U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-315/OL-91-02

Docket No.s. 50-315; 50-316 Licenses No. DPR-58; DPR-74

Licensee: Indiana Michigan Power Company
1 Riverside Plaza
Columbus, OH 43216

Facility Name: D. C. Cook

Examination Administered at: D. C. Cook Plant

Examination Conducted: Initial Examination and Requal Retakes

Chief Examiner: T Burdick for 8/2/91
J. R. Walker Date

Approved By: T Burdick 8/2/91
T. Burdick, Chief Date
Operator Licensing Section 1

Examination Summary

Examination administered on the week of June 24, 1991, (Report No. 50-315/OL-91-02 (DRS)).
Written and operating examinations were administered to five Reactor Operator (RO) candidates, three Senior Reactor Operator (SRO) upgrade candidates and one Senior Reactor Operator (SRO) instant candidate. Additionally, requalification retake examinations were administered to two SRO and one RO who had failed the most recent NRC administered requalification examination.
Results. Two RO candidates and all three SRO upgrade candidates who were administered initial examinations passed. One RO failed the written examination and two ROs failed the operating portion of the examination. The one SRO instant failed both the written and operating portions of the examination. Additionally, the two SROs and one RO who were administered the requalification retake examinations passed.

SAFETY CONCERN

During the administration of the Simulator portion of the examination one practice was identified by the examiners as a safety concern. When the facility was asked about it, it was noted as a accepted practice by the facility. The event is as follows:

During a simulator scenario with the normal feed to one emergency bus out of service and the diesel supplying that bus also out of service, the SRO on all three crews examined, ordered the second diesel tripped, placing it out of service. This was done after an attempt to secure the diesel and place it in standby during emergency procedures with a Safety Injection in progress due to a Faulted/Ruptured Steam Generator.

These actions were taken based upon a precaution which used to appear in the normal operating/surveillance procedures. This action does not seem to be critical to equipment protection, while maintaining an operable diesel under the as stated conditions is critical to Reactor Safety. Securing and defeating automatic safeguards functions seems unnecessary and ill-advised. This concern was first brought to the attention of the facility during a routine safety inspection conducted by Division of Reactor Projects during the fall of 1989. Inspection report numbers (50-315/89029(DRP) and 50-316/89029(DRP)).

REPORT DETAILS

1. Examiners

   *J. Walker, NRC
   J. Lennartz, NRC
   A. Lopez, PNL
   E. Benjamin, PNL

   *Chief Examiner

2.. Exit Meeting

   An exit meeting was held to discuss the examiner's observations on June 28, 1991, with facility management and training staff representatives.

   NRC representatives in attendance were:

   J. Walker
   J. Lennartz
   J. Isom, Senior Resident Inspector
   D. Passehl

   Facility representatives in attendance were:

   E. V. Kincheloe, Training Superintendent
   R. K. Anderson, Requalification Program Administrator
   W. Nelson, Instructor
   J. Stubblefield, Program Administrator
   B. Nichols, Operations Training Supervisor
   B. Burgess, Simulator Supervisor
   K. Baker, APM-Production
   M. Mierau, STA Supervisor
   G. Arent, Ops Training Specialist
   J. Sampson, Ops Superintendent

   The facility management acknowledged the examiner's observations discussed further in section 2 of this report.

   Examiner Observations

   a. Examination Development

      The reference material that the licensee sent to the NRC for examination preparation was all properly bound and labeled, and for the most part, the NRC examiners were able to extract the needed information.

      The pre-examination review conducted by the licensee on the written examinations was very productive. The licensee's input to the examination ensured that the terminology used on the examination was plant specific, thus avoiding confusion on the part of the candidates during the examination. In addition, the review process insured that the examinations were technically correct and appropriate for each license type as specified by the licensee's job description.

b. Operating Examination Administration

During the administration of the examinations, the examiners observed both deficiencies and strengths on the part of the Senior Reactor Operator and Reactor Operator candidates.

The following deficiency in the candidates' performance were observed in the majority of the candidates that were examined:

1. Failure to diagnose some of the events on the simulator portion of the exam as a crew.

2. Ability to interpret status messages provided by the Radiation Monitoring system.

3. Knowledge of what logics were involved in Reactor Protection and Safeguards actuation.

The following strengths in the candidates' performance were observed in the majority of the candidates that were examined.

1. All crews examined made good use of alarm response procedures.

2. Communications of crew members in the simulator was good utilizing a feedback method of communication.

3. Most candidates showed a thorough knowledge of the location of equipment on the control boards.

c. Safety Concerns

During the administration of the Simulator portion of the examination one practice was identified by the examiners as a safety concern, when the facility was asked about it, it was noted as a accepted practice by the facility. The event is as follows:

During a simulator scenario with the normal feed to one emergency bus out of service and the diesel supplying that bus also out of service, the SRO on all three crews examined, ordered the second diesel tripped, placing it out of service. This was done after an attempt to secure the diesel and place it in standby during emergency procedures with a Safety Injection in progress due to a Faulted/Ruptured Steam Generator.

These actions were taken based upon a precaution which used to appear in the normal operating/surveillance procedures. This action doesn't seem to be critical to equipment protection, while maintaining an operable diesel under the as stated conditions is critial to Reactor Safety. Securing and defeating automatic emergency response safeguards functions seem unnecessary and ill-advised. This concern was first brought to the attention of the facility during a routine safety inspection conducted by Division of Reactor Projects during the fall of 1989. Inspection report numbers (50-315/89029(DRP) and 50-316/89029(DRP)).

3. Written Examination Administration

The post-exam review identified the following deficiencies in the candidates' knowledge as evidenced by the majority of the candidates failing to provide the correct response for each particular knowledge area examined. This information is being provided as input to the licensee System Approach to Training (SAT) process. No response is required.

1. Failure to understand the relationship between the P-4 interlock, Reactor Trip Breakers and the timing sequence for resetting Safety Injection. (RO question 21, SRO question 12).

2. Failure to understand what condition would actuate steam dumps. (RO question 21, SRO question 13).

3. Failure to understand what signals will actuate a Main Feedwater Isolation. (RO question 37, SRO question 19).

4. When the NA notation is appropriate to be used in procedures. (RO question 96, SRO question 74).

5. Failure to understand what signal will cause an automatic start of the Auxiliary Feedwater Pumps. (RO question 39, SRO question 14).

6. Failure to demonstrate what criteria is required to start the first Reactor Coolant Pump. (RO question 8, SRO question 18).

7. Failure to understand the inputs to the Power Range Channel Comparator. (RO question 25, SRO question 26).

8. Failure to understand the results of connecting the SFP and the RWST during cleanup operations. (RO question 32, SRO question 28).

9. Failure to demonstrate what the Immediate Operator actions are for a loss of RHR. (RO question 61, SRO question 39).

10. Failure to recognize the indications of a failure of the number 2 seal on a Reactor Coolant Pump. (RO question 59, SRO question 52).

11. Failure to recognize what conditions required as emergency boration of the RCS. (RO question 60, SRO question 32).

12. Failure to recognize who could and could not operate reactor controls under the supervision of a licensed operator. (RO question 14, SRO question 71).

13. Inability to determine when a licensed operator can return to work. (SRO question 98).

4. Written Examination Review

Licensee representatives reviewed the written examination prior to administration and any accepted comments were incorporated into the examination at that time. Additionally, following the conclusion of the written examinations, the licensee was given a copy of the RO and SRO examinations and answer keys. The licensee then had until the end of the examination week to provide any additional comments in writing to the NRC along with justification references. These facility comments and the NRC resolutions are contained in Enclosure 2 of this report.

ENCLOSURE 2

FACILITY COMMENTS AND NRC RESOLUTION OF COMMENTS

Written Examination Review

Facility representatives reviewed the written examinations prior to the administration as discussed in Section 2.a of this report, 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 Senior Reactor Operator and Reactor Operator examinations and answer keys. The facility then had until the end of the examination administration week to provide any additional comments in writing to the NRC.

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

1. RO Question 3/SRO Question 17:
During operation at 100% power, the Startup Reset (Full Length Rod Control Reset) is inadvertently operated. WHICH ONE (1) of the following describes the affect of this action?

A. The plant will NOT respond to a dropped rod in control banks B, C, or D.

B. Rod stop interlock C-11 CANNOT prevent further control bank D outward rod motion if rods move in automatic.

C. "Urgent Failure" alarm will actuate.

D. Annunciator alarm "Rod Sequence Violation" will actuate.

Answer: B

Facility Comment:
Request that answer (D) be accepted for full credit in addition to the keyed answer. OHP-4024.110 Drop 29, "Rod Sequence Violation" states that this alarm is actuated with "Rod position deviation of +/- 12 steps from demand counter". With the demand counters reset to 0 and IRPI at their 100% value the alarm will be actuated. The actuation of this alarm was tested using our site specific simulator. A print of this test is enclosed.

NRC Resolution:

Comment accepted. Answer (d) will be accepted for full credit in addition to the keyed answer of (b).

2. RO Question 17:

WHICH ONE (1) of the following describes the plant response to pressurizer pressure channel NPP-152 failing HIGH, while selected as the bistable channel? ASSUME plant at 100% power and no operator action is taken.

A. The spray valves will shut when pressure reaches the interlock channel setpoint.

B. The PORV block valve will shut when pressure reaches the interlock channel setpoint.

C. PORV valve NRV-152 will open when another pressure transmitter exceeds the high pressure setpoint.

D. A pressurizer high pressure alarm actuates and the reactor will trip on low pressure.

Answer: D

Facility Comment:
Request that question and answer be deleted from the examination as there are no correct answers. Refer to the attached logic print. With NPP 152 selected for bistable control and failed high the only pressure reduction device affected is NRV 153. Answer (A) is incorrect because the spray valves are not affected by this failure. Answer (B) is incorrect because the PORV block valves receive NO automatic signals and will not change position as a result of this failure. Answer (C) is incorrect because two additional channels would have to reach their high setpoints in order for NRV 152 to actuate. Answer (D) is incorrect because the initial failure of NPP 152 will not affect spray or heater operation and therefore the reactor will not trip on low pressure.

NRC Resolution:
Comment accepted. Question will be deleted from the exam as there are no correct answers.

3. RO Question 18:

WHICH ONE (1) of the following is the set of pressures that will energize the proportional (variable) and backup heaters on Unit 2, respectively, on a DECREASING pressure signal? ASSUME pressure control is set for normal operation.

A. 2220 psig, 2210 psig.

B. 2250 psig, 2210 psig.

C. 2250 psig, 2218 psig.

D. 2220 psig, 2218 psig.

Answer: B

Facility Comment:
Request that answer (A) be accepted for full credit in addition to the keyed answer. If pressurizer pressure is set for normal operation as stated in the question then the variable heaters will already be energized at some nominal value. This condition would make answer (a) correct as the variable heaters would be fully energized when system pressure decreased to 2220 psig. Answer (B) would be correct if the student assumed that the RCS pressure was higher than 2250 at the start of the transient. Had the question stated "with pressurizer pressure at 2275 psig and slowly decreasing, WHICH ONE (1)..." OR if the "ASSUME pressurizer pressure control is set for normal operation" statement had been omitted, the keyed answer would be the only correct answer.

NRC Resolution:
Comment not accepted, question deleted. Due to the lack of clarity, as to what was being asked the question is being DELETED.

4. RO Question 21/SRO Question 12:

The following plant conditions exist:

- Reactor at full power
- Reactor protection system (RPS) testing in progress
- Train B reactor trip breaker open
- Train B bypass breaker closed
- Train A reactor trip breaker closed
- Train A bypass breaker open

WHICH ONE (1) of the following is the correct system response IMMEDIATELY following a spurious reactor trip signal and Bypass Breaker B fails to open?

A. If an SI were to occur, manual reset would NOT be possible.

B. Main Feed Pumps trip and feedwater regulating valves trip close.

C. The Turbine Generator must be manually tripped.

D. Steam dumps receive and open signal, but do NOT arm.

ANSWER: A

Facility Comment:
Request that answer (b) be accepted for full credit in addition to the keyed answer. One train of safety injection can be reset therefore keyed answer (a) is not fully correct as written. Answer (b) is as correct as answer (a) in that both main feed pumps will trip (P-4 train independent 98212) and the feed water regulating valves will close when RCS temperature decreases to the Low TAVE value of 554 degree F (P-4 independent). Drawing OP-98512 for answer (a) and OP-98508/OP-98212 for answer (b) are included and annotated to support the comments.

NRC Resolution:
Comment not accepted. Answer (b) will not be accepted for full credit in addition to keyed answer. Keyed answer (a) is correct as stated. SI cannot be reset IMMEDIATELY following a reactor trip due to a 60 second timer preventing reset.

5. RO Question 51/SRO Question 58:

The following plant conditions exist on unit 1:

- Reactor is at 100% power
- Bank D rods at 198 steps

- Shutdown rod K10 and control rod H8 have just dropped

WHICH ONE (1) of the following actions are you required to take?

A. Determine that the Shutdown Margin is greater than or equal to 1.6 % Delta k/k within 1 hour.

B. Take actions to shutdown the reactor to Hot Standby.

C. Take actions per 1-OHP 4022.012.005, "Full Length Rod Misalignment".

D. Restore rod H8 to within the insertion limit specified in the COLR within 1 hour.

ANSWER: C

Facility Comment:
Request that answer (b) be accepted for full credit. OHP 4022.012.004 step 5.1 (Subsequent Actions) states "If two or more rods have dropped, shut down to Hot Standby within 6 hours in accordance with OHP 4021.001.003, Power Reduction. Request that the keyed answer (c) not be accepted for full credit as the procedure OHP 4022.012.004 Full Length Rod Misalignment is not applicable for this situation. Request that Keyed answer (a) be accepted for full credit. OHP 4022.012.004 step 5.5 states, "Declare the rod inoperable, and verify shutdown margin within 1 hour..." Also refer to Technical Specification 3.1.3.1 action statement c.2. The question asks for required actions NOT which action should be accomplished first.

NRC Resolution:
Question will be deleted from exam due to three answers being correct. Keyed answer (c) was correct, but due to facility pre-exam review, a stem change resulted in keyed answer not being only correct answer. Answers (a) and (b) also correct.

6. RO Question 89/SRO Question 70:

WHICH ONE (1) of the following items describes one requirement at DC Cook for operations log entries?

A. Technical Specification entries must be in red ink with the exception of the isolation of the CO2/Halon suppression system for personnel access.

B. Errors in all log entries shall be voided by a single line and signed by the individual voiding the entry.

C. Action entries shall be noted on the unit status board and logged when plant conditions allow.

D. A late entry must be made within (1) hour of the time the event occurred and the initials LE shall be written next to the entry time.

Answer: A

Facility Comment:
Request that answer (b) be accepted as full credit in addition to the keyed answer (a). Voided entries are to be initialed in accordance with OHI 2211 step 3.1.3, however, signing the entry is acceptable as well. No references provided.

NRC Resolution:
Comment accepted.

7. RO Question 78/SRO Question 38:

The Unit 1 Main Generator does NOT trip on loss of 250 vdc bus 1CD because power is lost to WHICH ONE (1) of the following?

A. Exciter field breaker.

B. Master trip solenoids.

C. Overall and Unit Differential HEAs.

D. Control power for breakers K and K1.

Answer: C

Facility Comment:
Request that this question be deleted as there are not correct answers. OHP 4023.082.002CD step 2.4 (discussion) states "relays which signal the overall and Unit Differential HEAs of a turbine trip lose power and no automatic trip of the generator will occur. A generator trip can be accomplished by depressing the EMERGENCY UNIT TRIP PUSHBUTTON on the generator panel". The loss of 1CD DC power only affects the overall HEA. The unit HEA is powered from switchyard DC. Therefore, the generator failure to trip is a result of the loss of actuation signal NOT the loss of power to BOTH HEAs as stated in answer (c).

NRC Resolution:
Comment not accepted. Keyed answer (c) is correct as stated. On a loss of 250V DC bus 1CD, power is lost to the relays energizing the Overall and Differential HEAs preventing a generator trip. Procedure 1-OHP 4023.082.002CD LOSS OF POWER TO 250 VDC BUS 1CD step 2.4 states the following: " Relays which signal the Overall and Unit Differential HEAs of a turbine trip lose power and no automatic trip of the turbine generator will occur.

8. RO Question 52/SRO Question 46:

The following plant conditions exist on unit 1:

- Power range channel 42 upper drawer has failed high.
- Channel has been removed from service is accordance with OHP 4022.013.006 "Tripping of Protection Set Bistables".
- Pressurizer pressure channel NPP-151 has just failed low.

- Reactor power is at 85% and increasing.

WHICH ONE (1) of the following actions are you required to take?

A. Per Technical Specifications 3.1.2.4 "Charging Pumps" declare the East CCP inoperable due to Mini flow valve (QMO-225) inoperability.

B. Ensure all bistables associated with the pressure channel failure are placed in the trip position within 1 hour.

C. Turn off the heaters and manually open the spray valves to maintain pressure.

D. Initiate manual reactor trip, if unsuccessful, enter FR-S.1, "Response to Nuclear Power Generation/ATWS".

Answer: D

Facility Comment:
Request that answer (a) also be accepted as a correct response. In accordance with the "Tripping of Protection Set Bistables" procedure attachment 2 page 1 of 3 step 5.1.1a. This is true for any mode which the CCP is required to be operated. Answer (d) is correct because the trip logic for overtemperature delta temperature has been met. The questions asks for required actions NOT which required action should be accomplished first.

NRC Resolution:
Question will be deleted from exam due answer (b) also being correct as a subsequent action. This is a required step that must also be done. This gives the question 3 correct answers so question will be delated.

SIMULATION FACILITY REPORT

Facility Licensee: D. C. Cook Nuclear Power Plant

Facility Licensee Docket No. 50-315/50-316

Operating Tests Administered On: June 24-28, 1991

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

ITEM	DESCRIPTION
NONE	