January 23, 1997

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

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

AFFIDAVIT OF BRIAN HUGHES AND THOMAS A. PEEBLES

Brian Hughes (BH) and Thomas A. Peebles (TAP), having first been duly sworn, do hereby state as follows:

1(a). (BH) My name is Brian Hughes. I am employed as a Reactor Engineer (Examiner Qualified), in the Operator Licensing Branch, Division of Reactor Controls and Human Factors, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC), in Washington, D.C. A statement of my professional qualifications is attached hereto.

1(b). (TAP) My name is Thomas A. Peebles. I am employed as Chief, Operator Licensing and Human Performance Branch, Division of Reactor Safety, NRC Region II, in Atlanta, Georgia. A statement of my professional qualifications is attached hereto.

2. This Affidavit is prepared in response to the written presentation submitted by Ralph L. Tetrick on January 3, 1997, in support of his request for a hearing on the NRC Staff's denial of his application for a Senior Reactor Operator (SRO) license for use at the Turkey Point Nuclear Generating Plant, Units 3 and 4 ("Turkey Point"), operated by Florida Power and Light Company ("FP&L"). We have reviewed Mr. Tetrick's written presentation and are familiar therewith.

3. Mr. Tetrick's written presentation consists of a cover letter to Staff Counsel Sherwin E. Turk, dated December 30, 1996, and partial copies of the following documents:

* Letter from Ralph L. Tetrick to Director, Division of Reactor Controls and Human Factors, Office of Nuclear Reactor Regulation, NRC, dated July 30, 1996 (requesting an informal review of his answers to written examination questions 24, 63, 84 and 96) *[the section concerning Q 96 was not attached to Mr. Tetrick's written presentation]*

* Letter from NRC staff to Mr. Tetrick, dated September 12, 1996 *[this item was not attached to Mr. Tetrick's written presentation]*

* Letter from Ralph L. Tetrick to the Assistant General Counsel for Hearings and Enforcement, Office of the General Counsel, NRC, dated September 25, 1996 (requesting a hearing on questions 24, 63, 84, 90, and 96, and replying to NRC staff letter of September 12, 1996)

* Letter from Ralph L. Tetrick to the Secretary, NRC, with attached letter to the Assistant General Counsel for Hearings and Enforcement, Office of the General Counsel, NRC, dated September 25, 1996 (see above).

4. On November 7, 1996, the Staff transmitted the Hearing File to the Presiding Officer and Mr. Tetrick, along with a numbered index thereto. Items contained in the hearing file are herein referred to by their designated "item" number, as set forth in the hearing file index.

5. Mr. Tetrick holds an NRC operating license, License No. OP-20909, issued on December 3, 1991, which authorizes him to manipulate all controls at the Turkey Point Nuclear Plant, Units 3 and 4 (Item 2). That license expires on December 3, 1997, unless terminated, renewed or upgraded prior to that date. *Id.*

6. On February 16, 1996, the NRC Staff informed FP&L that licensing examinations for reactor operators (ROs) and senior reactor operators (SROs) would be conducted in June 1996 (Item 3). In May 1996, FP&L submitted an application for Mr. Tetrick to upgrade his license to senior reactor operator (Item 4).

7. On June 12, 1996, the Staff transmitted a letter to FP&L, in which the Staff (a) informed FP&L that it was authorized to administer the written portion of the licensing examinations on June 14, 1996; (b) provided instructions and guidelines to be followed in administering the written examinations; and (c) provided the written examinations to FP&L (Item 6). Also enclosed with that letter was a list of the license applicants, consisting of 12 applicants for reactor operator licenses and 7 applicants for senior reactor operator licenses, including Mr. Tetrick (Item 6, Enclosure 4).

8. Master copies of the Turkey Point written examinations, No. 96-300, for SRO and RO licenses administered on June 14, 1996 (NRC Forms ES-401), are included in the hearing file as Items 9 and 10, respectively. As an SRO applicant, Mr. Tetrick ("the Applicant") was administered the SRO written examination (Item 9). That examination consists of 100 multiple choice questions, the correct answer to each of which was to count as one point. If an applicant believed that the intent of a question was unclear, the applicant was directed to seek clarification from the examiner (Item 9,

at 6). Applicants were informed that in order to pass the examination, they must achieve a grade of 80% or greater (Item 9, at 7).

9. Each of the applicants was administratively assigned to one of various "crews" for purposes of the operating test only; the assignment of an individual to one or another crew did not affect the questions on the written examination. In this manner, Mr. Tetrick was administratively assigned to "crew" No. 2. Copies of the operating test "scenario events" and "operator actions" for Mr. Tetrick's crew are included in the hearing file as Items 11 and 12. The examination outlines provided to Mr. Tetrick ("Set 3") are included in the hearing file as Item 13.

10. Mr. Tetrick's written examination answer sheet is included in the hearing file as Item 17. As graded by the NRC Staff, Mr. Tetrick incorrectly answered a total of 22 questions: Questions 4, 17, 24, 30, 31, 32, 34, 47, 50, 51, 63, 70, 76, 84, 86, 87, 90, 91, 92, 96, 98, and 100. Accordingly, on July 3, 1996, NRC Region II personnel recommended that Mr. Tetrick's application to upgrade his RO license to SRO status should be denied, notwithstanding the fact that he had successfully passed the operating test which had been administered as part of the licensure examination (Item 18).

11. In a letter dated July 19, 1996, Thomas A. Peebles of NRC Region II informed Mr. Tetrick that the Staff proposed to deny his application for a SRO license, due to his having failed the written examination (Item 19). Mr. Tetrick was advised that if he chose to accept the proposed denial, it would become final within 20 days and he could reapply for a SRO license after two months -- in connection with which he would be required to retake the written examination, but the operating test could be waived

(*Id.*). Mr. Tetrick was also advised, on the other hand, that he could request an informal NRC staff review or a hearing within 20 days; if he requested an informal review, he was to indicate which answers he believed were incorrectly graded and to provide the basis with supporting documentation for his contentions. Upon receipt of that request and supporting information, the Staff would review his contentions, reconsider its grading and inform him of the results, and he could then request a hearing pursuant to 10 C.F.R. § 2.103(b)(2) (*Id.*).

12. Also on July 19, 1996, the Staff provided the results of the Turkey Point licensure examination to FP&L (Items 15 and 16). The Staff reported that Mr. Tetrick had received an examination score of 78 out of 100, and therefore failed the written examination; the other 18 applicants for RO and SRO licenses had passed the examination with a minimum score of 80 or above (Item 5, Enclosure 1; Item 16, Enclosure (NRC Report No. 50-250/96-300 and 50-251/96-3( ))).

13. On July 30, 1996, Mr. Tetrick responded to the NRC Staff's letter of July 19, 1996, and requested an informal review of his written examination (Item 20). In particular, Mr. Tetrick requested review of his answers to four examination questions: Questions 24, 63, 84, and 96 (*Id.*). An informal review of Mr. Tetrick's contentions was then undertaken by the NRC Staff in NRC Region II, as well as independently by the Office of Nuclear Reactor Regulation (*see* Items 22, 24, 25), in accordance with the procedures found in NUREG-1021, Operator Licensing Examiner Standards, Rev. 7, Supp. 1, § ES-502 (Attachment 1 hereto). During this review, information was also obtained from FP&L (Item 23).

14. On September 12, 1996, the Staff transmitted a letter to Mr. Tetrick, in which it informed him that it had reviewed the grading of his written examination in light of the information he supplied and concluded that he had not passed that examination (Item 26). In particular, the Staff concluded that no answers to Question 24 are completely correct, and the Staff determined that this question should be deleted from the examination. However, the Staff further found that Mr. Tetrick's answers to Questions 63, 84 and 96 were clearly incorrect and it sustained its grading of his answers to these questions. As a result, Mr. Tetrick was determined to have a final grade of 78.8%, which remained below the minimum passing grade of 80%, and the Staff therefore concluded that Mr. Tetrick had failed the written examination. Accordingly, the Staff determined that the proposed denial of Mr. Tetrick's SRO license application should be sustained, and advised him of his right to request a hearing in connection therewith (Item 26).

15. On September 25, 1996, Mr. Tetrick filed a request for hearing in connection with the proposed denial of his SRO license application. In that document, Mr. Tetrick also included a request for reconsideration of his answer to Question 90.

16. The Staff's views with respect to Mr. Tetrick's answers to Questions 63, 84, 90 and 96 are as follows.

EXAMINATION QUESTION 63

17. 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.

The correct answer to this question is "b" -- "Sound the containment evacuation alarm." Mr. Tetrick's answer was "a" -- "Verify alarms by checking containment sump level recorder and spent fuel level indication."

18. In support of his answer, Mr. Tetrick's contentions may be summarized as follows. The Applicant asserts that the reactor control operator (RCO) is required to respond to alarms per O-ADM-219, "Annunciator Response Procedure [ARP] Usage," by reading the ARP in effect and performing the event mitigation strategy for the alarms received in the control room. The Applicant also states that, individually, these alarms

are priority 3, which require prompt rather than immediate action, and that the appropriate response for each alarm, per the associated ARP and off-normal operating procedures (ONOPs), is to verify the alarm. Therefore, he asserts, answer "a" should be accepted as an additional correct answer.

19. The NRC Staff reviewed Mr. Tetrick's contentions concerning this answer during its reconsideration of his written examination, as he requested. The Staff's analysis and conclusion concerning this question are as follows.

20. Reactor operators and senior reactor operators are expected to analyze alarms and determine the appropriate course of action based upon the specific plant conditions and indications. This expectation is reflected in step 3.4.1 of procedure O-ADM-219 (Item 20), which directs the RCO to respond to alarms based on color code priority and plant conditions (emphasis added).

21. Furthermore, steps 5.1.15 and 5.6.8 of procedure O-ADM-200, "Conduct of Operations," direct on-shift licensed operators involved in abnormal or emergency operations to believe and respond to their instrument indications until the instruments are proven to be incorrect. 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" (Item 24). That procedure has only one IMMEDIATE ACTION - to sound the containment evacuation alarm. In accordance with step 5.2.1 of O-ADM-211, "Emergency and Off-Normal Operating Procedure Usage"

(Item 25), operators shall be capable of performing steps identified as IMMEDIATE ACTION steps from memory, and step 3.5.1 requires the RCO to ensure that all immediate operator actions of the procedure in effect are performed (Item 25). 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 the loss of water which normally acts as a radiation shield) unless they are promptly notified by a containment alarm to evacuate the area.

22. Furthermore, step 5.13 of O-ADM-211 states that plant operating procedures have the following order of priority: Functional Restoration Procedures (FRPs), Optimal Recovery Procedures (ORPs), and Off-Normal Operating Procedures (ONOPs) (Item 25). In other words, operators are initially directed to follow these procedures, including the Off-Normal Operating Procedures, rather than the Alarm Response Procedures (ARPs) -- contrary to Mr. Tetrick's understanding. For these reasons, the Staff concluded that answer "b" is the only correct answer to this question.

23. On September 25, 1996, Mr. Tetrick filed a reply to the Staff's assessment of this matter, attached to his request for a hearing. Therein, Mr. Tetrick asserted as follows:

> The NRC analysis and conclusion contends that reactor operators and senior reactor operators are expected to analyze alarms and determine the appropriate course of action based upon specific plant conditions and indications.
>
> Applicant contends that performing an action based solely on annunciation alone is not the proper way to operate. Even though SFP LOW LEVEL and CONTAINMENT SUMP HIGH LEVEL alarms are

> mutually supportive and sufficient to enter 3-ONOP-033.2 "REFUELING CAVITY SEAL FAILURE" The annunciators should be verified by additional supportive information to preclude the possibility of annunciator failure. Additionally CONTROL ROOM ANNUNCIATOR RESPONSE procedure 3-ARP-097.CR states that for all alarms the ARP shall be consulted. Applicant therefore contends that answer "A" verify alarms is also a correct answer.

24. These additional assertions by Mr. Tetrick do not alter the Staff's view of this matter, for the following reasons. The stem of the question asks for the IMMEDIATE ACTION in response to these conditions. Procedure 0-ADM-211, Emergency (EOP) and Off-Normal Operating Procedure (ONOP) Usage, Section 5.2.1, states, "Operators shall be capable of performing steps identified as IMMEDIATE ACTION steps from memory." Section 5.3.1 states, "When the immediate actions have been completed, the operator shall begin reading the procedure." Procedure 0-ADM-201, Operations Procedure Usage, Section 5.1.4.1.a., states, "Procedures for which actions should be committed to memory are Immediate Actions in Emergency Operating Procedures (EOPs) and Off-Normal Operating Procedures (ONOPs)." Neither Control Room Annunciator Response procedure for H-1/1 SFP LO LEVEL or G-9/5 CNTMT SUMP HI LEVEL have immediate actions that would be required to be conducted by an operator from memory. Therefore, the only correct answer would be immediate actions required by EOPs or ONOPs that would be entered based on the initial plant conditions provided in the stem of the question. Based on the symptoms given in the initial plant conditions in the stem of the question only one ONOP could be entered, 0-ONOP-033.2, Refueling Cavity Seal Failure. The only <u>immediate</u> action in 0-ONOP-033.2 is "Sound

containment evacuation alarm." While Mr. Tetrick is correct that, eventually, the annunciators should be verified to be correct in accordance with the ARP, an immediate action must nonetheless be taken, in accordance with O-ONOP-033.2.

25. For the reasons stated above, answer "b" is the only correct answer to this question.

EXAMINATION QUESTION 84

26. 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.

The correct answer to this question is "a" -- "To ensure that only decay heat and reactor coolant pumps are adding heat to the RCS." Mr. Tetrick's answer was "c" -- "To alert the operator to take further corrective action if the reactor is NOT tripped."

27. In support of his answer, Mr. Tetrick's contentions may be summarized as follows. The Applicant compares the corresponding steps and the difference in the basis documents for procedure FR-S.1, "Response to Nuclear Power Generation/ATWS," and procedure EOP-E-0, "Reactor Trip or Safety Injection." In FR-S.1, the control rods are manually inserted, but they are not inserted in EOP-E-0. The basis documents for both procedures discuss decay heat and reactor coolant pump heat, but only the basis document for FR-S.1 discusses the need for inserting control rods if the reactor is not tripped. Therefore, he asserts, the NRC should accept answer "c" as an additional correct answer.

28. The NRC Staff reviewed Mr. Tetrick's contentions concerning this answer during its reconsideration of his written examination, as he requested. The Staff's analysis and conclusion concerning this question are as follows.

29. The basis documents for both FR-S.1 and EOP-E-0 clearly indicate that the first step of each procedure (*i.e.*, "verify reactor trip") is necessary to ensure that the only heat being added to the reactor coolant system is from the reactor coolant pumps and the radioactive decay of fission products (Item 20). The safeguards systems that protect the plant during accidents are designed on the basis of that assumption. Although the basis document for FR-S.1 goes on to state that the control rods should be manually inserted into the core to decrease reactor power if the reactor cannot be tripped, this does not change the fact that the reason (i.e., basis) for the (automatic and manual) action associated with this step of FR-S.1 (i.e., tripping the reactor or inserting control rods)

is to reduce the heat load to within the capacity of the safeguards systems. Therefore, answer "a" is the only correct answer to this question.

30. On September 25, 1996, Mr. Tetrick filed a reply to the Staff's assessment of this matter, attached to his request for a hearing. Therein, Mr. Tetrick asserted as follows:

> The NRC contends that the basis for E-O step one and FR-S.1 step one are the same and that there is only one answer.
>
> The applicant contends that FR-S.1 is a FUNCTION RESTORATION PROCEDURE and that it gives guidance to restore CRITICAL SAFETY FUNCTIONS. Since the reactor was verified not tripped in E-O step one you are sent to FR-S.1 Where the operator is directed to insert rods because the reactor is not tripped. Because FR-S.1 is a FRP and gives guidance the applicant contends that the basis for FR-S.1 is twofold, (1) to ensure only decay heat is added and (2) To direct corrective actions. Therefore the applicant asks that answer "C" also be accepted as a correct answer.

31. These additional assertions by Mr. Tetrick do not alter the Staff's view of this matter, for the following reasons. The "Basis Document" for both FR-S.1 and EOP-E-O clearly indicate that the first step of each procedure (*i.e.*, "verify reactor trip") is necessary to ensure that the only heat being added to the reactor coolant system is from the reactor coolant pumps and the radioactive decay of fission products (*i.e.*, decay heat) (Item 20). Although the Basis Document for FR-S.1 goes on to state that the control rods should be inserted into the core (thereby decreasing reactor power) if the reactor cannot be tripped, this does not change the fact that the reason or basis for the (automatic and manual) action associated with this step -- *i.e.*, tripping the reactor or inserting control

rods -- is to reduce the heat load to within the capacity of the safeguards systems. Thus, the "Basis Document" for FR-S.1 makes it clear that operators are directed to take the action cited by Mr. Tetrick (*i.e.*, insert control rods if the reactor has not tripped), in order to ensure that the only heat being added to the reactor coolant system is from decay heat and RCP heat. Accordingly, Mr. Tetrick's reliance upon this action is misplaced, and answer "a" is the only correct answer to this question.

<u>EXAMINATION QUESTION 90</u>

32. 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.*

The correct answer to this question is "a" -- "A vacuum in the RCS loops will result in level indication being <u>lower</u> than actual levels." Mr. Tetrick's answer was "b" -- "A vacuum in the RCS loops will result in level indication being <u>higher</u> than actual levels." (emphasis added).

33. In support of his answer, Mr. Tetrick's contentions may be summarized as follows. He asserts that 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. The Applicant therefore requests that this question be deleted.

34. The NRC Staff has reviewed Mr. Tetrick's contentions concerning this answer, following receipt of his September 25, 1996 request for a hearing on this matter. The Staff's analysis and conclusion concerning this question are as follows.

35. The Applicant is correct in stating that the differential pressure detector uses a dry leg. Procedure 0-PMI-041.110, "RCS Drain Down Level Calibration", step 6.4.3, discusses the dry leg when placing the level transmitter in service. Plant Drawing 5613-M-3041 Sheets 1 & 2 shows the level system discussed in Question 90. These drawings show that the reference leg used for the differential pressure cell is dry. This does not resolve the question, however, as asserted by Mr. Tetrick.

36. The stem of the question states that the candidates should assume the reference leg "remains full." This statement (inserted at the facility's request during the facility's pre-examination technical review) thus served to inform applicants that the reference leg remained constant and was not a variable in the problem that could affect

one's answer. The question thus tested applicants on their knowledge of hydraulic effects and theory rather than plant configuration.

37. The Staff agrees that the statement in the question's stem, directing applicants to "assume the reference leg remains full," could have been written more clearly, so as not to be confusing.[1]. However, regardless of the fact that Mr. Tetrick may have found this question to be confusing, it simply does not matter what type of reference leg is used -- wet or dry -- the answer to the question will be the same. The question assumes that the RCS is not properly vented. Applying this situation to the two different reference leg designs, the following would apply:

(a) For a dry reference leg, the variable leg (high pressure) level starts to decrease from the draining of the loop. This causes a decrease in differential pressure (DP) across the DP cell and in indicated level. Decreasing pressure (creating a vacuum) would result in a further decrease in the pressure sensed by the variable leg. This would result in a lower indicated level than actual level.

(b) For a wet reference leg, the variable leg (low pressure) level starts to decrease from the draining of the loop. This causes an increase in differential pressure across the DP cell and a decrease in indicated level. Decreasing pressure (creating a

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[1] We note, however, that if Mr. Tetrick found this aspect of the question to be confusing, he could have requested clarification from an examination proctor, but did not do so. Candidates were instructed to seek clarification from the proctors of the written examination, concerning matters which they found to be unclear (Item 9 at 6). Although the proctors documented all such requests for clarification from the candidates, there were no clarification requests regarding this question.

vacuum) would further decrease the pressure sensed by the variable leg. This would cause a higher differential pressure and the indicated level would be less than actual level.

(c) Both systems -- wet or dry -- would result in a level indication output that would be erroneously low given the circumstances of draining the RCS without a proper vent path. Thus, there is no basis for Mr. Tetrick's selection of answer "b", which stated that "a vacuum in the RCS loops will result in level indication being higher than actual levels" (emphasis added).

38. This question tests 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 in this question.

39. In sum, the question was designed to measure the candidate's ability to analyze the effects on reactor vessel level indication if an adequate vent path was not provided while draining the RCS. The fact that the reference leg is dry as opposed to filled with water is immaterial.

40. For the reasons stated above, answer "a" is the only correct answer to this question. Based upon the above, no change to the grading of this question should be made.

EXAMINATION QUESTION 96:

41. 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*

The correct answer to this question is "b" -- "Assistant Nuclear Plant Supervisor." Mr. Tetrick's answer was "d" -- Nuclear Watch Engineer."

42. In support of his answer, Mr. Tetrick's contentions may be summarized as follows. The Applicant recommends that choice "d" be accepted as an additional correct answer because procedure O-ADM-200, "Conduct of Operations," makes the Nuclear Watch Engineer (NWE) responsible for routinely relieving the Assistant Nuclear Plant Supervisor (ANPS) of the control room command and control function to enable the ANPS to leave the control room. Consequently, the NWE becomes the lowest level responsible for making entries in the Technical Specification Related Equipment Out-of-Service Index.

43. The NRC Staff reviewed Mr. Tetrick's contentions concerning this answer during its reconsideration of his written examination, as he requested. The Staff's analysis and conclusion concerning this question are as follows.

44. Procedure 0-ADM-213, "Technical Specification Related Equipment and Risk Significant SSC Out-of-Service Logbook," states that the ANPS is the lowest level position responsible for entering inoperable equipment in the subject index (Item 24). When the NWE relieves the ANPS, he then assumes the position of the ANPS. The NWE is not authorized to make entries in the subject index unless he is acting in the capacity of the ANPS, any more than he would be able to exercise any other functions of the ANPS unless he is acting in the ANPS capacity.[2] At such times, his authority exceeds the authority he is able to exercise as the NWE. Therefore, answer "b" is the only correct answer to this question.

SUMMARY OF NRC REVIEW

45. In summary, the NRC Staff has concluded the following based upon its review of the Applicant's contentions:

QUESTION NO.	NRC CONCLUSIONS
24	This question was deleted upon the Staff's informal review of Mr. Tetrick's contentions, as stated in the Staff's letter of September 12, 1996.
63	No change, answer "b" is the only correct answer.
84	No change, answer "a" is the only correct answer.
90	No change, answer "a" is the only correct answer.
96	No change, answer "b" is the only correct answer.

[2] This was re-confirmed by William Lindsey, Operations Training Supervisor at Turkey Point, in a telephone conversation with Thomas A. Peebles on January 23, 1997.

<u>Applicant's original examination grade:</u> 78.0% (78 of 100)

<u>Final written examination grade:</u> 78.8% (78 of 99)

46. Based upon the above, we have concluded that the Applicant's final grade of 78.8% remains below the minimum passing grade of 80%. Therefore, the Applicant has failed the written examination. The NRC Staff's denial of the Applicant's application for a SRO license should therefore be sustained.

47. I hereby certify that the foregoing is true and correct to the best of my knowledge, information and belief.

Brian Hughes
Brian Hughes

Thomas A. Peebles

Subscribed and sworn to before me
this 23rd day of January, 1997.
AS TO BRIAN HUGHES ONLY

Malinda L. McDonald
Notary Public

My commission expires: 12/1/97

46. Based upon the above, we have concluded that the Applicant's final grade of 78.8% remains below the minimum passing grade of 80%. Therefore, the Applicant has failed the written examination. The NRC Staff's denial of the Applicant's application for a SRO license should therefore be sustained.

47. I hereby certify that the foregoing is true and correct to the best of my knowledge, information and belief.

____________________	x Thomas A Peebles
Brian Hughes	Thomas A. Peebles

Subscribed and sworn to before me
this 23rd day of January, 1997.

Rachel Faulding
Notary Public

My commission expires: Notary Public, DeKalb County, Georgia
My Commission Expires January 4, 1998

D. PROCEDURE FOR THE INFORMAL REVIEW OF EXAMINATION OR TEST RESULTS

The following actions shall be taken when an applicant requests the Director, DRCH, to conduct an informal review of his or her license examination. These actions should be completed within 45 days after receiving the applicant's request.

1. The Chief, Operator Licensing Branch (OLB), will notify the appropriate region and forward a complete copy of the review package to that region. The Chief, OLB, will also prepare a letter for the Director, DRCH's, signature, notifying the applicant that his or her examination is being reviewed, and inform the OLB licensing assistant to track the licensing action.

2. The region shall evaluate the applicant's contentions within 5 working days after receiving the review package. If the contentions justify overturning the failure, the region will inform the Chief, OLB, of its findings and issue the appropriate license (the license will not be backdated). OLB will prepare a letter in the format of Attachment 1 to notify the applicant that the proposed denial was overturned and that a license will be issued.

3. If the region sustains the original denial, it shall inform the Chief, OLB, of its decision within 5 working days. The region shall provide the Chief, OLB, with a written summary and explanation of the grading changes that were made as a result of its review.

4. If the region sustains the original denial, the Chief, OLB, will convene a three-person board to review the applicant's documented contentions. The board shall be impartial (i.e., it will not include anyone who was in any way involved with the applicant's licensing examination), will include at least 2 certified examiners, and will be chaired by a supervisor (written examination review panels may be chaired by a senior license examiner). To promote objectivity, the 3 appeal board members will be obtained from different offices, whenever possible. The board may conduct its review in the appropriate regional office, at the facility where the operating test was administered, or by mail or telephone, depending on the extent of the applicant's contentions and the need for access to reference material (or the simulation facility in the case of operating test reviews).

5. For written examinations, the board shall review the original grading of the applicant's examination, the reference material supplied by the facility licensee, and the contentions and supporting documentation provided by the applicant. The review shall focus on those portions of the examination that were contested by the applicant and were not regraded by the region as part of the informal review process.

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For operating tests, the review board shall evaluate the examiner's comments, the examination report, and the simulator scenarios that were administered. The board shall then review the applicant's contentions in light of the information and documentation provided for the review (e.g., plant system descriptions, operating procedures, logs, chart recorder traces, process computer printouts) to determine if the applicant's contentions have merit. The board should ensure that specific examples of unsatisfactory performance were used to document each unsatisfactory (U) rating and that all comments are technically and procedurally correct.

The board will thoroughly document its findings and recommendations on each of the applicant's contentions.

6. The review board should evaluate the results of the region's review conducted in accordance with Section D.3 above. If the board's findings and recommendations differ from those of the regional reviewer(s), the board chairman shall discuss the matter with the original examiner and the regional office to determine the cause of the disparity. The board chairman should brief regional management on its preliminary findings and recommendations. Regional management may provide any additional information at that time, and the board will consider those concerns in its final recommendation.

7. The board will submit its findings and a recommendation to sustain or overturn the license examination failure to the Chief, OLB. If the region continues to have concerns, it should raise them to the Chief, OLB. The Chief, OLB, will make a final recommendation to the Director, DRCH.

8. The Director, DRCH, will consider the findings and recommendations of the review board and make a decision whether to sustain or overturn the applicant's license examination failure. The Director, DRCH, will notify the applicant in writing that his or her proposed denial was overturned (Attachment 1 and direct region to issue license) or sustained (Attachment 2).

E. PROCEDURE FOR PROCESSING APPLICATION DENIALS

If the region reviews the NRC Form 398 and the NRC Form 396 submitted by an applicant to demonstrate eligibility for a license examination and determines that the application is incomplete or that the applicant does not meet the requirements in 10 CFR 55.31, it will note the deficiencies and contact the applicant and the facility licensee and give them the opportunity to supply additional information to complete the application. If after the additional information is supplied, the applicant still does not meet the eligibility

BRIAN HUGHES

General

Brian Hughes has over twenty years nuclear experience. He recently completed a rotational assignment as a Senior Staff Engineer to the Nuclear Reactors Branch of the Advisory Committee on Reactor Safeguards (ACRS). He is a former licensed Senior Reactor Operator on a 3000 MWT commercial PWR (Indian Point Unit 3) and has extensive experience in commercial and non-power reactors. Mr. Hughes has either worked, inspected or administered operator license examinations at over 40 reactors in the United States. Mr. Hughes served as an Engineering Watch Supervisor with nuclear-related duties on the attack submarine USS Sturgeon SS(N)-637. Mr. Hughes received a Bachelors Degree in Business Administration from Iona College in 1981. A brief summary of Mr. Hughes' work experience is provided below.

NRR 1992-Present

Mr. Hughes has served as a Reactor Engineer in the Operator Licensing Branch of the Office of Nuclear Reactor Regulation (NRR). He prepares exemptions, reviews waivers, is the lead panel member for operator licensing appeals, and is the technical monitor for contract operator license examiners. He was an original member of the Cost Beneficial Licensing Action (CBLA) task force.

NRC Region I 1988-1992

Mr. Hughes served as both a regional inspector and an operator license examiner. He received the Region I "Employee of the Month" award for determining that a facility could not meet its FSAR commitment for accident mitigation. Mr. Hughes acted as site resident inspector at several facilities when needed.

Prior to NRC Employment

Prior to joining the NRC, Mr. Hughes worked for Consolidated Edison of NY at Indian Point (1973-1981). He served as an auxiliary operator and rose through the ranks to become a Licensed Senior Reactor Operator on Indian Point Unit 3 (IP3). At IP3, he loaded the first fuel assembly into the reactor. He also obtained first-hand knowledge of various upset conditions including loss of electrical grid, loss of instrument air, turbine blade failure, S/G tube failure, MSIV failure, intersystem LOCA and various reactor trips.

From 1981 to 1988, Mr. Hughes served as a consultant to numerous electric utilities. In this capacity, he served as an on-shift advisor in the control room at the Byron Nuclear Plant. This included all activities from initial core load to 100 percent power.

Thomas A. Peebles
Chief, Operator Licensing and Human Performance Branch
REGION II
U.S. Nuclear Regulatory Commission
Atlanta, Georgia 30323

<u>Education</u>: Bachelor of Electrical Engineering, Ohio State Univ., 1969
Senior Reactor Operator License, Surry Nuclear Plant, 1976

<u>Experience</u>:

1989-pres. <u>Chief, Operations Branch, DRS, Region II, NRC</u>

Manages implementation of NRC programs for operational inspections and reactor operator licensing. Assists HQ with modification of Operator Licensing policy. Represents NRC to public and industry officials.

1986-1989 <u>Section Chief, Division of Reactor Projects, Region II, NRC</u>

Managed and implemented routine and reactive inspection programs at three operating reactor sites. Evaluated licensee performance and safety significance of plant events. Recommended NRC followup.

1982-1986 <u>Senior Resident Inspector, Region III and Region II, NRC</u>

Managed and conducted safety inspection program at sites in both Regions. Represented the NRC to public and utility officials.

1980-1982 <u>Project Engineer and Resident Inspector, Region II, NRC</u>

Assisted in managing the inspection and licensing program for Farley Nuclear Plant in Region. At the Farley site, was Resident Inspector.

1978-1980 <u>Supt. Tech. Svcs., Surry Nuclear Plt., VA. ELECT.& POWER CO.(VEPCO)</u>

During the first commercial steam generator replacement project, managed the disciplines of Chemistry, Site Engineering, Health Physics, Inst. & Control maintenance, and Non-destructive Testing.

1973-1978 <u>Design Control Engineer, Licensing Eng., Proj. Mgr. (VEPCO)</u>

Reviewed design changes for Code compliance; managed Operating License changes for Surry NP with NRC and vendors; managed construction of Surry $26 million construction addition.

1972-1973 <u>Electrical Engineer, Newport News Shipbuilding & DD Co.</u>

Coordinated system design interfaces for a Navy reactor prototype. Electrical Test Engineer for two nuclear submarine overhauls.

1969-1972 <u>Design-Test Engineer, McDonnell Douglas Corp.</u>

Designed and supervised the building, maintenance and use of aircraft simulators for man-machine interface studies.

<u>Professional Association:</u> Reg. Professional Engineer, VA and FL;
Certified NRC Inspector and Operator Licensing Examiner

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

'97 JAN 24 A11:30

In the Matter of)
)
RALPH L. TETRICK) Docket No. 55-20726-SP
)
(Denial of Senior Reactor)
Operator License))

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

CERTIFICATE OF SERVICE

I hereby certify that copies of "WRITTEN PRESENTATION OF NRC STAFF" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or as indicated by an asterisk through deposit in the Nuclear Regulatory Commission's internal mail system, or as indicated by a double asterisk by express mail this 23rd day of January 1997.

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Sherwin E. Turk

Sherwin E. Turk
Counsel for NRC Staff