U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-315/OL-89-01; 50-316/OL-89-01

Docket Nos. 50-315; 50-316 Licenses No. DRP-58; DRP-74

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

Facility Name: D.C. Cook

Examination Administered At: D.C. Cook

Examination Conducted: Senior Reactor Operator and Reactor Operator

T Burdick for | 8-31-89
Jay A. Lennartz, Examiner | Date

T Burdick for | 8-31-89
K. M. Shembarger, Examiner | Date

J. Hopkins | 8-31-89
J. Hopkins, Inspector/Examiner | Date

Chief Examiner: T Burdick for | 8-31-89
Douglas L. Shepard | Date

Approved By: Thomas M Burdick | 8-31-89
Thomas M. Burdick, Chief
Operator Licensing Section 2 | Date

Examination Summary

Examination administered on July 10-19, 1989 and inspection conducted August 14-17, 1989 (Report No. 50-315/OL-89-01; No. 50-316/OL-89-01)
Areas Inspected: Written and operating exams were administered to six senior reactor operators and eleven reactor operators. In addition to the examination a special inspection was conducted during the week of August 14, 1989, to review your initial license training program and candidate evaluation program.

Results: One senior reactor operator and nine reactor operators passed the examination. Generic weaknesses identified by the examiners were: (1) the delay in the performance of immediate actions during the emergency procedure implementation, (2) lack of implementation of caution statements in the emergency procedures, and (3) mis-diagnosis of events. Generic strengths identified by the examiners during administration of the exams were: (1) the candidates demonstrated good control of bistable tripping during the simulator exams, and (2) the candidates were very knowledgeable in Group 1 topics of the Administrative Section of the oral exams. The inspection findings included a potential programmatic weakness in the area of student evaluations.

REPORT DETAILS

1. Examiners

   N. Maguire-Moffitt, PNL
   J. Lennartz, NRC
   D. Shepard, NRC
   K. Shembarger, NRC
   T. Burdick, NRC

2. Exit Meeting

   Exit meetings were held on July 19, 1989, and August 17, 1989 with facility management and training staff representatives.

   NRC Representatives in Attendance Were:

   July 19, 1989

   D. Shepard, Chief Examiner
   J. Lennartz, Examiner
   K. Shembarger, Examiner
   N. Maguire-Moffitt, PNL Contractor Examiner
   B. Jorgensen, Senior Resident Inspector

   August 17, 1989

   J. Hopkins, Inspector/Examiner
   B. Jorgensen, Senior Resident Inspector

   Facility Representatives in Attendance Were:

   July 19, 1989
   R. Gillespie, Operations Training
   G. McCullough, Training/Simulator
   J. Stubblefield, Training
   S. Wolf, Quality Assurance Auditor
   R. Simms, STA Supervisor
   H. Runser, Operations Production Supervisor
   K. Baker, Operations Superintendent
   G. Arent, Operations Training Specialist
   J. Sampson, Safety and Assessment Superintendent
   W. Smith, Plant Manager

   August 17, 1989
   W. Smith, Plant Manager
   A. Blind, Assistant Plant Manager
   L. Matthias, Administrative Superintendent/Training Manager
   J. Kirt Kouski, Assistant Plant Manager

R. Simms, Station Supervisor
S. Wolf, AEPSC Site QA Auditor
J. Sampson, Safety & Assessment Superintendent
W. Nichols, Operations Training Supervisor

The examiners observed the following generic strengths during the administration of the examinations: (1) the candidates demonstrated good control of bistable tripping during the simulator exams, and (2) the candidates were very knowledgeable in Group 1 topics of the Administrative Section of the oral exams.

The examiners also observed the following generic weaknesses during the administration of the examinations: (1) the delay in the performance of immediate actions during emergency procedures implementation during the simulator exams, (2) lack of implementation of caution statements in the emergency procedures during the simulator exams, and mis-diagnosis of events.

To aid in the development of future examinations, a book of indexes should be included in the submittal of the examination reference material. In addition, mechanical and electrical drawings should have an index to allow efficient use of NRC resources during the development of examinations.

The examiners and facility training representatives identified simulator fidelity problems during scenario validation and during administration of the simulator portion of the operating examination. A list of the specific weaknesses was given to the facility training representatives and is included as enclosure 4 to this report.

The facility did not believe that the NRC should be asking questions regarding refueling for the Senior Reactor Operator candidates. The facility stated that only specific SRO's that have training in refueling are allowed to perform in the position of refueling SRO and that SRO candidates are not trained in refueling topics. The NRC's position is that the licenses which are issued to the D. C. Cook SRO's are not restricted licenses but conform to the regulations. 10 CFR 55.41(b)(13), 55.43(b)(6), 55.45(a)(10) and (12) apply to the operator's knowledge in this area. K/A references to fuel handling systems are found in NUREG 1122, 3.11.

3. Simulator Examinations

The simulator portion of the examinations conducted resulted in a 35 percent failure rate. The region conducted a follow-up inspection during the week of August 14, 1989 to determine the root cause of these failures. The inspection findings concluded that there was no programmatic breakdown or major flaws in the licensee's simulator training program. The inspector determined that the major weakness in the program could best be characterized by a need to "tune" the simulator training program and gain experience in NRC simulator examinations.

The inspection identified four minor weaknesses in the training program. While it is important to note that these weaknesses could not be directly linked to individual failures or to the generic weaknesses identified in paragraph 1 of this report they may have contributed to the failures.

1. The licensee conducted formal evaluations of the candidates during the last three weeks (weeks 5, 6, 7) of the simulator training. This evaluation consisted of two or three scenarios which reviewed training material presented in the previous weeks. The crew's performance (3 operators) was evaluated by one floor instructor and one simulator booth instructor during weeks 5 and 6 and by two floor instructors and one simulator booth instructor during week 7. Additionally, during weeks 5 and 6 the same two instructors acted as the evaluators at the end of the week. Week 7 used two different evaluators. The inspector concluded that all three weeks of evaluation should have at least two floor evaluators and that they should not have been instructors that week. This would permit a more detailed observation of each candidate's actions and encourage a more objective evaluation of the candidate's performance. The licensee has agreed to use more evaluators which were not the candidates' instructors as personnel scheduling permits.

2. The evaluation scenarios referred to above did not always expose the candidates to the same conditions an NRC examination would. The principal weaknesses were (1) the scenarios did not routinely exercise the CAUTIONS/NOTES and Response Not Obtained (RNO) Column of the Emergency Operating Procedures (EOPs) and (2) the Emergency Contingency Actions (ECAs) were not frequently included in scenario development. The licensee has agreed to review their scenarios and review other facility simulator examinations to broaden the scope of the evaluation scenarios.

3. Instant Senior Reactor Operator (SRO-I) license candidates were not routinely evaluated as control board operators. Two of the SRO-I candidates were evaluated as control board operators once (one during week 6 and the other during week 7) and one was never evaluated as a board operator. The licensee has agreed to review this area to determine if it is a programmatic deficiency.

4. The licensee evaluated the candidates performance using the criteria in NUREG-1021, "Operator Licensing Examiner Standards", revision 5, section 301. This evaluation process compares the candidates observed performance to descriptive behavioral anchors for each required competency. The floor evaluators would document and remediate any individual or crew deficiency during the evaluation debriefing after the scenario. However, the licensee's simulator training program did not group all of the operators evaluations to look for generic or programmatic weaknesses. For example, the inspector reviewed the individual simulator evaluation documentation for 14 of 17 of the candidates for weeks 5, 6 and 7. During week 5,

six candidates had minor problems with procedure usage; week 6 identified eight operators with procedure usage problems; week 7 had ten operators with procedure usage problems. Each of the identified problems with procedure usage was minor in nature with one or possibly two examples per operator and was remediated during the debriefing. This problem with procedure usage was identified by the examiners as a generic weakness. The inspector concluded that the licensee should group the candidates performance to identify possible generic weaknesses as well as continue with their individual evaluation. The licensee has agreed to review this area to determine if it is a programmatic deficiency.

4. <u>Examination Review</u>

At the completion of the written examination, the facility was provided a copy of the examination and answer key for both the SRO and RO examinations. The facility provided written comments concerning the examination to the NRC on July 14, 1989. The following enclosure contains the facility comments concerning the examinations, followed by the NRC resolution.

Enclosure 2

REACTOR OPERATORS EXAMINATION

QUESTION 2.06/5.02 (3.00)

Match the control rod malfunction event in COLUMN B with the associated symptoms of the event in COLUMN A. COLUMN A may have more than one answer. (3.00)

COLUMN A

a. Decreasing Tavg _____

b. Rod Insertion Limit Alarm _____

c. Tavg-Tref Deviation Alarm _____

d. High Pressurizer Pressure Alarm _____

e. Low Pressurizer Level Alarm _____

f. Power Range Over Power Rod Stop Alarm _____

COLUMN B

1. Increase in turbine load with failure of rods to move.

2. Increase in turbine load with failure of of one bank D rod to move.

3. Decrease in RCS boron concentration with failure of all rods to move

4. Decrease in RCS boron concentration with failure of one bank D rod to move.

ANSWER 2.06/5.02 (3.00)

a.	1.	Increase in turbine load with failure of rods to move.	(0.50)
b.	1.	Increase in turbine load with failure of rods to move.	(0.25)
	4.	Decreases in RCS boron concentration with failure of one bank D rod to move.	(0.25)
c.	1.	Increase in turbine load with failure of rods to move.	(0.25)
	3.	Decrease in reactor coolant boron concentration with failure of rods to move.	(0.25)
d.	3.	Decrease in reactor coolant boron concentration with failure of rods to move.	(0.25)
e.	1.	Increase in turbine load with failure of rods to move.	(0.50)
f.	1.	Increase in turbine load with failure of rods to move.	(0.25)
	3.	Decrease in reactor coolant boron concentration with failure of rods to move.	(0.25)

COMMENT 2.06/5.02

1. COLUMN A, symptoms "f" is not an identified symptom in accordance with OHP-4022.012.001. COLUMN B events No. 2 and No. 4 are not identified in OHP-4022.012.001, and as such, their affects are subjective.

2. If answered IAW OHP-4022.012.001, the key should say:

   a. 1
   b. 1
   c. 1, 3
   d. 3
   e. 1
   f. no response

3. Since the student must assume extreme conditions in order to come up with a response to "f," (violate 100% RTP for event No. 1 or indicated NI for event No. 3 due to increased leakage), the following additional responses should be considered such that the key will read:

   a. 1
   b. 1 (4 acceptable, but not required)
   c. 1, 3
   d. 3
   e. 1
   f. 1 or 2 or 3 (any one required for full credit)

4. If the question is used again at Cook, please reference the Abnormal Procedure(s) used in the question premise to minimize the necessary assumptions to answer the question.

NRC Response

Concur. Symptom f in Column A was deleted from the question.

Although event No. 4 is not identified in OHP-4022.012.001, it has the same affect on the primary system as event No. 1. Both events, an increase in turbine load with failure of rods to move and a decrease in RCS boron concentration with failure of one bank D rod to move, add positive reactivity to the RCS, and would result in a Rod Insertion Limit Alarm.

QUESTION 2.09 (2.00)

Indiana Michigan Power Company LER 88-009-00 was submitted as a result of discovering the existence of a direct pathway from the containment atmosphere to the auxiliary building for Unit 2 during refueling.

a. Which ONE of the following resulted in the release path: (1.0)

1. Particulate filters and iodine cartridges for the lower containment radiation monitors were changed without positive containment integrity established.

2. Both personnel airlock doors (inner and outer) were opened simultaneously while the reactor head was being removed from the vessel.

3. While performing local leak rate testing on a containment penetration, the leak rate was determined to exceed the maximum allowable by 10 CFR 20.

4. A containment penetration was found to be eroded during maintenance activities, resulting in a release path from containment to the annulus.

b. Which ONE of the following describes why the release path was a safety concern: (1.0)

1. The level of airborne radioactivity in the auxiliary building following a fuel handling accident would exceed the maximum allowable level without the use of respirators.

2. The radiation dose received by an employee located at containment entry during a fuel handling accident would exceed the maximum allowable quarterly dose.

3. The level of airborne radioactivity in the auxiliary building following a fuel handling accident would exceed the maximum allowable by the ALARA program.

4. The radiation dose received by a person offsite as a result of a fuel handling accident in containment would significantly increase.

<u>ANSWER 2.09</u> (2.00)

a. 1. Particulate filters and iodine cartridges for the lower containment radiation monitors were changed without positive containment integrity established. (1.0)

b. 4. The radiation dose received by a person offsite as a result of a fuel handling accident in containment would significantly increase. (1.0)

COMMENT 2.09

2.09.a. The question, as worded, requires the candidate to have memorized events by LER number. Three of the events described in the potential answers were LER's at the Cook Nuclear Plant

1. U-2 LER 88-009
2. U-1 LER 88-034
3. U-1 LER 89-004

All three resulted in actual or potential leakage paths from containment.

In part "b." all of the items listed as choices are real safety concerns for the postulated fuel handling accident provided. We request that the question be deleted.

NRC Response

Concur. Questions 2.09 and 5.04 have been deleted.

QUESTION 2.13 (1.00)

Which ONE of the following is the BEST way of determining the magnitude of a steam generator tube rupture following a reactor trip and safety injection (assuming RCS pressure is stable): (1.0)

a. Determining the rate of change in level of the ruptured steam generator.

b. Determining the difference in auxiliary feedwater flow rate to intact steam generators versus ruptured steam generator.

c. Determining the difference between charging and letdown flow rates.

d. Determining the safety injection flow rate into the reactor coolant system.

ANSWER 2.13 (1.00)

d. Determining the safety injection flow rate into the reactor coolant system. (1.0)

COMMENT 2.13

The referenced lesson plan (RO-C-TN12) does not discuss ANY method for determining the magnitude of a tube rupture. There is no known reference to determining the magnitude in E-3, its supplemental procedures, or the background documents. The "BEST" method would be strictly an opinion based on actual conditions present and all potential answers require assumptions on the part of the candidate.

We request this question be deleted from the exam.

NRC Response

The question was deleted.

QUESTION 3.03/6.02 (2.00)

Match the bases for the precaution in COLUMN B to the associated precaution for operation of the reactor coolant pumps in COLUMN A. COLUMN B bases may be used more than once or not at all. (2.0)

COLUMN A	COLUMN B
a. For RCP operations, a minimum pressure differential of 200 psid must be maintained across RCP No. 1 seals. _____	1. Prevents damage to the RCP motor bearings.
b. If CCW flow to the RCP is lost, the pump must be stopped within two minutes, or when either RCP motor bearing temperature reaches 185 degrees F. _____	2. Ensures the No. 1 seal is not riding on the runner.
c. Whenever a RCP is running, maintain a pressure of at least 15 psig in the Volume Control Tank. _____	3. Ensures cooling is provided for the lower radial bearing and the seal package.
d. RCP seal water injection flow or CCW to the RCP thermal barrier must be continuously supplied when RCS temperature exceeds 150F. _____	4. Provides overpressure protection.
	5. Ensures adequate seal water flow to the No. 2 seal.
	6. Ensures proper No. 1 seal leakoff flow is maintained.

ANSWER 3.03/6.02 (2.00)

a. 2. Ensures the No. 1 seal is not riding on the runner. (0.5)
b. 1. Prevents damage to the RCP motor bearings. (0.5)
c. 5. Ensures adequate seal water flow to the No. 2 seal. (0.5)
d. 3. Ensures cooling is provided for the lower radial bearing and the seal package. (0.5)

COMMENT 3.03/6.02

Precaution "a." in COLUMN A asks for the basis for maintaining 200 psid across RCP No. 1 seals. The keyed answer (Basis 2 in COLUMN B) states that it ensures the No. 1 seal is not riding on the runner. The answer is, without question, the most correct response of the basis given. However, without minimum differential pressure across No. 1 seal it would be difficult to attain (maintain) No. 1 seal leakoff flow and/or adequate flow for cooling the lower radial bearing and seal package.

We request that there be no penalty for COLUMN B bases No. 3 and No. 6 being identified for COLUMN A Precaution "a.".

NRC Response

Do not concur. Number 2 in COLUMN B is a required correct answer. Number 3 and 6 will be accepted as additional correct answers.

QUESTION 3.12/6.07 (3.00)

Match the Main Feedwater and Steam Dump System responses in COLUMN B to the associated events in COLUMN A. COLUMN B responses may be used more than once or not at all. COLUMN A events may have more than one answer. (3.0)

COLUMN A	COLUMN B
a. Loan rejection during a loss of offsite power. _____	1. The Feedwater System can continue to operate until condenser hotwell is depleted.
b: Plant trip during a loss of offsite power and unsuccessful load rejection. _____	2. The atmospheric steam dumps will actuate.
c. Plant trip with a loss of circulation water. _____	3. Turbine throttle steam is dumped directly to the main condenser.
	4. Feedwater flow is equal to the sum of steam bypass system flow and turbine steam flow required to carry auxiliary loads.
	5. The main feedwater pumps will trip on initiation of the event.

ANSWER 3.12/6.07 (3.00)

a. 3. Turbine throttle steam is dumped directly to the main condenser. (0.5)
   4. Feedwater flow is equal to the sum of steam bypass system flow and turbine steam flow required to carry auxiliary loads. (0.5)

b. 2. The atmospheric steam dumps will actuate. (0.5)

5. The main feedwater pumps will trip on initiation of the event. (0.5)

c. 2. The atmospheric steam dumps will actuate. (0.5)

5. The main feedwater pumps will trip on initiation of the event. (0.5)

COMMENT 3.12/6.07

Recently, both Units at Cook have reduced their condenser steam dump capacity from 21 valves to 9 valves per Unit. This has resulted in a reduction in the design load rejection capability to approximately 50% load (35% steam dumps, 10% rods, 5% house loads). Event "a.", Load rejection during a loss of offsite power, did not specify an initiating power level which forced the candidates to make an assumption. If the candidate assumed that the initial power level was 50% or less, then the keyed answers (3, 4) would be the correct responses. If the candidate assumed the initial power level was greater than 50% (e.g., 100%), then in order for the load rejection to be successful would require the atmospheric steam dumps to actuate to remove the excess heat generated in the RCS.

We request that there be no penalty for COLUMN B response No. 2 to Event "a."
Reference: 2-OHP 4022.001.002

NRC Response

Partially concur. A review of the question indicated that reactor power should have been given in the question. As power was not provided, the question has been deleted.

QUESTION 3.15/6.10 (1.00)

Which ONE of the following is correct concerning the spent fuel pool cooling system. (1.0)

a. With both cooling trains in operation, and a normal refueling off-load, the pool temperature will be maintained less then 100°F.

b. With both cooling trains in operation, and a full-core off load, the pool temperature will be maintained less than 120°F.

c. With a loss of one cooling train, and a normal refueling off-load, the pool temperature will be maintained less than 130°F.

d. With a loss of one cooling train, and a full-core off load, the pool temperature will be maintained less than 150°F.

ANSWER 3.15/6.10 (1.00)

d. With a loss of one cooling train, and a full-core off load, the pool temperature will be maintained less than 150°F. (1.0)

COMMENT 3.15/6.10

This question uses four potential answers which are similar to design cooling capacity bases identified in lesson plan RO-C-AS05. The differences are that:

1) The design basis specific flow rates and CCW temperature was not included, and

2) Three of the potential answers were either 10° or 20° different from the design bases.

Since the question does not refer to the design basis of the SFPCS and design basis is conservative with actual system operation, all four answers should be considered correct.

Further, the KA referenced (03300K.303) states: "Knowledge of the effect that a loss of the SFPCS will have on the following: . . . K.303 spent fuel temperature . . . " The question refers to both trains in operation and loss of one train, i.e., one train in operation. The lesson plan contains an objective (Objective 2) which requires knowledge of the consequences of a complete loss of SFPC. The knowledge required by the question is not identified by either K.303 or the objective.

We request that this question is deleted.

NRC Response

Concur. This question was deleted. K/A more accurately supporting it were determined to be of low importance.

QUESTION 3.18/6.11 (3.00)

Match the automatic reactor trip in COLUMN B with the associated protection the trip provides in COLUMN A. COLUMN B trips may be used more than once or not at all. COLUMN A may have more than one answer. (3.00)

COLUMN A	COLUMN B
a. Protects against a high startup rate. ________	1. Overtemperature delta Trip.
b. Protects against DNB. ________	2. Low Pressurizer Pressure Trip.
c. Protects against a rod ejection accident. ________	3. Pressurizer High Level Trip.
	4. Low-Low Steam Generator Level Trip.

5. Power Range Neutron Flux High Positive Rate Trip.

6. Source Range Neutron Flux Trip.

7. Intermediate Range Neutron Flux Trip.

8. Power Range Neutron Flux Trip.

ANSWER 3.18/6.11 (3.00)

a. 6. Source Range Neutron Flux Trip
   7. Intermediate Range Neutron Flux Trip
b. 1. Overtemperature Delta T Trip
   2. Low Pressurizer Pressure Trip
   8. Power Range Neutron Flux Trip
c. 5. Power Range Neutron Flux High Positive Rate Trip
   6. Source Range Neutron Flux Trip
   7. Intermediate Range Neutron Flux Trip
   (1.0 pts per section)

COMMENT 3.18/6.11

Part a: Tech Spec basis states that the "Power Range Neutron Flux High Positive Rate Trip" provides protection against rapid flow increases which are characteristic of rod ejection events . . ." and the Power Range Neutron High setpoint provides protection against reactivity excursions which are too rapid . . ."

Rapid flux increases and rapid reactivity excursions infer "high SUR."

Part "c." of this questions asks the candidates to identify, from the list of trips given, those that protect against a rod ejection accident. It does not specify whether the basis for the candidates' selection(s) should be from Technical Specifications (2.2.1 Bases) or the plant's training materials (RO-C-NSS11-SH03). In both documents, neither the Source Range Neutron Flux Trip or the Intermediate Range Neutron Flux Trip are credited with protecting the core against a rod ejection accident.

We request that there be no penalty for Answers No. 5 and No. 8 to Part "a." and that the <u>only</u> answer that be accepted for credit for Part "c." of the question be 5. Power Range Neutron Flux High Positive Rate Trip.

NRC RESPONSE

Concur. For Protection a. in COLUMN A, trip number 5 will be accepted as an additional REQUIRED correct answer. Trip number 8 will be accepted as an additional correct answer which is not required, since 8 did not specifically state Power Range Neutron High Range Flux Trip.

For Protection c. in COLUMN A, trip number 5. will be accepted as the only correct answer.

QUESTION 3.31 (1.00)

Fill in the blanks: (1.0)

No individual shall exceed ________ mR/quarter or a total of ______ mR/year, (D.C. Cook Plant administrative limits), including previous exposure at other facilities, while an employee at Cook Nuclear Plant.

ANSWER 3.31 (1.00)

2750 (0.5)
4750 (0.5)

COMMENT 3.31

The keyed answer is incorrect for the question asked. The keyed answer is lifted out of a CAUTION under the paragraph title "4.4 Administrative Limits" with the exception for (D.C. Cook Plant Administrative Limit). The procedure clearly states that the normal administrative limits is 1000 mRem/qt (see Paragraphs 4.4.2, 4.4.3, 4.4.1, Attachment 1).

During the exam, clarification was requested concerning this question by several candidates and at least two proctors (Mr. Lennartz and Ms. Schamberger) provided clarification to individuals. The clarification requested was whether the question referred to EXTENDED administrative limits or normal administrative limits. For at least one individual, the initial clarification was "NO, not extended" and then later reversed to "YES, extended." For at least one other individual, only the initial clarification of "NO" was given. No general announcement was made concerning this question, nor was a clarification written on the whiteboard provided in the room.

Due to the inadvertent differences in clarifications, we request that either 1000 mR or 2750 mR be accepted for the quarterly limit.

Reference: PMP 6010 RPP.102

NRC RESPONSE

Concur. Either 1000 mR or 2750 mR will be accepted as a correct answer.

QUESTION 3.33 (2.00)

Match the controlled access areas in COLUMN B with the associated area description in COLUMN A. COLUMN B areas may be used more than once or not at all. (2.0)

COLUMN A		COLUMN B
a. Area with a dose in 5 consecutive days of 200 mR ________.	1.	High Radiation Area
b. Barricaded and posted area as required by Tech Specs ________.	2.	Radiation Area
c. Area with a dose rate of 500 mR/hr ________.	3.	Extreme High Radiation Area
d. Area with a dose rate of 50 mR/hr ________.	4.	Unrestricted Area

ANSWER 3.33 (2.00)

a.	2.	Radiation Area	(0.5)
b.	1.	High Radiation Area	(0.5)
c.	1.	High Radiation Area	(0.5)
d.	2.	Radiation Area	(0.5)

COMMENT 3.33

Technical Specification 6.12.2 states that "The requirements of 6.12.1 shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mRem/hr." "Extreme High Radiation Area" is defined as "Any area, accessible to personnel in which there exists radiation at such levels that a major portion of the body could receive in any one hour a dose in excess of 1000 millirem." (See RO-C-RP02, Page 7 of 17.) Therefore, Extreme High Radiation Areas ("3." from COLUMN B) are a "Barricaded and posted area as required by Tech Specs." ("b." from COLUMN A)

We request that the answer key be modified to include "3." for Part "b."

NRC Response

Do not concur. Technical Specifications outline the minimum requirements for high radiation areas, which requires the area be barricaded and posted. Technical Specifications also state the areas where the intensity of radiation is greater than 1000 mR/hr, locked doors shall be provided to prevent unauthorized entry. Therefore, a barricade is not an acceptable means of preventing unauthorized entry into an extreme high radiation area.

QUESTION 3.34 (3.00)

Match the hazard in COLUMN B with the associated biological effects in COLUMN A. COLUMN B hazards may be used more than once. COLUMN A effects may have more than one answer. (3.0)

COLUMN A		COLUMN B
a.	Unable to penetrate through the skin ________.	1. gammas
b.	Internal hazard ________.	2. neutrons
c.	External hazard ________.	3. alphas
d.	Can cause damage to living tissue ________.	4. betas

ANSWER 3.34 (3.00)

a.	3.	alphas	(0.375)
	4.	betas	(0.375)
b.	1.	gammas	(0.18)
	2.	neutrons	(0.19)
	3.	alphas	(0.19)
	4.	betas	(0.19)
c.	1.	gammas	(0.25)
	2.	neutrons	(0.25)
	4.	betas	(0.25)
d.	1.	gammas	(0.18)
	2.	neutrons	(0.19)
	3.	alphas	(0.19)
	4.	betas	(0.19)

COMMENT 3.34

The keyed answer does not accurately reflect the information supplied in the reference, RO-C-RP01, Page 13, Paragraph V.A.

We request the answer key be modified such that alphas and betas (3 and 4 from COLUMN B) be accepted for full credit on part "b." Internal, and gammas and neutrons (1 and 2 from COLUMN B) be accepted for full credit on part "c."

NRC RESPONSE

Concur. The reference states:

> Because of their inability to penetrate the layers of skin, alpha and beta particles are generally referred to as INTERNAL HAZARDS. This means that the major hazard would be ingestion of material which could damage living tissue.
>
> Gamma and neutron are considered EXTERNAL HAZARDS because they can cause damage to living tissue without ingestion.

Gammas and neutrons are classified as external hazards in the reference material since ingestion is not required to cause damage. They will not be required for credit for part b.

Although betas cannot penetrate the skin, they can cause damage to the eye. Therefore, betas are also an external hazard and is required as a correct answer to Item c.

Question 3.11

This questions was deleted by the Region staff. It is more appropriate at the SRO level.

Question 3.26.a.

Part a. was deleted by the Region staff. A K/A could not be found to support that item specifically.

QUESTION 5.11 (1.50)

a. Which one of the following is the limit for the quantity of radioactivity which may be contained in each gas storage tank for curies of noble gas (considered as Xe 133)? (1.0)

1. 43500
2. 45000
3. 43800
4. 45500

b. The basis for the gas storage tank radioactivity limit provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest site boundary will not exceed _______.

ANSWER 5.11 (1.50)

a. 1. (1.0)
b. 0.5 rem (0.5)

COMMENT 5.11

(Note: Per Technical Specifications, No. 3 is correct)

An operator does not use the Technical Specification numerical value specified in the question in the performance of his/her duties. This T.S. is met, at Cook, in the following manner:

> In order to add 43,800 curies of noble gas to a gas decay tank (GDT), the RCS (both Units) gaseous activity would need to be 77 μc/cc, prior to discharging all of this gas to a GDT. We have imposed an administrative limit of 10% of this value (or 7.7 μc/cc), to limit the possibility of loading an excess amount of curies to a GDT. Chemistry Department is assigned the responsibility of ensuring the surveillance for T.S. 3.11.2.6 is performed and that the LCO is met. (Appropriate procedural pages are attached and highlighted to support this.)

Furthermore, NUREG/BR-0122 discourages the use of three distractors so similar to the correct answer. The range of difference between the distractors and the correct answer is -0.7% to + ≈3% (Refer to NUREG/BR-0122, Page 4-13, Item (8), last paragraph).

We also note that Part "b." of the question requires memorization of a T.S.

basis numerical value. This is a numerical value for a T.S. LCO that the operator is not responsible for verifying, as previously discussed.

We request that both Parts "a." and "b." of this question be deleted from the examination.

NRC Response

Partially concur. Part a. was deleted and part b. was retained. The particular basis number of 0.5 Rem relates to the unrestricted limit for the public and is not an obscure value.

QUESTION 6.18 (1.00)

Which ONE of the following statements correctly describes all of the alarms which would actuate for the indicated Spent Fuel Pool problems?

a. SFP low level: the "SFP Low Level" alarm in both control rooms would actuate.

b. SFP high temperatures: the "SFP Cooling System Abnormal" alarm in both control rooms would actuate.

c. SFP filter high differential pressure: the "SFP Cooling System Abnormal" alarm in both control rooms would actuate.

d. SFP high level: the "SFP Cooling System Abnormal" alarm in both control rooms would actuate.

ANSWER 6.18 (1.00)

d. (1.0)

COMMENT 6.18

Both answers "a." and "d." are true statements. Answer "b." and "c." are false statements. The only differentiation between the keyed answer "d." and the other true statement is contained in the word "all" of the question.

The K/A statement for this question is "Knowledge of SFPCS design feature(s) and/or interlock(s) which provided for maintenance of spent fuel level." There are approximately 1650 annunciators per unit control room at the Cook facility. Memorization of the actuation inputs, logics, and locations of these 1650 annunciators per unit is unrealistic, considering that the 4024 series (annunciator response) procedures exist for each of them in the control room.

We ask for deletion of this question from the standpoint that there must be a logical limit to the amount of required memorization necessary in order to successfully answer a test question.

NRC Response

Do not concur. The correct answer is differentiated from the incorrect by the key word "ALL." The reason this issue is significant is that the only control room indications for spent fuel pool level are the annunciators.

QUESTION 6.20 (1.00)

FILL in the blanks for the following with respect to Hydrogen Recombiner System. (Note: Blank may contain more than one word.)

> Operation of one Hydrogen Recombiner Unit in conjunction with ________ following a LOCA will prevent hydrogen concentration from reaching the lower flammability limit of ______ volume percent.

ANSWER 6.20 (1.00)

one CEQ fan
4

COMMENT 6.20

The CEQ fan will also frequently be referred to as a "skimmer fan." "CEQ" is the acronym used on control room switches, facility prints, etc. "Containment Equalization/Hydrogen Skimmer Fan" is the complete noun name but is more commonly referred to as just "skimmer fan" by facility personnel.

We ask that minor variations for this complete noun name also be accepted as an equal alternate substitute for answer to "part one" such as:

- Containment Equalization Fan
- Skimmer Fan
- Hydrogen Skimmer Fan
- Containment Skimmer Fan
- Hydrogen Recirculation Fan

NRC Response

Concur. Additional minor variations will be accepted.

Question 6.07

The region staff deleted this question. Power level was required to answer and was not given.

Question 6.10

The region staff deleted this question. The K/A's specifically supporting this item were of low importance.

Enclosure 4

SIMULATION FACILITY REPORT

Facility Licensee: D.C. Cook

Facility Licensee Docket Nos. 50-315; 50-316

Operating Tests Administered At: D.C. Cook

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

ITEM	DESCRIPTION
MSIV Switches	Two switch overrides for MSIV manual operation do not work.
Malfunction NI04A	Source range block failure causes power range N42 to oscillate causing PR flux deviation alarms and negative rate trip on N42.
IC-24	100% EOL IC set not available.
SI, CS, CCP	Pumps ran without a suction source (RWST empty) with no indications of loss of NPSH.
AFW Pumps	Pumps ran without suction source giving greater than 100,000 PPH.
Delta-I Curve	Do not have Delta-I curve for BOL IC sets.
Malfunction RD18A	During loss of Bus 21A numerous pump/valve lights flashed on and off until rods dropped.
Malfunction CV28A	Loss of No. 3 RCP seal malfunction did not work.
IC-20	PORV discharge temp high alarm came in at beginning of scenario.
Malfunction CW03	Condenser vacuum did not indicate properly (read too high) on gage on front panel (near turbine controls).
Radiation Monitor 2700	High value on last day's scenarios.
Iconics Display	Was skewed toward high radiation throughout entire scenario.

P-4 SI Logic	Per logic diagram, P-4 not being met should result in the inability to permanently reset SI. This is not true on the simulator.