenthesis in choice "b". Valves FCVs 1115 D, E, and F do receive a close signal upon SIS actuation, if they were open for any reason at all, testing etc. they would then close as stated in choice "d".

Had the proper valves been referenced as the Seal Injection Valves, FCVs 1115 A, B, and C been properly referenced or had no valve numbers been given the question would have had only one proper answer, choice "b", but as stated there are two correct choices "b" and "d."

The slight wording change allowed SROs to select the answer marked as the correct choice on the SRO key, i.e. answer "b", while ROs were faced with two possible choices based on their assumptions of what was being really said by the choices.

REFERENCE MATERIAL: The NRC examiner referenced L.P. L.P. 1XC207, objective 5.0, states "Explain how systems interface with the Sequencer Systems, including their purpose and effects upon the system operation." The terminal objective supporting this enabling objective states " Explain how the Containment Isolation Systems interface with the Sequencer Containment Pressure inputs, and the affects of testing Containment Isolation without the Normal-Test Toggle Switch."

The referenced objective is meant to show how other systems affect the sequencer due to system interfaces not how the sequencer affects other systems. This is an inappropriate application of this learning objective. But, the knowledge asked is an expected knowledge of operators at SONGS 1 and is covered in the lesson plan. The proper objective reference would be 3.4, "List and describe the inputs, outputs, associated setpoints and coincidence with regard to

sequencer operation." This is the reference listed as being applicable to this required knowledge in the lesson plan.

The body of the L.P. does support the discussion in the facility discussion section above. Page 17 of the LP discusses: 1) FCV 1112 goes shut, 2) FCV-1115 D, E, and F go closed, 3) FCV-1115 A, B, and C remain available, and Letdown isolates.

SD-SO1-630, Containment and Containment Isolation Systems, referenced by the NRC do not appear to apply to this question.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. L.P. 1XC207, Obj. 3.4, Pgs. 16 and 17
2. 1542 sheet 63C

SCE REQUESTED RESOLUTION: Accept both b and d as correct answers.

11. **QUESTION** 018 (1.00)

Given the following:

- Reactor power is 8%
- Plant startup is in progress.
- N-1203 - Intermediate Range channel 1 has failed.
- I&C is troubleshooting the drawer.
- I&C technician removes the control power fuses from the N-1203 instrument drawer.

WHICH ONE (1) of the following occurs when the fuses are pulled?

a. N-1203 bistables de-energize, reactor trips.

b. N-1203 indication lost at J console and remote shutdown panel, no trip occurs.

c. Instrument power supplies bistables, no trip occurs.

d. Source range re-energizes, reactor trips.

ANSWER:

a

NRC REFERENCE: (TV)

1. SO1-4-34, Reactor Plant Instrumentation Operation, Precaution 4.9, Pg 30 of 97
2. L.P. 1XC205, Obj. 1.5.2, Pg 51

SCE COMMENTS

FACILITY DISCUSSION: The initial conditions provided do not state that the Source Range High Voltage has been turned to OFF. There is no information provided that illustrates that this condition is present. The Source Range High Voltage for the failed channel must be turned to OFF. If this does not occur, as the control power fuses on the intermediate range channel are pulled the bistables will de-energize and the Source Range Channel associated with the NIS channel having its control power fuse pulled will energize. The reactor will trip.

SO1-4-34, Precaution 4.9, Pg. 30 referenced by the NRC Examiner does not discriminate against answer "d". It does

not elaborate on the operation of the system, only that de-energizing the TRIP Bistables, will cause them to TRIP, and that they can not be blocked below 10% power.

SO1-4-34, Section 6.7, Page 35 of 97, specifically discusses what must be done when an IR instrument is going to have maintenance or testing done on it with power ABOVE 3×10^{-4}% (nominal HV cutoff setpoint for the Source Range Instruments). Step 6.2.1.1 directs placing the HV MANUAL ON/OFF Switch on the appropriate Source Range Channel to the "OFF" position. The reason for doing this is that the Source Channel will re-energize if power is removed from the IR channel.

REFERENCE MATERIAL: L.P. 1XC205, Obj. 1.5.2, Pg 51 does not discuss what occurs if the Intermediate Range (IR) fails and the control power fuses are removed. It simply discusses the symptoms, automatic actions, technical specification implications, need for maintenance initiation, need for SS to determine classification and reportability and that depending on which way the IR fails if Source Range High Voltage fails to energize appropriately then it must be manually turned on. There is no discussion about pulling the control power fuses and the effect this has on the plant.

NRC Reference to SO1-4-34, Precaution 49, Pg. 30; see discussion above. If NRC Examiner would have read the section specifically related to maintenance of an IR channel in the same procedure, SO1-4-34, Section 6.7, Page 35 of 97, he would have realized to make choice "d" invalid the SRHV would have had to be manually turned off in the setup conditions.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-4-34 pages 30 & 35

SCE REQUESTED RESOLUTION: Accept both a and d as correct answers.

12. **QUESTION** 019 (1.00)

WHICH ONE (1) of the following will cause immediate actuation of both trains of Containment Spray?

a. Containment pressure 7 psig on 1/3 containment pressure detectors, SI signal present.

b. Containment pressure 7 psig on 2/3 containment pressure detectors, SI and LOP signal are present.

c. CSAS Train A and B pump pushbuttons depressed.

d. CSAS Train A and B pump and valve pushbuttons depressed.

ANSWER:

d

NRC REFERENCE: (TV)
1. 1XA208, Obj. 3.4, Pg 17
2. SD-SO1-580, Pg 76 of 90

SCE COMMENTS

FACILITY DISCUSSION: To utilize the LOP Signal as a discriminator in choice "b" is misleading and incorrect. All of the training material and plant information discuss Low Bus Voltage on busses 1C and 2C. None of the material references used by the NRC examiner reference an LOP Signal. This is because the input to the CSAS is not a LOP signal. The input to CSAS is independent voltage signals from each of the 4KV buses. A LOP Signal present on the sequencer will not always result in a delay of the CSAS initiation of components. There are scenarios in which a LOP signal can be present and CSAS components will initiate with no delay.

As an example:

A LOP occurs 1 minute prior to a LOCA. At time 30 seconds power is restored to either 4KV Bus 1C AND/OR 2C, the LOCA does not occur for 30 more seconds. When the LOCA occurs, there is still a LOP Signal at the sequencer because it is not yet RESET. During a LOCA, Containment pressure will reach 7 psig and each Containment Spray Train that has power available will initiate immediately with no delay. This could only be one train or both trains, it is dependent on which of the 4KV busses have power at the time.

By using the term "LOP signal", it does not mean that low volts on 1C & 2C exist, just that it did at one time.

The above is the purpose for the training and reference material specifically saying Low Voltage on Bus 1C and 2C, and not implying that it is the "LOP Signal."

Answer "d" would be correct as stated in the key.

Answer "b" is also correct based on the discussion above.

REFERENCE MATERIAL: L.P. 1XA208, Obj. 3.4, Pg. 17, does not support the NRC examiner's use of the term "LOP signal". It specifically discusses Low Voltage on Busses 1C and 2C. The examiner would have avoided the confusion with the test question had the proper terminology been utilized per the Lesson Plan. Pg. 18 further clarifies that system components with power will start when voltage is restored to either 4KV Bus 1C or 2C. Again it does not discuss resetting of the "LOP signal" because it is not a part of this circuit.

SD-SO1-580, Pg 76 of 90, referenced by the NRC examiner also supports that the condition needed is not related to a "LOP signal" but to UV condition on 1C and 2C Busses.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. 1XA208, Obj. 3.4, Pg 17 and 18
2. SD-SO1-580, Pg 76 of 90

SCE REQUESTED RESOLUTION: Accept both answers "b" and "d".

13. **QUESTION** 029 (1.00)

WHICH ONE (1) of the following will trip the Safety Injection pumps during an SI?

a. Time delay undervoltage Or Loss of DC power.

b. Time Delay overcurrent OR Loss of DC power.

c. Time Delay overcurrent OR Low-low RWST level.

d. Loss of DC power OR Low-low RWST level.

ANSWER:

d

NRC REFERENCE: (JB)
1. L.P. 1XA207, Obj. 1.3.1.6, Pg 18
2. Drawing 5102063

SCE COMMENTS

FACILITY DISCUSSION: There is no correct answer. Elementary diagrams 1542 Sheets 11 and 11A, SI Pump Elementaries Sheets 1 and 2, show that the only trips to an SI Pump during an SI are Instantaneous Overcurrent and Low-low RWST Level. Choice "c" states "Time Delay overcurrent OR Low-low RWST level." INSTANTANEOUS overcurrent or Low-Low RWST level will trip the safety injection pumps, not Time Delay overcurrent. The prints further show that if there is no DC Power available to the breaker the breaker becomes "solid", it can not be opened or closed electrically.

L.P. 1XA207, page 11 of 53 does discuss that the SI Pumps will auto trip on low-low RWST level OR loss of DC Power to the SI Pump. This statement is not correct. The proper tripping scheme that should have been mentioned was that the SI Pump will trip on either RWST Low-low Level or an Instantaneous Overcurrent.

REFERENCE MATERIAL: L.P. 1XA207, 1.3.1, pg 18, referenced by the NRC does not discuss the Safety Injection Pumps. It discusses the Feedwater Pumps, as they operate on an SI. There is no discussion in this section regarding a time delayed overcurrent trip. It only discusses an

Instantaneous Overcurrent Trip (150 Relay), and Low-low RWST level.

L.P. 1XA207, page 11 would have misled the NRC Examination writer. The examination writers did not have the individual elementaries available while writing the examination.

System Description SD-SO1-580, Safety Injection System, pages 11 and 12 discuss that the pumps will trip on Low-low RWST level, and that the Timed Delay Overcurrent is blocked on a SI, it is not clear from the description that the instantaneous overcurrent is still available.

Drawing 5106063 referenced by the NRC examiner was unable to be located at SONGS. Do not know what this is referencing.

CONCLUSION: The reference material sent to the NRC could have misled the NRC examination writer.

SCE REFERENCE:

1. L.P. 1XA207, 1.3.1, pg 18
2. 1542 Sheet 11, Safety Injection Pumps Elementary Sheet 1
3. 1542 Sheet 11A, Safety Injection Pumps Elementary Sheet 2
4. System Description SD-SO1-580, Pages 11 and 12.

SCE REQUESTED RESOLUTION: Delete question; no correct answer provided.

14. **QUESTION** 030 (1.00)

WHICH ONE (1) of the following meets the MINIMUM criteria for enabling the Overpressure Mitigation System?

	PRESSURIZER LEVEL	RCS PRESSURE	RCS TEMPERATURE
a.	60%	400 psig	365 degrees F
b.	50%	465 psig	360 degrees F
c.	40%	410 psig	350 degrees F
d.	30%	430 psig	320 degrees F

ANSWER:

a

NRC REFERENCE: (TV)

1. 1XI202, Obj 2.4, Pg 56 and 58
2. SO1-3-5, Plant Shutdown from Hot Standby to Cold Shutdown, Step 6.18
3. SO1-1.0-30, Loss of Secondary Coolant, Caution prior to Step 31, Pg 43.

SCE COMMENTS

FACILITY DISCUSSION: To make choice "a" be the correct choice the question would have to have been worded such that it stated "WHICH ONE (1) of the following meets the technical specification criteria for enabling the Overpressure Mitigation System?"

With the words provided in the test question stating "...meets the MINIMUM criteria...", both the procedures and technical specifications must be met.

Technical Specification 3.20, states that OMS is required whenever RCS Pressure is ≤ 400 psig and Pressurizer Level is > 50%. The TS clarifies that the 400 psig criteria assures protection is provided whenever temperature is below 360 °F.

Procedure SO1-3-5, Page 9 step 6.1.5, and Page 13, step 6.7.2 have the operators establish Pressurizer Level at ≈45%. The level from the above steps is maintained until SO1-3-5, Page 19, steps 6.20.3 and 6.20.3.1 which are:

6.20.3 When the RCS is < 350 °F, then perform the following:

.1 INCREASE Pressurizer level to < 80%

Procedure SO1-3-5, page 18 step 6.18 states "When RCS pressure is 350 - 400 psig and before RCS temperature is reduced to ≤ 360 °F, place the Overpressure Protection System in operation as follows:" At this time the operator has not reached step 6.20.3.1 which will have the pressurizer level raised above 45%.

As discussed above level is maintained at ≈ 45% until RCS temperature is < 350 °F.

Choice "a" is not correct because Pressurizer level is given at 60%, this is procedurally too high for the RCS Pressure and RCS Temperature conditions presented.

Choices "b", "c", and "d" are not correct because pressure is too high for all of these, i.e. > 400 psig. There are other conditions in each of these also.

To meet the MINIMUM criteria for enabling OMS per the procedure and technical specification, the following conditions would need to exist:

Pressurizer Level at ≈ 45%, RCS Pressure between 350 - 400 psig, and temperature is getting ready to be decreased below 360 °F.

REFERENCE MATERIAL: The NRC reference to 1XI202, Obj. 2.4, Pg 56 and 58 is appropriate and helps to supply the correct information and should have helped the NRC determine that there was no correct answer. Specifically on page 56, in Content Section there is a discussion of the Technical Specification Requirements for placing OMS in service. Directly across from this discussion in the Activities Section, there is the following statement: "Administratively we are more conservative than T.S. 3.20 because we do not invoke the PZR level criteria." This statement supports the discussion about the procedure above and lets the operator know that the technical specification requirements are not the MINIMUM criteria for enabling the OMS.

SO1-3-5, was referenced by the NRC but was not properly used and applied.

The caution in SO1-1.0-30, Loss of Secondary Coolant, on Page 45 (NRC referenced pg 43), was not properly applied to this situation.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-3-5 page 9, 13, 18 and 19.
2. Technical Specification 3.20.
3. 1XI202, Obj 2.4, Pg 56 and 58
4. SO1-1.0-30, Loss of Secondary Coolant, Caution prior to Step 31, Pg 45

SCE REQUESTED RESOLUTION: Delete question; no correct answer available.

15. **QUESTION** 032 (1.00)

WHICH ONE (1) of the following reflects the response to LC-430F, (Pressurizer level controller) as power increases from 15% to 85%?

a. Output decreases as power increases.

b. Output increases until 50% power.

c. Output increases to 85% power.

d. Maintains 35% output continuously.

ANSWER:

b

NRC REFERENCE: (TV)
1. L.P. 1XI202, Obj 3.3, Pg 48
2. SO1-3-3, Plant Operation from Minimum Load to Full Power, Pg 17

SCE COMMENTS

FACILITY DISCUSSION: Programmed Pressurizer Level ramp increases all the way to 100%. Pressurizer programmed level is directly a result of Tavg. The level ramps as Tavg changes. The Tavg program maintains Tavg at 535 °F from 0 to 15% power, Tavg is ramped from 535 °F to 551.5 °F, from 15 to 100%. See the attached Tavg program per SD-SO1-390, pg 43.

Pressurizer level will ramp from 25% to 37.5% as Tavg increases from 535 to 551.5 °F, with power going from 15% to 100%. The program level does not stop increasing at 50% as stated in the answer key's selected choice "b".

The correct answer is choice "c"

REFERENCE MATERIAL: L.P. 1XI202, Obj. 3.3, Pg 48 referenced by the NRC is correct and specifically states in the content and activities section the information discussed above about program level for the pressurizer.

SO1-3-3, Pg. 17 does not make any reference to the programmed operation of pressurizer level other then to tell

the operator in step 6.16.1.2 to ensure that Programmed Pressurizer Level in relation to RCS Tavg is maintained.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. L.P. 1XI202, pg 48
2. SD-SO1-390 pages 31 and 43.

SCE REQUESTED RESOLUTION: Change answer key from choice "b" to select choice "c" as the correct answer.

With LC-430F failing high, FCV-1112 will go full open, this will cause actual pressurizer level to increase, backup heaters to energize, pressure will increase as it always does when the level is increased quickly, pressure increasing will cause spray valves to open, and eventually the plant will trip on pressurizer high level. This meets all of the conditions of choice "b".

Choice "c" is not correct, charging flow will not decrease nor do any of the other conditions of choice "c" occur.

San Onofre Exam Bank, Exam No. 2161, Question 57, supplied to the NRC examined a LC-430F output signal failure high. It clearly indicates that as output signal fails high it appears that pressurizer level is low, which would cause charging flow to increase, not decrease.

All of the licensed operator candidates have operated LC-430F and watched it's output increase as level decreases and vice-versa.

REFERENCE MATERIAL: L.P. 1XI207, Obj 1.2.f, Pg 19 used by the NRC examiner is very specific when it describes LC430F failing high. The lesson content specifically states "If LC430F fails high, meaning level appears high. FC-1112 will ....." There was a clarifier of "MEANING LEVEL APPEARS HIGH" so as not to confuse the actions to be discussed with a failure of LC430F output being high. LC-430F is a reverse logic type controller, i.e., as level decreases, LC-430F's output signal increases. As level increases it's output signal decreases.

The L.P specifically clarifies how it is referencing both LC430F's high and low failures, by including the clarifying statement of "MEANING LEVEL APPEARS HIGH or LOW" as appropriate. IF the output failed in the same direction as the level indication there would be no need for this clarifying statement. Without the clarifying statement one must assume that when the statement LC-430F fails high it is referring to it's output.

SO1-2.3-4, does not address the control failure but is addressing an actual level problem.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. L.P. 1XI207, Obj 1.2.f, Pg 19

2. SO1-2.3-4, Abnormal Pressurizer Level, Pg 8
3. San Onofre Exam Bank, Exam No. 2161, Question 57

SCE REQUESTED RESOLUTION: Change answer key from "c" to "b".

17. **QUESTION** 035 (1.00)

Given the following:

- Reactor Power is 100%
- VLPT setpoint display is inoperable
- VLPT setpoint formula is 26.15(.894 Delta T + Tavg) - 14341

WHICH ONE (1) of the following is the current calculated VLPT trip setpoint?

a. 1156 psi

b. 1180 psi

c. 1203 psi

d. 1872 psi

ANSWER:

b

NRC REFERENCE: (TV)
1. L.P. 1XC204, Obj 1.2.1, Pg 21
2. SO1-13-7, Reactor Plant Partial Trip Matrix, Window 32

SCE COMMENTS

FACILITY DISCUSSION: Based on the candidates assumptions made when answering the questions it is possible to have two correct answers.

If the candidate read the question to ask him to strictly "plug and chug" and calculate the answer per the formula provided, this appeared to be very easy, he got an answer of 1179.5, or ≈ 1180 psi as provided in choice "b"

The VLPT calculator has a floor value for its calculations of 1872 psi. This confusion about what was actually wanted caused several of the students to ask the proctor what was wanted by the statement "calculated value". The proctor replied, "its simple, the lowest value you would see on the indications/recorder for the conditions given." This led some of these people to believe that the question wanted the lowest value calculated by the VLPT calculator in the plant which has a floor of 1872 psi.

If the candidate read the question as asking what would be the calculated value in the plant he would have chosen the value of 1872 psi as provided in choice "d".

REFERENCE MATERIAL: L.P. 1XC204, Obj. 1.2.1, Pg 21 specifically performs the same calculation asked for in the question. It calculates the raw "plug and chug" value given by the formula. The next statement is a reminder to ensure the operators know that the actual setpoint floor is 1840 psig + instrument error which is 1872 psig. It is enforced in several places that a floor limit is set on the VLPT calculator of 1872 psig.

The NRC examiner's reference to SO1-13-7, Window 32 also supports a minimum of 1872 psig as a possible correct answer.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. L.P. 1XC204, Obj 1.2.1, Pg 21
2. SO1-13-7, Reactor Plant Partial Trip Matrix, Window 32

SCE REQUESTED RESOLUTION: Accept both answers b and d.

18. **QUESTION** 036 (1.00)

WHICH ONE (1) of the following provides the power level input to the shutdown margin computer?

a. Autioneered high Power Range NI's

b. Auctioneered high Delta T.

c. Auctioneered high Tavg.

d. Auctioneered high Tref.

ANSWER:

b

NRC REFERENCE: (TV)
1. 1XA202, Obj. 5.2, 6.4, Pg 14-15
2. SD-SO1-390, Primary Process Instrumentation Systems, Pg. 35

SCE COMMENTS

FACILITY DISCUSSION: The only auctioneered high feature at SONGS 1 are the inputs to the subcooling margin monitor system. SONGS 1 does not utilize an Auctioneered ΔT, or any of the other Auctioneered instrumentation mentioned in this question.

None of the listed answers are design features at SONGS 1, therefore none of the answers input to the Shutdown Margin Computer, and there is no technically correct answer.

REFERENCE MATERIAL: 1XA202, Obj 5.2, 6.4 , Pgs 14-15 make no mention of an Auctioneered ΔT or Auctioneered Tavg circuit. Pg 14 specifically states under 6.2.6.2.1.2 that AVG ΔT is the input provided to the shutdown margin (SDM) computer.

SD-SO1-390, Primary Process Instrumentation Systems, Pg. 35, referenced by the NRC examiner, is the Tavg circuit and does not apply to this problem. Pg. 36 of this same document does discuss the ΔT circuit and illustrates that AVG ΔT is the input to the SDM computer.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SD-SO1-390, Primary Process Instrumentation Systems, Figures 1A (pg 31), 3 (pg 35), 4 (pg 36), and 8 (pg 40)

SCE REQUESTED RESOLUTION: Delete question; correct answer not provided.

19. **QUESTION** 050 (1.00)

WHICH ONE (1) of the following is the pressure at which the Turbine Generator Emergency DC bearing oil pump will start?

a. 10 psig

b. 9 psig

c. 8 psig

d. 7 psig

ANSWER:

c

NRC REFERENCE: (TV)
1. L.P. 1XT204, Obj 6.1, Pg 11
2. SO1-6-3, Main Turbine Oil System Operation, Section A, Pg 5

SCE COMMENTS

FACILITY DISCUSSION: The nominal value and that referenced in the lesson plan and system description referenced by the NRC is stated as 8 psig. This value is not what is normally expected to be seen in the plant. Procedure SO1-12.2-8, Pg 5, steps 2.1.5, 2.1.5.7, and 2.1.5.8, which is the part of the procedure discussing checking the auto start feature of the Emergency D.C. Bearing Oil Pump, as actuated by PS-42, states the following:

2.1.5 SLOWLY OPEN Pressure Switch Test Valve <u>and</u> perform the following:

.7 **RECORD** PS-42 close pressure:_____ psig (7-9 psig normal)

.8 Verify locally D.C. Emergency Bearing Oil Pump starts.

SO1-12.2-8, Page 8, step 3.2.3 has the test being accepted as satisfactory provided PS-42 actuates between 7 and 9 psig.

Due to the appearance of more than one correct answer, some of the students asked the proctor about this question. The

proctor replied to those that asked "What is the highest pressure you would expect this pump to start."

REFERENCE MATERIAL: L.P. 1XT204, Obj 6.1, Pg 11 does state the nominal pressure of 8 psig as the starting pressure for the pump.

SO1-6-3, Section A, Pg 5 does not apply to this question. It simply discusses closing the circuit breaker for the pump, it makes no mention of the auto start pressure for the pump.

CONCLUSION: The training reference material sent to the NRC could lead to them using the wrong information for this question.

SCE REFERENCE:

1. SO1-12.2-8 pages 5 & 8.

SCE REQUESTED RESOLUTION: Delete question, more than two correct answers.

20. **QUESTION** 055 (1.00)

WHICH ONE (1) of the following is the IMMEDIATE criteria for initiation of RCP/Reactor trip when responding to a loss of RCP motor cooling?

a. RO determination of severity.

b. RCP Oil bearing temperature 202 degrees F.

c. Containment area temperature 150 degrees F.

d. SRO determination of severity.

ANSWER:

d

NRC REFERENCE: (TV)
1. SO1-2.1-7, Step 7
2. L.P. 1AI709, Obj 1.1.1

SCE COMMENTS

FACILITY DISCUSSION: The question wording of "....IMMEDIATE criteria..." leads the operators to believe that there is suppose to be some type of criteria that requires action in a rapid fashion. There are no immediate operator actions associated with the procedure governing this condition, and no IMMEDIATE operator action required.

Criteria is usually reserved for fact based information or indications such as temperatures, pressures, levels, flows etc., not subjective information such as RO or SRO determination of severity.

If an RO determined that the plant was in a condition such that public health and safety, or important equipment were in danger it would be appropriate for him/her to trip the RCP/Reactor. This would make answer "a" the proper choice, but does not fit the concept of criteria.

If RCP Oil bearing temperature was too high, this would also be a reason to trip the RCP/Reactor. The temperature in choice "b" does not meet a severity that would probably warrant this but does provide criteria information.

If containment area temperature were at a 150 °F this is very hot and may well be considered proper criteria to trip the RCP/Reactor.

Containment temperature is a critical parameter, at Step 5 the operator is told to verify area temperature less than 150 °F. If the temperature is not, he must obtain Station Management approval for continued operation.

The SRO always has the job to determine if plant conditions are severe enough to warrant trip of the reactor or any component.

As can be seen, the question use of .."IMMEDIATE criteria.." makes the question very confusing. Criteria should be solid data by which a decision can be made.

L.P. 1AI709, Objective 1.1.1 referenced by the NRC examiner does not exist. In addition L.P. 1AI709 does not cover RCP Motor Cooling. L.P. 1AI708, does address this, but there is not an objective that requires the operators to memorize steps out of the procedure. Memorization of the procedure is not an expectation.

REFERENCE MATERIAL: L.P. 1AI708, is the proper L.P. for addressing this question, but there is not an objective that requires the operators to memorize steps out of the procedure. L.P. does not support the need for immediate operator actions.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-2.1-7, pgs 1 through 4
2. L.P. 1AI708, Loss of RCP Motor Cooling, page iv.

SCE REQUESTED RESOLUTION: Delete question; terms used prevented success.

21. **QUESTION** 061 (1.00)

Given the following:

- Steam line break inside containment has just occurred.
- SO1-1.0-30, "Loss of Secondary Coolant" is in use.
- AFW Pump G-10W RUNNING and G-10 in STANDBY.
- Core Exit Temperature is 395 degrees F.
- Containment pressure 6 psig.
- S/G "C" is faulted and dry.
- S/G's A/B are intact with NR level at 40%.

WHICH ONE (1) of the following is the APPROXIMATE allowable AFW flow to the non-faulted S/G's under these conditions?

a. 25 gpm

b. 50 gpm

c. 300 gpm

d. 450 gpm

ANSWER:

a

NRC REFERENCE: (TV)
1. SO1-1.0-30, Loss of Secondary Coolant, "Caution."

SCE COMMENTS

FACILITY DISCUSSION: The question asked for AFW flow to the non-faulted S/G's. It did not differentiate between total flow to the non-faulted S/G's or flow to each non-faulted S/G. Depending on how the question was read, it is reasonable to interpret the question requiring either one of these answers.

Choice "a" would be the correct choice if the candidate reading the question interpreted it to mean flow to each S/G. Allowable flow to each S/G is 25 gpm.

Choice "b" would be the correct choice if the candidate interpreted the question to want flow to both S/G's. Allowable flow to both non-faulted S/G's, at 25 gpm each, would be 50 gpm.

REFERENCE MATERIAL: SO1-1.0-30, Loss of Secondary Coolant, referenced by the NRC was the appropriate reference. Question wording did not allow for proper discrimination.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-1.0-30, Loss of Secondary Coolant, pg. 5

SCE REQUESTED RESOLUTION: Accept both answers a and b.

22. **QUESTION** 064 (1.00)

WHICH ONE (1) of the following will AUTOMATICALLY occur on a loss of the Utility Bus?

a. Pressurizer heaters trip OFF.

b. Charging and Letdown ISOLATE.

c. AFW Train "A" ACTUATES.

d. AFW Train "B" ACTUATES.

ANSWER:

a

NRC REFERENCE: (TV)

1. SO1-2.6-3, Loss of Vital or Utility Bus, Part H, Note 4.6, Pg 62
2. L.P. 1AI736, Obj 1.2, Pg 84

SCE COMMENTS

FACILITY DISCUSSION: Per the SO1-2.6-3, Attachment 8, Pgs 123 and 124 of 127, Section 3.0, the only actions that automatically occur are charging and letdown isolates. This is also supported by the NOTE following Part H, Step 2, Pg 72 of 127. This is choice "b".

Choice "a" as selected on the NRC answer key, pressurizer heaters trip Off, does not occur due to a loss of the Utility Bus.

REFERENCE MATERIAL: L.P. 1AI736, Obj 1.2, Pg 84 does not support the NRC's answer. It does support answer "b" Charging and Letdown Isolate.

The NRC's reference to SO1-2.6-3, Loss of Vital or Utility Bus, Part H, Note 4.6, Pg 62, does not make sense. Part H is the Section describing a Loss of the Utility Bus but there are no notes or other discussions supporting the NRC's chosen answer. On page 72, there is a note, but it supports answer "b", not "a".

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-2.6-3, Loss of Vital or Utility Bus, Part H, Note following step 2, Pg 72
2. SO1-2.6-3, Loss of Vital or Utility Bus, Attachment 8 Pages 123 and 124, Section 3.0
3. L.P. 1AI736, Obj 1.2, Pg 84

SCE REQUESTED RESOLUTION: Change correct answer from "a" to "b".

23. **QUESTION** 065 (1.00)

WHICH ONE (1) of the following plant conditions verifies adequate natural circulation cooling?

a. RCS subcooling 34 F.

b. RCS hot leg temperature trending with saturation temperature for Main Steam pressure.

c. Steam generator levels at or approaching 50% narrow range.

d. Core exit thermocouples stable and trending with RCS cold leg temperature.

ANSWER:

c

NRC REFERENCE: (JB)

1. SO1-1.0-60.1, Background Document For Loss of All AC, Rev. 3, Eff. 3/6/91, Step 27, Page 29
2. SO1-3-6, Plant Operation With Natural Circulation, Page 3, step 4.2

SCE COMMENTS

FACILITY DISCUSSION: Answer key's choice "c" by itself does not ensure natural circulation and heat removal. S/G level can be at 50% and natural circulation not exist. Raising S/G Level to 50% is what an operator does to help ensure natural circulation continues. Natural circulation can be present at SG Levels other than 50%. To verify adequate natural circulation cooling all of the substeps associated with step 4.2.4 (steps 4.2.4.1 through 4.2.4.3) need to be present. See the steps discussed below.

SO1-3-6, page 3 states the following:

4.2 During the course of natural circulation operation, adequate core cooling should be verified frequently by ensuring the following:

4.2.1 RCS subcooling ≥ 40 °F maintained;

4.2.2 Core Exit Thermocouples (TCs) are stable and varying with RCS hot leg temperature;

4.2.3 Core Exit Temperature Indicators TI-4185A and TI-4445A are stable and varying with RCS hot leg temperature;

4.2.4 Continued indications of natural circulation and heat removal;

.1 Loop ΔT stable at < 45 °F, immediately following a reactor trip the ΔT should drop to ≈ 15 - 25 °F.

.2 Steam Generator levels at or approaching 50% narrow range.

.3 RCS cold leg temperature trending with saturation temperature for Main Steam pressure.

REFERENCE MATERIAL: SO1-1.0-60.1, Background Document for Loss of All AC Power, referenced by the NRC does address natural circulation but does not discuss the 50% criteria used as the correct answer in the question.

SO1-3-6, pg 3, step 4.2 referenced by the NRC examiner was inappropriately interpreted.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-3-6, page 3, step 4.2.4 and all of its associated substeps.

SCE REQUESTED RESOLUTION: Delete question; no correct answer provided.

24. **QUESTION** 069 (1.00)

Given the following:

- A remote manual containment isolation valves actuator is disconnected and inoperable.

WHICH ONE (1) of the following meets the MINIMUM required conditions to satisfy containment integrity per Tech Spec 3.6.2, Containment Integrity?

a. Upstream isolation valve closed and locked.

b. Upstream and downstream isolation valves closed and locked.

c. Qualified operator at the valve with a radio and appropriate valve operator tool.

d. Qualified mechanic at the valve with a radio and appropriate valve operator tool.

ANSWER:

c

NRC REFERENCE: (TV)
1. SO1-2.1-6, Loss of Containment Integrity
2. L.P. 1XA200, Obj. 5.2, Pg 30

SCE COMMENTS

FACILITY DISCUSSION: Technical specification 3.6.2, "Containment Isolation Valves," does not directly address choice "c" the answer stated as correct. This answer is acceptable per SO1-2.1-6, Loss of Containment Integrity.

Technical Specification 3.6.2 simply requires that an isolation valve be closed, it does not require that the valve be locked. The statement that an isolation valve be shut makes choice "a" also a correct answer. The addition of "and locked" is really extraneous information, the valve being closed would satisfy the minimum requirement.

Using the word MINIMUM requirements does not discriminate against the answers provided as choice "a" and choice "c". The referenced procedure, lesson plan, and technical specification do not make any statement about a method of establishing a minimum requirement. Procedure SO1-2.1-6, pg

SCE REFERENCE:

1. Technical Specification, 3.6.2, Containment Isolation Valves.
2. SO1-2.1-6, Loss of Containment Integrity, pg 2
3. L.P. 1XA200, Obj. 5.2, Pg 30

SCE REQUESTED RESOLUTION: Accept both answers "a" and "c".

25. **QUESTION** 082 (1.00)

Given the following:

- Power is 10 E-5%.
- Permissive annunciator #4, "S/U RATE TRIPS ACTIVE" is energized.
- N-1204, Intermediate Range level and SUR indications peg high.

WHICH ONE (1) of the following describes the response of the Nuclear Instrumentation/Reactor Protection system to this failure?

a. N-1204 High SUR trip bistable trips, no reactor trip occurs.

b. N-1204 High SUR trip bistable trips, reactor trip occurs.

c. N-1202 Source Range High volts de-energize, reactor trip occurs.

d. N-1201 Source Range High volts de-energize, no reactor trip occurs.

ANSWER:

d

NRC REFERENCE: (TV)
1. SD-SO1-380, Rev 2
2. L.P. 1XC205, Rev 4, Obj 1.5.2, Pg 51

SCE COMMENTS

FACILITY DISCUSSION: The answer key chose "d" as the correct answer. The condition stated in "d" will not occur upon the described failure. Source Range N-1201 High Voltage de-energizes based on operation of Intermediate Range N-1203. N-1203 was not affected in this problem and would therefore not have any effect on N-1201 as stated in choice "d".

Answer "b" is correct, as N-1204 signal goes above 10^{-4}% the block of the High SUR trip is automatically defeated, and the reactor will trip on high SUR at 5 DPM. Normally the IR High SUR signal is blocked below 10^{-4} % power, this is

accomplished by the IR signal and by no other external circuitry or switches. A lesson plan used for 1991 Licensed Operator Requalification, 91RQ09, "Detailed NIS Operation: IR", discusses the details of the blocking and unblocking of the High SUR Trip circuit and clarifies this circuitry.

Answer "c" is correct, as N-1204 signal goes above 3 x 10^{-4}%, due to the failure, the high voltage for Source Range N-1202 will de-energize. The reactor will trip as described above on high SUR.

In addition, this question is very close to Question 18, and could be considered a double jeopardy question. The differences between the question are the power level at which occurs, the intermediate range that fails, and the removal of control power fuses. The question is testing the same type of information, how does the intermediate and source range interact with each other. In fact the reference provided for both of these questions are exactly the same 1XC205, Obj. 1.5.2, page 51.

REFERENCE MATERIAL: NRC's reference to LP 1XC205, Obj. 1.5.2, pg 51, provides no information to support the answer provided by the NRC as being the correct choice. The same referenced lesson plan, on page 17, directly conflicts with choice "d" as the selected answer and supports answer "c". It discusses the fact that IR N-1204 cuts off power to N-1202 at ≈ 3 x 10^{-4}% power, and that IR N-1203 cuts off power to N-1201.

L.P. 1XC205, page 15 discusses the fact that the IR High SUR trip is blocked below 10^{-4}% power. The System Description SD-SO1-380, pg 9, also mentions this fact. Neither of these discuss the details of how this is done.

L.P. 91RQ09 was not sent to the NRC because it has not yet been incorporated into the initial training program, it was scheduled to be incorporated into the initial training program prior to the next time the course was taught.

The reference material provided by SCE was adequate to allow addressal of this question and answer.

In attempting to utilize a fine discriminator for choice "b" they did not have enough information to know how the circuit for blocking the High SUR trip actually worked.

SCE utilized the same references as the NRC to determine that answer "d" was not correct. To determine that answer "b" was also correct SCE utilized a more detailed lesson

plan 91RQ09, which was scheduled to become part of the initial operator training program curriculum prior to the next time the system was taught.

SCE REFERENCE:

1. SD-SO1-380, Rev 2, pg 9
2. L.P. 1XC205, Rev 4, Obj 1.5.2, Pages 15, 17, and 51
3. L.P. 91RQ09, Detailed NIS Operation:IR, Pg 18

SCE REQUESTED RESOLUTION: Accept both answers b and c, change key from d.

26. **QUESTION** 085 (1.00)

Given the following:

- Preparations are being made to release 2 liquid radioactive sources concurrently.
- R-1218 is in service.
- Release permits have been prepared for each source indicating that the sources can be released concurrently.

WHICH ONE (1) of the following conditions must be met prior to the start of the current discharge?

a. Plant Manager approval, total number of MPC's of all sources released calculated to be less than ONE (1) prior to dilution.

b. Chemistry Department approval, total number of MPC's of all sources released calculated to be less than ONE (1) prior to dilution.

c. Plant Manager approval, total number of MPC's of all sources released calculated to be less than ONE (1) after dilution.

d. Chemistry Department approval, total number of MPC's of all sources released calculated to be less than ONE (1) after dilution.

ANSWER:

b

NRC REFERENCE: (TV)

1. SO1-5-16, Liquid Radioactive Waste Releases, Precaution 4.8
2. L.P. 1XR204, Obj 5.5, Pg 53

SCE COMMENTS

FACILITY DISCUSSION: The correct answer is d. The MPCs are calculated after dilution per SO1-5-16, Precaution 4.8, not prior to the dilution. SO1-5-16, Page 4, Precaution 4.8, specifically states: "The release of 2 or more sources of radioactive effluent concurrently is permissible with Chemistry Department approval. Release Permits for each source to be released must indicate other sources which may

be released concurrently and will show that the total number of MPCs after dilution is less than 1."

In addition, it does not appear appropriate that this would be an RO Level question, and maybe not even an SRO level question. A procedure and/or the SRO would tell the RO who to notify and what needs to be done. To have all of the details in all of the procedures memorized is not a reasonable expectation. References should have been provided if this level of detail is expected. This question would be an acceptable open reference test item, but does not require application or demonstration of any knowledge other than pure memorization. This question may be acceptable as an SRO question, but this is still questionable.

REFERENCE MATERIAL: LP 1XA204 Pg 53 referenced by the NRC does not specifically cover the information discussed in this question. It basically gives an overview of what is needed and uses the procedure to discuss this issue. The objective referenced does not require memorization of all of the precautions and limits, it only requires that the operator be able to explain the precautions and limits. The procedure does give the correct answer.

Procedure SO1-5-16, Precaution 4.8 is very specific and gives the correct answer as choice "d".

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-5-16, Liquid Radioactive Waste Releases, Pg 8, Precaution 4.8
2. L.P. 1XR204, Obj 5.5, Pg 53

SCE REQUESTED RESOLUTION: Change answer key from b to d as correct choice.

27. **QUESTION** 086 (1.00)

WHICH ONE (1) of the following is the reason for the difference in severity between a feedline break inside containment, upstream of the inside containment check valve, and a steam line break inside containment?

a. The feedline break will cause a less severe cooldown transient, and release less energy to the containment than a steam line break, due to the enthalpy difference between the feedwater and the steam.

b. The feedline break will cause the loss of feed to one steam generator, but no steam generator will blowdown and minimal RCS cooldown will occur.

c. The feedline break will cause one steam generator to blow down out the break, and steam the other generators dry. The energy lost, and cooldown associated with these occurrences is less severe than that caused by a steam line break.

d. The feedline break will result in an immediate loss of steam generator level, resulting in degraded heat transfer in that steam generator and a loss of heat sink.

ANSWER:

b

NRC REFERENCE: (TV)
1. SO1-1.0-30, Loss of Secondary Coolant
2. SO1-14-47, EOI Users guide, Pg 19
3. L.P. 1EI701, Obj 1.7, Pg 9

SCE COMMENTS

FACILITY DISCUSSION: Answer "b" as selected in the answer key is correct per the background document associated with SO1-1.0-30, Loss of Secondary Coolant. The background document is SO1-1.0-30.1.

Although not specifically detailed in the background document answer "a" is also a correct choice for the following reasons:

a. The feed break will cause a less severe cooldown.

b. Less energy will be released to the containment during a feedline break as compared against a steam line break, and

c. One of the reasons for less energy being released to the containment in a feedline break as compared to a steam line break is that the feed water enthalpy is lower.

- Enthalpy of a steam line break at full power (524 psi) is ≈ 1204 BTU/lbm
- Enthalpy of the feedwater at 600 psi and 400 °F, (normal conditions) would be < 471 BTU/lbm

In addition, detailed analysis associated with each of the accidents is more of what should be expected of a SRO candidate.

REFERENCE MATERIAL: L.P.1EI701, Obj 1.7, does not require the detailed background knowledge be known about each EOI. It only requires that if the operator is given a set of operational and/or procedural conditions, to explain how the EOI will be used. This objective is requiring that an operator be able to work through the EOI's for various situations, not explain in detail the background behind each EOI. The referenced Pg 9 of this LP does not discuss any information related to this question.

SO1-14-47, Pg 9, does not relate to this question. It discusses how the procedures are used. It discusses things like A/ER steps, RNO Steps, and Continuous Action Steps. There is no discussion of background information or this specific accident.

SO1-1.0-30.1, Background Document for Loss of Secondary Coolant, referenced by the NRC does not discriminate against either "a" or "b".

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-1.0-30.1, Background Document for Loss of Secondary Coolant, pg 5 and 6.

SCE REQUESTED RESOLUTION: Accept both answers a and b.

28. **QUESTION** 088 (1.0)

WHICH ONE (1) of the following is the preferred order of power sources for recovery from a loss of all AC power if the Main XFMR is not available?

a. Aux XFMR C, Emergency Diesel Generator, DSD

b. Aux XFMR C, Emergency Diesel Generator, SDG&E 12 KV

c. Emergency Diesel Generator, Aux XFMR C, DSD

d. Emergency Diesel Generator, Aux XFMR C, SDG&E 12 KV

ANSWER:

c

NRC REFERENCE: (TV)
1. SO1-1.0-60/60.1, Loss of All AC Power/Background Document

SCE COMMENTS

FACILITY DISCUSSION: The correct answer is dependent on the Mode of operation assumed at the time the LOP occurs. If the LOP occurs while the plant is in Modes 1, 2, or 3, SO1-1.0-60 is entered and "c" is the correct choice. If the LOP occurs while in Modes 4, 5, or 6 the Abnormal Operating Instruction SO1-2.6-14, is applicable and "b" is the correct answer.

REFERENCE MATERIAL: SO1-1.0-60, Loss of All AC Power, referenced by the NRC properly addressed one of the possible scenarios, i.e. Loss of All AC Power, during Modes 1, 2, or 3.

NRC did not reference SO1-2.6-14, Loss of All AC Power While in Modes 4-6, which would address the power return sequence during a Loss of All AC Power Modes 4, 5 and 6.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:
1. SO1-1.0-60/60.1, Loss of All AC Power/Background Document, Pg 62, Executive Summary

2. SO1-2.6-14, Loss of All AC Power While in Modes 4-6, Pages 1-9, Steps 3, 7, and 13

SCE REQUESTED RESOLUTION: Accept both answers b and c.

29. **QUESTION** 089 (1.00)

Given the following:

- Reactor trip and SIS/LOP has occurred.
- SO1-1.0-60 "Loss of All AC Power" is in progress.
- SI block does not function at Step 3.
- RNO directs the operator to de-energize the sequencer.

WHICH ONE (1) of the following is the reason for de-energizing the sequencer?

a. Locks SI signal into sequencer memory and allows SI reset.

b. Removes LOP signal from sequencer memory and allows SI reset.

c. Ensures all SI relays have actuated and allows SI reset.

d. Eliminates initiation signals from being generated and blocks SI.

ANSWER:

d

NRC REFERENCE: (TV)

1. SO1-1.0-60.1, Loss of All Power Background Document Pg 9
2. 1XC207, Obj 4.2, Pg 26

SCE COMMENTS

FACILITY DISCUSSION: By not providing SO1-1.0-60, the operators only have a general idea of what is being asked in this question. The operators are not required to memorize the details of each individual step in the EOI's, only the Immediate Operator Actions. Also it should not be an expectation that they have all of the information memorized as it is exactly stated in the background documents for each individual part of a step. The background document, SO1-1.0-60.1 does support both answers "b" and "d" if read in its full context.

The purpose of Step 2 in SO1-1.0-60 is to get the sequencer reset such that when power is back the plant can be brought back to its desired condition in a controlled manner. NOTE:

This was step 3 in the old EOI, the background document still references step 3, and was referenced by the NRC examination writer.

SO1-1.0-60.1, page 9, paragraphs 2, 3, and 4 explain that there are two parts to getting the sequencer reset, one is the ability to first block SI, without SI being blocked and the other then is to physically reset the sequencer. The ultimate purpose of the step is to get the sequencer reset.

Answer "d" is correct, de-energizing the sequencer will eliminate the initiation signals from being generated and will block SI. This will then allow the sequencer to be reset.

Answer "b" is also correct. De-energizing the sequencer will remove the LOP signal from the sequencer memory, although this part is not essential for the resetting of SI, it does occur. De-energizing the sequencer allows SI to be reset, because it is now blocked as discussed for choice "d" being correct. The ultimate goal of Step 2 in the EOI, as stated previously is to get SI Reset, and this makes answer "b" correct as well as answer "d".

REFERENCE MATERIAL: The NRC reference to L.P. 1XC207, Obj. 4.2, Pg. 26 is not appropriate for the question asked. Pg. 26 only discusses the process for de-energizing the sequencer, it does not discuss the results or the effects of de-energizing the sequencer for the conditions provided in this question. The discussion in the referenced section also does not say it is applicable to the referenced objective 4.2, but is applicable to objective 4.1.

SO1-1.0-60.1 referenced by the NRC when read completely described the overall purpose of the step being discussed by the question and should have let the examination writer know that he had provided two correct answers to the question asked.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-1.0-60.1, B/D for Loss of All Power.

SCE REQUESTED RESOLUTION: Accept both answers b and d.

COMMENTS ON SRO EXAMINATION
(Administered April 27, 1992)

1. **QUESTION:** 001 (1.00)

Given the following conditions:

Make-up to the RCS has increased to 95 gpm, and ONLY the following alarms are received:

"RC PUMP A NO. 1 SEAL LOW DELTA P"
"RC PUMP A NO. 1 SEAL LEAKOFF FLOW OFF NORMAL"
"RC PUMP A SEAL WATER HI FLOW"
"RC PUMP A NO. 2 SEAL HI FLOW"

WHICH ONE (1) of the following has occurred to the A RCP?

a. #1 seal has failed.

b. #1 and #2 seals have failed.

c. All the seals have failed.

d. Seal injection has failed.

ANSWER:

a

NRC REFERENCE: (JB)
1. Lesson Plan 1XA203, Obj. 7.5, page 14, para. 6.2.2.1.2
2. SONGS Exam Bank, Review Section 2, 2 of 15
3. System Description SD-SO1-300, Reactor Coolant Pump System, Page 27, step 3.2.2
4. SO1-13-4, "Reactor Plant No. 1 Annunciator", Windows 32, 52, 72

SCE COMMENTS

FACILITY DISCUSSION: The conditions/symptoms provided to determine the problem would be indicative of both answers "a" and "b". The only difference between a number 1 and number 2 seal failure and a number 2 seal failure only is the amount of flow past the number 2 seal. A small increase in flow would take place in a number 1 seal failure, or a large increase in flow which would occur for a number 1 and 2 seal failure.

Procedure SO1-2.1-8, Section C, RCP Seal Trouble, does not differentiate between a failure of the number 1 seal and a combined failure of the number 1 and 2 seals. The procedure

simply treats the symptoms. Once number 1 seal has failed it is hard to determine if number 2 seal has failed also. The only means to identify if the number 2 seal has also failed is to have some idea of the amount of relative leakage past the number 2 seal. For either a number 1 seal failure only or a number 1 and 2 seal, the operator would expect to see the "RC PUMP A NO.2 SEAL HI FLOW" alarm, which is typically used to determine a number 2 seal problem. This alarm can only be accurately used when there is a problem only with the number 2 seal. If there is a problem with the number 1 seal, then the alarm can indicate either a failed number 1 seal or a failure of the number 1 and number 2 seals. Therefore, both answers "a" and "b" should be correct.

The NRC Referenced SONGS Exam Bank, Review Section 2, 2 of 15. A search of the Examination Bank indicated that this question appears to have come from examination number 2228, review section 2, and was question number 2. The NRC examiner changed the question by adding a 95 GPM makeup and a statement that alarm "RC PUMP A NO. 2 SEAL HI FLOW" was present. The SONGS Examination had only the following alarms: "RC Pump A No. 1 Seal Low ΔP"; "RC Pump A No. 1 Seal Leakoff Flow Off Normal"; and "RC Pump A Seal Water Hi Flow". The NRC examiner's use of the additional alarm created a situation where there were two correct answers.

NRC REFERENCE PROBLEMS: The NRC Examination reference L.P. 1XA203, Obj 7.5, page 14, para. 6.2.2.1.2 does not support the conclusion provided by the examination writer.

The NRC Examination references L.P. 1XA203, Obj. 7.5, page 14, para 6.2.2.1.2. There is no objective 7.5 in this L.P., an assumption is made that the examiner made a typographical error, and meant to reference 1.7.5, which states "State the symptoms of a failure of #1 seal for a single RCP. Describe the immediate operator actions and identify the applicable AOI." The referenced page 14, and paragraph 6.2.2.1.2 are not referenced in the lesson plan as supporting this objective. The LP only discusses that the purpose of the floating ring seal is to limit leakage to 100 GPM @ 100 psid, in the event #1 seal is lost. None of the other conditions described in the plant conditions are described by this part of the lesson plan. The only reference in the question that appears to touch on this topic is that RCS makeup has increased to 95 gpm. This flowrate was not the referenced number in the lesson plan and does not discriminate between a number 1 seal failure and a number 1 and 2 seal failure.

The initial conditions referenced that there is a "RC PUMP A NO.2 SEAL HI FLOW" alarm annunciated. The same L.P. referenced by the NRC examiner L.P. 1XA203, page 23, 6.2.4.1.11.5 states "Abnormal - High level may be indication of failure of #2 seal or both #1 and #2 seals". High level in the Vapor Seal Head Tank referenced above is indicated by a "RC PUMP A NO.2 SEAL HI FLOW" alarm. This section of the L.P. does reference the appropriate objective of 1.7.5. This section supports again that there is not enough information provided to discriminate between a #1 seal failure only or a #1 and #2 seal failure.

The NRC Referenced SONGS Exam Bank, Review Section 2, 2 of 15, was inappropriately modified. See previous discussion.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SD-SO1-300, Reactor Coolant Pump System, page 27, step 3.2.2.
2. SO1-2.1-8 page 21, step 7.
3. SO1-2.1-8 background document, section G, page 4 of 6.
4. SONGS Exam Bank, Review Section 2, 2 of 15

SCE REQUESTED SOLUTION: Accept "a" and "b" as correct answers.

2. **QUESTION:** 010 (1.00)

WHICH ONE (1) of the following will trip the Safety Injection pumps during an SI?

a. Time delay undervoltage OR Loss of DC power.

b. Time Delay overcurrent OR Loss of DC power.

c. Time Delay overcurrent OR Low-low RWST level.

d. Loss of DC power OR Low-low RWST level.

ANSWER:

d

NRC REFERENCE: (JB)
1. L.P. 1XA207, Obj. 1.3.1.6, Pg 18
2. Drawing 5102063

SCE COMMENTS

FACILITY DISCUSSION: There is no correct answer. Elementary diagrams 1542 Sheets 11 and 11A, SI Pump Elementaries Sheets 1 and 2, show that the only trips to an SI Pump during an SI are Instantaneous Overcurrent and Low-low RWST Level. Choice "c" states "Time Delay overcurrent OR Low-low RWST level." INSTANTANEOUS overcurrent or Low-low RWST level will trip the safety injection pumps, not Time Delay overcurrent. The prints further show that if there is no DC Power available to the breaker the breaker becomes "solid", it can not be opened or closed electrically.

L.P. 1XA207, page 11 of 53 does discuss that the SI Pumps will auto trip on low-low RWST level OR loss of DC Power to the SI Pump. This statement is not correct. The proper tripping scheme that should have been mentioned was that the SI Pump will trip on either RWST Low-low Level or an Instantaneous Overcurrent.

NRC REFERENCE PROBLEMS: L.P. 1XA207, 1.3.1, pg 18, referenced by the NRC does not discuss the Safety Injection Pumps. It discusses the Feedwater Pumps, as they operate on an SI. There is no discussion in this section regarding a time delayed overcurrent trip. It only discusses an

Instantaneous Overcurrent Trip (150 Relay), and Low-low RWST level.

L.P. 1XA207, page 11 would have misled the NRC Examination writer. The examination writers did not have the individual elementaries available while writing the examination.

System Description SD-SO1-580, Safety Injection System, pages 11 and 12 discuss that the pumps will trip on Low-low RWST level, and that the Timed Delay Overcurrent is blocked on a SI, it is not clear from the description that the instantaneous overcurrent is still available.

Drawing 5102063 referenced by the NRC examiner was unable to be located at SONGS. Do not know what this is referencing.

CONCLUSION: The reference material sent to the NRC could have misled the NRC examination writer.

SCE REFERENCE:

1. L.P. 1XA207, 1.3.1, pg 18
2. 1542 Sheet 11, Safety Injection Pumps Elementary Sheet 1
3. 1542 Sheet 11A, Safety Injection Pumps Elementary Sheet 2
4. System Description SD-SO1-580, Pages 11 and 12.

SCE REQUESTED SOLUTION: Delete question; no correct answer provided.

3. **QUESTION:** 019 (1.00)

WHICH ONE (1) of the following is the reason for the difference in severity between a feedline break inside containment, upstream of the inside containment check valve, and a steam line break inside containment?

a. The feedwater break will cause a less severe cooldown transient and release less energy to the containment than a steam line break, due to the enthalpy difference between the feedwater and the steam.

b. The feedwater break will cause the loss of feed to one generator, but no steam generator will blowdown and minimal RCS cooldown will occur.

c. The feedwater break will cause one steam generator to blow down out the break, and steam the other generators dry. The energy lost and associated cooldown is less severe than that caused by steam line break.

d. The feedwater break will result in an immediate loss of steam generator level, resulting in degraded heat transfer in that steam generator and a loss of heat sink.

ANSWER:

b

NRC REFERENCE: (JDB)
1. SO1-1.0-30.1, Background Document For the Loss of Secondary Coolant

SCE COMMENTS

FACILITY DISCUSSION: Answer "b" as selected in the answer key is correct per the background document associated with SO1-1.0-30, Loss of Secondary Coolant. The background document is SO1-1.30.1.

Although not specifically detailed in the background document answer "a" is also a correct choice for the following reasons:

a. The feed break will cause a less severe cooldown.

b. Less energy will be released to the containment during a feedline break as compared against a steam line break, and

c. One of the reasons for less energy being released to the containment in a feedline break as compared to a steam line break is that the feed water enthalpy is lower.

- Enthalpy of a steam line break at full power (524 psi) is ≈ 1204 BTU/lbm
- Enthalpy of the feedwater at 600 psi and 400 °F, (normal conditions) would be < 471 BTU/lbm

NRC REFERENCE PROBLEMS: SO1-1.0-30.1, Loss of Secondary Coolant Background Document, referenced by the NRC does not discriminate against either "a" or "b".

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-1.0-30.1, Loss of Secondary Coolant Background Document, pg 5 and 6.

SCE REQUESTED SOLUTION: Accept both answers a and b.

4. **QUESTION:** 020 (1.00)

WHICH ONE (1) of the following plant conditions verify adequate natural circulation cooling?

a. RCS subcooling 34 F.

b. RCS hot leg temperature trending with saturation temperature for Main Steam pressure.

c. Steam generator levels at or approaching 50% narrow range.

d. Core exit thermocouples stable and trending with RCS cold leg temperature.

ANSWER:

c

NRC REFERENCE: (JB)
1. SO1-1.0-60.1, Background Document For Loss of All AC, Rev. 3, Eff. 3/6/91, Step 27, Page 29
2. SO1-3-6, Plant Operation With Natural Circulation, Page 3, step 4.2

SCE COMMENTS

FACILITY DISCUSSION: Answer key's choice "c" by itself does not ensure natural circulation and heat removal. S/G level can be at 50% and natural circulation not exist. Raising S/G Level to 50% is what an operator does to help ensure natural circulation continues. Natural circulation can be present at SG Levels other than 50%. To verify adequate natural circulation cooling all of the substeps associated with step 4.2.4 (steps 4.2.4.1 through 4.2.4.3) need to be present. See the steps discussed below.

SO1-3-6, page 3 states the following:

4.2 During the course of natural circulation operation, adequate core cooling should be verified frequently by ensuring the following:

4.2.1 RCS subcooling ≥ 40 °F maintained;

4.2.2 Core Exit Thermocouples (TCs) are stable and varying with RCS hot leg temperature;

4.2.3 Core Exit Temperature Indicators TI-4185A and TI-4445A are stable and varying with RCS hot leg temperature;

4.2.4 Continued indications of natural circulation and heat removal;

.1 Loop ΔT stable at < 45 °F, immediately following a reactor trip the ΔT should drop to ≈ 15 - 25 °F.

.2 Steam Generator levels at or approaching 50% narrow range.

.3 RCS cold leg temperature trending with saturation temperature for Main Steam pressure.

NRC REFERENCE PROBLEMS: SO1-1.0-60.1, Background Document for Loss of All AC Power, referenced by the NRC does address natural circulation but does not discuss the 50% criteria used as the correct answer in the question.

SO1-3-6, pg 3, step 4.2 referenced by the NRC examiner was inappropriately interpreted.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-3-6, page 3, step 4.2.4 and all of its associated substeps.

SCE REQUESTED SOLUTION: Delete question; no correct answer provided.

5. **QUESTION:** 029 (1.00)

Given the following:

- Power level 80%
- Pressurizer level is at program setpoint.
- Tavg recorder pens all indicate approximately 549 degrees F.
- Tavg/Tref are matched on recorder TR-405
- Rods start stepping in at high speed
- No turbine runback is in progress

WHICH ONE (1) of the following would cause these indications?

a. TM-407 - Avg Tave module to Rod Control loss of power.

b. TE-401A, Loop A hot leg temperature failed high.

c. TM-405 - Avg Tave summing computer failed low.

d. PT-415 - 1st Stage pressure failed high.

ANSWER:

a

NRC REFERENCE: (TV)
1. SO1-13-3, "Annunciator Response", Pg 13
2. L.P. 1XC206, Obj 2.2, Dwg RPS-1-7

SCE COMMENTS

FACILITY DISCUSSION: The sequence of the information presented in the initiating conditions resulted in confusion for the licensed operator candidates. The indications provided imply the rods start stepping in after all of the other information is provided. The setup of the question made it unclear as to whether the failure just occurred, with the conditions provided as the starting point or if the failure referenced was already present with the conditions provided.

Due to this confusion the students requested clarification as to which was the correct condition, were the conditions presented at time 0, or after the failure. The response to the question was to consider the conditions at time 0. In addition, the students are not expected to know the failure

mode of all components in a given circuit, this caused some to question what the result would be if TM-407 were to lose power.

Answer a is correct. If TM-407 loses power, it fails such that Tavg appears high, thus resulting in rods moving in.

Answer b is correct. If the conditions are assumed to be at time 0. At time 0, with all systems working properly, the power level provided does not make a difference. Pressurizer level will be on program as stated in the initial conditions, Tavg recorder pens would all indicate approximately 549 degrees F, and Tavg/Tref would be matched on the recorder TR-405. Turbine runback would be irrelevant at this time, and as stated in the conditions, rods would have just started stepping in at high speed after the failure initiated.

The initial condition of pressurizer level at program setpoint further led the candidates to believe that the conditions provided in the question marked time 0 vs. some time after the initiation of the failure. Pressurizer level would not be at program setpoint if the failure had occurred prior to the conditions given. Pressurizer level and pressure are normally seen prior to actual temperature changes on the recorders, if the conditions provided represented a time after time 0, the pressurizer level should have been less then program level, due to RCS temperature decrease, resulting in system shrinkage and an outsurge from the pressurizer.

This question was not reviewed during the examination pre-review and had not been previously validated. The SCE reviewer believes it was added to the examination after the pre-review.

NRC REFERENCE PROBLEMS: L.P. 1XC206, Objective 2.2, Dwg. RPS-1-7, has no application to this question. Objective 2.2 states " Identify the physical location of all indicators, recorders, and controls for the Reactor Control System."

It is not an expectation that an operator know the failure mode of each part of a circuit in the plant, only that he/she can respond to the symptoms as they occur. To know that loss of power to Tavg module TM-407 would result in Tavg appearing high is not an expectation of the lesson referenced, and is not covered by any of the objectives in the referenced lesson plan, 1XC206. Drawing RPS-1-7 does not indicate/label TM-407 Tavg module and does not discuss or indicate how it will fail on a loss of power.

None of the references used by the NRC examiner were appropriate for this question, or could have provide the NRC examiner the information needed to correctly answer this question.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

To answer this question takes an integrated knowledge of the entire Reactor Control System, and Nuclear Plant Fundamentals, there is no single set of references available that can be used to answer this question

1. 1542 sheet 102C

SCE REQUESTED SOLUTION: Accept both a and b.

6. **QUESTION:** 037 (1.00)

WHICH ONE (1) of the following will AUTOMATICALLY occur on a loss of the Utility Bus?

a. Pressurizer heaters trip OFF.

b. Charging and Letdown ISOLATE.

c. AFW Train "A" ACTUATES.

d. AFW Train "B" ACTUATES.

ANSWER:

a

NRC REFERENCE: (TV)
1. SO1-2.6-3, Loss of Vital or Utility Bus, Part H, Note 4.6, Pg 62
2. L.P. 1AI736, Obj 1.2, Pg 84

SCE COMMENTS

FACILITY DISCUSSION: Per the SO1-2.6-3, Attachment 8, Pgs 123 and 124 of 127, Section 3.0, the only actions that automatically occur are charging and letdown isolates. This is also supported by the NOTE following Part H, Step 2, Pg 72 of 127. This is choice "b".

Choice "a" as selected on the NRC answer key, pressurizer heaters trip Off, does not occur due to a loss of the Utility Bus.

NRC REFERENCE PROBLEMS: L.P. 1AI736, Obj 1.2, Pg 84 does not support the NRC's answer. It does support answer "b" Charging and Letdown Isolate.

The NRC's reference to SO1-2.6-3, Loss of Vital or Utility Bus, Part H, Note 4.6, Pg 62, is not appropriate. Part H is the Section describing a Loss of the Utility Bus but there are no notes or other discussions supporting the NRC's chosen answer. On page 72, there is a note, but it supports answer "b", not "a".

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-2.6-3, Loss of Vital or Utility Bus, Part H, Note following step 2, Pg 72
2. SO1-2.6-3, Loss of Vital or Utility Bus, Attachment Pages 123 and 124, Section 3.0
3. L.P. 1AI736, Obj 1.2, Pg 84

SCE REQUESTED SOLUTION: Change correct answer from a to b.

7. **QUESTION:** 039 (1.00)

Given the following conditions:

- Preparations are being made to release 2 liquid radioactive sources concurrently.
- R-1218 is in service.
- Release permits have been prepared for each source indicating that the sources can be released concurrently.

WHICH ONE (1) of the following conditions must be met prior to the start of the concurrent discharge?

a. Plant Manager approval, total number of MPC's of all sources released calculated to be less than ONE (1) prior to dilution.

b. Chemistry Department approval, total number of MPC's of all sources released calculated to be less than ONE (1) prior to dilution.

c. Plant Manager approval, total number of MPC's of all sources released calculated to be less than ONE (1) after dilution.

d. Chemistry Department approval, total number of MPC's of all sources released calculated to be less than ONE (1) after dilution.

ANSWER:

b

NRC REFERENCE: (TV)
1. SO1-5-16, "Liquid Radioactive Waste Releases", Precaution 4.8
2. L.P. 1XR204, Obj 5.5, Pg 53

SCE COMMENTS

FACILITY DISCUSSION: The correct answer is d. The MPCs are calculated after dilution per SO1-5-16, Precaution 4.8, not prior to the dilution. SO1-5-16, Page 8, Precaution 4.8, specifically states: "The release of 2 or more sources of radioactive effluent concurrently is permissible with Chemistry Department approval. Release Permits for each source to be released must indicate other sources which may be released concurrently and will show that the total number of MPCs after dilution is less than 1."

NRC REFERENCE PROBLEMS: LP 1XR204 Pg 53 referenced by the NRC does not specifically cover the information discussed in this question. It basically gives an overview of what is needed and uses the procedure to discuss this issue. The objective referenced does not require memorization of all of the precautions and limits, it only requires that the operator be able to explain the precautions and limits. The procedure does give the correct answer.

Procedure SO1-5-16, Precaution 4.8 is very specific and gives the correct answer as choice "d".

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. SO1-5-16, Liquid Radioactive Waste Releases, Pg 8, Precaution 4.8
2. L.P. 1XR204, Obj 5.5, Pg 53

SCE REQUESTED SOLUTION: Change answer key from b to d as correct choice.

8. **QUESTION:** 046 (1.00)

Given the following:

- Power is 10 E-5%.
- Permissive annunciator #4, "S/U RATE TRIPS ACTIVE" is energized.
- N-1204, Intermediate Range level and SUR indications peg high.

WHICH ONE (1) of the following describes the response of the Nuclear Instrumentation/Reactor Protection system to this failure?

a. N-1204 High SUR trip bistable trips, no reactor trip occurs.

b. N-1204 High SUR trip bistable trips, reactor trip occurs.

c. N-1202 Source Range High volts de-energize, reactor trip occurs.

d. N-1201 Source Range High voltage de-energize, no reactor trip occurs.

ANSWER:

d

NRC REFERENCE: (TV)
1. SD-SO1-380, Rev 2
2. L.P. 1XC205, Rev 4, Obj 1.5.2, Pg 51

SCE COMMENTS

FACILITY DISCUSSION: The answer key chose "d" as the correct answer. The condition stated in "d" will not occur upon the described failure. Source Range N-1201 High Voltage de-energizes based on operation of Intermediate Range N-1203. N-1203 was not affected in this problem and would therefore not have any effect on N-1201 as stated in choice "d".

Answer "b" is correct, as N-1204 signal goes above 10^{-4}% the block of the High SUR trip is automatically defeated, and the reactor will trip on high SUR at 5 DPM. Normally the IR High SUR signal is blocked below 10^{-4} % power, this is accomplished by the IR signal and by no other external circuitry or switches. A lesson plan used for 1991 Licensed

Operator Requalification, 91RQ09, "Detailed NIS Operation: IR", discusses the details of the blocking and unblocking of the High SUR Trip circuit and clarifies this circuitry.

Answer "c" is correct, as N-1204 signal goes above 3×10^{-4}%, due to the failure, the high voltage for Source Range N-1202 will de-energize. The reactor will trip as described above on high SUR.

NRC REFERENCE PROBLEMS: NRC's reference to LP 1XC205, Obj. 1.5.2, pg 51, provides no information to support the answer provided by the NRC as being the correct choice. The same referenced lesson plan, on page 17, directly conflicts with choice "d" as the selected answer and supports answer "c". It discusses the fact that IR N-1204 cuts off power to N-1202 at $\approx 3 \times 10^{-4}$% power, and that IR N-1203 cuts off power to N-1201.

L.P. 1XC205, page 15 discusses the fact that the IR High SUR trip is blocked below 10^{-4}% power. The System Description SD-S01-380, pg 9, also mentions this fact. Neither of these discuss the details of how this is done.

L.P. 91RQ09 was not sent to the NRC because it has not yet been incorporated into the initial training program, it was scheduled to be incorporated into the initial training program prior to the next time the course was taught.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

However, in attempting to utilize a fine discriminator for choice "b" they did not have enough information to know how the circuit for blocking the High SUR trip actually worked.

SCE utilized the same references as the NRC to determine that answer "d" was not correct. To determine that answer "b" was also correct SCE utilized a more detailed lesson plan 91RQ09, which was scheduled to become part of the initial operator training program curriculum prior to the next time the system was taught.

SCE REFERENCE:

1. SD-S01-380, Rev 2, pg 9
2. L.P. 1XC205, Rev 4, Obj 1.5.2, Pages 15, 17, and 51
3. L.P. 91RQ09, Detailed NIS Operation:IR, Pg 18

SCE REQUESTED SOLUTION: Accept both answers b and c, change key from d.

9. **QUESTION:** 050 (1.00)

Given the following:

- Turbine Runback from Under frequency has reduced power to 60% (Overshoot on turbine controls)
- Rod Control System in manual for duration of runback
- 10.5 degree F Tavg-Tref mismatch

WHICH ONE (1) of the following rod speeds would be observed if the Rod Control Selector switch was taken to the AUTO position? (Figure 5 of SD-SO1-400, "Rod Control System" is attached)

a. 8.75 inches/min

b. 9.25 inches/min

c. 11.25 inches/min

d. 13.0 inches/min

ANSWER:

c

NRC REFERENCE: (TV)
1. SD-SO1-400, Rod Control System, Pg 6 of 52, Figure 5
2. L.P. 1XI203, Obj 1.3.2, Pg 26

SCE COMMENTS

FACILITY DISCUSSION: The graph removed from system description SD-SO1-400 which was attached to the test does not agree with the formula in system description SD-SO1-400. Choice "c" is correct using the formula, which is correct. Answer "d" is correct using the graph. The graphic artist when drawing the graph made a mistake and slipped the endpoint of the graph to the left one unit. The slope for Rod Speed rose from 5 in/min to 15 in/min from 8 to 11 degrees vs. the proper slope from 8 to 12 degrees.

NRC REFERENCE PROBLEMS: L.P. 1XI203, Rod Control System, Obj 1.3.2 , Pg 26, does not address this question clearly. Page 12, in lesson plan 1XI203 is the proper location to address this question, it discusses the fact that the rod speed is variable between 5" to 15"/min and provides a handout/TP entitled Figure 1-7. This figure is similar to

the graph referenced in the system description but is drawn correctly.

SD-SO1-400, Rod Control System, Pg. 6, properly discussed the correct Tavg program, this is the choice that the NRC chose as correct. The NRC examiner did not check the accuracy of SD-SO1-400, Figure 5, but included it on the examination and this is what led to their being two correct answers.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:

1. L.P. 1XI203, Obj 1.3.2, Pgs 12 and 26
2. L.P. 1XI203, Student Handout Figure 1-7
2. SD-SO1-400, Rod Control System, Pg 6 and Figure 5

SCE REQUESTED SOLUTION: Accept both answers c and d

10. **QUESTION:** 085 (1.00)

Given the following information:

- Pressurizer Level Controller LC-430F output has failed high.
- Assume no operator action is performed, and all controls are in auto.

WHICH ONE (1) of the following will be the first auto action to occur?

a. FCV-1112 will go full open.

b. A PZR high level trip will occur.

c. Letdown isolation LCV-1112 closes.

d. Pzr heaters energize at +4% program level.

ANSWER:

c

NRC REFERENCE: (JB)
1. L.P. 1XI207, Obj 1.2.f
2. SONGS Exam Bank 2161 - 57
3. SO1-4-34, Reactor Plant Instrumentation Operation, Section B, Step 6.9

SCE COMMENTS

FACILITY DISCUSSION: This question is very similar to that given on the RO Examination as question number 33. This question is testing the same knowledge. Failure of level controller LC-430F high results in a full open demand signal to Charging Flow Control Valve, FCV-1112.

With LC-430F failing high, the first auto action that will occur will be FCV-1112 going full open.

The answer key's choice "c" is not correct. Charging flow will not decrease, pressurizer level will not go low and letdown will not isolate.

The NRC examination references SONGS Examination Bank, Exam 2161, Question 57. This question is the identical question to that asked on the NRC Examination. The answer chose on

the SONGS Examination was "a". This is the same answer that the answer key is being requested to accept as correct on the answer key.

Answer "a" should be the correct answer.

NRC REFERENCE PROBLEMS: L.P. 1XI207, Obj 1.2.f, Pg 19 used by the NRC examiner is very specific when it describes LC430F Failing High. The lesson content specifically states "If LC430F fails high, meaning level appears high. FC-1112 will ....." There was a clarifier of "MEANING LEVEL APPEARS HIGH" so as not to confuse the actions to be discussed with a failure of LC430F output being high. LC-430F is a strange controller, it is a reverse logic type controller, as level decreases, LC-430F's output signal goes up. As level decreases it's output signal decreases. No increases

The L.P specifically clarifies how it is referencing both LC430F's high and low failures, by including the clarifying statement of "MEANING LEVEL APPEARS HIGH or LOW" as appropriate. IF the output failed in the same direction as the level indication there would be no need for this clarifying statement. Without the clarifying statement one must assume that when the statement LC-430F Fails high it is referring to it's output.

SONGS Examination Bank, Examination 2161, question 57, was very clear about the proper answer.

The reference material provided by SCE was adequate to allow proper addressal of this question and answer.

SCE REFERENCE:
1. L.P. 1XI207, pg 19
2. SONGS Examination Bank No. 2161, Question 57

SCE REQUESTED SOLUTION: Change answer key from "c" to "a".