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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:
Peter B. Bloch, Presiding Officer
Peter Lam, Special Assistant

SERVED FEB 28 1997

In the matter of	Docket No. 55-20726-SP
RALPH L. TETRICK	Re: Operator License
(Denial of Application for Reactor Operator License)	ASLBP No. 96-721-01-SP

INITIAL DECISION

Ralph L. Tetrick, a reactor operator at the Turkey Point Nuclear Generating Plan, Units 3 and 4 ("Turkey Point"), operated by Florida Power & Light Company ("Florida Power"), is an applicant for a senior reactor operator's (SRO's) license. On October 21, 1996, I granted Mr. Tetrick's request for a hearing concerning whether he had passed his SRO license examination.[1] An SRO is defined in 10 C.F.R. § 55.4 as "any

[1]This is an informal hearing under 10 C.F.R. Part 2, Subpart L. *See* 10 C.F.R. § 2.1201(a)(2). By letter of November 7, 1996, the NRC Staff ("Staff") submitted the Hearing File pursuant to 10 C.F.R. § 1.1231. On December 30, 1996, Mr. Tetrick filed his written presentation in this proceeding, pursuant to 10 C.F.R. § 2.1233 (Tetrick Presentation). Staff replied, pursuant to this same section of the regulations, on January 23, 1997 (Staff Presentation).

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individual licensed under this part to manipulate the controls of a facility and to direct the licensed activities of licensed operators." (Emphasis added.)

The Nuclear Regulatory Commission (NRC) has jurisdiction of this request for a hearing, in which Mr. Tetrick appeals the result of his written examination The NRC helps to assure the health and safety of the public by requiring reactor operators to successfully demonstrate their knowledge of nuclear power plant operation before they are licensed. See *Alfred J. Morabito* (Senior Operator License for Beaver Valley Power Station, Unit 1), LBP-88-10, 27 NRC 417 (1988) and LBP-88-16, 27 NRC 583 (1988); *Roger W. Ellingwood* (Senior Operator License for Catawba Nuclear Station), LBP-89-21, 30 NRC 68 (1989).

The Commission's regulations in 10 C.F.R. §§ 55.43 and 55.45 require that an applicant for a senior reactor operator license pass both a written examination and an operating test. Written examinations taken by applicants for senior reactor operator licenses are developed and administered by the licensee, in this case Florida Power & Light Company, and are governed by 10 C.F.R. § 55.43. Written examination questions test "the knowledge, skills, and abilities needed to perform licensed senior operator duties." 10 C.F.R. § 55.43(a). In addition to information contained in a facility's training program, knowledge of "information in the Final Safety Analysis

Report, system description manuals and operating procedures, facility license and license amendments [and] Licensee Event Reports" may properly be tested. *Id.* Written examinations for senior operators include a representative sample of questions from 14 subject areas specified for operator license applicants in 10 C.F.R. § 55.41(b)(1-14). In addition, written examinations for senior operators are to include a representative sample of questions from the seven areas specified in 10 C.F.R. § 55.43(b)(1-7).[2]

In addition to the written test, Mr. Tetrick took and passed the operating test, which involves a plant walkthrough and dynamic simulator evaluation during which various plant tasks, scenarios and questions are presented to the applicants. *See* 10 C.F.R. § 55.45.

On the written examination, Mr. Tetrick was scored by the examiner as correctly answering 78 of 100 multiple choice questions, for a score of 78%, which does not meet the 80% minimum score required to pass. *See* NUREG-1021, page 5 of 6. In response to Mr. Tetrick's request, the Staff completed an informal review that confirmed his failing grade. Hearing File item 21, attachment at 2-7.

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[2]*See* NUREG-1021, "Operator Licensing Examiner Standards," for further guidance on the administration and grading of the senior reactor operator written test.

Initially, Mr. Tetrick challenged the grading of Questions 24, 63, 84 and 96 on his examination. In its review, the Staff determined that Question 24 was invalid and should be deleted from the examination. However, the result of this determination was that Mr. Tetrick's score was raised only to 78.8%, which is short of the 80% required to pass. Mr. Tetrick continues to contest the scoring of his answers to questions 63, 84 and 96 and he also is contesting the scoring of his answer to Question 90. Mr. Tetrick must be sustained in at least one of the four remaining challenges to pass the examination. Below, the challenges are considered one at a time.

I. Question 63

A. The Question

Examination Question 63 stated as follows:

Plant conditions:

- *Preparations are being made for refueling operations.*
- *The refueling cavity is filled with the transfer tube gate valve open.*
- *Alarm annunciators H-1/1, SFP LO LEVEL and G=9/5, CNTMT SUMP HI LEVEL are in alarm.*

Which ONE of the following is the required IMMEDIATE ACTION in response to these conditions?

a. *Verify alarms by checking containment sump level recorder and spent fuel level indication.*

b. *Sound the containment evacuation alarm.*

c. *Initiate containment ventilation isolation.*

d. Initiate control room ventilation isolation.

B. Staff Position

Staff contends[3] that the correct answer to this question is "*b. Sound the containment evacuation alarm.*" It relies on Procedure 0-ADM-219, § 3.4.1 (Hearing File #20, attachment 2), which states: "Respond to alarms on color code priority *and plant conditions.*" [Emphasis added.] Staff argues that:

> The plant conditions and indications specified in this question (*i.e.*, the refueling cavity filled and the transfer tube gate valve open with coincident SFP LOW LEVEL and CONTAINMENT SUMP HIGH LEVEL alarms) *are mutually supportive and confirmatory*, and require entry into Off-Normal Operating Procedure 3-ONOP-033.2, "Refueling Cavity Seal Failure" (Hearing File #24[4]). [Emphasis added.]

Staff further argues that there is only one immediate action specified for a Refueling Cavity Seal Failure. That action, which the Operator must be able to perform *from memory* and *before opening and reading the emergency procedures*, is to sound the containment evacuation alarm. Hearing File #24, 3-ONOP-033.2, p. 5 at § 4.1; Hearing File #25 at 0-ADM-211, p. 11, § 5.2.1; and Hearing File #25 at 3-BD-EOP-E-O "BASIS DOCUMENT".

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[3]Affidavit of Brian Hughes and Thomas A. Peebles, January 23, 1997 (Staff Affidavit), Attachment 2 to Staff's Presentation, at p. 8, ¶ 20.

[4]Staff refers to "Item 24", which I have change solely for the purpose of complying with the style used in this document.

Staff stresses the importance of this immediate action. It states, at Staff Affidavit p. 9, that:

> Significantly, the need for such immediate action results from the fact that under the stated conditions, personnel located in the containment would quickly be exposed to high levels of radiation (due to loss of water which normally acts as a radiation shield) unless they are promptly notified by a containment alarm to evacuate the area.

Furthermore, Staff indicates that Off-Normal Operating Procedures have a high priority among plant operating procedures. Hearing Record #25, at 0-ADM-211, p. 25, § 5.13.1.

Staff also points out that the question explicitly asks for "the IMMEDIATE ACTION". Staff Affidavit at 10.

C. Mr. Tetrick's Position

Mr. Tetrick's answer was "*a. Verify alarms by checking containment sump level recorder and spent fuel level indication.*" He relies on the CONTROL ROOM ANNUNCIATOR RESPONSE procedure 3-ARP-097.CR to support his belief that, "The annunciators should be verified by additional supportive information to preclude the possibility of annunciator failure." Hearing File #20, discussion of Exam Question #63; *see also* Tetrick Request for Hearing, September 25, 1996.

D. Conclusion

The Staff has persuaded me that when two concurrent annunciators sound, indicating that there is an off-normal event that could cause harmful radiation within the containment, that the Operator should take the required IMMEDIATE ACTION. Given the important safety problem that is being indicated by two different annunciators, that is not the time to verify that each of the annunciators is working properly. That they sound *together* is enough corroboration to act immediately to prevent injury to the health of plant employees.

Mr. Tetrick has had this Staff response available to him for some time and has never directly addressed it. In consequence, he continues to argue for an examination answer that could delay his action in preventing unnecessary exposure of his co-workers. I find that Mr. Tetrick's answer to this question was not correct.

I note, as well, that Mr. Tetrick is incorrect in stating that 3-ARP-097.CR states "that for all alarms the ARP shall be consulted." *See* the ARP at 8, "NOTES," at the bottom of the box. Step 2 in the notes requires that immediate corrective actions be taken as necessary. I interpret this to require that the immediate action of 3-ONOP-033.2 should be taken. The language quoted by Mr. Tetrick is from a bulleted paragraph that is part of paragraph "3) Daily Annunciator Response Procedure

Usage." I do not interpret that language to supersede or qualify in any way plant procedures that require immediate action.

II. Question 84

A. The Question

Examination Question 84 stated as follows:

Which ONE of the following is the basis for step 1, "Verify Reactor Trip", of FR-S.1, Response to Nuclear Power Generation/ATWS?

a. To ensure that only decay heat and reactor coolant pumps are adding heat to the RCS.

b. To ensure shutdown margin is within Technical Specifications limits for HOT STANDBY.

c. To alert the operator to take further corrective action if the reactor is NOT tripped.

d. To verify that all automatic reactor protective features have functioned as designed.

B. Staff Position

Staff states that the correct answer is "a." Staff argues that the question requests the basis (or reason) for step 1, Verify Reactor Trip, of Fr-S.1, Response to Nuclear Power Generation/ATWS. To determine the basis for step 1, I first examine Step 1 in the following Table Box:

Verify reactor trip:	
* Rod bottom lights -- ON * Reactor trip and bypass breakers - OPEN * Rod position indicators - AT ZERO * Neutron flux - DECREASING	Manually trip reactor. If reactor will <u>NOT</u> trip, <u>THEN</u> manually insert control rods.

Staff asserts that the reason or basis for this step (e.g., the reason the step is required) is: "*a. To ensure that only decay heat and reactor coolant pumps are adding heat to the RCS [reactor coolant system].*" In support of this basis, Staff states that the reactor safeguard systems that protect the plant during an accident are designed on the basis that there are no additional sources of heat other than those mentioned in the correct answer, *a*. Staff Affidavit at 11-12, ¶¶ 26-27; Hearing File #20, "Page 9", 3-BD-EOP-E-0, "Basis Document".

C. Mr. Tetrick's Position

Mr. Tetrick asserts that a correct answer to Question #84 is, "C. To alert the operator to take further corrective action if the reactor is not tripped."

D. Conclusion

I conclude that the basis or "reason" for Step 1 has been correctly specified by the Staff as specified in File #20, 3-BD-EOP-E-O, "Basis Document". Since the procedure correctly states the "basis," a student could have answered correctly merely by learning what the procedure stated. The answer given by Mr. Tetrick is not the "basis" for Step 1. It is a follow-up action that might be taken after performing Step 1 but it is not the "basis" for that step.

III. Question 90

A. The Question

Examination Question 90 stated as follows:

When draining the RCS using 3-OP-041.9, REDUCED INVENTORY OPERATIONS, the reactor vessel head and pressurizer are both vented to containment atmosphere.

Which one of the following describes the effects on reactor vessel indication if an adequate vent path is not provided? (Assume the reference leg remains full).

a. *A vacuum in the RCS loops will result in level indication being lower than actual levels.*

b. *A vacuum in the RCS loops will result in level indication being higher than actual levels.*

c. *A positive pressure in the RCS loops will result in level indication being lower than actual levels.*

d. *The level instruments automatically compensate for positive or negative pressure.*

B. Mr. Tetrick's Position

Mr. Tetrick's argument is simple. He states:

> The assumption that the reference leg remains full makes this question invalid. At Turkey Point the drain down level indication has dry reference legs. This condition is verified by 0-PMI-041.110. Applicant requests that this question be deleted.

C. Staff Position

Staff states that the correct answer is:

> *a. A vacuum in the RCS loops will result in level indication being lower than actual levels.*

Staff concedes that at Turkey Point the drain down level indication has a dry reference leg and that the assumption that the reference leg remains full is contrary to fact. Staff Affidavit at 15, ¶¶ 33, 35. Nevertheless, the Staff asserts that the question remains valid because "the fact that the reference leg is dry as opposed to filled with water is immaterial." Staff Affidavit at 17, ¶ 39.

The purpose of this question, according to the Staff, was to test an understanding of a basic hydraulic principle, that if a vacuum is drawn above the water level in the reactor pressure vessel that will affect the instrument that measures water level because it will reduce the pressure exerted by the water in the pressure vessel.

The important leg to consider here is the variable leg of the water level instrument. When there is a vacuum above the

water in the pressure vessel, there will be less pressure on the variable leg than if the space above the water were filled by air at atmospheric pressure. The purpose of the "reference leg" of the pressure indicator is to measure the height of water that corresponds to the pressure on the variable leg. Providing that there is no malfunction affecting the reference leg, it does not matter whether the design uses a wet or a dry reference leg. The answer will be the same: an accurate measurement of the height of the water in the variable leg. (Staff Affidavit at 16-17, ¶¶ 37-39.

Staff states that:

> 38. This question test applicants on their understanding of the hydraulic effects on level indication during mid-loop operations (*i.e.*, water level in the loop piping is less than full) and other draining operations if a vacuum is drawn while lowering water level. **Numerous incidents have occurred within the nuclear industry which involved draining reactor coolant systems. A lack of understanding of the hydraulic effects on level indications by operators has been a prime contributor to many of these events.** Therefore, it is important that applicants demonstrate an understanding of this problem, as examined on this question.

[Emphasis added.]

D. Conclusion

On this question, I agree with the Staff. The question is poorly worded, containing an assumption that is different from the plant configuration. This could have been somewhat confusing to Mr. Tetrick.

However, I have decided that if Mr. Tetrick had a basic knowledge of the principles that affect water level indication, he should have realized that the entire purpose of the reference leg of the water level indicator is to measure the height of water in the variable leg. Since the pressure exerted by the column of water in the variable leg would be reduced by the vacuum above the water in the reactor pressure vessel, *the water level indicated by the instrument would be lower than the water level in the reactor vessel.* Given the importance of this principle, I conclude that Mr. Tetrick should be able to understand it and answer the question correctly. There is no explanation for the answer he gave: that the water level indication would be higher than actual levels.

I conclude that, despite the contrary-to-fact predicate that makes this question more difficult than intended, Mr. Tetrick should have answered it correctly. The question is valid and Mr. Tetrick's answer is wrong.

IV. Question 96

A. The Question

Examination Question 96 stated as follows:

> *Which ONE of the following is the lowest level position responsible for ensuring entries are made in the Technical Specification Related Equipment Out-of-Service Index?*

a. Nuclear Plant Supervisor

b. Assistant Nuclear Plant Supervisor

c. Senior Nuclear Plant Operator

d. Nuclear Watch Engineer

B. Staff Position

Staff states that the correct answer is "*b. Assistant Nuclear Plant Supervisor.*" Staff states that

> Procedure 0-ADM-213, "Technical Specification Related Equipment and Risk Significant S.C. Out-of-Service Logbook," states that the ALPS is the lowest level position responsible for entering inoperable equipment in the subject index (Item 24). When the NWE [Nuclear Watch Engineer] relieves the ALPS, he then assumes the position of the ALPS. The NWE is not authorized to make entries in the subject index unless he is acting in the capacity of the ALPS, any more than he would be able to exercise any other functions of the ALPS unless he is acting in the ALPS capacity.

C. Mr. Tetrick's Position

Mr. Tetrick states that "*d. Nuclear Watch Engineer*" is also correct because procedure 0-ADM-200 makes the Nuclear Watch Engineer (NWE) responsible "for *routinely* relieving the Assistant Nuclear Plant Supervisor (ALPS) of the control room command and control function to enable the ALPS to leave the control room." [Emphasis added.] Staff does not question Mr. Tetrick's statement that this substitution is authorized and routine.

D. Conclusion

I conclude that the question is ambiguous and should be struck.

Mr. Tetrick has reasonable ground to consider his answer to be correct. I do not think it necessary to address the following metaphysical question: Is the Nuclear Watch Engineer still at least in part a Nuclear Watch Engineer when he relieves the Assistant Nuclear Plant Supervisor? Staff apparently thinks that the Nuclear Watch Engineer completely loses his ordinary job identity when he acts as a substitute for the Assistant Nuclear Plant Supervisor. While that is a plausible way to view what happens, I do not think it fair to require Mr. Tetrick to adopt that view of the use of words in order to pass his examination. The question in its current form is ambiguous and invalid.

Mr. Tetrick has answered correctly 78 of 98 questions. His score, rounded to the nearest tenth of a percent is 79.6%.

I note that for the examination question to have the unambiguous meaning given to it by the Staff, it could have said: "*Which ONE of the following is the lowest level position that one must have (or be acting as) for ensuring entries are made in the Technical Specification Related Equipment Out-of-Service Index?*"

V. Overall Conclusion

I have determined that Mr. Tetrick was correct in 78 of 98 valid questions on his examination. Staff has not addressed the question of the number of digits in the examination score that should be considered significant. Because I have not been directed to any governing guidance or regulation, I have decided that it is appropriate to round up the answer to the nearest integer. These tests are not so precise that tenths of a percent have any meaning. Consequently, Mr. Tetrick's score is 80 percent, which is a passing score. He shall, therefore, be granted a license as a Senior Reactor Operator.

VI. ORDER

For all the foregoing reasons and upon consideration of the entire record in this matter, it is this 28th day of February, 1997, ORDERED, that:

1. The Staff of the Nuclear Regulatory Commission may issue to Mr. Ralph L. Tetrick a Senior Reactor Operator License for Turkey Point Nuclear Generating Plant, Units 3 and 4.

2. Pursuant to 10 C.F.R. § 2.1251, this initial decision constitutes the final action of the Commission thirty (30) days after the date of issuance, unless any party petitions for Commission review in accordance with § 2.786 or the Commission takes review of the decision sua sponte. If there is no petition for review, the date on which this decision will become final is Monday, March 31, 1997.

3. Pursuant to 10 C.F.R. § 2.786, a petition for review must be filed within fifteen (15) days after service of this decision, which is considered served on the date it is mailed, pursuant to 10 C.F.R. § 2.712(e). However, since service of this decision is by mail, five days shall be added to the prescribed period of response, pursuant to 10 C.F.R. § 2.710, which governs the computation of time. Consequently, the date the petition for review must be served is Thursday, March 20. Service of the petition for review must, pursuant to this Order, be made by express mail.

4. A petition for review and a response to a petition for review must meet the requirements of 10 C.F.R. § 2.786.

5. If a petition for review is filed, the answer must be filed within 10 days. Since the petition for review shall be filed by express mail, two days shall be added to the period of response pursuant to 10 C.F.R. § 2.710, which governs the computation of time. Consequently, the date the answer must be served is Tuesday, March 16, 1997.

Service of the answer must, pursuant to this Order, be made by express mail.

Peter B. Bloch
Presiding Officer

Rockville, Maryland

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of

RALPH L. TETRICK

(Denial of Senior Reactor Operator's License)

Docket No.(s) 55-20726-SP

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing LB INITIAL DECISION - LBP-97-2 have been served upon the following persons by U.S. mail, first class, except as otherwise noted and in accordance with the requirements of 10 CFR Sec. 2.712.

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Dated at Rockville, Md. this
28 day of February 1997

Office of the Secretary of the Commission