



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

ENCLOSURE 1

EXAMINATION REPORT - 50-416/OL-89-01

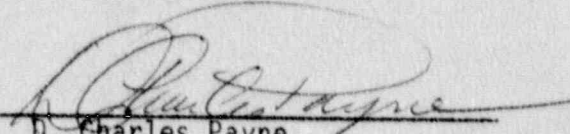
Facility Licensee: System Energy Resources, Inc.
P. O. Box 756
Port Gibson, MS 39150

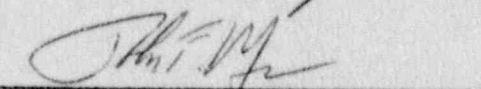
Facility Name: Grand Gulf 1

Facility Docket No.: 50-416

Facility License No.: NPF-29

Requalification written examinations and operating tests were administered at the Grand Gulf Nuclear Station near Port Gibson, Mississippi.

Chief Examiner:  4/10/90
D. Charles Payne Date Signed

Approved By:  4/13/90
John F. Munro, Chief Date Signed
Operator Licensing Section 1

SUMMARY

Examinations were administered during the weeks of November 13 and 27, 1989.

Requalification written examinations and operating tests were administered to 11 Senior Reactor Operators (SROs) and 9 Reactor Operators (ROs). Of the 11 SROs tested, 10 passed the examination. Of the 9 ROs tested, 7 passed the examination. All others failed.

Based upon the above described results, 17 of 20 licensed operators (85 percent) passed the examination.

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REPORT DETAILS

1. Facility Employees Contacted During the Examination

- *C. R. Hutchinson, General Plant Manager
- *J. W. Yelverton, Manager, Plant Operations
- *L. B. Moulder, Operations Superintendent
- *C. O. Hicks, Operations Assistant
- *J. L. Robertson, Training Manager (Acting)
- *W. M. Shelley, Operations Training Superintendent
- K. A. Bowden, Operations Training
- C. R. Roberts, Operations Training

*Attended Exit Meeting

2. Examiners

- *D. C. Payne, RII
- L. L. Lawyer, RII
- M. W. Parrish, EG&G
- C. J. Tyner, EG&G

#J. F. Munro, RII
#L. Vick, HQ

+H. O. Christensen, SRI
+J. Mathis, RI

*Chief Examiner
#Observed examination process
+Attended exit meetings only

3. Examination Changes

Modifications to some of the examination questions, Job Performance Measure (JPM) steps or Individual Simulator Critical Tasks (ISCTs) were proposed by the facility training staff during and after administration of the exam. The utility changes have been consolidated within Enclosure 3. Some of the changes were accepted, based on their particular technical merit; others were not allowed. The NRC evaluation of the facility comments is included in the enclosure.

4. Exit Meeting

At the conclusion of each week of the site visit, the examiners met with representatives of the plant staff to discuss the results of the

examinations. Operators that the facility had failed during the week were discussed as well as actions taken to remove them from shift. During Week 1, the NRC noted a potential operator failure which the facility had not identified. The facility voluntarily removed this individual from licensed duties pending final evaluation of his performance. This operator was subsequently evaluated as being satisfactory and he was allowed to resume licensed duties.

The examiners made the following observations concerning your plant in general:

- a. The team noted the high degree of cleanliness and abundant lighting in most parts of the plant. The licensee is encouraged to continue this effort in other vital plant areas, particularly in the diesel generator buildings.
- b. The facility should review the human factors design of the Diesel Generator auxiliary control panel. It is difficult to associate a set of indications with a given control switch. Additionally, the color scheme of the indicating lights seems to be inconsistent with that used in the rest of the plant. The training department evaluators were aware of this design weakness and indicated that it was a known problem to plant management. During the exit, management committed to investigating the issue and correcting any deficiencies identified.
- c. The color scheme for the labeling of Division 1 of the Shutdown LSS System (white lettering on a bright yellow background) is very difficult to read, especially in poor lighting. Both the operators and the evaluators had trouble reading the nameplates of switches on this cabinet.
- d. SOI 041-01-R21-1, Load Shedding and Sequencing System, System Shutdown, directs the operator to place the associated LSS AUTO TEST handswitch to the BLOCK/RESET position. This is, in fact, a toggle switch vice a handswitch. As a result, several operators became temporarily confused while they searched the cabinet for this switch. Improved nomenclature in the procedure may be appropriate.

The examiners made the following observations concerning your training program:

- a. The use of Motor Operated Valve (MOV) test switches was poorly applied and not well understood by the operators being tested. Only rarely during the course of a JPM were these switches manipulated when expected. Also, they occasionally were used for valves unrelated to the switches. Additional operator training in this area as well as consideration of procedural clarification should be made by the facility.

- b. The use of Alteration Tracking Sheets was poorly applied by the operators being tested. Many operators did not consider completion of this form following JPMS which required it to be used, such as installing jumpers or lifting leads. Additional training on this requirement should be considered.

The cooperation given to the examiners and the effort to ensure an atmosphere in the control room conducive to oral examinations was also noted and appreciated.

The licensee did not identify as proprietary any material provided to or reviewed by the examiners.

ENCLOSURE 3
Exam Report 50-416/OL-89-01

Grand Gulf Requalification Examination
Response to Facility Changes to Answer Key
Week I

Question 282:

Facility deleted "the total of which does not exceed the steady state rating of Div. III diesel generator." and redistributed the points. Justification was "Loads included on list guarantees not exceeding rating of Div II D/G therefore this statement should not be required for full credit."

NRC Response:

The question required the operator to "...determine if all these loads can be started and allowed to run without exceeding the Div. III diesel generator steady state rating?" In order to explain that the loads can be started and allowed to run without exceeding the steady state limit the operator must describe the concept that the total loads on the table do not exceed the steady state limit so any load on the table can be started.

NRC will not accept the facility's change.

Question 317:

Facility changed the answer to part a from 50.0 to 52.9 due to a revision to the procedure.

NRC Response:

NRC will accept the change and grade accordingly.

Question 356:

The facility changed the answer to "also accept 4.1.2.d as note is between d & e (Ref 01-S-02-) does not require note to be placed before the step that is applicable."

NRC Response:

Step 4.1.2.d by itself will not be accepted, this indicates that the operator associated the note with that step. The note is associated with 4.1.2.e. Answers indicating the physical location of the note will be accepted.

Question 397:

Facility deleted part a because "Insufficient information given in part a to classify event as General Emergency. In order to classify as General Emergency core damage must have occurred or is predicted to occur within two hours as reported by reactor engineering (10-S-01-1 Attachment 1 page 17 of 27). This information not given. Change point value to .75."

NRC Response:

Based on the information given the NRC examiner determined the proper classification to be an Alert. The answer key was changed accordingly.

Question 235:

Facility also accepted "PA" in lieu of C. W. Thompson Paging System.

NRC Response:

Agree with comment and will grade accordingly.

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5.2.5 (Cont.)

- g. Have an electrician lift the lead from Terminal No. 3 of Relay 27N1 located on E22-S004, Cubicle 2 (Bus 17AC).
- h. Set the synch-check relay 125 located on E22-S004 to HBDL.
- i. Place the synch switch for Breaker 152-1704 (P601-16C) to ON.
- j. Close Breaker 152-1704.
- k. Close Breaker 152-1903 (P807-1C).
- l. Close Breaker 152-1904.
- m. Shut down LES by placing the local test switch on Panel P331 (P332), to the BLOCK/RESET position; and the ESF 125 Vdc keylock switch to OFF.
- n. Rack down Breaker 152-1702 (HPCS pump motor).
- o. Reset the BUV lockouts for Breakers 152-1511 (152-1611).
- p. Close Breaker 152-1511 (152-1611) to energize Bus 15AA (16AB).

CAUTION

Ensure the SSW pump is started before starting the RHR pump to prevent simultaneous starting of both pumps.

- q. The loads listed in Table 1 can now be started as required.

5.2.6 Return to Section 5.1.

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TABLE 1
ESSENTIAL LOADS

LOAD	LOAD kW
SSW Pump	997
RHR Pump	803
MOVs	185
Control Room Standby Fresh Air	20
ESF Battery Charger	110
Drywell Coolers	120
FPCCU Pump	124
SSW O/A Fan	40
Control Room A/C	120
SSW Cooling Tower Fans	244
ESF Switchgear Room Cooler	12
Instrument Air Compressor	230
Safeguard Switchgear/Battery Fans	122
ECCS Room Cooler	8
HPCS Auxiliaries	154
Total	3291

Title: HPCS Diesel Generator 13 Functional Test	No.: 06-OP-1P81-M-0002	Revision: 31	Page: 3
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- 2.1.9 After changing oil filter elements or draining the accessory oil system of the diesel generator, for any reason, and upon refilling of the system, make sure that the soakback pumps have been in operation for at least 30 minutes and that the strainer boxes are full before starting the engine. The engine should then be brought to an idling condition to ensure complete filling of accessories before any subsequent fast start is made.
- 2.1.10 Any small leak at the top of the oil cooler should be corrected.
- 2.1.11 Any time the diesel generator is paralleled with an offsite power source, attempting to adjust generator voltage will change the reactive loading (VARS) on the generator. When the diesel generator is the only supply to the bus, the reactive load (VARS) is determined by the equipment supplied by the bus.
- 2.1.12 When diesel engine is running, oil addition is through the strainer box. In addition, oil can be added at the plug in the soakback pump suction.
- 2.1.13 Steps marked with a pound sign (#) within the body of the procedure require information to be recorded on data sheets.
- 2.1.14 Steps denoted by a dollar sign (\$) are those items which are required to be completed for Technical Specifications Acceptance Criteria.
- 2.1.15 Steps denoted by an I are those items which are required to be completed for ISI Acceptance Criteria and are to be recorded on either Data Sheet IV or V.
- 2.1.16 Equipment numbers appearing within this procedure have an MPL prefix of 1P81, unless otherwise noted.
- 2.1.17 The HPCS Diesel Generator has a continuous service rating of 3300 kW with 110% (3630 kW) overload permissible for 2 hours out of every 24 hours.
- 2.1.18 Avoid operation of the diesel unloaded for greater than 5 minutes.
- 2.1.19 For the ISI program, if no local position indicator exists, the local valve position indication is observation of indirect evidence such as changes in system pressure or flow.
- 2.1.20 If a valve fails the operability test, refer to the ASME Code Section XI, IWV-3417 or IWV-3514 and the applicable Technical Specifications for corrective action and time requirements for return to service. In any case, any abnormality or erratic action shall be reported to the Shift Supervisor.

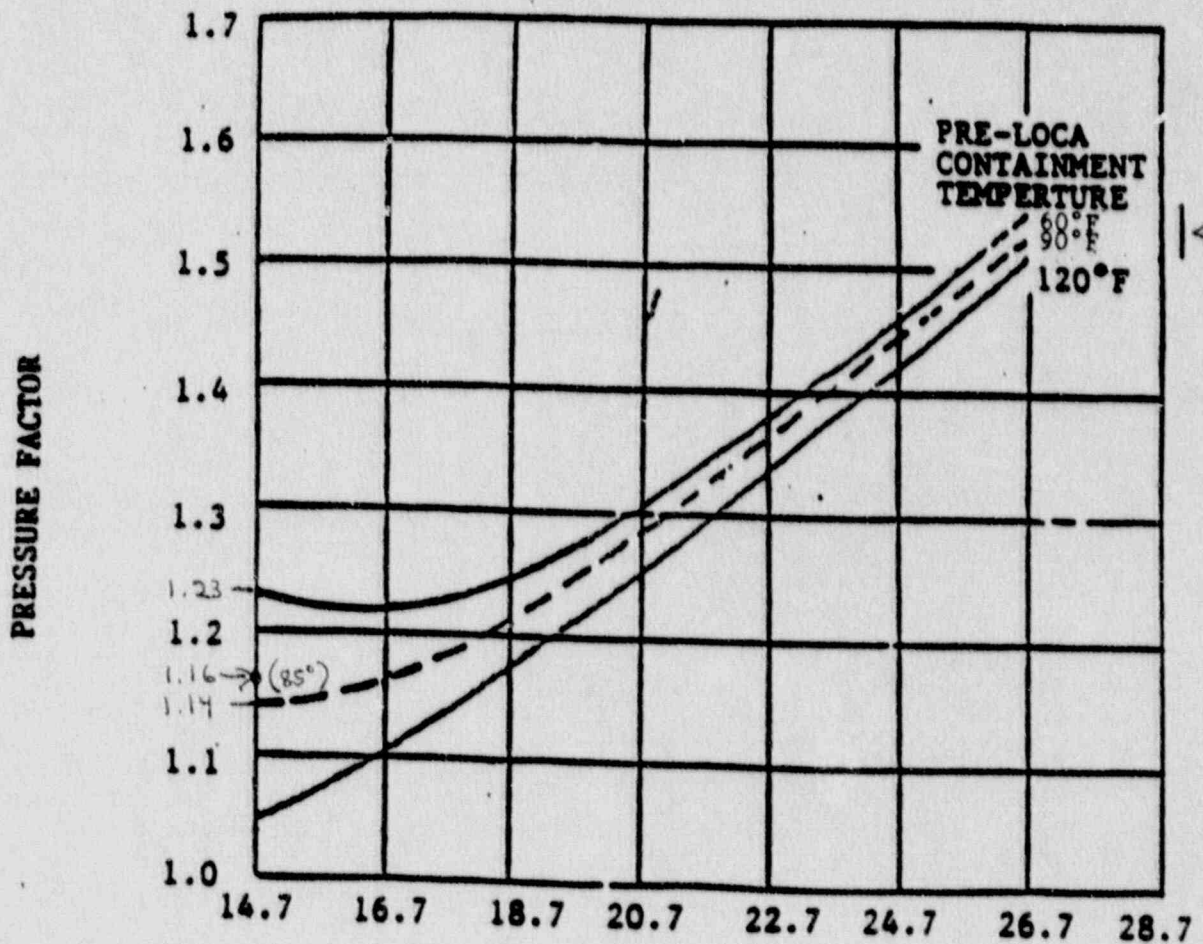
Title: Combustible Gas Control System	No.: 04-1-01-E61-1	Revision: 22	Page: 13
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REC ID 317

Figure 1

DW Containment Recombiner Power Correction Factor versus Containment Pressure.
(14.7 PSIA INITIAL PRESSURE)

(procedure revised 9-21-89)



1.16 thru interpolation

Post LOCA Containment Pressure (Psia)

Title: Drywell Chilled Water System	No.: 04-1-01-P72-1	Revision: 18	Page: 4
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REC ID 356

4.1.2 (Cont.)

- d. Locally at all four (4) drywell chillers check the following:
- (1) POWER ON light is On.
 - (2) The compressors are reset by pressing the STOP-RESET pushbutton.
 - (3) All other lights are Off.

NOTE

If the chillers were previously stopped locally, Control Room handswitch must be placed to STOP prior to placing in START in order to clear logic.

- e. Start DRWL CHILL A(B). Observe both A(B) units running lights On.
- f. Place DRWL CHILL B(A) in STBY after temperature of chilled water supply to drywell drops below 53°F on Computer Point P72-N003. Observe white standby light On.
- g. Locally check that the following occurs for the running units:
- (1) The No. 1 compressor starts, ten (10) seconds later the No. 2 compressor starts, and in an additional ten (10) seconds the No. 3 compressor starts.
 - (2) After approximately one (1) minute the compressors will start to load up. Check for decreasing suction pressure and increasing discharge pressure. Suction pressure should settle out somewhere between 65 to 75 psig and discharge pressure between 200 to 250 psig depending on heat load.

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6.5.3 (Cont.)

- (2) CAUTION - Describes equipment hazards or adverse operations

NOTE

Notes are for information only; they should not contain sequential instructions related to the activity being performed.

- (3) NOTE - Clarifies instructions

b. Placement

- (1) Center, capitalize and underline the word NOTE, CAUTION or WARNING.
 - (2) NOTES must be blocked.
 - (3) CAUTIONS or WARNINGS must be blocked and directly precede the steps referenced.
 - (4) WARNINGS and CAUTIONS and the impacted steps are placed on the same page.
 - (5) WARNINGS precede CAUTIONS if both refer to the same text step.
- c. If several NOTES, WARNINGS or CAUTIONS are grouped within one box, use only one heading; however, separate the statements.

6.5.4 Coordinated Actions

- a. Indicate coordinated actions of two or more persons by a NOTE. Indicate the person responsible for coordination and the means of communications to use.

6.5.5 Checkoff and Hold Points

- a. Denote any Quality Witness Points, Quality Hold Points and other checkpoints in the left margin of the instruction adjacent to the applicable step.

REC ID
397

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EMERGENCY CLASSIFICATIONS

CATEGORY	INITIATING CONDITION	EMERGENCY ACTION LEVEL	EMERGENCY CLASSIFICATION
12. Plant Shutdown Function (Cont.)	4. Transient requiring operation of shutdown systems with failure to scram and continued power generation but <u>NO CORE DAMAGE immediately evident</u>	1. Scram conditions confirmed with inability to manually scram <u>AND</u> 2. All control rods <u>NOT inserted</u> to between 00 and 06 <u>AND</u> 3.a. Reactor power \geq 5% on APRMs <u>OR</u> b. Reactor power \geq 20 on range 8 of inserted IRMs <u>AND</u> 4. No core damage immediately evident	<u>SITE AREA EMERGENCY</u>
	5. Reactor shutdown <u>AND</u> Loss of decay heat removal capability <u>AND</u> <u>CORE DAMAGE</u> predicted within several hours <u>AND</u> Subsequent containment failure	1. All rods fully inserted <u>AND</u> 2. No method is available for decay heat removal <u>AND</u> 3. Core damage has occurred or is predicted to occur within two hours as reported by Reactor Engineering <u>AND</u> 4. Containment pressure is $>$ 17.25 psig or containment is breached	<u>GENERAL EMERGENCY</u>

APRM - Average Power Range Monitor
IRM - Intermediate Range Monitor

Title: Use of Communications Equipment	No.: 01-S-06-14	Revision: 10	Page: 3
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6.0 DETAILS

6.1 Normal Operations

6.1.1 Communications Systems

a. C.W. Thompson Paging System

- (1) The Paging System can be used for plant alarms, plant announcements and paging personnel.
- (2) The Paging System consists of five party lines. Line 1 is for Operations use only. All other lines are available for plantwide use.

NOTE

A separate PA System, independent of the main PA System, is provided as a direct line of communication between the Control Room and the refueling areas of the Containment and Auxiliary Building. The separate system permits extended fuel-handling communications without disrupting communications on the remainder of the PA System.

A third independent PA System is also available for communication between the Emergency Shutdown Panel and the Control Room.

- (3) The C.W. Thompson Public Address Stations that are located in high noise areas have been modified to accommodate headsets to ensure adequate communications between the area and the Control Room. These headsets should be used when background noise in these areas warrants their use.

b. Public Telephone System

The Public Telephone System provides for both offsite and onsite communications. The telephones also provide access to the SERI Microwave System.

c. Radio System

The Radio System provides a means of communicating with remote areas of the site and of notifying personnel offsite (within a limited range) of plant events.

Grand Gulf Requalification Examination

Response to Facility Changes to Static Exam Answer Key (week #1)

Question 243:

Facility added "30 seconds following LOCA signal" to the portion of the answer not required for credit. Justification was that the auto initiation is delayed for 30 seconds and credit should not be deducted if this information is included.

NRC Response: Addition is noted and will be taken into account during grading.

Question 18:

Facility changed the answer to delete the requirement for "Div I" to be included for full credit. Justification was that "Div I" is given in the question and that just stating "Containment Spray Initiation Signal" is sufficient to demonstrate adequate knowledge.

Facility added "also accept 15AA deenergizing and subsequently reenergizing". Justification was a print showing the overriding of the manual overriding of the pump on power loss/return.

NRC Response: NRC will accept the both changes and grade accordingly.

Question 241:

Facility changed answer in part (a) to 490 +/- 10 psia to correct original value on answer key. Justification was steam tables.

NRC Response: NRC will accept the change and grade accordingly.

Question 164:

Facility added "also accept Emergency Stop P. B. (Local & Remote)" to the answers in part (a). Justification was the ARI for "HPCS DSL ENG TRIP" annunciator.

Facility added "or generator lockout" as correct answer for "generator differential" in part (a). Justification was to give terminology as found in the ARI.

NRC Response: NRC agrees that the ARI states the Emergency Stop Pushbuttons are still "operational" with a LOCA signal present but the question specifically asks for "Which D/G AUTOMATIC trips are currently active?". The Emergency Stop Pushbuttons are NOT automatic trips (they require operator action) while the other two trips do not require any operator action. NRC will not accept the facility change.

NRC accepts the second facility change and will grade accordingly.

Question 163:

Facility changed the grid location number for the "LPCS B/C INJ VLV RPV PRESS LO" annunciator from "G3" to "G2". Justification is to correct error in answer key.

NRC Response: NRC will accept the change and grade accordingly.

Question 165:

Facility changed "and the system will inject into the reactor" to "not required for full credit". Justification was that this is not and "effect" on the system as the question asks.

NRC Response: NRC will accept the change and grade accordingly.

Grand Gulf Requalification Examination

NRC Changes to the Static Exam Answer Key (week #1)

Question 244:

NRC changed the answer key to require the word "HIGH" be included only if the setpoint was not given in the answer. Justification is based on just stating the "condition" without stating either a value or "high or low" does not determine if the candidate knows why EP-3 should be entered.

Question 159:

NRC changed the answer to key to accept "FCV Lockup" or "FCV Motion Inhibit" as acceptable answers. Justification is that this is also acceptable terminology for what occurs on the trip of an HPU.

HPCS DSL ENG TRIP

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Safety Related

Alarm Device E22-XA-L618

REC ID 164

1.0 POSSIBLE CAUSES

- 1.1 Emergency stop pushbutton (Local and Remote)
- * 1.2 Low lube oil pressure
- 1.3 Generator trip lock-out (see 04-1-02-1H13-P601-1A-A1)
- * 1.4 Jacket water temperature high
- 1.5 Engine overspeed
- * 1.6 Crankcase pressure high
- * Signals bypassed with LOCA signal present

2.0 AUTOMATIC ACTION

- 2.1 Diesel engine trips
- 2.2 Trips diesel generator output breaker 152-1701 (BKR #1)

3.0 IMMEDIATE OPERATOR ACTION

- 3.1 Check that automatic actions occur.
- 3.2 If diesel generator was supplying bus 17AC, reenergize bus 17AC from another source (ESF XFMR 11, 12 or 21)
 - 3.2.1 Refer to EP-2, KPV Control, for other water sources if the HPCS system was maintaining reactor water level and power cannot be established to bus 17AC.

4.0 SUBSEQUENT OPERATOR ACTION

- 4.1 Determine cause of generator trip and return to STANDBY as soon as possible.
- 4.2 Ensure compliance with Technical Specifications 3.8.1.1, 3.5.1, 3.5.2.

REC ID 235

C.W. Thompson Paging System is the GGNS public address (P.A.) system. Examinees do not need to know manufacturers' name to demonstrate knowledge asked by question (see 01-S-06-14)

REC ID 243

Auto initiation is time delayed 30 seconds. Information is correct but not required for full credit since this is in parentheses, credit should not be deducted if this information is given (See 04-1-01-E61-1)

REC ID 18

"Div I" not required for full credit.

Examinees would demonstrate adequate knowledge of question by simply stating containment spray initiation signal. Div I is stated in question.

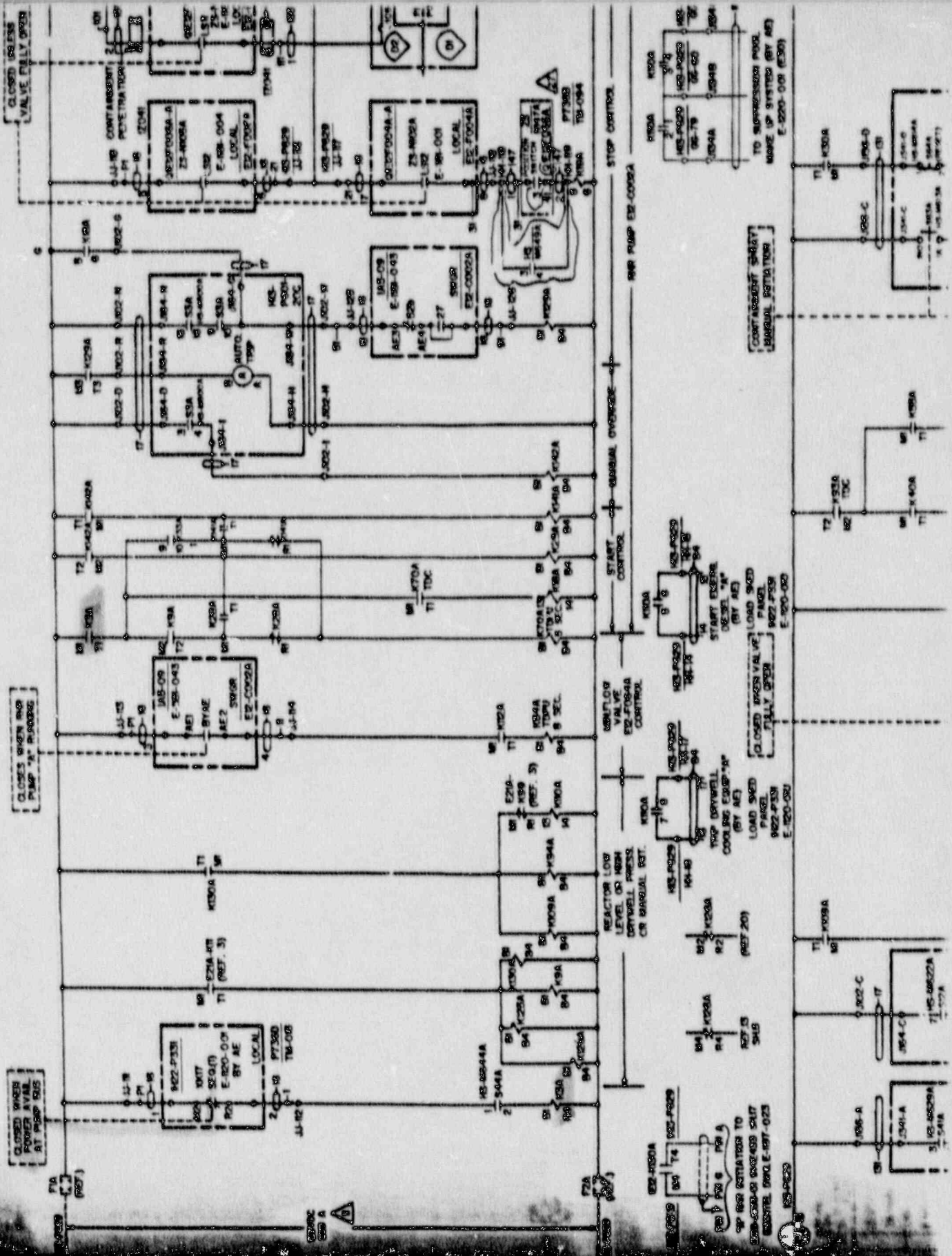
REC ID 380

"Step 21" not required for full credit.

Question ask to include reference but not a particular step of reference.

Stating trip one pump demonstrates Examinees are at proper step of EP-2A

REC ID 18
 K3A dropping out would break
 seal-in on manual override of pump



3-11-63 410 1048

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

LOAD SHED PANEL

CLOSED PRESSURE VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

LOAD SHED PANEL

CLOSED PRESSURE VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

LOAD SHED PANEL

CLOSED PRESSURE VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

LOAD SHED PANEL

CLOSED PRESSURE VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

LOAD SHED PANEL

CLOSED PRESSURE VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

LOAD SHED PANEL

CLOSED PRESSURE VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

LOAD SHED PANEL

CLOSED PRESSURE VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

LOAD SHED PANEL

CLOSED PRESSURE VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

LOAD SHED PANEL

CLOSED PRESSURE VALVE FULLY OPEN

CLOSED WHEN PUMP "A" IN PROGRESS

CLOSED WHEN VALVE FULLY OPEN

CLOSED WHEN POWER AVAIL AT PUMP BUS

COMPRESSOR START MANUAL ROTATION

REACTOR LOG LEVEL OR HIGH DRYWELL PRESSURE OR MANUAL RT.

START EMERGENCY DESEL

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REC ID 243

- 3.6 During hydrogen recombiner operation, maintain recombiner temperature between 1225°F and 1450°F to ensure hydrogen/oxygen recombination.
 - 3.7 Do not exceed a maximum hydrogen recombiner temperature of 1450°F as read on TI-R603A(B) on 1H13-P870.
 - 3.8 Do not exceed a maximum hydrogen recombiner power output of 75 kW as read on JI-R604A(B) on 1H13-P870.
 - 3.9 A 30-second time delay is imposed by the Drywell Purge Control System on the various Combustible Gas Control subsystem controls upon receipt of a LOCA signal. This allows for the isolation of the Combustible Gas Control subsystems during reactor blowdown. The Combustible Gas Control systems then automatically start after the time delay.
 - 3.10 E61-F009 PURGE SPLY CTMT OTBD ISOL VLV and E61-F010 PURGE SPLY CTMT INBD ISOL VLV are interlocked Closed for 24 hours following a LOCA. If the LOCA signal clears before 24 hours, the valves can be operated normally.
 - 3.11 Start H₂ igniters before hydrogen concentration reaches 3.5% to prevent the possibility of hydrogen explosion.
 - 3.12 Anytime control power is lost to one of the CGCS divisions, the logic must be reset to place system in STANDBY.
- 4.0 NORMAL OPERATIONS
- 4.1 Placing System in Standby
 - 4.1.1 Prerequisites
 - a. Attachment I, Manual Valve Lineup Checksheet, completed.
 - b. Attachment II, Remote Operated Valve and Switch Lineup Checksheet, completed.
 - c. Attachment III, Electrical Lineup Checksheet, completed.
 - d. Instrument air is available per SOI 04-1-01-P53-1.
 - e. Standby Service Water System running or in STANDBY in accordance with SOI 04-1-01-P41-1. (Drywell purge compressor only)
 - f. Domestic Water System in operation in accordance with SOI 04-1-01-P66-1. (CTMT purge compressor only)

Grand Gulf Requalification Examination
Response to Facility Changes to Answer Key
Week II

Question 495:

Facility changed supplementary information in the answer. Did not change answer.

NRC Response:

Noted change.

Question 26:

Facility changed part a of the answer to "also accept 18.2 only." Changed part b to "also accept 16.3 only." Changed part c from "Unusual Event, 2.1.1" to "Unusual Event, 2.1.2." Changed part d to also accept 19.7 also. Also added supplemental information indicating why certain references were incorrect for part d.

NRC Response:

Agree with comments and will grade accordingly.

Question 270:

Facility changed the TSPS reference from #68 to #125.

NRC Response.

The operators were not required to state the TSPS number so change does not affect grading.

Question 55:

Facility changed answer to require "Supt./Supervisor" instead of "Supt."

NRC Response:

The procedure indicates that it is the responsibility of the Supervisor to mark up the line-up before it commences, however it indicates that if problems are encountered during performance of the lineup it is the superintendent's responsibility. The question will be graded per the answer key.

Rec. I.D.
495

10-S-01-12	Revision 15
Attachment VI	Page 1 of 2

DOSE CALCULATION WORKSHEET

EMERGENCY CLASSIFICATION

- UNUSUAL EVENT
- SITE AREA EMER.
- ALERT
- GENERAL EMER.

- ESCALATED
- DE ESCALATED
- UNCHANGED

Notification Form # _____
Calculation # _____

METEOROLOGICAL INFORMATION

TIME OF DATA _____ DATE _____
 ΔT _____ °F
 $\sigma \theta$ _____ °
 STABILITY CLASS _____
 WIND SPEED _____ MPH
 WIND DIRECTION _____ ° (FROM)
 _____ ° (TO)
 BLOWING INTO _____ SECTORS
 (Minimum of 3)

RECOMMENDED PROTECTIVE ACTIONS

- None
- Evacuate 2 mile radius; Evacuate 5 mile radius;
- Evacuate 5 mile radius; 10 mile downwind sectors;
- Other - Specify _____

Comments: _____

RADIOLOGICAL INFORMATION

Release: Began _____ hrs. date _____
 Ended _____ hrs. date _____
 TIME OF DATA _____ DATE _____
 Source of data
 Effluent Monitor
 Default Source Term
 Field Data
 Field Location _____
 Whole Body Dose Rate _____ mRem/hr
 Radioiodine Airborne Conc. _____ uCi/cc
 Field Contamination _____ DPM/100 sq. cm
 Particulate Airborne Conc. _____ uCi/cc

OTHER INFORMATION

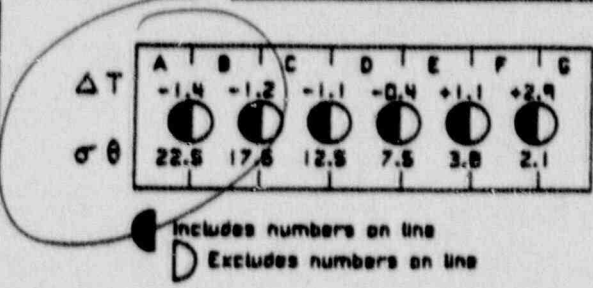
Time of Rx Shutdown _____
 Time After Shutdown _____

Mile or Sector Location	X/B [F] x E-6	G _{NG} (Ci/sec)	X _{NG} (Ci/m ³)	NGCF	Rem to mRem	Whole Body Dose Rate (mRem/hr)	HR. W.B. Dose Comm. (mRem)
.4	E-6				1000		
2	E-6				1000		
5	E-6				1000		
10	E-6				1000		
	E-6				1000		
	E-6				1000		

W.B. Dose Rate = (1) x (2) x (3)
 $X_{NG} = [F] \times E-6 \times (G_{NG})$
 $X_1 = [F] \times E-6 \times (G_1)$

(Child) Thyroid Dose Comm. (PROJECTED)
 $(X_1) (1.33E9) \text{ EXPOSURE TIME} = \text{mRem}$
 2

Mile or Sector Location	X/B [F] x E-6	G ₁ (Ci/sec)	X ₁ (Ci/m ³)	HR (Child) Thyroid Dose Comm. (mRem)
.4	E-6			
2	E-6			
5	E-6			
10	E-6			
	E-6			
	E-6			



TIME AFTER SHUTDOWN (HRS)	0.0	2.5	4	5.5	7	9	11	14	18	26	35	44	60	96
NGCF (Rem/h ² / (Ci/m ³))	530	430	340	270	220	170	140	110	88	68	52	41	34	28

*** **NOTES FOR NGCF DETERMINATION**

- a) If Time falls between listed times, record the higher conversion factor.
- b) If Time is greater than 96 Hrs. use Conversion Factor for 96 Hrs.

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ATMOSPHERIC DISPERSION MULTIPLICATION FACTORS (F)

INSTRUCTIONS FOR USE OF TABLES

- 1) Select the table for the Stability Class determined per Section 6.2.
- 2) If the Wind Speed is not shown on the table, use the (F) factor for the next slowest Wind Speed (use the higher or more conservative F factor). If the Wind Speed is >40 mph, use the (F) factors for 40 mph.
- 3) Record (F) factors for SB, 2, 5 and 10 miles on the Dose Calculation Worksheet.

A STABILITY

Windspeed Mi/Hr	SB	2 mi	5 mi	10 mi
	0.4 mi			
1	21.93	1.32	0.58	0.31
2	10.96	0.66	0.29	0.16
3	7.31	0.44	0.19	0.10
4	5.48	0.33	0.15	0.08
5	4.39	0.26	0.12	0.06
6	3.65	0.22	0.10	0.05
7	3.13	0.19	0.08	0.04
8	2.74	0.16	0.07	0.03
9	2.44	0.15	0.06	0.03
10	2.19	0.13	0.06	0.03
12	1.83	0.11	0.05	0.02
14	1.57	0.09	0.04	0.02
16	1.37	0.08	0.04	0.02
18	1.22	0.07	0.03	0.02
20	1.10	0.07	0.03	0.01
25	0.88	0.05	0.02	0.01
30	0.73	0.04	0.02	0.01
40	0.55	0.03	0.01	0.01

Rec. I.D. 26

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EMERGENCY CLASSIFICATIONS

CATEGORY	INITIATING CONDITION	EMERGENCY ACTION LEVEL	EMERGENCY CLASSIFICATION
2. Abnormal Primary Leak Rate	1. Exceeding primary coolant system leak rate	1. > 0 pressure boundary leakage <u>OR</u> 2. > 30 gpm total leakage	<u>UNUSUAL EVENT</u> <i>MUST SPECIFY 1 or 2 for full credit</i>
	2. Coolant leak rate > 50 gpm	1. Total leakage calculated to be > 50 gpm while in Plant Operational Condition 1, 2 or 3	<u>ALERT</u>
	3. LOCA > makeup pump capacity	1. Reactor vessel water level at or below the top of active fuel as indicated on fuel zone level indicator (-167" fuel zone) <u>AND</u> 2. Makeup capacity unable to increase reactor vessel level	<u>SITE AREA EMERGENCY</u>
3. Abnormal Coolant Temperature/Pressure/Safety Limit	1. Abnormal reactor coolant pressure and/or temperature <u>OR</u> Abnormal fuel temperatures	1. Reactor vessel steam dome pressure > 1325 psig <u>OR</u> 2. Reactor/thermal power > 25% rated (958 MWt) <u>AND</u> < 785 psig dome pressure <u>OR</u> < 10% core flow <u>OR</u> 3. MCPR < 1.06 (Single Recirc Loop Ops - MCPR < 1.07) <u>AND</u> > 785 psig dome pressure and > 10% core flow <u>OR</u> 4. Temperature to the left of the applicable curve on Tech Spec Figure 3.4.6.1-1	<u>UNUSUAL EVENT</u>

LOCA - Loss of Coolant Accident
MCPR - Minimum Critical Power Ratio

MWt - Megawatt Thermal

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EMERGENCY CLASSIFICATIONS

CATEGORY	INITIATING CONDITION	EMERGENCY ACTION LEVEL	EMERGENCY CLASSIFICATION
4. Core Fuel Damage (Cont.)	4. Loss of 2 of 3 fission product barriers with a potential loss of 3rd barrier	1. Radiation monitoring team reports indicate (at the site boundary): a. > 50 mR Whole Body OR b. > 3.75 E-7 µCi/cc Iodine AND 2. Containment pressure exceeds 17.25 psig or containment is breached	GENERAL EMERGENCY NOT CORRECT FOR PART D
5. Steam Leaks	1. Main steam line break outside the containment with significant MSIV leakage.	1. Isolation initiated and abnormal leakage down stream of MSIVs (> 10 gpm or 5000 lbm/hr)	ALERT
	2. RCIC steam line break outside the containment with significant isolation valve leakage	1. Isolation initiated and abnormal leakage down stream of isolation valves (> 10 gpm or 5000 lbm/hr)	
	3. Main steam line break outside of containment which <u>cannot be isolated.</u>	1. Isolation required due to confirmed steam line break AND One or more main steam lines fails to isolate	SITE AREA EMERGENCY
	4. RCIC steam line break outside of containment which <u>cannot be isolated.</u>	1. Isolation required due to confirmed steam line break AND RCIC steam line fails to isolate	

MSIV - Main Steam Isolation Valve
 RCIC - Reactor Core Isolation Cooling

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EMERGENCY CLASSIFICATIONS

CATEGORY	INITIATING CONDITION	EMERGENCY ACTION LEVEL	EMERGENCY CLASSIFICATION
12. Plant Shutdown Function (Cont.)	4. Transient requiring operation of shutdown systems with failure to scram and continued power generation but <u>NO CORE DAMAGE immediately evident</u>	1. Scram conditions confirmed with inability to manually scram <u>AND</u> 2. All control rods <u>NOT inserted</u> to between 00 and 06 <u>AND</u> 3.a. Reactor power \geq 5% on APRMs <u>OR</u> b. Reactor power \geq 20 on range 8 of inserted IRMs <u>AND</u> 4. No core damage <u>immediately evident</u>	<u>SITE AREA EMERGENCY</u>
	5. Reactor shutdown <u>AND</u> Loss of decay heat removal capability <u>AND</u> <u>CORE DAMAGE predicted within several hours</u> <u>AND</u> Subsequent containment failure	1. All rods fully inserted <u>AND</u> 2. No method is available for decay heat removal <u>AND</u> 3. Core damage has occurred or is predicted to occur within two hours as reported by Reactor Engineering <u>AND</u> 4. Containment pressure is $>$ 17.25 psig or containment is breached	<u>GENERAL EMERGENCY</u>

NOT CORRECT FOR PART D

APRM - Average Power Range Monitor
IRM - Intermediate Range Monitor

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EMERGENCY CLASSIFICATIONS

CATEGORY	INITIATING CONDITION	EMERGENCY ACTION LEVEL	EMERGENCY CLASSIFICATION
16. Security Threat	1. Security threat <u>OR</u> Attempted entry <u>OR</u> Attempted sabotage	1. Based upon the assessment of the alarm or the event reported by Security. Actual threat must be determined prior to establishing an emergency classification	<u>UNUSUAL EVENT</u>
	2. On-going Security compromise	1. Identification of adversaries attempting to command areas of plant, but not <u>controlling shutdown capability</u> <u>OR</u> vital areas	<u>ALERT</u>
	3. Imminent loss of physical control of plant	1. Physical attack on the plant involving imminent occupancy of the Control Room or Remote Shutdown Panel.	<u>SITE AREA EMERGENCY</u>
	4. Loss of physical control of the facility	1. Physical attack on the plant has resulted in unauthorized personnel occupying the Control Room or Remote Shutdown Panel	<u>GENERAL EMERGENCY</u>

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EMERGENCY CLASSIFICATIONS

CATEGORY	INITIATING CONDITION	EMERGENCY ACTION LEVEL	EMERGENCY CLASSIFICATION
18. Natural Events	1. Natural events being experienced or projected beyond usual levels	1. A verified earthquake detected by in-plant seismic instrumentation OR 2. A tornado observed onsite OR 3. A hurricane warning issued that includes the site area	<u>UNUSUAL EVENT</u>
	2. Severe natural event near site being experienced or projected	1. A verified earthquake detected by in-plant seismic instrumentation \geq OBE levels OR 2. A tornado causing damage to Safety Related structures OR 3. Sustained winds \geq 73 mph onsite	<u>ALERT</u>
	3. Severe natural event near site being experienced or projected with plant in Modes 1, 2, or 3	1. A verified earthquake detected by in-plant seismic instrumentation \geq SSE levels OR 2. Sustained winds \geq 90 mph onsite	<u>SITE AREA EMERGENCY</u>
	4. Major internal <u>OR</u> external events	1. Fires, earthquakes, etc., substantially beyond design basis which could or have caused massive common damage to plant systems	<u>GENERAL EMERGENCY</u>

OBE - Operating Earthquake
SSE - Safe Shutdown Earthquake

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EMERGENCY CLASSIFICATIONS

CATEGORY	INITIATING CONDITION	EMERGENCY ACTION LEVEL	EMERGENCY CLASSIFICATION
19. Other (Cont.)	5. Loss of <u>ALL</u> <u>annunciators</u> (conditions not stable)	1. Loss of all annunciators in the controlled area <u>AND</u> 2. The plant is <u>NOT</u> in a <u>stable condition</u> (a significant transient initiated or in progress)	<u>SITE AREA EMERGENCY</u>
	6. Other plant conditions exist that warrant activation of Emergency Facilities	1. Emergency Director determines that plant conditions exist that warrant: a. The activation of the EOF and SAP <u>OR</u> b. A precautionary notification to the public near the site	
	7. Other plant conditions exist that make <u>release of large amounts of radio-activity</u> in a short time possible	1. Reactor vessel water level at or below the top of active fuel <u>AND</u> 2. Containment pressure is > 17.25 psig or containment is breached <u>AND</u> 3. Core damage is predicted to occur (within 2 hours) as reported by Reactor Engineering	<u>GENERAL EMERGENCY</u>

EOF - Emergency Facility
SAP - Site Access Point

Rec. ID 270

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TECHNICAL SPECIFICATION POSITION STATEMENT

TSPS Number: 125

Part 1 Originator: J. L. Robertson 3/16/89
Name Date

Technical Specification: 3.5.2

References: AECM 89/0052, 89/0053. Supercedes TSPS 068 & 121

Technical Specification Change Required* (X) NO YES () If Yes CR #

Position: See Page 2

Position Affect/Comments: This position statement provides ECCS requirements in OPERATIONAL CONDITIONS 4 and 5 until a permanent resolution is reached.

*Under evaluation.

Part 2 Plant Licensing Superintendent [Signature] 3-16-89
Signature Date

Table with 4 columns: Review and Approval, Approve, Disapprove. Rows include System Engineering Superintendent, Operations Superintendent, PSAC Meeting Number, OGNS General Manager, OGNS Site Director, and Manager, Nuclear Licensing.

SERI committed to apply the following ECCS restrictions in OPERATIONAL CONDITIONS 4 and 5 when two ECCS systems are required OPERABLE:

1. At least one of the two ECCS required OPERABLE by Tech Spec 3.5.2 must be capable of automatic initiation and injection to the reactor vessel (i.e., without reliance on manual realignment)
2. This OPERABLE automatic initiation and injection ECCS must be associated with an OPERABLE diesel/generator.

If either of the above conditions are not met, in addition to any other action requirements, immediately suspend operations that have a potential for draining the reactor vessel. "Immediately" shall not be interpreted to preclude completion of actions to place the operation that has the potential to drain the vessel in a safe condition.

In addition, any OPERABLE ECCS requiring manual realignment must be capable of being realigned from the Control Room panels within 25 minutes.

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6.12.4 The normal completion of a checksheet will be as follows:

CAUTION

Checksheets are normally performed as one step of the SOI. The applicable section of the SOI must be referred to, to ensure the desired system condition is achieved or maintained.

- Rec.
IP
53
- a. For operating checksheets, the Shift Superintendent will notify the Control Room Operator that an evolution must be performed, obtain the latest revision of the required checksheets following the guidelines of 6.11 and have personnel assigned to perform the evolution.
 - b. For system lineups the Shift Supervisor will obtain a copy of the lineup checksheets. This may be a copy from the controlled set of SOIs in the Control Room. He will verify that the lineup checksheet is complete and legible. He will also verify the copy to be the latest revision with all applicable TCNs entered. He will stamp the top of the first page of each set of checksheets with a stamp that says CONTROLLED COPY. In the event a controlled copy stamp is not available, he may write the words CONTROLLED COPY on the checksheets using red ink. He will then sign and date the checksheet beneath the CONTROLLED COPY stamp. This signature and date will indicate that he has verified the proper revision and applicable TCNs. The stamp and signature will be good for the entire time the checksheet is in use. The Shift Supervisor will then determine if there are any required deviations from the SOI lineup by checking Clearance Logs and the Temporary Alteration Log or NA any portion of the lineup that is not to be performed.
 - c. All required deviations will be entered on the checksheet as follows:

NOTE

The following is allowed only if a permanent change to the checksheet is not required. If a permanent change to the checksheet is required, a TCN will be used in accordance with Reference 3.2.

- (1) To change the "Req. Pos." of a valve or breaker on a checksheet, the Shift Supervisor will write in the new position in the "Deviation" column.

Title: Control & Use of Operations Section Directives	No.: 02-S-01-2	Revision: 17	Page: 12
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6.12.4 (Cont.)

- (2) To delete a portion of a checksheet, the Shift Supervisor will place "NA" in the "1st Ck" and "2nd Ck" block for each item not required to be done. If an entire page is "NA," the page may be omitted from the checksheet if so denoted in the "Note Exceptions" portion of the checksheet.
 - (3) The Shift Supervisor will write in the reason for the deletion(s)/deviation(s) in the "Note Exceptions" section of the checksheet. This may include clearance numbers, if applicable.
 - d. The designated person will perform the lineup with the checksheet in hand and initial each step as it is completed. If a component cannot be placed in the required position, the Shift Superintendent shall be notified. The Shift Superintendent will resolve the problem by clearance tag removal, repairs, etc., or by changing the required position or deleting the component.
 - e. Attachment III provides acceptable methods for determination of positions for various components.
 - f. The completed checksheet will be signed by the performer. In cases where more than one operator performs the checksheet only one operator must sign the last page of the checksheet. This signature indicates the lineup is complete.
 - g. The completed checksheet will be returned to the Shift Supervisor, who reviews and signs it. This signature indicates approval of all deviations and deletions in the lineup and indicates that the Shift Supervisor has re-verified the checksheet for proper revision and that all appropriate TCNs have been incorporated.
 - h. After the Shift Supervisor signs a checksheet, any items deviated on the checksheet due to equipment clearances will be controlled per Reference 3.8, and will not be controlled by the checksheet.
 - i. The completed checksheets are placed in the System Status File.
- (1) When two or more partial lineup checksheets have been completed which together have included lining up every component listed on the checksheet, this set of partial checksheets may be considered an entire checksheet.

Combination of
Shift Supv. / Supt.

Grand Gulf Requalification Examination

Response to Facility Changes to Static Exam Answer Key (week #2)

Question 604:

Facility added additional correct answers, i.e. the downstream LCCs and MCCs powering the rod position indications in RC & IS. Justification was the electrical lineup checksheet from the RC & IS SOI.

NRC Response: NRC will accept the changes and grade accordingly.

Question 629:

Facility deleted the question from the exam. Justification is that the question was requesting what actions would or would not occur based on the loss of power and not on the fact that reactor pressure is still too high for any actions to occur per the ONEP for Automatic Isolations.

NRC Response: NRC will agree to delete the question and modify the total points for the exam accordingly.

Question 627:

Facility deleted "Must override the High Drywell pressure signal" portion of the answer. Justification is that the question asks for operator action, not justification or clarification.

NRC Response: NRC will not accept the change and does not agree that the deleted portion is a justification or clarification of what operator actions are required in the question especially since the words "HPCS initiation logic" are included in the question tending to lead the candidate to the correct actions required. Answer will be graded from the key as originally written.

Question 580:

Facility deleted part (a) of the question and changed the point values of the remaining two parts. Justification is that the initial conditions specified N094B as being valved out while the question stated that N094F was valved out, negating the point of the question.

NRC Response: NRC will agree to delete part (a) from the question and will grade accordingly.

Question 583:

Facility changed the answer to "accept arm and depress F only." Justification is that the "B" subchannel logic is sealed-in from the valved out transmitter so "F" is the only button required to initiate the "B" ADS.

NRC Response: NRC will agree to the change and will grade accordingly.

Question 585:

Facility split the point value for one part of the answer for part (a) into two separate values. Justification is that the answer consists of an electrical bus and a time limitation and should be graded separately.

NRC Response: NRC will agree to the change and will grade accordingly.

Question 586:

Facility added "can accept D/G in maintenance mode" to the answer. Justification is the ARI lists this as a possible cause for the D/G tripping.

NRC Response: NRC will not accept the change. The question specifically asks for "possible AUTOMATIC" causes for the D/G trip. The engine being in the maintenance mode is done by operator actions only, with no automatic features.

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REC ID 629

AUTOMATIC ISOLATION CHECKLIST (Continued)Reactor Water Cleanup Group 8Division 2

1H13-P680

G33-F001	RWCU PMP SUCT DRWL INBD ISOL	
G33-F028	RWCU BLWDN CTMT INBD ISOL	
G33-F040	RWCU RTN CTMT INBD ISOL	
G33-F053	RWCU PMP DISCH CTMT INBD ISOL	
G33-F251	RWCU SPLY TO RWCU HXS	1H13-P870
G33-F252	RWCU RTN TO RWCU PMPS	

RCIC Exhaust Vacuum Breakers Group 9Isolation Signals (need both signals)

Drywell Pressure - High 1.39 psig

RCIC Steam Supply Pressure - Low 60 psig

Valves ClosedDivision 1

1H13-P601

E51-F077 RCIC TURB EXH OTBD VAC BRKR

Division 2

E51-F078 RCIC TURB EXH INBD VAC BRKR

Reactor Sample Isolations Group 10Isolation Signals

Reactor Vessel Water Level - Low Low (Level 2) -41.6

Main Steam Line Radiation - High 3.0 X Full Power Background

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System Rod Control and Information System MPL No. 1C11

Checksheet Name Electrical Lineup Checksheet

Instruction Step 4.1.1a, 4.2.2.a

REC ID 604

COMPONENT NO.	COMPONENT DESCRIPTION	MCC, LCC PANEL NO.	BREAKER NO.	REQUIRED POSITION	BREAKER DEV.	INITIALS	
						1st CK	2nd CK
	5 VDC POWER SUPPLY						
	18 VDC POWER SUPPLY						
	CRD TEMPERATURE RECORDER	15P41	52-1P54109	CLOSED			
	5 VDC POWER SUPPLY						
	28 VDC POWER SUPPLY						
	PGCC PANEL 1H13-P651						
	ROD PATTERN CONTROLLER MULTIPLEXER	15P61	52-1P56138	CLOSED			
	5 VDC POWER SUPPLY						
	28 VDC POWER SUPPLY						
	PGCC PANEL 1H13-P652						
	ROD PATTERN CONTROLLER MULTIPLEXER	16P61	52-1P66138	CLOSED			
	RC&IS						
	1H13-P653 C11	1Y78	08-1Y78-29	CLOSED			
	ROD CONT & INFO SYS						
	1H13-P680 C11	1Y71	08-1Y71-04	CLOSED			
	PGCC PANEL 1H13-P610						
	P53 24 VDC POWER SUPPLY	16P61	52-1P66104	CLOSED			

Note Exceptions: _____

Performed by: _____ Date _____

Reviewed by: _____ Date _____

DIV 1 DSL GEN
TRIP

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Safety-Related

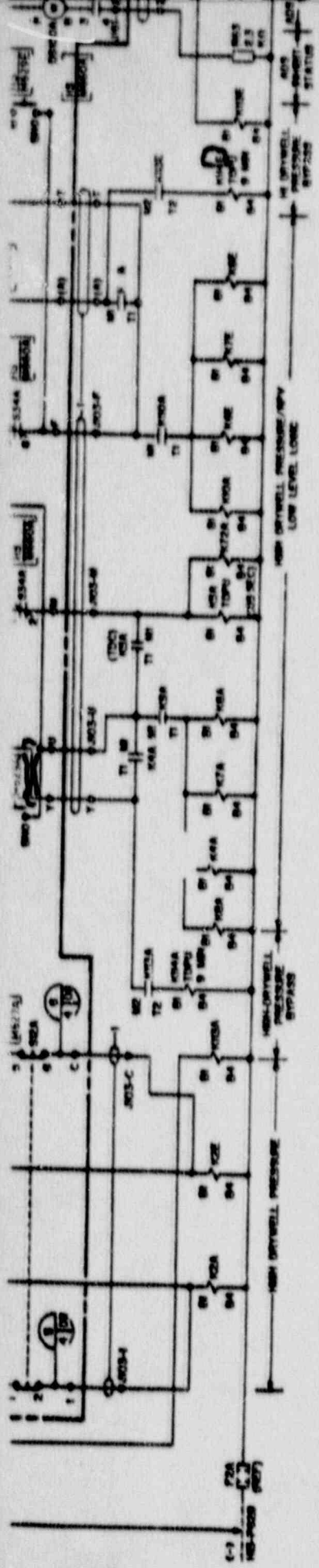
Alarm Device 1P75-XA-L605A

1.0 POSSIBLE CAUSES

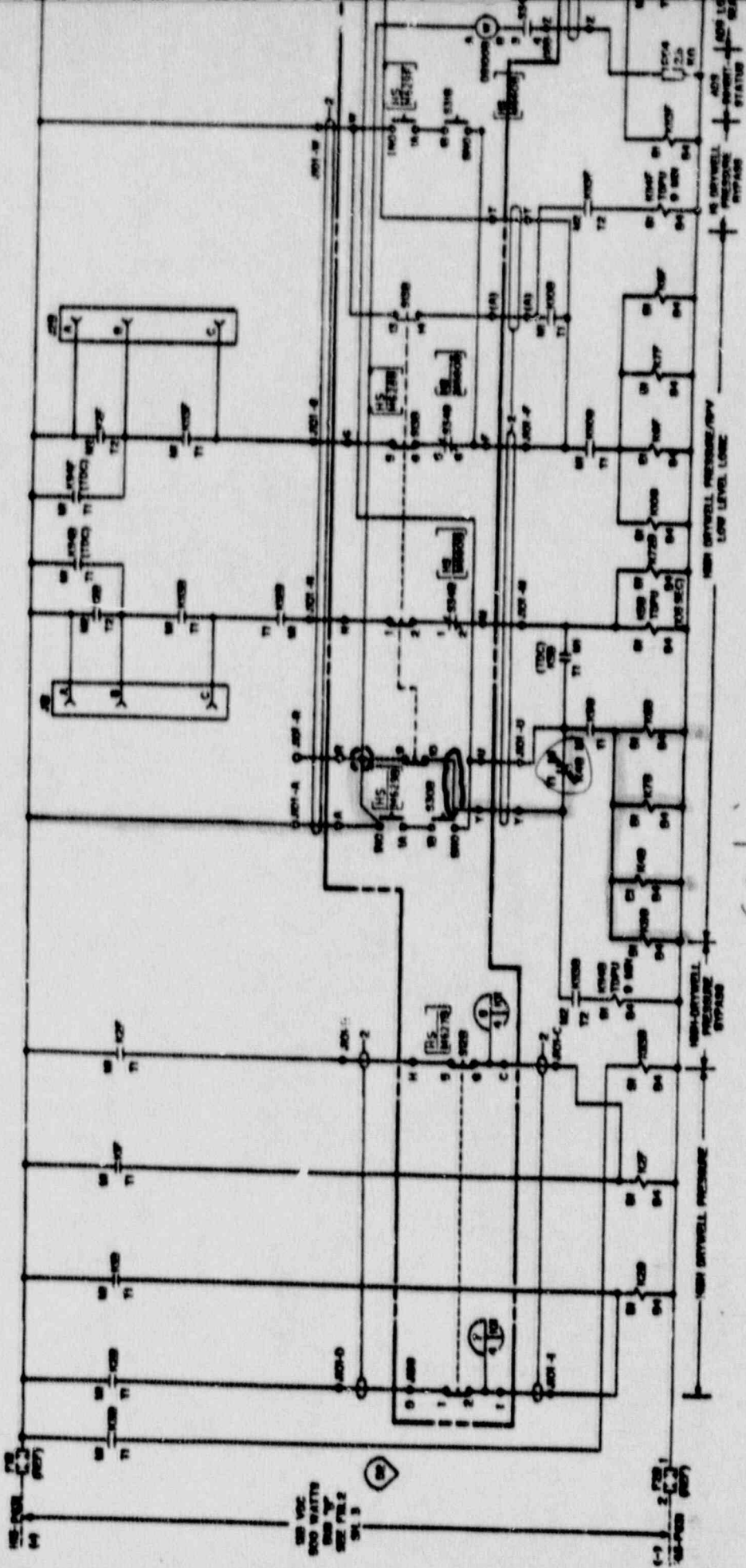
- 1.1 Diesel overspeed trip at 517.5 RPM.
- 1.2 Diesel lube oil pressure low, less than 30 psig (2 of 3).
- * 1.3 Diesel crankcase pressure high, greater than 3.0 psig (2 of 3)
- 1.4 Remote Manual Diesel Stop pushbutton on 1H13-P864 (with Master Selector Switch in STDBY mode).
- 1.5 Local Manual Diesel Guarded Stop pushbutton on 1H22-P400.
- 1.6 Diesel Generator Lockout Relays 186D1 and 186D2 operated.
- * 1.7 Turbocharger lube oil pressure low, left or right.
- * 1.8 Diesel Engine high vibration at 12 mils.
- * 1.9 Diesel Turbocharger high vibration at 12 mils.
- * 1.10 Diesel engine main bearing temperature greater than 228°F.
- * 1.11 Jacket Water temperature greater than 200°F.
- * 1.12 Diesel engine lube oil temperature greater than 200°F.
- 1.13 STOP/RUN valve in PUSH TO STOP position.
- 1.14 Diesel in Maintenance Mode.
- 1.15 Barring Device not Locked Out.
- * These trips are blocked when an ECCS Signal or local Emergency Manual start signal exists.

2.0 AUTOMATIC ACTION

- 2.1 Diesel Generator A trip.
- 2.2 Diesel Generator A output Breaker 152-1508 trips open.
- 2.3 Lube Oil Heater Pump C009A starts.
- 2.4 Jacket Water Heater Circulating Pump C006A starts.

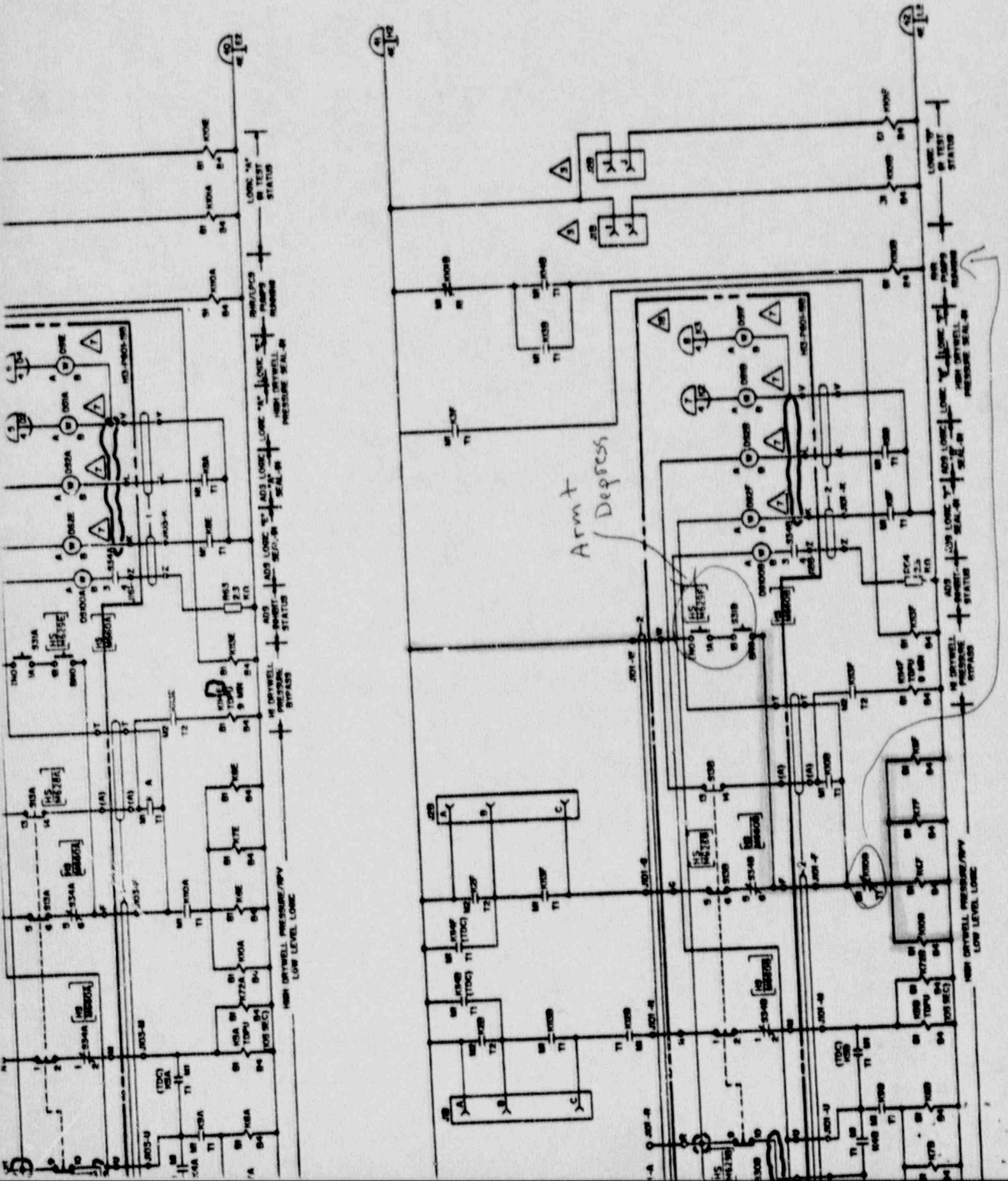


REC'D 583



Seal in on
Subchannel 'B' logic

SEE VICE
800 WATTS
800 V
SEE P&L
DL 3



Arm + Depress

Part A-1
Week 1

EXAMINER GUIDE
GRAND GULF NUCLEAR STATION
STATIC SIMULATOR EXAM
SCENARIO 3

GOOD KEY

I. SIMULATOR SET-UP

- A. Initialize the simulator to IC-17.
- B. Set all recorders to the correct time, rotating each chart a minimum of 3 inches and ensuring sufficient paper for the examination.
- C. Establish exam conditions as follows:

<u>Event</u>	<u>Time</u>	<u>Malf. No.</u>	<u>Description</u>
1	0	63,,,10%	Recirculation Loop Break, 10% severity
2	0	40A	Division 1 Drywell Pressure transmitter fails as is

D. Operator Actions

- 1. Perform Scram ONEP including placing the Reactor Mode Switch to shutdown, selecting all SRMs and IRMs and driving them in, Selecting IRMs on the APRM recorders.
- 2. At problem time 1, secure the B RFPT.
- 3. At problem time 3, restore Instrument Air to the Auxiliary Building and Containment and Drywell, and restore Drywell Cooling, restore PSW to the Auxiliary Building.
- 4. At problem time 3, restore MCC 16B42 using Remote Function 100 on page R20.
- 5. At problem time 5, reset RFPT High Level trips if necessary and restart the A RFPT and align for the Startup level Controller and restore Recator level to between 30 and 40 inches.
- 6. At problem time 7, secure both the A and B Heater Drain Pumps, C Condensate Pump and C Condensate Booster Pump.
- 7. At problem time 10, place the simulator in FREEZE.

II. VALIDATION

- A. For the malfunctions listed, verify the following desired results are obtained:
1. For the Reculation loop break in the Drywell, Drywell pressure is > 1.39 Psig.
 2. For the Drywell pressure transmitter failure, the "A" High Drywell Pressure signal is not received by lack of the appropriate white status light above the ADS High Drywell pressure reset pushbuttons.
 3. Both annunciators RECIRC FCV A/B PARTIAL CLOSE/TRIP are sealed in on the P680 panel.
- B. For the examination, the following parameters must be in the allowed band:
1. Drywell pressure > 1.39 psig
 2. Containment pressure < 3 psig
 3. Reactor pressure > 550 psig
 4. Reactor level between 30 and 40 inches.
- C. Check all recorder traces and parameters for reasonable values and trends for existing simulator conditions.
- D. If any parameter is outside the allowed band, the simulator must be re-initialized, beginning at Step I. A.
- E. If all indications are satisfactory, the exam may begin.

III. QUESTION AVAILABILITY

- A. The following questions were developed specifically for use with Scenario 6:

<u>REC ID</u>	<u>LESSON PLAN</u>	<u>OBJECTIVES</u>	<u>TIME (min.)</u>
159	OP-LO-SYS-LP-B33-2	9	5
160	OP-LO-SYS-LP-E30	5,8b	10
161	OP-LOR-SYS-LP-ECCS	4	10
162	OP-LOR-SYS-LP-ECCS	15	10
163	OP-LOR-SYS-LP-ECCS	12A	10
164	OP-LOR-AC-LP-002	4	10
165	OP-LOR-SYS-ECCS	23,26F,27	10
166	OP-LOR-SYS-LP-E51	5.A.2	5
244	OP-LOR-EP-LP-001	1	5

SCENARIO 3

INITIAL CONDITIONS

Approximately 10 minutes ago, plant conditions were as follows:

100% steady state power
1280 MWE
All controls in AUTO/NORMAL
No active or potential LCOs existed

Then, a transient occurred resulting in a scram and isolation. Initial corrective operator actions were taken in accordance with the OFF-NORMAL and EMERGENCY Procedures. The Shift Superintendent is currently reviewing the EPPs for event classification.

***** ANNUAL REQUALIFICATION EXAM ANSWER KEY *****
SCENARIO 3

QUESTION: 1 REC ID: 159 PT VAL: 1.00
LP: OP-LO-SYS-B33-2 Objective: 9

SCENARIO 3

Both "RECIRC FCV A/B PARTIAL CLOSE/RFP TRIP" annunciators are sealed in. However, the flow control valves are at approximately 70% open. Explain why the flow control valves did not run back. Include in your explanation any signal(s) that prevented the run back.

ANSWER: 1

High drywell pressure (0.5) caused the HPUs to trip (0.5) (preventing any further valve movement.) (also accept "HPU lockup" for "HPUs to trip)

SHOULD ALSO ACCEPT FCV LOCKUP

Ref. SOI 04-1-01-B33-1, Schematic Diagram E-1163

OR "MOTION INHIBIT"

Edited 11/13 - modified question to include initiating signal
- added point breakdown for answer
- added HPU lockup as acceptable answer

QUESTION: 2 REC ID: 161 PT VAL: 1.00
LP: OP-LOR-SYS-LP-ECCS Objective: 4

SCENARIO 3

- a. Why did DIVISION I ECCS not initiate?
- b. What action(s) would be required to initiate Division I ECCS?

ANSWER: 2

a. The "A" or Division I High Drywell Pressure signal was not received. (0.5) *(RPV LEVEL OF -150.3 " WAS NOT REACHED)*

b. Arm and depress the "LPCS/RHR A MAN INIT" pushbutton. (0.5)
(also accept "Div I ECCS" for "LPCS/RHR A")

Ref. Schematic Diagram E-1182

Edited 11/13 - modified part b of the question
- changed point total to 1.0 vice 1.5
- added Div I ECCS as acceptable answer

QUESTION: 3 REC ID: 163 PT VAL: 2.00
LP: OP-LOR-SYS-LP-ECCS Objective: 12a

SCENARIO 3

- a. Injection valve E12-F042B is currently CLOSED. What signal(s) are NOT yet present for an automatic opening of E12-F042B?
- b. What annunciation would indicate that the required signal(s) was/were present?

ANSWER: 3

- a. Reactor pressure is not below 476 psig. (Also accept 516 psig T.S. 3.3.3) (1.0)
- b. Annunciator "LPCI B/C INJ VLV RPV PRESS LO" (1.0)
(also accept "P601-17A-G~~2~~" for annunciator name)
2 MCB 11/20/89

Ref. SOI 04-1-01-E12-1, Schematic Diagram E-1181

Edited 11/13 - modified part b of question
- added annunciator location as acceptable answer

QUESTION: 4 REC ID: 164 PT VAL: 1.50
LP: OP-LOR-AC-LP-002 Objective: 4

SCENARIO 3

Concerning the Division III Diesel Generator (D/G):

- a. Which D/G automatic trips are currently active?
- b. What is the maximum time the D/G can run in its current condition?
- c. *Assume* ~~After~~ the D/G has run unloaded for 10 minutes, what additional actions must be taken prior to shutting it down?
- d. What operator action must be taken PRIOR to paralleling and loading the D/G?

ANSWER: 4

- a. Overspeed
Generator Differential (*or generator lockout*) ARI 04-1-02-1H13-P601-16A-A2
(0.25 pts ea)
~~also accept Emergency stop P.B (Local and Remote) ARI 04-1-02-1H13-P601-16A-A2~~
- b. Four hours. (0.25 pts)
- c. It must be loaded to 50% load for at least 30 minutes. (0.25 pts)
- d. The HPCS initiation logic has to be reset. (0.5 pts)

Ref. SOI 04-1-01-P81-1, Schematic Diagram E-1188

Edited 11/13 - modified parts c and d of question

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SCENARIO 3

QUESTION: 5 REC ID: 165 PT VAL: 1.50
LP: OP-LOR-SYS-LP-ECCS Objective: 27,26f,23

SCENARIO 3

Consider the High Pressure Core Spray (HPCS) System:

- a. Identify the effects on the system if the HPCS High Level Reset pushbutton is depressed.
- b. What signal caused the HPCS suction to realign to the suppression pool?
- c. If level subsequently decreases to -70 inches, explain how the HPCS System will respond and at what reactor level will this response occur? (Include specific valve(s) movement(s))

ANSWER: 5

- a. Injection valve E22-F004 will open ^{0.5} (0.25) ~~(and the system will inject into the reactor)~~ (0.25) *not required for full credit since this is not an effect on the HPCS system.* MLB
11/20/89
- b. Suppression Pool Level (0.25)
- c. Level reaches -41.6 inches (0.25), the injection valve E22-F004 will open (allowing injection into the reactor) (0.25) and the minimum flow valve will close (0.25).

(also accept "level 2" for "-41.6 inches")

Ref. SOI 04-1-01-E22-1, Schematic Diagram E-1183

Edited 11/13 - modified part c of question and answer
- added level 2 as an acceptable answer

QUESTION: 6 REC ID: 166 PT VAL: 1.00
LP: OP-LO-SYS-LP-E51 Objective: 5a2

SCENARIO 3

The Shift Supervisor directs you to manually initiate the RCIC system, and you erroneously depress the "RCIC MAN ISOL" pushbutton. State why RCIC failed to isolate.

Facility deleted but failed to note it here

ANSWER: 6

(The isolation pushbutton is only) active when an initiation signal is present. (1.0)

Ref. SOI 04-1-01-E51-1, Schematic Diagram E-1185

Edited 11/13 - modified question

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SCENARIO 3

QUESTION: 7 REC ID: 244 PT VAL: 1.50
LP: OP-LOR-EPS-LP-001 Objective: 1

SCENARIO 3

Based on current plant conditions, which plant conditions require entry into EP-3?

ANSWER: 7

- High Suppression Pool Level (> 18.81 feet) (0.5 pts)
- High Drywell Temp (> 135°F) (0.5 pts)
- High Drywell Pressure (> 1.23 psig) (0.5 pts)

Reference: 05-S-01-EP-3

Modified 11/13 - answer to not require setpoints

SETPOINTS REQUIRED IF "HIGH" NOT USED

**** END OF EXAM ****

9.50 POINTS 100% VALUE
6.650 POINTS 70% VALUE

7.600 POINTS 80% VALUE

Part A-2
Week 1

**EXAMINER GUIDE
GRAND GULF NUCLEAR STATION
STATIC SIMULATOR EXAM
SCENARIO 6**

GOOD KEY

I. SIMULATOR SET-UP

- A. Initialize the simulator to IC-17.
- B. Set all recorders to the correct time, rotating each chart a minimum of 3 inches and ensuring sufficient paper for the examination.
- C. Establish exam conditions as follows:

<u>Event</u>	<u>Time</u>	<u>Malf. No.</u>	<u>Description</u>
1	0	127	Feedwater master controller fails open
2	10 sec.	139, 17AC	Phase Overcurrent Trip of Bus 17AC
3	30 sec.	65,,,20	Steam line rupture, 20% severity

- D. Operator Actions
 - 1. Place Mode Switch to SHUTDOWN
 - 2. Once drywell pressure is >1.39 psig and reactor water level is >+54", place the simulator in FREEZE.

II. VALIDATION

- A. For the malfunctions listed, verify the following desired results are obtained:
 - 1. For the feedwater master controller failure (#127), level exceeds +54".
 - 2. For the loss of Bus 17AC (#139), there is no power to HPCS.
 - 3. For the steam leak in the drywell (#65), drywell pressure is >1.39 psig and containment pressure is <2.2 psig.

- B. For the examination, the following parameters must be in the allowed band:
1. Drywell pressure, >1.39 psig
 2. Containment pressure, <2.2psig
 3. Reactor pressure, >650 psig
 4. Reactor level, >+54" and <+90"
 5. Suppression pool level, >18.81'
 6. Suppression pool temperature, <95°F
- C. Check all recorder traces and parameters for reasonable values and trends for existing simulator conditions.
- D. If any parameter is outside the allowed band, the simulator must be re-initialized, beginning at Step I. A.
- E. If all indications are satisfactory, the exam may begin.

III. QUESTION AVAILABILITY

- A. The following questions were developed specifically for use with Scenario 6:

<u>REQ ID</u>	<u>LESSON PLAN</u>	<u>OBJECTIVES</u>	<u>TIME (min.)</u>
236	OP-LOR-SYS-LP-ECCS	10a, 14, 15	7
237	OP-LOR-SYS-LP-M71	9, 10	7
238	OP-LOR-EP-LP-001	2a	8
239	OP-LOR-EP-LP-001	1	6
240	OP-LOR-SYS-LP-DCP	H4	5
241	OP-LOR-IE-LP-006	A5	5
242	OP-LOR-EP-LP-001	2a	6
243	OP-LOR-MCD-LP-009	2a, 4a	5
576	OP-LOR-SYS-LP-R21-1	4, 5	9

- B. Select other Category A Questions, as required, to provide an examination length of 1 to 1½ hours.

SCENARIO 6

INITIAL CONDITIONS

Approximately 4 minutes ago, plant conditions were as follows:

- 100% steady state power
- 1275 MWE
- All controls in AUTO/NORMAL
- No active or potential LCOs existed

Then, a transient occurred. Thus far, the only operator action taken has been the placing of the reactor mode switch in the SHUTDOWN position.

Those questions identified with SCENARIO 6 at the top are related to these initial conditions. All others are NOT related to the scenario.

The SPDS is selected to the OVERVIEW screen and should NOT be changed at any time during this exam.

All questions related to the initial conditions of this scenario are labeled SCENARIO 6 at the top. Questions not labeled with this title are NOT related to the above initial conditions.

QUESTION: 1 REC ID: 237 PT VAL: 1.00
LP: OP-LOR-SYS-LP-M71 Objective: 9,10

SCENARIO 6

- a. Determine the operator actions required to allow re-opening of each of the following valves:
 - 1. P53-F026A, IA SPLY HDR TO AUX BLDG
 - 2. P53-F001, IA SPLY HDR TO CTMT
 - 3. P44-F121, PSW SPLY HDR TO AUX BLDG
- b. You are directed to close feeder breakers to MCCs 15B42 and 16B42. What specific plant parameter caused the feeder breakers to open? Setpoint required for full credit.

ANSWER: 1

- a.
 - 1. Take AUX BLDG ISO BYPASS DIV 1 keylock switch to BYPASS
 - 2. None (30 second timer already timed out)
(or also accept)
Take valve handswitch to open
 - 3. Take AUX BLDG ISO BYPASS DIV 1 keylock switch to BYPASS
- b. 1.39 psig high drywell pressure
(4 parts at 0.25 each)

Ref. ONEPs 05-1-02-III-5 and 05-1-02-V-16, SOI 04-1-01-R21-1

QUESTION: 2 REC ID: 238 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 2a

SCENARIO 6

Based on the present plant conditions, if reactor water level begins decreasing, which of the following systems can be used for level control?

- a. CRD
- b. HPCS
- c. Condensate
- d. LPCI

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SCENARIO 6-1

ANSWER: 2

a. CRD

Ref. EP-2

QUESTION: 3 REC ID: 243 PT VAL: 0.50
LP: OP-LOR-MCD-LP-009 Objective: 2a, 4a

SCENARIO 6

Based on present plant conditions, determine when a Drywell Purge Compressor automatic initiation would occur.

ANSWER: 3

30 seconds following LOCA signal

(Auto initiation will occur once) drywell pressure drops to within 0.87 psi (0.25) of containment pressure (0.25).

Ref. ~~ONEP 05-1-02-III-11~~, SOI 04-1-01-E61-1

*148 11-20-89
(ONEP deleted)*

QUESTION: 4 REC ID: 576 PT VAL: 1.00
LP: OP-LOR-SYS-LP-R21-1 Objective: 4, 5

SCENARIO 6

Based on present plant conditions, specify which of the following actions should have been automatically initiated:

- a. Align SSW to supply cooling to the drywell chillers.
- b. Start diesel generators with all protective trips active.
- c. Shed 15B42 feeder breaker.
- d. Isolate CCW to the RWCU non-regenerative heat exchangers.

ANSWER: 4

c. (1.0)

Ref. SOI 04-1-01-P75-1, SOI 04-1-01-R21-1, E-1039, E-1120

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
 SCENARIO 6-1

 QUESTION: 5 REC ID: 643 PT VAL: 1.00
 LP: OP-LOR-MCD-LP-009 Objective: 4

SCENARIO 6

If drywell pressure drops from present value to 0.5 psig, which logic system(s) must be reset in order to remove the automatic initiation signal to the Combustible Gas Control System?

ANSWER: 5

1. Div. I and II ECCS logic (0.5) (must be reset in order to reset LSS)
2. Div. I and II LSS panels (0.5) (must then be reset).

Ref. Schematic Diagram E-1186, SOI 04-1-01-R21-1

 QUESTION: 6 REC ID: 18 PT VAL: 1.00
 LP: OP-LOR-SYS-LP-ECCS Objective: 17a

With a LOCA initiation signal sealed in on Div. I ECCS and with the RHR pump A start signal manually overridden:

What conditions, if any, will cause an automatic start of the RHR "A" pump?

ANSWER: 6

(Div. I) Ctmt. Spray Initiation (1.0)
 also accept ISAA deenergizing and subsequently reenergized (E-1181-67)
 Ref. 04-1-01-E12-1

 QUESTION: 7 REC ID: 44 PT VAL: 1.00
 LP: OP-LOR-SYS-LP-ECCS Objective: 11,14

The RHR-A loop is running in Suppression Pool cooling mode when a LOCA signal is received.

- a. Describe the effect of the LOCA signal on the following valves:
 1. RHR-A Test Return Valve, E12-F024A
 2. RHR-A Heat Exchanger Bypass Valve, E12-F048A.
- b. Describe the operator actions required to restore RHR-A to the Suppression Pool Cooling Mode. Include time limits on actions (if applicable).

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SCENARIO 6-1

ANSWER: 7

- a. 1. E12-F024A (automatically) closes. (0.25)
- 2. E12-F048A is (interlocked) opened (for 10.85 minutes). (0.25)
- b. 1. E12-F042A can be manually overridden closed (0.17),
- 2. E12-F024A can be manually overridden open (0.17), and
- 3. E12-F048A can be closed (0.08) after 10.85 minutes (0.08).

Ref. SOI 04-1-01-E12-1, Schematic Diagram E-1181

QUESTION: 8 REC ID: 241 PT VAL: 1.00
LP: OP-LOR-IE-LP-006 Objective: A5

Assuming the initial steady state reactor pressure was 1000 psia, you are ordered to reduce reactor pressure to the minimum allowable.

- a. To what value can reactor pressure be lowered and still maintain the administrative cooldown limit?
- b. To what value can reactor pressure be lowered and still maintain the Tech. Spec. cooldown limit?

ANSWER: 8

- a. ⁴⁹⁰~~470~~ ± 10 psia (0.5)
- b. 400 ± 10 psia (0.5)

Ref. EP-2, IOIs 03-1-01-3, 03-1-01-4

**** END OF EXAM ****

7.50 POINTS 100% VALUE
5.250 POINTS 70% VALUE

6.000 POINTS 80% VALUE

***** ANNUAL REQUALIFICATION EXAM ANSWER KEY *****
RO-2-1

QUESTION: 1 REC ID: 52 PT VAL: 1.00
LP: OP-LOR-SYS-LP-DCP Objective: G37

During rated power operations, Recirc. Pump "A" trips and is not recoverable. The A Recirc. Pump discharge valve is closed and flow in the B recirc. loop is verified to be 42,000 gpm. After taking these actions, you observe the following plant parameters:

Total core flow = 62 Mlbm/hr
Indicated thermal power = 64% of rated

Determine which, if any, of the following actions are required based on present plant conditions:

- a. Reduce "B" loop flow to less than 50% of rated loop flow.
- b. Reduce reactor power by driving control rods to less than approximately 44% within two hours.
- c. Reopen Recirc. Pump A discharge valve, after 5 minutes.
- d. Reduce total core flow to less than 39 Mlbm/hr.

ANSWER: 1

c. (1.0)

Ref: ONEP 05-1-02-III-3

QUESTION: 2 REC ID: 250 PT VAL: 0.50
LP: OP-LOR-LL-LP-001 Objective: 10

The Shift Superintendent asks you to ensure that RHR Shutdown Cooling Isolation Valve E12-F009-B does not isolate upon re-energization of bus 16 AB. How can the isolation of E12-F009-B be prevented?

(NOTE: The isolation logic is initiated.)

ANSWER: 2

By opening the MCC breaker for the MOV (0.5).

Ref. 01-S-07-1

QUESTION: 3 REC ID: 527 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 1,2a,2d

Several hours following a LOCA in which the core was calculated to have uncovered, indicated plant conditions are as follows:

Reactor Power	0%
Reactor Pressure	20 psig
Reactor Water Level	-18"
Containment Temperature	225°F
Containment Pressure	8 psig
Containment Hydrogen Concentration	7.0%

With the offsite radioactive release rate expected to remain below the LCO limit, evaluate present plant conditions and determine which statement must be performed:

- a. Perform containment vent and purge.
- b. Isolate containment vent and purge.
- c. Secure the DW purge compressors.
- d. Secure and prevent operation of the hydrogen igniters.

ANSWER: 3

a. (Perform CTMT vent and purge) (1.00)

Ref. EP-3

QUESTION: 4 REC ID: 503 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 1,2d,2h

Shortly after a reactor scram, the following plant conditions exist:

Reactor Power	0%
Reactor Pressure	500 psig
Reactor Water Level	+180"
Average Suppression Pool Water Temperature	140°F
Suppression Pool Water Level	18'5"

Based on these conditions, the suppression pool Heat Capacity Level Limit (HCLL) is:

- a. 14.5'
- b. 16'
- c. 17'
- d. 18.5'

ANSWER: 4

c. (17') (1.0)

Ref. EP-3, Figures 2 and 7

QUESTION: 5 REC ID: 474 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 2a,2d

The following plant conditions exist after a LOCA:

- Wide range reactor water level = +10" and slowly increasing due to MPCs and RCIC injection from the CST.
- Suppression pool level = 29'
- Suppression pool temperature = 120°F
- Reactor pressure = 400 psig

Based on the given conditions, determine the highest acceptable reactor pressure.

ANSWER: 5

290 ± 10 psig (1.00)

Ref. EP-2 and EP-3, Figures 2 and 3

QUESTION: 6 REC ID: 153 PT VAL: 1.50
LP: OP-LOR-AD-LP-005 Objective: 5

Answer the following concerning the Protective Tagging System:

- a. If red tags are to be removed in any particular order, where is the tagging removal order indicated?
- b. How are Safety Related valves in Non-Safety Related systems determined?
- c. When isolating rotating machinery, such as a pump motor, in addition to tagging out the pump motor, what other source of energy should be isolated and danger tagged?

ANSWER: 6

- a. In the Special Instruction/Remarks block. (0.5)
- b. By the use of the "Q-Valve" list. (also accept) Valve QTS (0.5)
- c. Pump suction and discharge valves (to prevent possible rotation due to fluid flow). (0.5)

Ref. 01-S-06-1

QUESTION: 7 REC ID: 235 PT VAL: 0.50
LP: OP-LOR-PR-LP-001 Objective: 1d

During a routine shift, while at 100% power, an HCU accumulator trouble alarm is received for Control Rod 32-33. The Auxiliary Building operator is summoned to investigate. While he is on that elevation of Containment, he must report that the accumulator has low pressure.

- a. What mode of communication is he prohibited from using due to his present location?
- b. What is the highest priority means of communication than he can use?

ANSWER: 7

- a. Radio
- b. C. W. Thompson Paging System (or P.A.) (0.25 each)

*agreed with comment
new p*

Ref. 01-S-06-14

QUESTION: 8 REC ID: 57 PT VAL: 1.00
LP: OP-LOR-AD-LP-005 Objective: 9,10

You are in Operational Condition 4, Cold Shutdown. With five of the six SRMs inoperable, answer each of the following:

- a. In what position is the Reactor Mode Switch required to be?
- b. Where must the Reactor Mode Switch KEY be for this condition?
- c. What type tag needs to be placed on the Reactor Mode Switch?
- d. What information should be included on this tag?

ANSWER: 8

- a. Locked in the shutdown position (0.25)
- b. In the custody of the Shift Supv or Supt. (0.25)
- c. Information tag (0.2)
- d. LCO number (0.1), date (0.1), required position (0.1)

Ref: 01-S-06-2

QUESTION: 9 REC ID: 62 PT VAL: 1.00
LP: OP-LOR-AD-LP-005 Objective: 6

In addition to breaker position, what four (4) items should an operator check on a breaker, if applicable, during the performance of a system lineup cheksheet when the breaker is required to be RACKED IN. Consider local checks only, and a 4.16 KV I.T.E. circuit breaker as an example.

ANSWER: 9

1. Control Power fuses installed.
2. Spring charging motor switch to ON.
3. Closing springs charged.
4. Door bolted and closed. (0.25 each)

Ref: 02-S-01-2, Att. III (III.A)

QUESTION: 10 REC ID: 282 PT VAL: 1.00
LP: OP-LOR-DT-LP-003 Objective: 17d

A loss of off-site power has occurred. Busses 15AA and 16AB are still deenergized due to failures of DG 11 and 12. DG 13 is running and tied to Bus 17AC. RCIC has been initiated and is currently maintaining RPV level at approximately +20". SRVs are being used to control RPV pressure. The Shift Superintendent has determined that, in order to maintain the plant in a safe shutdown condition, Bus 16AB must be connected to the Div III diesel generator.

- a. Upon energizing Bus 16AB from Div. III diesel generator, the Shift Supervisor directs the restarting of drywell coolers, instrument air compressors and a loop of suppression pool cooling. How do you determine if all these loads can be started and allowed to run without exceeding the Div. III diesel generator steady state rating? Include the procedure section referenced for your decision.
- b. When the CRO places suppression pool cooling in service, state why the SSW pump should be started first.

ANSWER: 10

(0.5)

- a. (All are listed essential loads) in Table 1 of ONEP 05-1-02-I-4 (0.25), ~~the total of which does not exceed the steady state rating of Div. III diesel generator (0.25)~~ (Loads included on list gaurentees not exceeding rating of Div III D/G) 10/27/89
- b. to prevent simultaneous starting (0.25) of the RHR and SSW pumps (0.25) [(also accept) minimize starting current surge on the D/G (0.5)] 10/27/89

Ref. ONEP 05-1-02-I-4

a. Comment not agreed with grade per answer key. MWP

therefore this statement should not be required for full credit) 10/27/89

QUESTION: 11 REC ID: 317 PT VAL: 1.50
LP: OP-LOR-MCD-LP-009 Objective: A15g,h,j

With the plant initially operating at 100% power, a LOCA occurs. Just prior to the LOCA, the roving NOA had recorded the following containment parameters in the Daily Ops Log:

Containment pressure	0.01 psig
Containment average temperature	85°F

Several days following the LOCA, containment pressure is 0.0 psig with hydrogen concentration at 2.8%, the Shift Superintendent directs the starting of the "B" hydrogen recombiner. Based on the information provided,

- a. Determine the required recombiner setting.
- b. How long will it take to reach the final power setting?
- c. If hydrogen concentration subsequently increases to 3.4% over the next 24 hour period, how much, if any, should the power setting be changed?

ANSWER: 11

- a. ^{22.9}~~20.9~~ + 2 kW ^{1.16}~~1.11~~ x 45.6 kW (0.5) (Reference 04-1-01-E61-1 Figure 1. This Figure revised between writing of question and test. Revised 9-21-89)
 - b. 25 minutes (0.5)
 - c. Increase by 4 kW (0.5)
- Ref. 04-1-01-E61-1
- a. agree with comment. grade as revised mwp*

QUESTION: 12 REC ID: 380 PT VAL: 0.50
LP: OP-LO-SYS-LP-C41 Objective: 9b

As directed by Emergency Procedures during a failure to SCRAM, the Standby Liquid Control (SLC) System is initiated and both pumps are verified to be injecting into the RPV. The CRO, while monitoring injection of SLC, states that the tank level has decreased to 1000 gallons. What actions are required? Explain, including the applicable reference(s).

ANSWER: 12

(As specified in) EP-2A, (Step 21) (0.25), only one pump should be tripped (0.25).

Ref. EP-2A, Step 21

QUESTION: 13 REC ID: 378 PT VAL: 0.75
LP: OP-LOR-SYS-LP-N31 Objective: A16

The reactor has been operating at rated conditions for several weeks when annunciator TURB LUBE OIL TEMP HI (P680, 10A-D3) alarms. The oil temperature, as indicated at P680, is 122°F. The TURB LUBE OIL TEMP CONT (1N32-TK-R602) is in AUTO and set to control temperature at 113°F. The TBCW control valve signal deviation (vertical scale) is near zero. Which of the following is the most likely cause of the alarm?

- a. Turbine lube oil temperature controller 1N34-TK-R602 requires adjustment or has malfunctioned.
- b. Improper valve lineup of the Lube Oil System and/or TBCW coolers.
- c. Air in the turbine lube oil coolers (lube oil and/or TBCW side).
- d. Turbine lube oil cooler fouling.

ANSWER: 13

- a. Turbine lube oil temperature controller 1N34-TK-R602 requires adjustment or has malfunctioned. (0.75)

Ref: ARI 04-1-02-1H13-P680-10A-D3

QUESTION: 14 REC ID: 138 PT VAL: 1.00
LP: OP-LOR-SYS-LP-DC Objective: 5b

While checking for grounds on Bus 11DD, you observe the following:

- Initially, V1, V2, V3 and V4 all indicate 62 volts
- When PB-1 is depressed, V1 indicates 0 volts, V2 indicates 125 volts
- When PB-2 is depressed, V3 indicates downscale, V4 indicates 250 volts

You interpret these readings as (Choose one of the following):

- a. No grounds
- b. "E" bus dead ground on negative bus
- c. "D" bus dead ground on negative bus
- d. "D" bus dead ground on positive bus

ANSWER: 14

d (1.0)

Ref. SOI 04-1-01-L11-1

QUESTION: 15 REC ID: 194 PT VAL: 0.50
LP: OP-LOR-RSD-LP-001 Objective: 7b2

Having evacuated the Main Control Room due to a fire and upon arriving at the Remote Shutdown Panel, if indicated Reactor water level is +40 inches and Reactor pressure is 450 psig, what is the actual water level?

ANSWER: 15

+22.5 ± 1.0 inch (0.5)

Ref. ONEP 05-1-02-II-1

QUESTION: 16 REC ID: 252 PT VAL: 0.50
LP: OP-LOR-SYS-LP-F11/17 Objective: A5a

The cable on the fuel handling platform main hoist has gone completely slack due to the slack cable interlock not stopping the downward motion of the hoist. What action is required prior to attempting to raise the hoist?

ANSWER: 16

The main hoist's cable drum must be checked for proper lay (0.5).

Ref. SOI 04-1-01-F11-3

QUESTION: 17 REC ID: 271 PT VAL: 1.00
LP: OP-LO-SYS-LP-E12-1 Objective: 14

Prior to opening E12-F066A when aligning ADHR for reactor to reactor cooling, the RHR A ADHR MODE TRIP ENABLE SW (on panel 1H13-P629) is placed in the ADHR position. Why is this necessary?

ANSWER: 17

To prevent RHR pump A from operating without a suction path (with F066A open only). (1.0)

Ref. 04-1-01-E12-1

QUESTION: 18 REC ID: 356 PT VAL: 0.75
LP: OP-LO-SYS-LP-M51/P72 Objective: 10b

The reactor is in Mode 4 for a two-week maintenance outage. An operator has been dispatched to perform a local shutdown of Drywell Chiller A. Unfortunately, he shuts down Drywell Chiller B instead. After verifying power lights ON, having Drywell Chiller B compressors reset locally, and verifying all other lights OFF, the CRO attempts to restart the chiller by taking the control room handswitch (P72-M600B) from NORM to START. Drywell Chiller B failed to start. Explain why, providing appropriate procedural references.

ANSWER: 18

(In accordance with) 04-1-01-P72, Section 4.1.2.e NOTE (0.25), (since the chiller was shut down locally), the control room switch must be cycled from NORM to STOP to START (0.25) in order to reset system logic and allow startup of the chiller (0.25).

also accept 4.1.2.d as note 1.5 between d & e (Ref 01-5-02- does not require note to be placed before step that is applicable)

comment not agreed with. mwp

QUESTION: 19 REC ID: 562 PT VAL: 1.00
LP: OP-LOR-IE-LP-006 Objective: A.3

- a. With the plant shut down in preparation for a refueling outage, if a loss of forced core circulation occurs, to what level must reactor water be raised?
- b. Four days following a scheduled plant shutdown, a total loss of RHR Shutdown Cooling and ADHRS capability occurs. RWCU reject is increased to maximum and condensate and CRD are being used for makeup. With RPV water level initially at +32" and RPV water temperature at 130°F, determine the estimated time till boiling occurs.

ANSWER: 19

- a. 615" from Vessel zero or 82" indicated
- b. 1.3 hours ± 0.1 hours (0.5 pts. each)

Ref. ONEP 05-1-02-III-1

**** END OF EXAM ****

17.00 POINTS 100% VALUE
11.900 POINTS 70% VALUE

13.600 POINTS 80% VALUE

***** ANNUAL REQUALIFICATION EXAM ANSWER KEY *****
SRO-2-1

QUESTION: 1 REC ID: 490 PT VAL: 1.00
LP: EPTS-3 Objective: A17

The reactor was operating at rated conditions when events occurred which resulted in the offsite release of radionuclides. Evaluate the following information and determine the Whole Body Dose Commitment for a two-hour exposure at the site boundary:

delta T = -1.15 Wind Speed = 15 mph
Time After Shutdown = 3 hours Q(NG) = 10 Ci/sec

- a. 91 mrem
- b. 104 mrem
- c. 115 mrem
- d. 131 mrem

ANSWER: 1

- d. 131 mrem (1.0)

The following is provided for information purposes only:

delta T = -1.15 » Stability Class C (from Dose Calc. W/S)
 [F] = 15.28 (from C Stability; SB,14)
 X/Q = [F] x E-6
 NGCF = 430 (from Dose Calc. W/S)

Whole Body Dose Commitment =

$$X/Q \times Q(NG) \times NGCF \times 1000 \text{ mrem/rem} \times \text{Exposure Time} =$$

$$15.28E-6 \times 10 \text{ Ci/sec} \times 430 \times 1000 \times 2 \text{ hours} = 131 \text{ mrem}$$

Ref: EPP 10-S-01-12, Section 6.3

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SRO-2-1

QUESTION: 2 REC ID: 397 PT VAL: ~~1.00~~ 0.75 WJB 11-22-89
LP: EPTS-6 Objective: A1

Evaluate the following conditions and determine the appropriate Emergency Classification, if any. (Note: Each answer may be used more than once.)

~~WJB 11-22-89 a. All rods have been verified fully inserted following a LOCA. There is NO available method of decay heat removal. Containment pressure is 18 psig.~~

- b. The primary coolant system leak rate is 27 gpm (4 gpm unidentified and 23 gpm identified) with no indication of pressure boundary leakage.
- c. Health Physics reports that the air dose due to noble gases in gaseous effluents, for the calendar quarter just ended, is 6 mrad for gamma and 8 mrad for beta.
- d. Unknown, armed personnel appear in the Control Room and state that they are now in control of operations.

- | | |
|------------------------|----------------------|
| 1. UNUSUAL EVENT | 4. GENERAL EMERGENCY |
| 2. ALERT | 5. NONE |
| 3. SITE AREA EMERGENCY | |

ANSWER: 2

- ~~a. 4 (General Emergency) (0.25)~~ WJB 11-22-89 a. 2 Alert (.25) mwp
- b. 5 (None) (0.25)
- c. 1 (Unusual Event) (0.25)
- d. 4 (General Emergency) (0.25)

Ref: EPP 10-S-01-1, Att. I

Question 2 IP 397 Delete part a. Insufficient information given in part a to classify event as General Emergency. In order to classify as General Emergency core damage must have occurred or is predicted to occur within two hours as reported by reactor engineering (10-S-01-1 Attachment 1 page 17 of 27) This information not given. Change point value to .75

WJB 11-22-89

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SRO-2-1

QUESTION: 3 REC ID: 52 PT VAL: 1.00
LP: OP-LOR-SYS-LP-DCP Objective: G37

During rated power operations, Recirc. Pump "A" trips and is not recoverable. The A Recirc. Pump discharge valve is closed and flow in the B recirc. loop is verified to be 42,000 gpm. After taking these actions, you observe the following plant parameters:

Total core flow = 62 Mlbm/hr
Indicated thermal power = 64% of rated

Determine which, if any, of the following actions are required based on present plant conditions:

- a. Reduce "B" loop flow to less than 50% of rated loop flow.
- b. Reduce reactor power by driving control rods to less than approximately 44% within two hours.
- c. Reopen Recirc. Pump A discharge valve, after 5 minutes.
- d. Reduce total core flow to less than 39 Mlbm/hr.

ANSWER: 3

c. (1.0)

Ref: ONEP 05-1-02-III-3

QUESTION: 4 REC ID: 336 PT VAL: 0.75
LP: OP-LOR-IE-LP-003 Objective: A3e

A loss of feedwater heating occurs due to a level control problem in the 6A High Pressure Feedwater Heater. Feedwater flow through the high pressure feedwater heater is not isolated. Following a power reduction to 70%, the feedwater temperature is determined to be 325°F. The results of the next POWERPLEX MON indicate the following:

CMAPRAT	=	0.957
CMFLCPR	=	0.995
CFDLRX	=	0.975
CFDLRC	=	0.375

Evaluate present plant conditions and determine what additional operator actions, if any, are required.

ANSWER: 4

Reduce reactor power until back in Region I or II (0.5) and perform 06-RE-1J11-V-0001 (0.25).

Ref. ONEP 05-1-02-V-5

QUESTION: 5 REC ID: 418 PT VAL: 1.50
LP: OP-LOR-IE-LP-005 Objective: B.1, B.3

While operating at rated conditions, a reactor coolant chemistry sample indicates an activity level of 0.18 $\mu\text{Ci}/\text{gram}$ Dose Equivalent I-131 and rapidly increasing.

- a. Once the activity increases to greater than 0.2 $\mu\text{Ci}/\text{gram}$, how long may power operations continue before Tech. Spec. actions are required?
- b. If, upon performing a sample 4 hours later, the activity level has increased to 4.1 $\mu\text{Ci}/\text{gram}$, what actions are required?
- c. Based on the latest coolant sample results, CLASSIFY the event, if required.

ANSWER: 5

- a. For up to 48 hours (0.5).
- b. Be in at least Hot S/D with the MSIVs closed (0.25) within 12 hours (0.25)
- c. Unusual Event (0.5).

Ref. T.S. 3.4.5, EPP 10-S-01-1, Att. 1, EAL 4.1.4

QUESTION: 5 REC ID: 527 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 1,2a,2d

Several hours following a LOCA in which the core was calculated to have uncovered, indicated plant conditions are as follows:

Reactor Power	0%
Reactor Pressure	20 psig
Reactor Water Level	-18"
Containment Temperature	225°F
Containment Pressure	8 psig
Containment Hydrogen Concentration	7.0%
Drywell Hydrogen Concentration	8.0%

With the offsite radioactive release rate expected to remain below the LCO limit, evaluate present plant conditions and determine which statement must be performed:

- a. Perform containment vent and purge.
- b. Isolate containment vent and purge.
- c. Secure the DW purge compressors.
- d. Secure and prevent operation of the hydrogen igniters.

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SRO-2-1

ANSWER: 6

a. (Perform CTMT vent and purge) (1.00)

Ref. EP-3

QUESTION: 7 REC ID: 503 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 1,2d,2h

Shortly after a reactor scram, the following plant conditions exist:

Reactor Power	0%
Reactor Pressure	500 psig
Reactor Water Level	+180"
Average Suppression Pool Water Temperature	140°F
Suppression Pool Water Level	18'5"

Based on these conditions, the suppression pool Heat Capacity Level Limit (HCLL) is:

- a. 14.5'
- b. 16'
- c. 17'
- d. 18.5'

ANSWER: 7

c. (17') (1.0)

Ref. EP-3, Figures 2 and 7

QUESTION: 8 REC ID: 474 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 2a,2d

The following plant conditions exist after a LOCA:

- Wide range reactor water level = +10" and slowly increasing due to HPCS and RCIC injection from the CST.
- Suppression pool level = 29'
- Suppression pool temperature = 120°F
- Reactor pressure = 400 psig

Based on the given conditions, determine the highest acceptable reactor pressure.

ANSWER: 8

290 ± 10 psig (1.00)

Ref. EP-2 and EP-3, Figures 2 and 3

QUESTION: 9 REC ID: 153 PT VAL: 1.50
LP: OP-LOR-AD-LP-005 Objective: 5

Answer the following concerning the Protective Tagging System:

- a. If red tags are to be removed in any particular order, where is the tagging removal order indicated?
- b. How are Safety Related valves in Non-Safety Related systems determined?
- c. When isolating rotating machinery, such as a pump motor, in addition to tagging out the pump motor, what other source of energy should be isolated and danger tagged?

ANSWER: 9

- a. In the Special Instruction/Remarks block. (0.5)
- b. By the use of the "Q-Valve" list. (also accept) Valve QTS (0.5)
- c. Pump suction and discharge valves (to prevent possible rotation due to fluid flow). (0.5)

Ref. 01-S-06-1

QUESTION: 10 REC ID: 57 PT VAL: 1.00
LP: OP-LOR-AD-LP-005 Objective: 9,10

You are in Operational Condition 4, Cold Shutdown. With five of the six SRMs inoperable, answer each of the following:

- a. In what position is the Reactor Mode Switch required to be?
- b. Where must the Reactor Mode Switch KEY be for this condition?
- c. What type tag needs to be placed on the Reactor Mode Switch?
- d. What information should be included on this tag?

ANSWER: 10

- a. Locked in the shutdown position (0.25)
- b. In the custody of the Shift Supv or Supt. (0.25)
- c. Information tag (0.2)
- d. LCO number (0.1), date (0.1), required position (0.1)

Ref: 01-S-06-2

QUESTION: 11 REC ID: 62 PT VAL: 1.00
LP: OP-LOR-AD-LP-005 Objective: 6

In addition to breaker position, what four (4) items should an operator check on a breaker, if applicable, during the performance of a system lineup cheeeksheet when the breaker is required to be RACKED IN. Consider local checks only, and a 4.16 KV I.T.E. circuit breaker as an example.

ANSWER: 11

1. Control Power fuses installed.
2. Spring charging motor switch to ON.
3. Closing springs charged.
4. Door bolted and closed. (0.25 each)

Ref: 02-S-01-2, Att. III (III.A)

QUESTION: 12 REC ID: 282 PT VAL: 1.00
LP: OP-LOR-DT-LP-003 Objective: 17d

A loss of off-site power has occurred. Busses 15AA and 16AB are still deenergized due to failures of DG 11 and 12. DG 13 is running and tied to Bus 17AC. RCIC has been initiated and is currently maintaining RPV level at approximately +20". SRVs are being used to control RPV pressure. The Shift Superintendent has determined that, in order to maintain the plant in a safe shutdown condition, Bus 16AB must be connected to the Div III diesel generator.

- a. Upon energizing Bus 16AB from Div. III diesel generator, the Shift Supervisor directs the restarting of drywell coolers, instrument air compressors and a loop of suppression pool cooling. How do you determine if all these loads can be started and allowed to run without exceeding the Div. III diesel generator steady state rating? Include the procedure section referenced for your decision.
- b. When the CRO places suppression pool cooling in service, state why the SSW pump should be started first.

ANSWER: 12

comment not agreed with mwp.

- a. (All are listed essential loads) in Table 1 of ONEP
05-1-02-I-4 (0.25), ~~the total of which does not exceed the steady state rating of Div. III diesel generator (0.25)~~ *148 11-22-87 (If loads listed in Table 1 of ONEP this guarantees max load on D/G no exceed, this info not reported for full credit 148 11-22-87)*
- b. to prevent simultaneous starting (0.25) of the RHR and SSW pumps (0.25) [(also accept) minimize starting current surge on the D/G (0.5)]

Ref. ONEP 05-1-02-I-4

 QUESTION: 13 REC ID: 246 PT VAL: 1.00
 LP: OP-LOR-RF-LP-001 Objective: B6

During refueling operations, the refueling bridge's travel interlock has malfunctioned and is allowing the refueling bridge to enter the horizontal fuel transfer area with the carrier in the horizontal position. What action must you take concerning the horizontal fuel transfer system?

ANSWER: 13

Suspend HFTS operation (0.5) with the HFTS at either the Spent Fuel Building pool (0.25) or the Reactor Building pool terminal point (0.25).

Ref. Tech. Spec. 3.9.12

 QUESTION: 14 REC ID: 317 PT VAL: 1.50
 LP: OP-LOR-MCD-LP-009 Objective: A15g,h,j

With the plant initially operating at 100% power, a LOCA occurs. Just prior to the LOCA, the roving NOA had recorded the following containment parameters in the Daily Ops Log:

Containment pressure	0.01 psig
Containment average temperature	85°F

Several days following the LOCA, containment pressure is 0.0 psig with hydrogen concentration at 2.8%, the Shift Superintendent directs the starting of the "B" hydrogen recombiner. Based on the information provided,

- a. Determine the required recombiner setting.
- b. How long will it take to reach the final power setting?
- c. If hydrogen concentration subsequently increases to 3.4% over the next 24 hour period, how much, if any, should the power setting be changed?

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SRO-2-1

ANSWER: 14

comment agreed with inf

- a. ~~50.0~~ ^{52.9} + 2 kW (~~2.7~~ ^{1.16} x 45.6 kW) (0.5) Reference 04-1-01-E61-1 Figure 1
~~NO 11-22-89~~ ~~NO 11-22-89~~
 - b. 25 minutes (0.5)
 - c. Increase by 4 kW (0.5)
- Ref. 04-1-01-E61-1

QUESTION: 15 REC ID: 371 PT VAL: 1.00
LP: OP-LO-SYS-LP-C41 Objective: 10

During a plant startup with the plant currently at 5% reactor power and following the addition of water to the Standby Liquid Control System Storage Tank, the on-shift Chemistry Technician reports the SLC System sampling results as follows:

Available volume of solution = 4875 gallons
% by weight solution concentration = 13%

Based on these results, the plant can NOT enter into Mode 1. Explain why.

ANSWER: 15

T.S. 3.0.4 applies (1.00)
(also accept)
Can't change operational modes while relying on an action statement of a LCO.

Ref. T.S. 3.0.4, 3.1.5; 01-S-06-5, Attachment 3 & 5

QUESTION: 16 REC ID: 47 PT VAL: 1.00
LP: OP-LOR-SYS-LP-DC Objective: 4

The plant is operating at 100% power with battery charger 1C4 INOPERABLE due to electrical failure. All other systems are OPERABLE. The electricians perform the required Tech. Spec. surveillances and inform the Shift Superintendent that the terminal voltage for battery 1C3 is 128 VDC. Determine the required Tech. Spec. actions, if any.

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SRO-2-1

ANSWER: 16

Declare battery 1C3 INOP (0.5) and

Declare HPCS system INOP (0.25), and

take the actions required by Tech Spec 3.5.1. (0.25)

(also accept the following for Tech. Spec. 3.5.1 actions)

Restore HPCS to OPERABLE status within 14 d:ys. Otherwise, be in HOT SHUTDOWN within the next 12 hours and in CO D SHUTDOWN within the following 24 hours.

Ref: T/S 3.8.2.1.c Action b.

QUESTION: 17 REC ID: 41 PT VAL: 1.00
LP: OP-LOR-SYS-LP-M71 Objective: 6

During a reactor plant heatup, MSIVs were stroked at 800 psig in accordance with a plant surveillance procedure. Both B21-F022A and B21-F028A are closed. The recorded closure times were:
B21-F022A = 5.0 seconds, B21-F028A = 2.8 seconds.

Evaluate the above information and determine which of the following actions most correctly complies with Tech Spec requirements:

- a. Immediately enter Tech. Spec. 3.0.3.
- b. Restore B21-F028A closure time to within 3 to 5 seconds within 8 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.
- c. Reduce Reactor pressure to less than 600 psig. Correct problems before continuing Heatup. There are no time limits below 600 psig.
- d. Within 1 hour initiate measures to place the unit in Hot Shutdown within the next 6 hours.

ANSWER: 17

b (1.0)

Ref: T/S 3.4.7

***** ANNUAL REQUALIFICATION ANSWER KEY CONTINUED *****
SRO-2-1

QUESTION: 18 REC ID: 138 PT VAL: 1.00
LP: OP-LOR-SYS-LP-DC Objective: 5b

While checking for grounds on Bus 11DD, you observe the following:

- Initially, V1, V2, V3 and V4 all indicate 62 volts
- When PB-1 is depressed, V1 indicates 0 volts, V2 indicates 125 volts
- When PB-2 is depressed, V3 indicates downscale, V4 indicates 250 volts

You interpret these readings as (Choose one of the following):

- a. No grounds
- b. "E" bus dead ground on negative bus
- c. "D" bus dead ground on negative bus
- d. "D" bus dead ground on positive bus

ANSWER: 18

d (1.0)

Ref. SOI J4-1-01-L11-1

QUESTION: 19 REC ID: 194 PT VAL: 0.50
LP: OP-LOR-RSD-LP-001 Objective: 7b2

Having evacuated the Main Control Room due to a fire and upon arriving at the Remote Shutdown Panel, if indicated Reactor water level is +40 inches and Reactor pressure is 450 psig, what is the actual water level?

ANSWER: 19

+22.5 ± 1.0 inch (0.5)

Ref. ONEP 05-1-02-II-1

**** END OF EXAM ****

19.75 POINTS 100% VALUE
13.825 POINTS 70% VALUE

15.800 POINTS 80% VALUE

Part A-1
Week 2

GODD
KEY

EXAMINER GUIDE
GRAND GULF NUCLEAR STATION
STATIC SIMULATOR EXAM
SCENARIO 9

0934

I. SIMULATOR SET-UP

- A. Initialize the simulator to IC-17.
- B. Set all recorders to the correct time, rotating each chart a minimum of 3 inches and ensuring sufficient paper for the examination.
- C. Establish exam conditions by inserting the following malfunctions simultaneously, using Malfunction Tableau Inactive:

<u>Event</u>	<u>Time</u>	<u>Malf. No.</u>	<u>Description</u>
1	0	63,,,100	Recirc Line Rupture (100% severity)
2	0	133U2	Service Transformer 21 Lockout
3	0	52	HPCS Pump Trip
4	0	40F	Drywell Pressure Transmitter Failure
5	0	141A	Div. I D/G Trip

D. Operator Actions

- 1. Take NO operator actions!
- 2. Allow Div. II low pressure ECCS to restore level to approximately -100", then place the simulator in FREEZE.
- 3. Acknowledge annunciators on all panels, but DO NOT reset.

II. VALIDATION

- A. For the malfunctions listed, verify the following desired results are obtained:
1. For the Recirc. Line Rupture (#63), verify that level has dropped below TAF on the Fuel Zone/Wide Range Recorders.
 2. For the Drywell Pressure Transmitter Failure (40F), verify that Drywell Pressure F light is not lit on the ADS status lights.
 3. ADS 105 second timers timed out.
- B. For the examination, the following parameters must be in the allowed band:
1. Drywell temperature at Elev. 166', >195°F
 2. Reactor water level, <-41.6"
- C. Check all recorder traces and parameters for reasonable values and trends for existing simulator conditions.
- D. If any parameter is outside the allowed band, the simulator must be re-initialized, beginning at Step I. A.
- E. If all indications are satisfactory, the exam may begin.

III. QUESTION AVAILABILITY

- A. The following questions were developed specifically for use with Scenario 9:

<u>REC ID</u>	<u>LESSON PLAN</u>	<u>OBJECTIVES</u>	<u>TIME (min.)</u>
577	OP-LOR-AC-LP-002	4, 5	6
578	OP-LOR-EP-LP-001	2a	8
579	OP-LOR-EP-LP-001	2a	8
580	OP-LOR-EP-LP-001	2a	7
582	OP-LOR-SYS-LP-DCP	H8	5
583	OP-LOR-SYS-LP-ECCS	20, 21	8
584	OP-LOR-SYS-LP-M71	2, 3	7
585	OP-LOR-SYS-LP-R21-1	3, 5	7
586	OP-LOR-SYS-LP-R21-1	3, 6	7

- B. Select questions for Scenario 9 from the above list and/or other non-scenario related Category A questions, as required, to provide an examination length of approximately forty-five minutes.

SCENARIO 9

INITIAL CONDITIONS

Approximately 5 minutes ago, plant conditions were as follows:

- 100% steady state power
- 1275 MWE
- All controls in auto/normal
- No active or potential LCOs existed

An I&C technician, in preparing to perform Surveillance Procedure 06-IC-1B21-M-2008, Drywell High Pressure Functional Test, had just reported that pressure transmitter B21-N094B was found isolated. A transient has occurred. NO operator actions have been taken!

The Shift Superintendent is currently reviewing the EPPs for event classification.

The SPDS is selected to the OVERVIEW screen and should NOT be changed at any time during this exam.

QUESTION: 1 REC ID: 578 PT VAL: 1.25
LP: OP-LOR-EP-LP-001 Objective: 2a

SCENARIO 9

- a. The Shift Supervisor has just ordered the hydrogen ignitors to be started. What parameters (including their REQUIRED values) were used to determine the need to operate the hydrogen ignitors?
- b. Determine the maximum hydrogen concentration allowable in containment to remain in the SAFE zone of the HDOL curve.

ANSWER: 1

- a. Containment hydrogen concentration is in the SAFE zone of the HDOL curve (0.25), drywell hydrogen concentration is less than 9% (0.25) and RPV water level has dropped below -167" (0.25).
- b. 8.6% ± 0.1% (0.5).

Ref. EP-3

QUESTION: 2 REC ID: 579 PT VAL: 1.50
LP: OP-LOR-EP-LP-001 Objective: 2a

SCENARIO 9

Concerning vessel level indication:

- a. Assume the level instruments read:

Wide Range Level = +50"
Narrow Range Level = +30"
Shutdown Range Level = +120"
Upset Range Level = +140"
Fuel Zone Level = > -117"

Based on the above level readings and current plant conditions, which level instruments can NOT be used to determine level?

- b. Which level instrument is closest to its calibration conditions?

ANSWER: 2

- a. Shutdown range (0.5) and Upset range (0.5)
- b. Fuel Zone range (0.5)

Ref. EP-2, Caution 1; P&ID M-1077C

QUESTION: 3 REC ID: 580 PT VAL: 1.50
LP: OP-LOR-EP-LP-001 Objective: 2a

SCENARIO 9

Concerning containment pressure control,

- a. What effect does pressure transmitter B21-N094F being valved out have on the Containment Spray Mode of RHR?
- b. At what pressure in the containment is containment spray required to be manually initiated? Assume adequate core cooling exists.
- c. At what pressure in the containment should containment spray be secured?

ANSWER: 3 MUB 12/4/89

- ~~a. None (0.5)~~ Deleted Initial conditions specified N094B was valved out vs. N094F. (See initial conditions sheet given to examinees)
- b. 2.2 psig ~~(0.5)~~ 0.75 MUB NRC AGREES
- c. 1.23 psig ~~(0.5)~~ 0.75 12/4/89

Ref. EP-3, Schematic Diagram E-1181

QUESTION: 4 REC ID: 583 PT VAL: 1.50
LP: OP-LOR-SYS-LP-ECCS Objective: 20,21

SCENARIO 9

Based on present plant conditions:

- a. Why didn't Div. I ADS logic open the ADS valves?
- b. Why didn't Div. II ADS logic open the ADS valves?
- c. How could the ADS logic be manually initiated with present plant conditions?

ANSWER: 4

- a. No Div. I low pressure pumps running (0.5)
- b. No high drywell pressure signal (0.25) in subchannel F logic (0.25)
- c. Arm and depress Div. II manual initiation pushbuttons (B & F) (0.5)
also accept
Manually initiate 'B' ADS

Ref. Schematic Diagram E-1161

NRC AGREES
go

Accept arm and depress
F only. 'B' subchannel
logic is sealed-in so this
action only would initiate
ADS (Reference E-1161)

QUESTION: 5 REC ID: 584 PT VAL: 1.00
LP: OP-LOR-SYS-LP-M71 Objective: 2,3

SCENARIO 9

RCIC received an initiation signal but is isolated.

- a. List any RCIC isolation valve(s) that can NOT be verified in the proper position.
- b. Given that HPCS suction has swapped to the suppression pool, why didn't the RCIC suction swap to the suppression pool?

ANSWER: 5

- a. 1. E51-F064 (Steam supply outboard isolation) (0.25)
2. E51-F077 (RCIC turbine exhaust outboard vacuum breaker isolation) (0.25)
- b. E51-F031 (RCIC pump suction from suppression pool) is interlocked closed (0.25) by the RCIC isolation signal. (0.25)

Ref. 05-1-02-III-5, Schematic Diagram E-1185

QUESTION: 6 REC ID: 585 PT VAL: 1.25
LP: OP-LOR-SYS-LP-R21-1 Objective: 3,5

SCENARIO 9

Both incoming feeder breakers to BOP busses 11HD, 12HE, 13AD and 14AE are open:

- a. What signal(s) automatically opened the feeder breakers to the above busses. Include any applicable setpoints.
- b. Concerning the re-energization of the above busses, which one of the following is correct:
 - 1) Busses can not be re-energized until 15AA is energized.
 - 2) Busses can be re-energized from BOP transformers 11A and 11B.
 - 3) Busses can be re-energized from BOP transformers 12A and 12B.
 - 4) Busses will automatically re-energize when power is restored to ST21 transformer.

ANSWER: 6

a. 80% BUV (0.25) on bus 15AA for 0.5 seconds (0.25) concurrent with a LOCA signal (0.25).

b. 2 (0.5).

NRC agrees of (.125 for "bus 15AA" .125 for "for 0.5 seconds")

Ref. SOI 04-1-01-R21-1, Schematic Diagram E-1039

QUESTION: 7 REC ID: 586 PT VAL: 1.00
LP: OP-LOR-SYS-LP-R21-1 Objective: 3,6

SCENARIO 9

a. Determine the possible automatic causes of the Div. I D/G trip, given present plant conditions.

b. Determine the cause of the Div. I LSS System Failure alarm.

ANSWER: 7

NRC DOES NOT AGREE of

- a. Engine overspeed, low lube oil pressure, generator differential (0.25 each). also accept "generator lockout relay" for "generator differential" can accept D/G in maintenance mode (Ref 04-1-02-1H13-P864-A-B)
- b. BUV (70%) on bus 15AA for greater than 12 seconds (0.25).

Ref. SOI 04-1-01 Schematic Diagram E-0014

Part 'b' should have been 0.5 pt vs 0.25

EXAM *****

9.00 POINTS 100% VALUE
6.300 POINTS 70% VALUE

7.200 POINTS 80% VALUE

Good Key Start 0812
Part A-2 Week 2

**EXAMINER GUIDE
GRAND GULF NUCLEAR STATION
STATIC SIMULATOR EXAM
SCENARIO 12**

I. SIMULATOR SET-UP

- A. Initialize the simulator to IC-17.
- B. Set all recorders to the correct time, rotating each chart a minimum of 3 inches and ensuring sufficient paper for the examination.
- C. Establish exam conditions as follows:
 - 1. Insert the following malfunctions to activate simultaneously:

<u>Event</u>	<u>Time</u>	<u>Malf. No.</u>	<u>Description</u>
1	0	135	Total Loss of Offsite Power
2	0	140A	DG 11 Failure to Start
3	0	140B	DG 12 Failure to Start

- D. Operator Actions:
 - 1. Place the Reactor Mode Switch to shutdown.
 - 2. Initiate RCIC; shutdown HPCS and reset its initiation logic once level is restored above level 2 (if it initiates).
 - 3. When reactor level is recovered to about -30 inches, HPCS and RCIC suction valves have aligned to the Suppression Pool, and SRVs are closed, FREEZE the simulator.

II. Validation

- A. For the malfunctions listed and operator actions taken, verify the following desired results are obtained:
 - 1. RCIC is injecting with a flow rate of approximately 800 GPM.
 - 2. All BOP busses and ESF busses 15AA and 16AB are deenergized; bus 17AC is reenergized by DG13.

3. All SRVs are currently closed with reactor pressure between 926 and 1033 psig.
- B. For the examination, the following parameters must be in the allowed band:
1. RCIC Flow is between 775 and 825 GPM
 2. Reactor pressure is between 926 and 1033 psig.
 3. Suppression Pool level is above 18.75 feet with alarms SUPP POOL LVL HI (P601-1A-C5 and P601-16A-C5) sealed in.
- C. Check all recorder traces and parameters for reasonable values and trends for existing simulator conditions.
- D. If any parameter is outside the allowed band, the simulator must be re-initialized, beginning at Step I. A.
- E. If all indications are satisfactory, the exam may begin.

III. QUESTION AVAILABILITY

- A. The following questions were developed specifically for use with Scenario 12:

<u>REC ID</u>	<u>LESSON PLAN</u>	<u>OBJECTIVES</u>	<u>TIME (min.)</u>
604	OP-LOR-DT-LP-003	A.2.o	5
607	OP-LOR-DT-LP-003	A.14	5
610	OP-LOR-DT-LP-003	A.14	10
613	OP-LOR-DT-LP-003	B.1.C	5
626	OP-LOR-DT-LP-003	A.17.E	10
627	OP-LOR-DT-LP-003	A17D	5
628	OP-LOR-DT-LP-003	A1D	10
629	OP-LOR-DT-LP-003	A10B	5

- B. Select other Category A Questions, as required, to provide an examination length of 1 to 1½ hours.

SCENARIO 12

INITIAL CONDITIONS

Approximately 10 minutes ago, plant conditions were as follows:

- 100% steady state power
- 1275 MWe
- All other controls in AUTO/NORMAL
- No active or potential LCOs existed
- The plant had just been issued a Heavy Weather warning for the next several hours

Then, a transient occurred, initiated by a tornado touching down in the 500 KV switchyard. The only operator actions taken were to place the Reactor Mode Switch to Shutdown and to initiate RCIC.

The Shift Superintendent is currently reviewing the EPPs for event classification. The Shift Supervisor is directing actions to control Reactor level and pressure per EP flowcharts.

The SPDS is selected to the OVERVIEW screen and should NOT be changed at any time during this exam.

Those questions identified with SCENARIO 12 at the top are related to these initial conditions. All others are NOT related to the scenario.

***** ANNUAL REQUALIFICATION EXAM ANSWER KEY *****
SC 12-1

PAGE 1

QUESTION: 1 REC ID: 604 PT VAL: 0.50
LP: OP-LOR-DT-LP-003 Objective: A.2.o

SCENARIO 12

Which plant load center(s) have lost power resulting in the operator-at-the-controls being unable to determine whether all rods have inserted following the transient?

ANSWER: 1 *also accept 15B61 or 15P61* *also accept 16B61 or 16P61*
LCC 15BA6 (0.25 pts) and LCC 16BB6 (0.25 pts) *AGREE w/COMMENT. gjo*

Reference: SOI 04-1-C11-2, Rod Control and Information System *gjo*
Attachment III

QUESTION: 2 REC ID: 607 PT VAL: 0.50
LP: OP-LOR-DT-LP-003 Objective: A.14

SCENARIO 12

Assuming HPCS is not needed for reactor level control and the Shift Superintendent has stated that secondary containment integrity must be maintained, what is preventing HPCS from being used to lower suppression pool level?

ANSWER: 2

F11-F064 (0.25 pts) and F065 (0.25 pts) are isolated (or closed due to a loss of instrument air).

Reference: *↳ or loss of power* ONEP 05-1-02-V-9, Loss of Instrument Air, Automatic Action 3.30; SOI 04-1-01-E51-1, Precaution 3.16

QUESTION: 3 REC ID: 629 PT VAL: 0.50
LP: OP-LOR-DT-LP-003 Objective: A10B

SCENARIO 12

If Drywell pressure increases to 1.40 psig, which RCIC isolation valves will not isolate due to present plant conditions?

DELETE *gjo*
~~Delete question 3~~

Since Rx Press is @ 950# no RCIC valves would isolate. Question is incorrect.

ANSWER: 3

E51-F077, RCIC TURB EXH OTBD VAC BRKR (0.25 pts) and E51-F078 RCIC TURB EXH INBD VAC BRKR (0.25 pts) will not isolate (due to loss of power.)

Reference: SOI 04-1-01-E51-1, Electrical lineup
ONEP 05-1-02-III-5, Automatic Isolations

DELETED
gg

QUESTION: 4 REC ID: 626 PT VAL: 1.00
LP: OP-LOR-DT-LP-003 Objective: A17E

SCENARIO 12

- a. For the present plant conditions, which SRVs are recommended to be used to control reactor pressure?
- b. Given that the present power outage has the potential for lasting several hours and possibly longer, what can be done to ensure ADS SRVs will continue to operate?

ANSWER: 4

- a. Operate the 11 SRVs other than ADS SRVs and B21-F051D (since these SRVs have only one accumulator good for at least one actuation each while ADS and B21-F051D have sufficient air supply from receivers and accumulators that will allow many actuations.) (0.5 pts)
- b. Maintenance can install N2 gas bottles and regulators (to establish makeup air to the ADS system.) (0.5 pts)

Reference: ONEP 05-1-02-I-4, Loss of AC Power, Section 5.2
ONEP 05-1-02-V-9, Loss of Instrument Air, Section 5.6

QUESTION: 5 REC ID: 627 PT VAL: 1.00
LP: OP-LOR-DT-LP-003 Objective: A17D

SCENARIO 12

ASSUME the HPCS motor breaker is racked down.

While preparing to crosstie the Division III diesel generator to either the Division I or II busses, Drywell Pressure increases to 1.4 psig. What operator actions must be performed in order to ensure that the HPCS initiation logic does not prevent Division III diesel generator from being crosstied to the Division I or II bus?

ANSWER: 5

(1.0)
100 12-4-84
~~Must override the High Drywell pressure signal (0.5) by depressing the~~
HPCS initiation logic *(0.5)* (Otherwise the DG 13 output breaker
cannot be closed at the same time as an ESF Bus 17AC feeder breaker,
thus preventing the cross-tie operation.)

Reference: SOI 04-1-01-P81-1, Precaution 3.26
SOI 04-1-01-E22-1, Precautions 3.7 and 3.10

NRC DOES NOT AGREE. GRABE AS ORIGINAL WRITER. Q/D
Question asks for operator action, not any justification or clarification

QUESTION: 6 REC ID: 635 PT VAL: 1.00
LP: OP-LOR-SYS-R21-1 Objective: 3e

SCENARIO 12

How does the Load Shedding and Sequencing System sense a loss of offsite power condition? (Setpoints NOT required)

ANSWER: 6

Feeder undervoltage (0.5) on the three ESF feeders upstream of the respective feeder breakers to the ESF bus (0.5).

Ref. Schematic Diagrams E-1120-11, E-1109-26 and E-1008

QUESTION: 7 REC ID: 337 PT VAL: 1.75
LP: OP-LOR-DT-LP-003 Objective: 1b,d,e

SCENARIO 12

Considering current plant conditions:

- a. Specify the available AC and DC power sources. Specific bus numbers are NOT required.
- b. Determine which systems are available to control reactor water level.
- c. Specify the methods available to remove decay heat from the reactor.

ANSWER: 7

- a. Div. III DG, all DC busses, UPS (0.25 ea.)
- b. RCIC and HPCS (0.25 ea.)
- c. RCIC and SRVs (0.25 ea.)

Ref. ONEP 05-1-02-I-4

QUESTION: 8 REC ID: 33 PT VAL: 1.00
LP: OP-LOR-SYS-LP-DCP Objective: B4

SLC System A is in the normal STANDBY lineup with one systematic deviation; the TEST TANK OUTLET VALVE (F031) is OPEN.

Which one of the following most accurately describes the effects on the STORAGE TANK OUTLET VALVE (F001) and SLC PUMP A when the SLC keylock control switch for pump A is placed to START?

- a. Valve F001 opens, SLC pump A starts after the valve reaches full open.
- b. Valve F001 opens, SLC pump A starts concurrently with the valve opening.
- c. Valve F001 does not open, SLC pump A starts.
- d. Valve F001 does not open, SLC pump A does not start.

ANSWER: 8

- c. (1.0)

Ref. E-1169, SOI 04-1-01-C41-1

**** END OF EXAM ****

6.75
~~7.25~~ POINTS 100% VALUE
~~5.075~~ POINTS 70% VALUE
4.725

5.400
~~5.800~~ POINTS 80% VALUE

***** ANNUAL REQUALIFICATION EXAM ANSWER KEY *****
RO 1-1

QUESTION: 1 REC ID: 128 PT VAL: 1.00
LP: OP-LOR-SYS-LP-DCP Objective: G37a

One of the symptoms of inadequate decay heat removal is "steaming evidenced from head vents, even though reactor coolant temperature indicates less than 212°F".

- a. With reactor coolant temperature less than 212°F, how can "steaming" occur?
- b. If forced core circulation is lost, to what point must reactor water level be increased to allow natural circulation to occur through the core and feedwater annulus?

ANSWER: 1

- a. (Loss of recirculation causing) thermal stratification (0.25) (can result in the surface) coolant temperature increasing above 212°F (0.25) (while recirculation temperature indication may remain at 212°F.)
- b. 82" indicated (615" above vessel zero). (0.5)

Ref: ONEP 05-1-02-III-1

QUESTION: 2 REC ID: 341 PT VAL: 1.00
LP: OP-LOR-IE-LP-003 Objective: A3b,e

A loss of feedwater heating occurred due to a level control problem in the 6A High Pressure Feedwater Heater while at 100% reactor power. After the immediate actions are taken, reactor power is 67% and feedwater temperature is 340°F.

- a. If control rods need to be adjusted, who must approve all rod movements?
- b. Present estimates of time for repair for the 6A High Pressure Feedwater Heater is 48 hours and requires feed flow through the heater to be isolated. What actions, if any, should be taken?

ANSWER: 2

- a. A reactor engineer (also accept "reactor engineering") (0.5).
- b. Reduce load to ≤ 50% electrical power (0.5)

Ref. ONEP 05-1-02-V-5, 04-1-01-N23-1 step 3.9

Edited 11/10/89 - revised part b of question/answer.

QUESTION: 3 REC ID: 712 PT VAL: 1.00
LP: OP-LOR-LL-LP-001 Objective: 15

During an outage it is observed that suppression pool level is increasing due to leakage through the closed HPCS minimum flow valve. It is determined that the valve should be locally stroked and torqued closed to prevent leakage. How is this operation required to be documented?

- a. Use of a Special Test Instruction written for this purpose.
- b. Use of a Maintenance Work Order.
- c. Use of a Temporary Alteration.
- d. Issuance of a Caution Tag.

ANSWER: 3

- b. (Maintenance Work Order) (1.0)

Reference:

QUESTION: 4 REC ID: 525 PT VAL: 1.00
LP: OP-LOR-LL-LP-001 Objective: 1,2a,2d,2e

Several hours following a LOCA in which the core was calculated to have uncovered, indicated plant conditions are as follows:

Reactor Power	0%
Reactor Water Level	-15"
Reactor Pressure	30 psig
Containment Temperature	225°F
Containment Pressure	5 psig

Hydrogen concentrations are increasing in both the drywell and containment. Evaluate present plant conditions and specify two (2) conditions under which the hydrogen igniters must be secured, assuming the conditions stated above do not change.

ANSWER: 4

- a. If containment hydrogen concentration increases to $8.5 \pm 0.2\%$ (0.5) also accept "exceeding the HDOL curve"
- b. If drywell hydrogen concentration increases to 9%. (0.5)

Ref. EP-3

QUESTION: 5 REC ID: 524 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 1,2a,2d

Several hours following a LOCA in which the core was calculated to have uncovered, indicated plant conditions are as follows:

Reactor Power	0%
Reactor Pressure	30 psig
Reactor Water Level	-10"
Containment Temperature	238°F
Containment Pressure	10 psig
Containment Hydrogen Concentration	8.5%
Drywell Hydrogen Concentration	10%

With the offsite radioactive release rate NOT expected to remain below the LCO limit, evaluate present plant conditions and specify which one of the following actions is NOT required to be performed: (ASSUME adequate core cooling exists)

- a. Secure and prevent operation of the Drywell Purge Subsystems.
- b. Operate the Div. I and II Hydrogen Igniters.
- c. Vent and purge the containment.
- d. Initiate Containment Sprays.

ANSWER: 5

- b. (Operate Div. I & II Hydrogen Igniters) (1.0)

Ref. EP-3

QUESTION: 6 REC ID: 482 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 2a,2d

Answer the following concerning the graphs of EP-2, RPV Control:

- a. For Figure 1, with a RCIC suction pressure (RCIC secured and suction aligned to the suppression pool) of 19.2 psig and a containment pressure of 2.5 psig, calculate the containment water level.
- b. If suppression pool/containment water level had been calculated to be 27 feet and suppression pool temperature is 130°F, in order to maintain RPV pressure in the SAFE zone of Figure 2 and Figure 3, RPV pressure must be less than approximately:

- 1. 450 psig
- 2. 750 psig
- 3. 800 psig
- 4. 925 psig

ANSWER: 6

- a. 44.5 ± 0.5 feet (0.5)
- b. 2 (750 psig) (0.5)

Ref. EP-2, Figures 1, 2 and 3 and Att. 29

QUESTION: 7 REC ID: 72 PT VAL: 1.00
LP: OP-LOR-AD-LP-005 Objective: 3

For the following sets of conditions (a and b), identify which criteria of the maximum work hour limits have been exceeded for each situation below.

a. Your work schedule is as follows:

- Day 1 - 1515 to 1530 Shift turnover
 1530 to 0330 Held shift position
- Day 2 - 1115 to 1130 Shift turnover
 1130 to 2330 Held shift position

b. The first four days of your work schedule are as follows:

- Day 1 - 0715 to 0730 Shift turnover
 0730 to 1530 Held shift position
- Day 2 - 0715 to 0730 Shift turnover
 0730 to 2030 Held shift position
- Day 3 - 0715 to 0730 Shift turnover
 0730 to 2030 Held shift position
- Day 4 - 0715 to 0730 Shift turnover
 0730 to 1530 Held shift position

ANSWER: 7

- a. Had less than 8 hours off including turnover time between work periods. (0.5)
- b. Worked greater than 24 hours in 48 hour period. (0.5)

Ref: T/S 6.2.2.f
01-S-06-2 (6.6)

QUESTION: 8 REC ID: 55 PT VAL: 1.00
LP: OP-LOR-AD-LP-005 Objective: 6

You are performing an electrical lineup on the TBCW system. When you get to 152-1313 (TBCW Pump A breaker), it is racked-out and red-tagged under number 19950 with work still in progress on the breaker. What must be done in order to complete the lineup for that component?

ANSWER: 8

Supervisor
Supervisor
12/3/89
(Take the lineup to the) Shift *Sup.* (who) will write-in the deviation (0.5) and note the reason in the exception area (0.5).

Ref: 02-S-01-2 (6.12.4.c)

Don't agree with comment; only accept Sup. mmj

QUESTION: 9 REC ID: 152 PT VAL: 1.00
LP: OP-LOR-AD-LP-005 Objective: 5

Identify which of the following is an acceptable means of "red tagging" systems/components under Red Equipment Clearances:

- a. Tagging a remote handswitch is adequate if a component is being tagged for personnel protection.
- b. When a motor-operated valve is to be a fluid system boundary point, the power supply is tagged, but the local operator need not be tagged.
- c. An air-operated valve that fails open will be considered closed for clearance purposes if, once it is closed, its air supply is opened and appropriately tagged to maintain the air supply to the valve.
- d. When isolating a pump motor, the pump suction and discharge valves should be shut and tagged to prevent possible rotation due to fluid flow.

ANSWER: 9

d. (1.0 ea)

Ref. 01-S-06-1

QUESTION: 10 REC ID: 125 PT VAL: 1.00
LP: OP-LO-DT-531 Objective: 1

In the event of resin intrusion into the vessel,

- a. Indicate which one of the following reactor water sample parameters would decrease:
1. Reactor Water Conductivity
 2. Reactor Water Activity
 3. Reactor Water pH
 4. Reactor Water Chlorides
- b. The most serious resin injection could result in a large release of nitrogen-16 into the steam lines resulting in a trip of the main steam line radiation monitors on HI-HI radiation. Which one of the following could also result in a main steam line radiation monitor trip on HI-HI radiation?
1. Hi reactor water level
 2. Fuel element failure
 3. Reactor recirc pump trip
 4. Turbine bypass/control valve failed open

ANSWER: 10

a. 3. (Reactor Water pH) (0.5)

b. 2. (Fuel element failure) (0.5)

Ref: ONEP 05-1-02-I-5 (2.0)

QUESTION: 11 REC ID: 312 PT VAL: 0.50
LP: OP-LOR-MCD-LP-009 Objective: A15g

The plant was operating at rated conditions. The following containment parameters were recorded on the Daily Ops logs just prior to a LOCA:

Containment Pressure = 0.01 psig
Containment Average Temperature = 90° F

The hydrogen igniters were secured during the shift following the LOCA, while the hydrogen analyzers were kept on-line. Two days later, annunciator "CTMT H2 HI" at panel P870 alarms. Current conditions indicate:

Containment Hydrogen Concentration = 2.9%
Containment Pressure = 0.0 psig.

The Shift Superintendent orders that the Combustible Gas Control System "A" Hydrogen Recombiner be started. Based on the above information and data found in applicable plant procedures, determine the required recombiner power setting.

ANSWER: 11

52.0 kW ± 2.0 kW (0.5)

(1.14 [CP from SOI Fig. 1] X 45.6 kW = 52.0 kW)

Ref. 04-1-01-E61-1

QUESTION: 12 REC ID: 364 PT VAL: 1.00
LP: OP-LOR-SYS-LP-N31 Objective: A10c

The reactor has been shutdown for critical repair of the main turbine. The main turbine is still on the turning gear and HP turbine casing temperature is 220°F. Mechanical maintenance requests that the turning gear be shut down in order to make the needed repairs to the main turbine. What conditions must be established to secure turning gear operation with the HP turbine casing temperature at the above value? Include applicable procedural references in your response.

ANSWER: 12

In accordance with SOI 04-1-01-N34-2 (0.25), (turning gear operation may be stopped with HP casing temperatures in excess of 212°F if all of the following requirements are met):

- 1) Shaft lift oil pump remains in operation. (0.25)
- 2) Auxiliary lube oil pump remains in operation (may be secured at UPC site representative discretion). (0.25)
- 3) Turbine shaft is manually rotated 180 degrees every 15 to 30 minutes until HP casing temperature is less than 212°F. (0.25) (Manual rotation may be secured earlier at UPC site representative discretion).

(Seal Steam System secured from the main turbine.)

QUESTION: 13 REC ID: 50 PT VAL: 1.00
LP: OP-LOR-SYS-LP-M71 Objective: 4

Concerning P601 annunciators, select the most correct statement:

- a. Both amber CTMT CLG EXH RAD HI-HI/INOP (18A-D5 and D6) annunciators MUST be in the alarm condition to obtain a Div. 1 containment/drywell ventilation system isolation from a containment high exhaust radiation condition.
- b. The red CTMT VENT RAD HI-HI/INOP (19A-G9) alarm means a containment/drywell ventilation isolation signal is present.
- c. The amber CTMT CLG EXH DIV 1, 4 RAD HI-HI/INOP (18A-D5) alarm means that one/or both of the trip signals required for a Div. 1 containment/drywell ventilation isolation on high exhaust radiation is present.
- d. Any time the amber CTMT CLG EXH DIV 1, 4 RAD HI-HI/INOP (18A-D5) annunciator alarms, the OUTBOARD containment/drywell ventilation dampers will isolate.

ANSWER: 13

c. (1.0)

Ref: ARI 04-1-02-1H13-P601-18A-D5

QUESTION: 14 REC ID: 48 PT VAL: 1.50
LP: OP-LOR-SYS-LP-M71 Objective: 1

The plant electrical system is in the normal SOI lineup when breaker 152-1501 trips and locks out bus 15AA. The Division 1 diesel generator starts and reenergizes bus 15AA. With power thus restored, you observe that annunciator MSIV/DRN VLV TRIP INIT (19A-E4) on P601 has sealed in.

- a. What caused this alarm? Be specific.
- b. Which, if any, of the valves associated with this alarm closed?
- c. What actions must be taken to clear this alarm?

ANSWER: 14

- a. Loss of power (0.25) to the Division 1 leak detection system temperature monitoring panel (0.25) (MAIN STEAM TUNNEL HIGH TEMPERATURE and HIGH DELTA T TRIPS) *- 1501 bus temp panel*
- b. No valves closed (G33-F004 did close but is not associated with the alarm) (0.5)
- c. Depress the NSSSS reset pushbutton for Division 1 (accept also BOTH pushbuttons) (0.5)

Ref: Dwg. E-1187-05
ARI 04-1-03-1H13-P601-19A-E4

QUESTION: 15 REC ID: 11 PT VAL: 1.00
LP: OP-LOR-SYS-LP-DC Objective: 5b

While checking for grounds on bus 11DH, you see the following indications:

- Initially V1, V2, V3 and V4 all read 12 volts
- When PB-1 is depressed V1 reads 24 volts, V2 reads 0 volts
- When PB-2 is depressed V3 reads 0 volts, V4 reads 24 volts

These readings indicate (Choose One)

- a. No grounds
- b. Dead ground on neutral bus
- c. Dead ground on negative bus
- d. Dead ground on positive bus

ANSWER: 15

b. ~~40.57~~ 1.0 ^{MLB} 12/2/89 ⇒ Reflects assigned pt. value @ heading & in all totals.

Ref: SOI 04-1-01-L11-1 (5.1.2)

QUESTION: 16 REC ID: 178 PT VAL: 1.00
LP: OP-LOR-RSD-LP-001 Objective: 7c

The control room is declared uninhabitable due to a fire and evacuated. The immediate and subsequent operator actions for "Shutdown from the Remote Shutdown Panel", ONEP 05-1-02-II-1, are completed. Upon exiting the control room, you determined that RCIC has isolated. Which one of the following statements is TRUE concerning RCIC operation from the Remote Shutdown Panel following a RCIC isolation?

- a. RCIC will restart automatically at Level 2.
- b. Close RCIC TURB TRIP/THROT VLV; Place RCIC TURB FLO CONT in manual at minimum setting; Re-open RCIC TURB TRIP/THROT VLV and establish flow.
- c. Close RCIC TURB TRIP/THROT VLV; reset RCIC TURB TRIP logic; RCIC will now restart automatically.
- d. RCIC can NOT be restarted from the Remote Shutdown Panel after an isolation.

ANSWER: 16

d (1.0)

Ref. ONEP 05-1-02-II-1

QUESTION: 17 REC ID: 269 PT VAL: 0.50
LP: OP-LOR-SYS-LP-F11/17 Objective: A5a

An emergency stop of the Main Hoist on the Fuel Handling Platform has occurred. The NOB operating the platform calls the Control Room and reports that, when he tries to lower the hoist, a loud chattering noise is emitted. He asks for your advice. You suspect that the emergency brake has not released. What actions should be taken?

ANSWER: 17

Raise the Main Hoist (to release the emergency brake) (0.5).

Ref. 04-1-01-F11-3, Step 3.11

**** END OF EXAM ****

16.50 POINTS 100% VALUE
11.550 POINTS 70% VALUE

13.200 POINTS 80% VALUE

***** ANNUAL REQUALIFICATION EXAM ANSWER KEY *****
SRO 1-1

QUESTION: 1 REC ID: 495 PT VAL: 1.00
LP: EPTS-3 Objective: A17

The reactor was operating at rated conditions when events occurred which led to an offsite release of radionuclides. Evaluate the following information and determine the Whole Body Dose Commitment for a two-hour exposure at the site boundary:

Whole Body Dose Rate at 2 miles (centerline) = 500 mrem/hr
delta T = -1.45 Wind Speed = 11 mph Wind Direction = 90°
Time After Shutdown = 3 hours

- a. 8.32 rem
- b. 10.42 rem
- c. 14.67 rem
- d. 16.85 rem

ANSWER: 1

- d. 16.85 rem (1.0)

The following is provided for information only:

delta T = -1.45 » Stability Class ~~B~~ A MCB 12/3/89
 [F] @ 2 miles = 0.13 (from A Stability; 2 mi, 10)
 [F] @ SB = 2.19 (from ~~B~~ A Stability; SB, 10)

Whole Body Dose Commitment =

$$[F]1 / [F]0 \times DRO \times \text{Exposure Time} =$$

$$2.19 / 0.13 \times 500 \text{ mrem/hr} \times 2 \text{ hours} = 16.85 \text{ rem}$$

Agree with comment
MVP

Ref: EPP 10-S-01-12, Section 6.3

Edited 11/10/89 - expanded ranges on possible answers

 QUESTION: 2 REC ID: 26 PT VAL: 2.00
 LP: EPTS-6 Objective: 1

Classify each of the following. Include in your answer the reference number for the specific Emergency Action Level to support your classification.

- a. Verified earthquake worse than an OBE but less than an SSE.
- b. Armed personnel are trying to break down the Control Room door from the Turbine deck. Security is responding; however, it appears unlikely that they will be able to stop the saboteurs.
- c. While operating, total leakage exceeds 32 gpm.
- d. Core is uncovered and damaged. Containment and Drywell integrity are in jeopardy due to overpressurization in excess of 20 psig in the Containment.

ANSWER: 2

Agree with comment must

- a. Alert, 18.2.1. also accept 18.2 only
- b. Site Area Emergency, 16.3.1. also accept 16.3 only
- c. Unusual Event, 2.1.2 MUB 2/3/89
- d. General Emergency, 19.7.1, 2, 3 → also accept 19.7 only (8 parts at 0.25 ea.)

Comments } For 4.4.1 and 2, no real monitoring team reports given.
 For 12.5, not enough info. provided.

} see justification sheets attached

Ref: EPP 10-S-01-1, Att. I

 QUESTION: 3 REC ID: 128 PT VAL: 1.00
 LP: OP-LOR-SYS-LP-DCP Objective: C37a

One of the symptoms of inadequate decay heat removal is "steaming evidenced from head vents, even though reactor coolant temperature indicates less than 212°F".

- a. With reactor coolant temperature less than 212°F, how can "steaming" occur?
- b. If forced core circulation is lost, to what point must reactor water level be increased to allow natural circulation to occur through the core and feedwater annulus?

ANSWER: 3

- a. (Loss of recirculation causing) thermal stratification (0.25) (can result in the surface) coolant temperature increasing above 212°F (0.25) (while recirculation temperature indication may remain at 212°F.)
- b. 82" indicated (615" above vessel zero). (0.5)

Ref: ONEP 05-1-02-III-1

 QUESTION: 4 REC ID: 341 PT VAL: 1.00
 LP: OP-LOR-IE-LP-003 Objective: A3b,e

A loss of feedwater heating occurred due to a level control problem in the 6A High Pressure Feedwater Heater while at 100% reactor power. After the immediate actions are taken, reactor power is 67% and feedwater temperature is 340°F.

- a. If control rods need to be adjusted, who must approve all rod movements?
- b. Present estimates of time for repair for the 6A High Pressure Feedwater Heater is 48 hours and requires feed flow through the heater to be isolated. What actions, if any, should be taken?

ANSWER: 4

- a. A reactor engineer (also accept "reactor engineering") (0.5).
- b. Reduce load to ≤ 50% electrical power (0.5)

Ref. ONEP 05-1-02-V-5, 04-1-01-N23-1 step 3.9

Edited 11/10/89 - revised part b of question/answer.

 QUESTION: 5 REC ID: 420 PT VAL: 1.50
 LP: OP-LOR-IE-LP-005 Objective: B.1, B.3

Several hours ago the STA discovered a mispositioned control rod. The reactor is operating at ≈ 90% power. Based on the following data, answer the questions below:

TIME	EVENT	OFF-GAS PRETREATMENT READINGS	OFF-GAS CHEMISTRY SAMPLE RESULTS
0100	Mispositioned rod discovered		
0500		15 mR/hr	5,000 μCi/sec
0530	Received Off-Gas Pretreat HI Rad Annunciator	300 mR/hr	150,000 μCi/sec

A sample of the reactor coolant system taken at 0530 indicates activity levels of 3.8 $\mu\text{Ci}/\text{gram}$ Dose Equivalent I-131 and 150/E gross Beta and Gamma.

- a. Briefly explain how long power operations could continue under present conditions.
- b. How often must the reactor coolant be sampled for activity?
- c. Based on the data provided, CLASSIFY this event, if required.

ANSWER: 5

- a. Must be in Hot Shutdown with the MSIVs closed within 12 hours (0.5)
- b. Every 4 hours (0.5)
(also accept)
Initially between 2 & 6 hours & every 4 hours thereafter
- c. Unusual Event (0.5)

Ref. T.S. 3.4.5; EPP 10-S-01-1, Att. 1, EAL 4.1.1

Edited 11/10/89 - added additional acceptable answer to part b
- restructured wording of the question in general

QUESTION: 6 REC ID: 525 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 1,2a,2d,2e

Several hours following a LOCA in which the core was calculated to have uncovered, indicated plant conditions are as follows:

Reactor Power	0%
Reactor Water Level	-15"
Reactor Pressure	30 psig
Containment Temperature	225°F
Containment Pressure	5 psig

Hydrogen concentrations are increasing in both the drywell and containment. Evaluate present plant conditions and specify two (2) conditions under which the hydrogen igniters must be secured, assuming the conditions stated above do not change.

ANSWER: 6

- a. If containment hydrogen concentration increases to $8.5 \pm 0.2\%$
(0.5) also accept "exceeding the HDOL curve"
- b. If drywell hydrogen concentration increases to 9%. (0.5)

Ref. EP-3

QUESTION: 7 REC ID: 524 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 1,2a,2d

Several hours following a LOCA in which the core was calculated to have uncovered, indicated plant conditions are as follows:

Reactor Power	0%
Reactor Pressure	30 psig
Reactor Water Level	-10"
Containment Temperature	238°F
Containment Pressure	10 psig
Containment Hydrogen Concentration	8.5%
Drywell Hydrogen Concentration	10%

With the offsite radioactive release rate NOT expected to remain below the LCO limit, evaluate present plant conditions and specify which one of the following actions is NOT required to be performed: (ASSUME adequate core cooling exists)

- a. Secure and prevent operation of the Drywell Purge Subsystems.
- b. Operate the Div. I and II Hydrogen Igniters.
- c. Vent and purge the containment.
- d. Initiate Containment Sprays.

ANSWER: 7

- b. (Operate Div. I & II Hydrogen Igniters) (1.0)

Ref. EP-3

QUESTION: 8 REC ID: 482 PT VAL: 1.00
LP: OP-LOR-EP-LP-001 Objective: 2a,2d

Answer the following concerning the graphs of EP-2, RPV Control:

- a. For Figure 1, with a RCIC suction pressure (RCIC secured and suction aligned to the suppression pool) of 19.2 psig and a containment pressure of 2.5 psig, calculate the containment water level.
- b. If suppression pool/containment water level had been calculated to be 27 feet and suppression pool temperature is 130°F, in order to maintain RPV pressure in the SAFE zone of Figure 2 and Figure 3, RPV pressure must be less than approximately:
 1. 450 psig
 2. 750 psig
 3. 800 psig
 4. 925 psig

ANSWER: 8

- a. 44.5 ± 0.5 feet (0.5)
- b. 2 (750 psig) (0.5)

Ref. EP-2, Figures 1, 2 and 3 and Att. 29

QUESTION: 9 REC ID: 72 PT VAL: 1.00
LP: OP-LOR-AD-LP-005 Objective: 3

For the following sets of conditions (a and b), identify which criteria of the maximum work hour limits have been exceeded for each situation below.

- a. Your work schedule is as follows:

Day 1 - 1515 to 1530	Shift turnover
1530 to 0330	Held shift position
Day 2 - 1115 to 1130	Shift turnover
1130 to 2330	Held shift position

- b. The first four days of your work schedule are as follows:

Day 1 - 0715 to 0730	Shift turnover
0730 to 1530	Held shift position
Day 2 - 0715 to 0730	Shift turnover
0730 to 2030	Held shift position
Day 3 - 0715 to 0730	Shift turnover
0730 to 2030	Held shift position
Day 4 - 0715 to 0730	Shift turnover
0730 to 1530	Held shift position

ANSWER: 9

- a. Had less than 8 hours off including turnover time between work periods. (0.5)
- b. Worked greater than 24 hours in 48 hour period. (0.5)

Ref: T/S 6.2.2.f
01-S-06-2 (6.6)

QUESTION: 10 REC ID: 152 PT VAL: 1.00
LP: OP-LOR-AD-LP-005 Objective: 5

Identify which of the following is an acceptable means of "red tagging" systems/components under Red Equipment Clearances:

- a. Tagging a remote handswitch is adequate if a component is being tagged for personnel protection.
- b. When a motor-operated valve is to be a fluid system boundary point, the power supply is tagged, but the local operator need not be tagged.
- c. An air-operated valve that fails open will be considered closed for clearance purposes if, once it is closed, its air supply is opened and appropriately tagged to maintain the air supply to the valve.
- d. When isolating a pump motor, the pump suction and discharge valves should be shut and tagged to prevent possible rotation due to fluid flow.

ANSWER: 10

d. (1.0 ea)

Ref. 01-S-06-1

QUESTION: 11 REC ID: 125 PT VAL: 1.00
LP: OP-LO-DT-531 Objective: 1

In the event of resin intrusion into the vessel,

a. Indicate which one of the following reactor water sample parameters would decrease:

1. Reactor Water Conductivity
2. Reactor Water Activity
3. Reactor Water pH
4. Reactor Water Chlorides

b. The most serious resin injection could result in a large release of nitrogen-16 into the steam lines resulting in a trip of the main steam line radiation monitors on HI-HI radiation. Which one of the following could also result in a main steam line radiation monitor trip on HI-HI radiation?

1. Hi reactor water level
2. Fuel element failure
3. Reactor recirc pump trip
4. Turbine bypass/control valve failed open

ANSWER: 11

- a. 3. (Reactor Water pH) (0.5)
- b. 2. (Fuel element failure) (0.5)

Ref: ONEP 05-1-02-I-5 (2.0)

QUESTION: 12 REC ID: 247 PT VAL: 1.00
LP: OP-LOR-RF-LP-001 Objective: B6

Core Alterations are in progress in the quadrant of the core where SRM "A" is located. The Electrical Department is trouble-shooting a problem with the "A" SRM drive motor. Determine which of the following actions can be performed without violating technical specifications.

- a. SRM "A" detector can be de-energized and red tagged.
- b. SRM "A" detector can be withdrawn to assist in the trouble-shooting effort.
- c. SRM "A" drive motor can be de-energized with SRM "A" detector fully inserted.
- d. SRM "A" drawer can be placed in STBY using the SRM "A" Mode Switch.

ANSWER: 12

c. (Drive motor can be de-energized with SRM "A" fully inserted) (1.0)
Tech Spec 3.9.2

QUESTION: 13 REC ID: 312 PT VAL: 0.50
LP: OP-LOR-MCD-LP-009 Objective: A15g

The plant was operating at rated conditions. The following containment parameters were recorded on the Daily Ops logs just prior to a LOCA:

Containment Pressure = 0.01 psig
Containment Average Temperature = 90° F

The hydrogen igniters were secured during the shift following the LOCA, while the hydrogen analyzers were kept on-line. Two days later, annunciator "CTMT H2 HI" at panel P870 alarms. Current conditions indicate:

Containment Hydrogen Concentration = 2.9%
Containment Pressure = 0.0 psig.

The Shift Superintendent orders that the Combustible Gas Control System "A" Hydrogen Recombiner be started. Based on the above information and data found in applicable plant procedures, determine the required recombinder power setting.

ANSWER: 13

52.0 kW ± 2.0 kW (0.5)

(1.14 [CP from SOI Fig. 1] X 45.6 kW = 52.0 kW)

Ref. 04-1-01-E61-1

QUESTION: 14 REC ID: 140 PT VAL: 1.00
LP: OP-LOR-SYS-LP-DC Objective: 4

While in Cold Shutdown, Surveillance Procedure 06-EL-1L11-Q-001, 125 Volt Battery Bank All Cell Check, is performed with the following parameters for Battery 1C3 being recorded:

- Average corrected specific gravity of all connected cells is 1.22
- Lowest corrected specific gravity of any connected cells is 1.201
- Lowest float voltage of any connected cell is 2.10 volts
- Electrolyte level is at normal level in all cells

From the parameters above, determine if any Technical Specification actions are required. Include all required times.

ANSWER: 14

The battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and are restored to within limits within 7 days. (1.0)

Ref. SOI 04-1-01-L11-1

QUESTION: 15 REC ID: 48 PT VAL: 1.50
LP: OP-LOR-SYS-LP-M71 Objective: 1

The plant electrical system is in the normal SOI lineup when breaker 152-1501 trips and locks out bus 15AA. The Division 1 diesel generator starts and reenergizes bus 15AA. With power thus restored, you observe that annunciator MSIV/DRN VLV TRIP INIT (19A-E4) on P601 has sealed in.

- a. What caused this alarm? Be specific.
- b. Which, if any, of the valves associated with this alarm closed?
- c. What actions must be taken to clear this alarm?

ANSWER: 15

- a. Loss of power (0.25) to the Division 1 leak detection system temperature monitoring panel (0.25) (MAIN STEAM TUNNEL HIGH TEMPERATURE and HIGH DELTA T TRIPS)
- b. No valves closed (G33-F004 did close but is not associated with the alarm) (0.5)
- c. Depress the NSSSS reset pushbutton for Division 1 (accept also BOTH pushbuttons) (0.5)

Ref: Dwg. E-1187-05
ARI 04-1-02-1M13-P601-19A-E4

QUESTION: 16 REC ID: 178 PT VAL: 1.00
LP: OP-LOR-RSD-LP-001 Objective: 7c

The control room is declared uninhabitable due to a fire and evacuated. The immediate and subsequent operator actions for "Shutdown from the Remote Shutdown Panel", ONEP 05-1-02-II-1, are completed. Upon exiting the control room, you determined that RCIC has isolated. Which one of the following statements is TRUE concerning RCIC operation from the Remote Shutdown Panel following a RCIC isolation?

- a. RCIC will restart automatically at Level 2.
- b. Close RCIC TURB TRIP/THROT VLV; Place RCIC TURB FLO CONT in manual at minimum setting; Re-open RCIC TURB TRIP/THROT VLVE and establish flow.
- c. Close RCIC TURB TRIP/THROT VLV; reset RCIC TURB TRIP logic; RCIC will now restart automatically.
- d. RCIC can NOT be restarted from the Remote Shutdown Panel after an isolation.

ANSWER: 16

d (1.0)

Ref. ONEP 05-1-02-II-1

QUESTION: 17 REC ID: 270 PT VAL: 1.00
LP: OP-LO-SYS-LP-E12-1 Objective: 12

During the ADHR - Flush/Test Mode of ADHR operation, LPCI "C" is considered inoperable. Explain why LPCI "C" was declared inoperable.

ANSWER: 17

Manual realignment for a suppression pool suction (as required by T. S. 3.5.2) (0.5) cannot be performed from the control room panels (~~as~~ ¹⁴⁰ required by TSPS #68) (0.5). (as required by TSPS #125) ¹²⁻⁵⁻⁸⁹

Can also accept "because F066C and F057 are open". See attached TSPS 125 for manual re-alignment
Ref. 04-1-01-E12-1
*agree with comment
NWP.*

***** END OF EXAM *****

18.50 POINTS 100% VALUE
12.950 POINTS 70% VALUE

14.800 POINTS 80% VALUE

ENCLOSURE 5

REQUALIFICATION PROGRAM EVALUATION REPORT

Facility Generated Reference Material

The reference material supplied by the facility was reviewed to determine its adequacy for examination development and administration. The facility supplied an adequate number of open reference questions and their supporting static simulator scenarios for development of the written portion of the examination. Seventy six JPMs and 16 dynamic simulator scenarios were provided for the development of the operating portion of the examination. A sufficient amount of additional reference material was provided to the examination team.

Examination Preparation

NRC examiners met with members of the facility training staff on several separate occasions for the purpose of constructing the examination. This included the entire week of October 23, which was devoted exclusively to examination development activities.

Specific lessons learned from this portion of the examination process include:

A. Dynamic Simulator

1. While the simulator scenarios were generally good in exercising the EOPs, additional effort should be made to exercise infrequently used portions of the flowpath through use of passive and active failures. Conversely, return of tripped or damaged equipment later in the scenario can test the operators ability to adapt to enhanced mitigation ability from a degraded state.
2. In general, identification of Individual Simulator Critical Tasks (ISCTs) was well done by the training staff. The facility developed a comprehensive guideline for defining and determining an ISCT. Some recalibration of training personnel in the philosophy of ISCT use was necessary, but they adapted well and exam administration was not adversely impacted. Clarification by the NRC of ISCT development and use will be provided in Revision 6 to the Examiner Standards.

B. Written

The NRC examiners expressed concern at the validation times provided on the written examination question bank. Validation should be based on the time a competent operator would take to correctly answer the question given his knowledge level, use of only those references required to specifically answer the question and the circumstances of the situation presented in the question.

Time should not be factored in for the operator to check every potential reference to verify his answer. For example, if reactor pressure is 550 psig during a LOCA and the Shift Supervisor asks why the RHR injection valve is not open, it should not take 5 minutes for him to get an answer from the panel operator. Likewise, time validating this question at 5 minutes is inappropriate. The facility needs to revisit their process of time validation with the above guidance in consideration.

Examination Administration

The facility's administration of the examination was acceptable. Plans for maintaining examination security were thorough and well executed. Some areas of exam administration were noted which require improvement.

A. General

1. Operators exhibited poor understanding and use of the MOV test switches. They often did not use these switches even when directed by the procedure and sometimes they would use them for valves that did not require their use.
2. Operators paid little attention to the use of Alteration Tracking Sheets and surveillance test results sheets during JPMs where their use would be appropriate.
3. Operators did not always obtain or demonstrate the need for special equipment, keys, tools, communication devices, etc. during JPMs where such items would be critical to successful completion of the task. Such items should be included as steps in the JPM evaluation package.
4. During performance of JPM OP-LOR-CRO-JPM-004-01, Scram Control Rods Locally at the HCU it was noted that the scram test switches were located at the top of the HCU. Switch "A", in particular, was very difficult reach by an average height person and shorter operators would be required to climb on this contaminated equipment to perform the task. The facility should review the human factors nature of this task and determine whether corrective action is appropriate.

B. Dynamic Simulator

1. Critiques of operator performance were generally good, however, at times some information about a particular action had to be solicited by the NRC. For example, the question arose of whether a CRO properly secured RCIC (during an ATWS) per operating procedure when directed to lower reactor pressure vessel (RPV) water level by terminating and preventing all injection into the RPV except from the Boron Injection System and CRD. The facility evaluators appeared reticent to state such until directly asked by the Chief Examiner. Since part of the examination involves an evaluation of the facility's requal program, it is important that the facility

evaluators cover all potentially controversial activities that occur during a scenario to demonstrate to the NRC that no errors in operator performance were overlooked during their evaluation.

2. The facility should ensure that identified critical tasks are specific enough that all evaluators will grade by the same criteria and obtain the same result. For example, if a critical task requires entering EP-3 even though EP-3 does not require any operator actions based on the current conditions, then the operator will be graded as UNSAT if he doesn't enter that emergency procedure and verify no action is required. The evaluators should not infer that the operator "knew" no action was required and therefore did not need to enter the procedure when the appropriate entry conditions were met or when directed by another EP.
3. During the first week of exams, the facility elected to utilize only one instructor in the control booth to manage the scenario. This person was quickly overwhelmed with normal scenario communications and thus was hampered in operating the control console. The facility recognized this weakness and corrected the difficulty by assigning another instructor to assist.
4. During the exams for the second crew in the second week, the simulator hardware/software encountered a significant difficulty and subsequently "crashed". As a result, one scenario had to be deleted and another substituted for it. Fortunately, the exam team had prepared standby scenarios so when the simulator was repaired, the exams were immediately recommenced. Unfortunately during this sequence of events, the simulator operator did not reload one of the scenarios per the guide and as a result a critical task specifically included by the NRC did not get tested. The simulator operators must not be in such a rush to keep the exams on schedule that they incorrectly set up the simulator and do not run the exams as they are written.

C. JPMs

1. JPM OP-LOR-CRO-JPM-041-01, Lower Suppression Pool Water Level with the Reactor Core Isolation Cooling System. This JPM has an evaluation step for placing the MOV test switch to TEST. However, there is no comparable step to return the test switch to NORMAL.
2. JPM OP-LOR-NOB-JPM-010-00, Local Start of the Standby Diesel Generator 11. Item 10 of this JPM concerns starting of the aux lube oil pump. The diesel start procedure specifies that turbo oil pressure should be greater than or equal to 20 psig. This requirement should be made part of the JPM standard.
3. JPM OP-LOR-NOB-JPM-010-00, Local Start of the Standby Diesel Generator 11. Question #2 of this JPM is a direct look up and should be revised.

4. JPM OP-LOR-CRO-JPM-007-01, Defeat RCIC Low RPV Pressure Isolation Interlocks. This JPM assumed that the Shift Supervisor (SS) made a controlled copy of the applicable attachments to the EPs for the operator to use to perform this task. This was an unreasonable assumption since the SS is very involved in managing the casualty in this type situation. This concern was born out by SRO performance during the dynamic simulator portion of the requal exam where obtaining a copy of the procedure was left to the operator. The facility must ensure that all aspects of the JPM reflect the operational manner in which business is conducted at Grand Gulf to the greatest extent possible.
5. JPM OP-LOR-CRO-JPM-054-00, Startup of Suppression Pool Cooling. Item 9 was a step for verifying that F064B was closed. The standard for this item was in error in that correct performance of this step was indicated by the green light being lit and the red light being out.
6. JPM OP-LOR-NOB-JPM-004-00, Defeat ARI/RPT Logic Trips Per Emergency Procedure. Item 3 of this JPM required the operator to "locate DC Distribution Panel 1DK1 and place breaker 72-11K26 to its 'OPEN' position". The actual panel designation where this breaker was located was 11DK1.
7. JPM OP-LOR-CRO-JPM-005-01, Defeat HPCS High SP Water Level Suction Transfer Interlock. At least one facility evaluator did not have the correct attachment to perform this task. As a result, the examination was unnecessarily delayed while the facility evaluator reproduced the appropriate attachment from control room material. The facility should ensure that all evaluators are provided the tools required to perform the examination and the evaluators must be adequately prepared to give the examination.
8. JPM OP-LOR-NOB-JPM-017-00, Shutdown of 1Y87 Inverter. Item 3 of this JPM directed the operator to "open INV. DC INPUT breaker". As a comment to this item, a cue was provided to tell the operator that the INVERTER AC VOLTS (METER 2VM) goes to "zero" when breaker 3CB was opened. This cue was inappropriate to give unless requested for by the operator since no alarms sound when the breaker is opened.
9. JPM OP-LOR-CRO-JPM-012-01, Operate SSW Pumps from the Remote Shutdown Panels. Question #1 of this JPM should have its answer key expanded to include an explanation of the operator's response. Simple yes/no answers are not adequate for this portion of the examination.
10. JPM OP-LOR-CRO-JPM-002-01, Operate the Control Rod Drive System to Bring the Reactor Critical. Item 4 of this JPM requires the operator to calculate reactor period, however, no stop watch was available in the simulator for the operator to use (as he normally would in the control room) to measure the time required for power to double.

11. JPM OP-LOR-CRO-JPM-002-01, Operate the Control Rod Drive System to Bring the Reactor Critical. Question #1 of this JPM is a direct look up and should be revised.
12. JPM OP-LOR-CRO-JPM-002-01, Operate the Control Rod Drive System to Bring the Reactor Critical. As part of the JPM turnover, the operator was never made aware of which IRM and SRM chart recorders were in fast speed. This should be made a part of the initial conditions since the operator is placed in the middle of an evolution in order to perform the JPM.
13. JPM OP-LOR-CRO-JPM-058-00, Paralleling and Loading Division I Diesel Generator. Question #2 of this JPM is a direct look up and should be revised.
14. JPM OP-LOR-CRO-JPM-058-00, Paralleling and Loading Division I Diesel Generator. Item 10 of this JPM specifies that the operator should adjust diesel generator VARs per the Table on Step 3.29 of the procedure. The meter to measure this adjustment has a scale from -10 to +10 MVARs with 1 MVAR graduations. Correct performance of this step requires the operator to adjust diesel generator VARs to between 0.43 and 0.6 MVARs. The NRC does not understand how any operator could hope to attain such a precise adjustment given the scale of the meter. The procedure should be reviewed from this human factors standpoint.
15. On several occasions, the facility evaluator permitted the operator to observe while the simulator was being set up for performance of a JPM. While normally this would not present a problem, two difficulties arise with this practice. First, an unfair advantage could be claimed since not all candidates are presented the initial conditions at the same time. Second, the candidates could be misled or confused by what they observe during the simulator setup that may adversely affect their performance. As a result, the practice of secluding the operators from the simulator floor until the setup is complete and the JPM ready to begin should be adopted.
16. JPM questions need to be time validated so that examination time can be effectively utilized. Additionally, these questions need to be in a short answer format - multiple choice format in this circumstance is inappropriate. Some existing questions were too lengthy for easy comprehension when read aloud to the operators.
17. The facility should ensure that the JPMs are time validated consistent with the initial conditions given in the JPM. For example, if reactor water level is dropping while in the Shutdown Cooling mode and manual initiation of LPCI is required, then the time to perform this task should be consistent with not allowing core damage or violating any other site specific operating requirements.

18. JPM OP-LOR-NOB-JPM-011-00, RPS Motor/Generator Startup. For question #2, depending on which procedures were referenced, there were two answers. This was apparently due to a recent procedure revision which had not been incorporated into all references affected by the change. Also question #3 should state whether setpoints are required for full credit.
19. JPM OP-LOR-CRO-JPM-047-00, Performance of Control Room Items Prior to Departing During a Shutdown from the Remote Shutdown Panel. Item 2 of this JPM has an additional correct method for accomplishing the intent of this step. Using the "All Rods" button without "Raw Data" selected will show any control rods that are not full in by displaying their notch position.
20. JPM OP-LOR-CRO-JPM-054-00, Startup of Suppression Pool Cooling. Question #2 had an answer that had a tolerance band of +/- 1 minute which is outside the allowable tolerance band of Tech Specs. Tolerance bands are recommended but they need to be consistent with plant directives.
21. JPM OP-LOR-CRO-JPM-030-01, Determine Reactor Water Level from the Remote Shutdown Panels. Question #2 had an answer that had a tolerance band outside the allowable tolerance band of Tech Specs. Tolerance bands are recommended but they need to be consistent with plant directives.
22. The JPM cover sheets were not identical on all JPMs. One version was missing a place to record the performance time. Also, there were spelling errors, grammatical errors and missing words in most of the JPMs.

D. Written

During their grading of the written exam, the facility deleted question #166 from scenario #3 but did not mark up the answer key to reflect that fact. As a result, it was only by comparing differences in grading that the NRC realized that there was a problem with one of the questions on the test. The facility is reminded to transmit to the NRC all changes made or other significant factors that the facility has taken into account during their grading of the examination.

Evaluator Performance

An evaluation of the facility's evaluators was conducted. The evaluation consisted of assessing the evaluator's skills in the following areas:

- Detection Skills,
- Probing of Examinee Weaknesses, as required,

- Properly Grading JPMs,
- Differentiating Training Knowledges,
- Judgement (Pass/Fail Decisions), and
- Cues (Verbal and Non-verbal).

Mr. Rasch of the Training Department did a very good job of simulating various auxiliary operators and other plant personnel during the dynamic simulator examinations. The NRC determined that no licensee evaluator was unsatisfactory; however, several evaluator shortcomings were noted during the operating portion of the examination. In general, several of the facility examiners were rather tentative and unsure of themselves during the exams. Exceptions to this conclusion would be Mssrs. Roberts, Bowdon and Bottemiller. This improved over the two week exam period as the evaluators gained experience. Those weaknesses of specific note were:

1. Not all evaluators started timing the performance of the JPMs at the same point in the JPM. This deficiency was addressed in the second week JPMs so the evaluators were consistent. The NRC starts the clock once the initial cue has been given.
2. During Week 1, some evaluators paraphrased the initial conditions and initiating cues of the JPMs which led to some confusion for the operators. The JPM information should be read verbatim the first time through and clarification provided if requested by the operator. This problem improved during the second week of exams.
3. Some evaluators were not prepared to give expected plant responses to the operator during the course of the JPM. As a result, critical feedback to the operator was missing which in some situations affected task performance.
4. The evaluators should be careful to not provide cues that are not solicited by the operator. For example, if a meter reading changes as a result of the operator's action, the evaluator should not volunteer that information unless an alarm would have sounded or the operator expresses an interest in it.
5. Some evaluators did not present the initiating cues at a location consistent with the task being performed. For example, restoration of AC power to the drywell unit coolers is performed per Attachment 10 to EP-2. Since the operator would receive the order and attachment in the control room, it would not be appropriate to provide the initiating cue out in the turbine building.

6. During the JPMs, several evaluators prompted the operator to obtain necessary room keys prior to leaving the control room. This action was inappropriate and the operator should have been allowed to recognize his mistake, in due course, at the price of a wasted trip out to the equipment room.
7. The number of evaluators utilized during the dynamic simulator portion of the examination varied from that specified in the facility Training Policy Procedure. The NRC is concerned that the exams are not conducted in a consistent manner whether the NRC is present or not. The facility should follow their procedures as written and change them in the approved manner if they are not adequate.
8. During one dynamic simulator scenario, the facility evaluators allowed the STA to enter the simulator control booth to speak with the simulator operator. The simulator operator quickly led the STA out of the booth without allowing exam compromise. However, the facility needs to exercise more diligence in maintaining the sanctity of the control booth and any other areas where exam integrity could be compromised.

Requalification Program Evaluation

A satisfactory requalification program must meet each of the following criteria:

- (1) A 90 percent pass/fail decision agreement between the NRC and the facility grading of the written and operating examinations.
- (2) At least 75 percent of all operators must pass the examination.
- (3) No crews rated as unsatisfactory by the NRC, but rated as satisfactory by the facility.
- (4) No more than one third of the crews are determined to be unsatisfactory on the simulator evaluations.

In addition, if three or more of the following are applicable to the requalification program, then the program shall be determined to be unsatisfactory. If one or two of the following are applicable, then the program may be determined to be unsatisfactory:

- (5) The same common JPM is missed by at least 50 percent of the examinees.
- (6) The same question on the same JPM is missed by at least 50 percent of the examinees.

- (7) The facility failed to train and evaluate operators in all positions permitted by their individual licenses.
- (8) The facility failed to train operators on "in-plant" JPMs.
- (9) Less than 75 percent of the examinees correctly answered 80 percent of the common JPM questions.
- (10) More than one facility evaluator is determined to be unsatisfactory.

Based on the information provided in the report above, the Grand Gulf Requalification Program met all evaluation criteria in an acceptable manner and therefore is rated as SATISFACTORY.

ENCLOSURE 6

SIMULATOR FACILITY FIDELITY REPORT

Facility Licensee: System Energy Resources, Inc.

Facility Docket No.: 50-416

Operating Tests Administered on: November 14 and November 28, 1989

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

<u>ITEM</u>	<u>DESCRIPTION</u>
Drywell Temperature	When eight ADS valves are opened, drywell temperature is noted to increase contrary to expectation.
Service Air Compressor	The plant operates the Service Air Compressor in the local mode which means the operators normally direct a Plant Equipment Operator (PEO) to start the compressor in plant when needed. However, the simulator has no capability to perform this remote function and the operators must be guided by the simulator operator to start the compressor from the Control Room thus providing negative training.
SPDS	The licensee recently implemented Revision 4 to the Emergency Operating Procedures (EOPs), however, the SPDS displays in the simulator have not been updated to reflect these changes.
RC&IS	The second column of rod full out and full in LED indicators from the right Display would not illuminate. This problem occurred while correcting another problem with the rod position indicators two columns further to the left.
SRM Chart Recorder 1C51-NR-R602A	The chart recorder for SRMs A, C, and E had incorrectly scaled chart paper installed.
IRM Chart Recorder 1C51-NR-R603A	This recorder did not function thus preventing the recording of trend information.
IRM Chart Recorder 1C51-NR-R603B	The chart paper take up wheel for this recorder did not function and thus the paper eventually jammed preventing the recording of trend information.

Simulator The simulator operator's booth is not sufficiently sound proofed to prevent the examinees from overhearing discussions in the booth. Additionally, the glass of the booth is not one way and the operators tend to rely on eye to eye contact with the operator in the booth to enhance communications.

Audio/Visual Recording Recording capability is limited in that the TV equipment only records in black and white (many annunciators have different colors for different states of the same alarm), definition of control panel indications/alarms/status is not sufficient to assist in making pass/fail decisions, and area microphones cannot pick up all operator conversations (particularly when speaking in small groups or on the phone).

Communication Systems The page system at each end of the P680 panel is not functional. As a result only one pager (P601) is available for use in the simulator. The simulator radio system does not duplicate that available in the control room. The simulator telephone system only has one outgoing line while the control room telephone system has three separate lines available. This inhibits the operators from responding as they would in the plant.

DG II O/A Fan This fan did not auto start with the diesel generator running and ready to load. It should auto start when the diesel is greater than 200 rpm. The facility reported that they were aware of this problem and that a discrepancy report had already been written.

Feedwater Line Break During a feedwater line break malfunction (MALF #171 at 100%) if the operator closes B21-F065B in an attempt to isolate the leak, the software program allows RCIC to begin injecting into the reactor vessel vice continue pouring out the break.

Annunciator P601-19A-E2 This annunciator "ADS B RHR B/RHR C PERM" is supposed to clear as RHR pump discharge pressure decreases during performance of JPM task OP-LOR-CRO-JPM-054-00, Startup of Suppression Pool Cooling. It did not do so.