U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-483/OL-89-02

Docket No. 50-483 License No. NPF-30

Licensee: Callaway Plant
RR 1 Box 66
Steedman, MO 65077

Facility Name. Callaway Plant

Examination Administered At: Callaway

Examination Conducted: Written and operating examinations for three Reactor Operator and four Senior Reactor Operator candidates.

RIII Examiners: T Burdick for — D. Shepard — Date 11/8/89

K. Shembarger — K. Shembarger — Date 11/8/89

Vincent Loughney — V. Loughney — Date 11/8/89

Chief Examiner: J. Lennartz — Date 11/8/89

Approved By: T. Burdick, Chief
Operator Licensing Section 2 — Date 11/8/89

Examination Summary

Examination administered on October 17-19, 1989 (Report No. 50-483/OL-89-02).
Areas Inspected: Written and operating examinations were administered to three Reactor Operator candidates and four Senior Reactor Operator candidates.
Results: Three Reactor Operator candidates and three Senior Reactor Operator candidates passed the examinations. One Senior Reactor Operator candidate failed both the written and operating examinations.

REPORT DETAILS

1. Examiners

*J. Lennartz, NRC
V. Loughney, NRC
K. Shembarger, NRC
D. Shepard, NRC

*Chief Examiner

2. Examiner Observations

During preparation of the written and operating examinations, the examiners found the reference material provided by the facility to be well indexed and properly labeled which allowed the examiners to efficiently locate and extract necessary information. However, the examiners found the system lesson plans, in general, to be lacking the in-depth description of system operations, interlocks, and interconnections with other systems as well as containing techncially incorrect information on a few occasions. In addition, one complete section of simulator malfunctions was missing from the provided material. The following are a few specific examples of reference material deficiencies:

- System 17, Safety Injection, Lesson Plan did not contain detailed information concerning system operations, associated system interlocks, and interconnections with other plant systems.

- The entire section of malfunctions pertaining to the Main Steam System was missing from the Simulator Information Book. This section of malfunctions was provided to the examiners upon request.

- System 19, Reactor Protection, Lesson Plan, page 19 incorrectly states that the setpoint for a steam generator low level reactor trip is 23.5 percent. The actual setpoint is 14.8 percent. This was identified by the facility representatives during the pre-examination review of the written examinations.

The facility is encouraged to ensure that System Lesson Plans are technically correct, and consistent with Plant Operating Procedures to preclude training on technically incorrect and contradictory information.

During the quality assurance review of the grading of the written examinations, one generic knowledge deficiency on the part of the candidates was identified concerning the requirements and control of entry into radiologically controlled areas.

A common question to the RO and SRO examinations concerning requirements for entry into a radiologically controlled area was incorrectly answered by five out of seven candidates. The question as stated on the SRO examination was as follows:

Which ONE of the following correctly describes the entry precautions and limitations/requirements in accordance with HTP-ZZ-06001, "High Radiation (HRA)/Very High Radiation Areas (VHRA)"?

A. Personnel escorted by Health Physics are exempt from having a RWP for entry into an HRA.

B. All entries to areas inside the Biological Shield at power require a SRWP written specifically for that entry.

C. Entries to VHRA must have continuous Health Physics coverage.

D. Any individual that is permitted to enter a HRA is required to only have a TLD and a self-reading pocket dosimeter in their possession.

In accordance with HTP-ZZ-06001 Rev. 11, the only correct response is Choice A. Contrary to this, the candidates incorrectly chose a variety of the remaining three choices.

Additionally, three out of four SRO candidates incorrectly answered a question concerning the issuance of keys required to enter locked radiologically controlled areas. This question, which was used exclusively on the SRO examination, was stated as follows:

Which ONE of the following correctly describes how keys are controlled for issuance to locked areas in accordance with HTP-ZZ-06001, "High Radiation (HRA)/Very High Radiation Area (VHRA) Access"?

A. The Shift Supervisor shall authorize issuance of the key needed for entry into the incore instrument tunnel only after he verifies that the incore thimbles are fully inserted into the core and that the moveable incore detectors are in their stored positions, and WPA is hung on the supply breaker to the incore detector drive units.

B. Plant personnel may be issued keys to a HRA at the Key Issue Station after receiving verbal authorization from Health Physics.

C. A VHRA Access Request Form must be approved by the Health Physics Supervisor and the Shift Supervisor prior to issuance of a VHRA key for the Reactor Building.

D. Distributions of keys to areas under HRA key control shall be restricted to Health Physics personnel and the Shift Supervisors key ring.

In accordance with HTP-ZZ-06001, Rev. 11, the only correct response is Choice B. Contrary to this, the candidates incorrectly chose either Choice A or C.

The above two examples demonstrates a knowledge deficiency on the part of the candidates of the requirements and control of entry into radiologically controlled areas. The facility is encouraged to cover this area during scheduled requalification training to ensure that this knowledge deficiency is not generic to all licensed plant personnel.

During the administration of the Operating Examinations the following strengths in the candidates' performance were observed by the examiners in the majority of the candidates that were examined in each particular ability:

- The candidates' ability to reference and utilize system flow diagrams was very good.

- The candidates' ability to keep plant personnel outside of the control room informed of plant status by use of the plant page system was very good.

An additional comment is that the examiners felt that the plant general housekeeping program is doing a good job at maintaining plant appearance as well as plant cleanliness.

3. Exit Meeting

An exit meeting was held to discuss the aforementioned examiner observations as well as the simulation facility report, as contained in Section 5 of this report, on October 20, 1989, with facility management and training staff representatives.

NRC representatives in attendance were:

J. Lennartz, Examiner
B. Little, Senior Resident Inspector
V. Loughney, Examiner
K. Shembarger, Examiner
D. Shepard, Examiner

Facility representatives in attendance were:

G. Randolf, General Manager Nuclear Operations
D. Heinlein, Assistant Superintendent, Operations
J. Blosser, Manager Callaway Plant
W. Robinson, Assistant Manager, Operations and Maintenance
C. Naslund, Manager, Operations Support
N. Lombardi, Quality Assurance Engineer
M. Evans, Training Superintendent
D. Neterer, Senior Training Supervisor
P. McKenna, Training Supervisor
P. Shannon, Training Supervisor

The facility management acknowledged the examiners observations and simulation facility report as contained in Section 2 and Section 5, respectively, of this report.

4. Examination Review

Facility representatives were allowed to review the written examinations prior to their administration, and any applicable comments from the review were incorporated into the examinations.

Following the conclusion of the written examinations, the facility was given a copy of the RO and SRO examinations and answer keys. The facility then had until the end of the week of the examination administration to provide any additional comments in writing to the NRC.

The following paragraphs contain the facility comments concerning the examinations followed by the NRC response.

REACTOR OPERATOR EXAMINATION

Emergency and Abnormal Plant Evolutions Section

QUESTION: 10 (1.00)

Which ONE of the following describes the bases for manual rod insertion during an Anticipated Transient Without Scram (ATWS) event: (1.0)

a. Rod insertion is quicker in manual than in automatic.

b. Once the RCS temperature is below the reference temperature, rods will not step in automatically.

c. Manual rod insertion requires operator attention, which ensures rods are stepping into the core.

d. The design bases ATWS event assumed automatic rod control was inoperable. Therefore, manual rod insertion is required.

ANSWER: 10 (1.00)

b. (1.0)

CALLAWAY COMMENT:

The correct bases for manual control rod insertion is "if the reactor cannot be tripped, then the control rods should be manually inserted into the core in order to decrease reactor power." None of the four responses support this.

CALLAWAY RECOMMENDATION

Delete the question.

REFERENCE: Lesson Plan D-7, FR-5.1, Pg. 18

NRC RESPONSE:

Comment accepted. The question has been deleted from the examination.

QUESTION: 22 (1.00)

Which ONE of the following describes the operator actions to be taken based on the following plant parameter/indications: (1.0)

1. Reactor Power - 72%

2. RCP A vibration - 5 mils on the frame, 16 mils on the shaft

3. SEAL INJ TO RCP FLOW LO annunciator - lit

4. RCP No. 1 SEAL FLOW HI annunciator - lit

5. RCP A No. 1 seal inlet temperature - 213 degrees F

6. CCW inlet temperature to thermal barrier on RCP A - 95 degrees F

7. RCP A motor bearing temperature - 165 degrees F

a. Close RCP A No. 1 seal leak-off isolation valve BB-HV-8141A and continue pump operation.

b. Reduce power to less than 48% and trip RCP A.

c. Trip the rector and turbine, trip RCP A and refer to E-0, Reactor Trip or Safety Injection.

d. Attempt to reopen CCW isolation valve to RCP A motor.

ANSWER: 22 (1.00)

a. (1.0)

CALLAWAY COMMENT

Under indication No. 2, the 5 mils when read on a meter that is oscillating, may be read as a band .5 mils wide or more.

CALLAWAY RECOMMENDATION

1. Accept answer c. as the correct answer.

2. Change the parameters to either 4 or 6 as desired for the questions.

REFERENCE: OTO-BB-00002, Pgs. 3-5

NRC RESPONSE:

Comment not accepted.

The supplied reference is a systems lesson plan which is not controlled material, and not used to operate the plant. The off normal operating procedure OTO-BB-00002, "Reactor Coolant Pump Off-Normal," was used by the NRC to develop the question. Step 5.1.1 of OTO-BB-00002 requires the operator to trip the reactor and turbine and trip the affected RCP (answer Choice c) if vibration exceeds 5 mils on the frame, or 20 mils on the shaft. Contrary to this, indication No. 2 clearly states that RCP frame vibration is at 5 mils and an oscilalting band of .5 mils wide or more should not be assumed. Therefore, Choice c is incorrect and will not be accpepted. The facility is encouraged to ensure that system lesson plan technical content is consistent with actual plant operating procedures.

Consideration will be given to recommended rewording of the question for future examinations.

PLANT SYSTEMS AND PLANT-WIDE GENERIC RESPONSIBILITIES SECTION

QUESTION: 01 (1.00)

Which ONE of the following Intermediate Range (IR) Nuclear Instrumentation Channel indications would be the FIRST to allow the Source Range Channels to be blocked on a power increase: (1.0)

a. N35 reads 10 (-11) amps
   N36 reads 10 (-11) amps

b. N35 reads 10 (-10) amps
   N36 reads 10 (-11) amps

c. N35 reads 10 (-10) amps
   N36 reads 10 (-09) amps

d. N35 reads 10 (-09) amps
   N36 reads 10 (-09) amps

ANSWER: 01 (1.00)

c. (1.0)

CALLAWAY COMMENT:

Correct answer should be $\geq 10^{-10}$ amps.

CALLAWAY RECOMMENDATION:

1. Accept answer b. as the correct answer.

REFERENCE: Technical Specification, Pg. 2-6, Tab 6, 2.2-1

NRC RESPONSE:

Comment accepted. The answer key has been revised to accept b. as the only correct answer.

QUESTION: 04 (3.00)

Match the following NIS failures in COLUMN A with the associated affects on plant operation in COLUMN B. COLUMN B affects may be used more than once or not at all. COLUMN A failures may have more than one answer. (3.0)

COLUMN A	COLUMN B
a. 1 of 4 Power Range Channels Fail High _____.	1. High range reactor trip
b. 1 of 4 Power Range Channels Fail Low _____.	2. Low range reactor trip
c. 2 of 4 Power Range Channels Simultaneously Fail High _____.	3. High power rate trip
d. 1 Power Range Channel Fails High Simultaneously with 1 Power Range Low Channel _____.	4. Automatic rod withdrawl is prevented
	5. Manual rod withdrawl is prevented
	6. Will not affect plant operation

ANSWER: 04 (3.00)

a.	4.	(0.5),	5.	(0.5)
b.	6.	(0.5)		
c.	1.	(0.5)	3.	(0.5)
d.	3.	(0.5)		

CALLAWAY COMMENT:

1. Answer c., if reactor power is about 10% and power range low range reactor trip has not been blocked, then as power passes through 25%, a reactor trip will occur.

2. Answer c., as reactor power increases past 103% then automatic and manual rod withdrawal is prevented.

3. Answer d., as reactor power increases past 103% then automatic and manual rod withdrawal is prevented.

CALLAWAY RECOMMENDATION:

1. Accept answer #2 as correct for c.
2. Accept answer #4 and #5 as correct for c.
3. Accept answer #4 and #5 as correct for d.

REFERENCE: System 28, Excore Nuclear Instrumentation, Pg. 11-12

NRC RESPONSE:

Comment not accepted. The fact that initial power level was not stated combined with the ambiguous wording of the stem of the question, a variety of the recommended responses could be correct depending on the assumptions made by the applicant. Since the question, as written, does not adequately discriminate the intended knowledge that was being tested, Parts c and d have been deleted from the examination. The answer key has been revised to reflect this.

QUESTION: 16 (3.00)

Match the Emergency Diesel Generator component in COLUMN A with the associated purpose/function in COLUMN B. COLUMN B purposes/functions may be used more than once or not at all. COLUMN A components may have more than one associated purpose/funciton. (3.0)

COLUMN A	COLUMN B
a. Static Exciter Voltage Regulator _____.	1. Controls generator output voltage in Isochronous or Voltage Droop mode.
b. Air Shutoff Relay _____	2. Moves the fuel racks to the no fuel position.
c. Overspeed Trip Mechanism _____.	3. Secures starting air to the engine.
d. Manual Trip Lever _____.	4. Automatically Trips the generator output breaker after an emergency start.
e. Differential Current Relay _____	5. Controls the strength of the rotor field.
	6. Opens the generator room ventilation damper.
	7. Energizes the shutdown solenoid valve to stop the diesel.

ANSWER: 16 (3.00)

a. 5. (0.5), 1. (0.5)
b. 3. (0.5)
c. 2. (0.5)
d. 2. (0.5)
e. 4. (0.5)

CALLAWAY COMMENT

1. The Overspeed Trip Mechanism specifically does not move the fuel racks. It ports oil to move the fuel racks. If an overspeed occurs then the fuel racks move to no fuel, the output breaker trips and the shutdown solenoid valve energizes.

2. The Differential Current Pelay energizes the Engine Shutdown Relay. The Engine Shutdown Relay trips the breaker and energizes the shutdown solenoid valve.

CALLAWAY RECOMMENDATION

1. Accept answers 4 and 7 for c. also.
2. Accept answer 7 for e. also.

REFERENCE: System 3, Standby Generation, Pg. 23

NRC RESPONSE:

Comment not accepted. Since the Overspeed Trip Mechanism, Column A, Part C, specifically does not move the fuel racks to the no fuel position, Column B, Item 2 is an incorrect response for Part C. In addition, recommended correct responses 4 and 7 of Column B are functions of the Engine Shutdown Relay and not the Overspeed Trip Mechanism. Since there are no correct responses from Column B for the Overspeed Trip Mechanism, Part C has been deleted. The answer key has been revised to reflect this.

In accordance with System 3, Standby Generation, Page 20 and TP-22, the generator output breaker will trip automatically following an emergency start by either the Differential Current Relay or the Engine Shutdown Relay. Since recommended correct response 7 of Column B is a function of the Engine Shutdown Relay and not a direct function of Differential Current Relay, it will not be accepted as a correct response. Therefore, the only correct response that will be accepted for Column A, Item C will be Column B, Item 4. No revisions to the answer key are required for this Item.

QUESTION: 17 (1.00)

Which ONE of the following describes the effect of a loss of DC power to the Emergency Diesel Generator: (1.0)

a. On a diesel generator automatic start, the generator would not come up to rated voltage, since the generator field would not be established.

b. The voltage droop mode of operation would be inoperable. Therefore, reactive load sharing with the generator in parallel with off-site power would not be allowed.

c. Manual shutdown of the diesel generator with the exciter shutdown pushbutton on the Local Generator Control Panel would not be possible.

d. Starting air from the air receivers could not be manually supplied to the engine cranking. Therefore, the diesel generator would not start.

ANSWER: 17 (1.00)

a. (1.0)

CALLAWAY COMMENT:

1. There are 4 DC control power circuits on the Emergency Diesel.

2. When there is no DC control power to the diesel the Diesel Start Circuit is disabled.

CALLAWAY RECOMMENDTION:

Void the question as there is no correct answer.

REFERENCE: OTA-KJ-00121

NRC RESPONSE:

Comment accepted. The question has been deleted.

SENIOR REACTOR OPERATOR EXAMINATION

EMERGENCY AND ABNORMAL PLANT EVOLUTIONS SECTION

QUESTION: 09 (1.00)

The URO manually opened the "A" Emergency Diesel Generator (EDG) output breaker following a start of the EDG due to a pressurizer pressure of 1829 psig. Thirty seconds later the "A" EDG received a start signal due to an undervoltage condition on bus NB01. Bus NB01.....(Choose One)

A. .....will be energized by automatic closure of the "A" EDG output breaker if the local AUTO/MANUAL transfer switch is in AUTO.

B. .....will be energized by automatic closure of the "A" EDG output breaker after the local AUTO/MANUAL transfer switch is placed in MANUAL.

C. .....can be energized by manually closing the "A" EDG output breaker from the Control Room after the local AUTO/MANUAL transfer switch is placed in MANUAL.

D. .....can be energized by manually closing the "A" EDG output breaker locally at the breaker after the local AUTO/MANUAL transfer switch is placed in MANUAL.

ANSWER: 09 (1.00)

D. (1.0)

CALLAWAY COMMENT:

The EDG breaker does not close on a SIS.

CALLAWAY RECOMMENDATION:

Void the question because the EDG breaker could not have been shut to allow the RO to open it.

REFERENCE:

E23-KJ01A T61/0810.8 Lesson Plan TP
E23-NE01 (18x24 sheets located at end of package)

NRC RESPONSE

Comment accepted. The question has been deleted.

QUESTION: 10 (1.00)

While the plant was operating at 75% power, a turbine trip coincident with a Pressurizer Pressure Control System malfunction resulted in an increase in pressurizer pressure to 2745 psig. (Assume that all other plant systems responded as designed.) In accordance with Technical Specifications .....(Choose One)

A. .....within 1 hour reduce pressure to within its limit and be in HOT STANDBY.

B. .....within 5 minutes reduce pressure to within its limit and be in HOT STANDBY.

C. .....within 1 hour reduce pressure to within its limit and be in HOT SHUTDOWN.

D. .....within 5 minutes reduce pressure to within its limit and be in HOT SHUTDOWN.

ANSWER: 10 (1.00)

A. (1.0)

CALLAWAY COMMENT:

The question states "Assume that all other plant systems respond as designed." Therefore, the reactor has received 2 valid trip signals; (a) the turbine tripped with Rx power > P-9; and (b) RCS pressure is > 2385 PSIG. The question states "Pressure Control Malfunction not Protection CKTS. The Rx would be in MODE 3 after the reactor trip.

CALLAWAY RECOMMENDATION

1. Selection B should be the correct answer for the given conditions.

2. Recommend reword stem to make it clear that the reactor remains in Mode 1 and 2. Possibly include ATWS.

REFERENCE:

1. Technical Specification 2.1.2, Mode 3, Rev. 1 (Attached)

2. E-0 Rx Trip or Safety Injection, Attachment 1, Page 1 of 2, Rev. 1A0 (Attached)

NRC RESPONSE

Comment accepted. The answer key has been modified to reflect this. Consideration will be given to the recommended rewording of the question for future examinations.

QUESTION: 22 (1.00)

The following pertinent plant parameters exist:

Reactor Power - 72%
RCP A Vibration - Stable at 5 mils frame, 16 mils shaft
SEAL INJ TO RCP FLOW LO annunciator - Lit
RCP No.1 SEAL FLOW HI annunciator - Lit
RCP A No. 1 Seal Inlet Temperature - 213 Deg. F
CCW Inlet Temperature to Thermal Barrier on RCP A - 95 Deg. F
RCP A Motor Bearing Temperature - 165 Deg. F

The IMMEDIATE ACTIONS that should be taken are . . . . (Choose One)

A. Close RCP A No. 1 seal leak-off isolation valve BB-HV-8141A and continue pump operation.

B. Reduce power to less than 48% and trip RCP A.

C. Trip the reactor and turbine, trip RCP A, and refer to E-0, "Reactor Trip or Safety Injection."

D. Attempt to re-open CCW isolation valve to RCP A motor.

ANSWER: 22 (1.00)

A. (1.0)

CALLAWAY COMMENT

Under indication No. 2, the 5 mils when read on a meter that is oscillating, may be read as a band .5 mils wide or more.

CALLAWAY RECOMMENDATION

1. Accept answer C. as the correct answer.

2. Change the parameters to either 4 or 6 as desired for the questions.

REFERENCE:

System 9, Reactor Coolant, Pg. 18 (Attached).

NRC RESPONSE

Comment not accepted. The supplied reference is a systems lesson plan which is not controlled material, and not used to operate the plant. The off normal operating procedure OTO-BB-00002, "Reactor Coolant Pump Off-Normal," was used by the NRC to develop the question. Step 5.1.1 of OTO-BB-00002 requires the operator to trip the reactor and turbine and trip the affected RCP (answer Choice C) if vibration exceeds 5 mils on the frame, or 20 mils on the shaft. Contrary to this, indication No. 2 clearly states that RCP frame vibration is stable at 5 mils and an oscillating band of .5 mils wide or more should not be

assumed. Therefore, Choice C is incorrect and will not be accepted. The facility is encouraged to ensure that system lesson plans technical content is consistent with actual plant operating procedures.

Consideration will be given to recommended rewording of the question for future examinations.

PLANT SYSTEMS AND PLANT WIDE GENERIC RESPONSIBILITIES SECTION

QUESTION: 09 (2.00)

MATCH each of the automatic actions listed in COLUMN A with its associated VCT level setpoint listed in COLUMN B. (NOTE: Items listed in COLUMN B may be used more than once or not at all.)

COLUMN A	COLUMN B
_____a. LCV-112A (Divert Valve) FULL OPEN TO RHT	1. 5%
_____b. LCV-112A (Divert Valve) STARTS TO OPEN to RHT	2. 6%
_____c. VCT Low Level Alarm	3. 10%
_____d. Permits RESET of Charging Pump Suction Valve Logic	4. 20%
	5. 22%
	6. 25%
	7. 70%
	8. 75%
	9. 78%
	10. 90%
	11. 95%
	12. 97%

ANSWER: 09 (2.00)

a. 12
b. 7
c. 6
d. 2 (0.5 each)

CALLAWAY COMMENT:

The low level alarm on the VCT is 22%.

CALLAWAY RECOMMENDATION

Make the correct answer for c No. 5.

REFERENCE:

OTA-RL-RK042, Att. B (Attached).

NRC RESPONSE

Comment accepted. The answer key has been revised to reflect this.

QUESTION: 14 (2.00)

MATCH each of the Reactor Trips listed in COLUMN A with the associated protection(s) it is designed to provide listed in COLUMN B. (NOTE: Items in COLUMN B may be used more than once or not at all.)

COLUMN A	COLUMN B
_____a. OPDeltaT	1. Single Rod Drop
_____b. PZR High Level	2. Single Rod Ejection
_____c. High Negative Flux Rate	3. Multiple Rod Drop
	4. Excessive KW/ft
	5. DNB
	6. Pressurizer Safety Valves Integrity
	7. Pressurizer PORVs Integrity
	8. Startup Accident
	9. RCS Over Pressure
	10. Over Power

ANSWER: 14 (2.00)

a. 4
b. 6
c. 3, 5 (0.5 each)

CALLAWAY COMMENT:

1. Pressurizer level is a backup trip for the high pressure reactor trip.

CALLAWAY RECOMMENDATION

1. Add to the correct answers on b No. 9.

REFERENCE:

Lesson Plan No. 27, Reactor Protection, Pg. 18.

NRC RESPONSE

Comment accepted. The answer key has been revised to accept No. 9 as an additional required correct response for b. Each correct response is worth 0.5 points.

QUESTION: 29 (2.00)

Match each of the Work Request types listed in COLUMN A with their correct definition(s) in COLUMN B. (There may be more than one correct definition in COLUMN B for each Work Request type listed in COLUMN A. Some of the definitions in COLUMN B may not be used at all.)

COLUMN A	COLUMN B
1. Priority 1	a. maintenance actions taken to protect property
2. Priority 2	b. work requests that require a load reduction to complete
3. Priority 3	c. repairs needed to eliminate a personnel safety hazard
4. Priority 4	d. applies to scheduled work that needs to be planned in accordance with the normal target schedule weeks
	e. repairs required based on an existing condition which has a high probability of impacting a systems function.
	f. repairs needed to restore a system to operable status to decrease the risk of radiation exposure to the public.

ANSWER: 29 (2.00)

1. c, f (0.5 each)
2. d (0.5)
3. a (0.5)
4. b (0.5)

5. SIMULATION FACILITY REPORT

Facility Licensee: Callaway

Facility Licensee Docket No. 50-483

Operating Tests Administered At: Callaway Training Center

During the conduct of the simulator portion of the operating tests, the following items were observed (if none, so state):

ITEM	DESCRIPTION
1.	Simulator setup initial conditions (IC) 29 and 30 would result in the Heater Drain Pumps tripping immediately upon taking the simulator to run, making these ICS too unstable to use for the examinations as programmed.
2.	Simulator malfunction EPS-7 provides for a loss of 125 VDC distribution bus NK04. This action should result in a loss of 120 VAC Instrument Bus NN04 also, since NK04 supplies the inverter which feeds instrument bus NN04. Contrary to this, the loss of NK04 did not result in a loss of NN04.
3.	Low failure of radiation process monitors incorrectly resulted in interlock functions such as Control Room Ventilation Isolation, Containment Purge Isolation, and Fuel Handling Building Ventilation Isolation to occur. This incorrect simulator response had already been identified by the facility and corrective actions were taken prior to administration of the simulator portion of the operating examinations.