February 23, 2001

Mr. John K. Wood
Vice President - Nuclear
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT - INITIAL LICENSE EXAMINATION REPORT 50-440/01-301(DRS)

Dear Mr. Wood:

On January 19, 2001, the NRC completed initial operator licensing examinations at your Perry Nuclear Power Plant. The enclosed report presents the results of the examination which were discussed on February 16, 2001, with Mr. B. Boles.

Perry Nuclear Power Plant training department personnel administered the written examination on January 19, 2001 and NRC examiners administered the operating examinations during the weeks of January 8 and January 15, 2001. Five reactor operator and seven senior reactor operator applicants were administered license examinations.

The results of the examinations were finalized on February 16, 2001. Six applicants passed all sections of their respective examinations resulting in the issuance of one reactor operator license and five senior reactor operator licenses. A seventh applicant passed all sections of the examinations; however, will not be issued a senior reactor operator license until the conditions of a pre-approved eligibility waiver are complete. An eighth applicant passed all sections of the examinations; however, will not be issued a reactor operator license until potential examination appeals are resolved. Four applicants demonstrated unsatisfactory performance on the written examination and were not issued their respective operator licenses.

The NRC staff considered four examination failures out of a total of twelve applicants examined to be an abnormally high failure rate. Your staff is expected to evaluate these failures to determine whether deficiencies exist in your initial licensed operator training program. In addition, the number of post-examination comments was considered high for an examination authored by a facility. We understand that your staff initiated a condition report to address this issue.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning this examination.

Sincerely,

/RA/

David E. Hills, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-440
License No. NPF-58

Enclosures:
1. Operator Licensing Examination Report 50-440/01-301(DRS)
2. Facility Comments and NRC Resolutions
3. Simulation Facility Report
4. Written Examinations and Answer Keys (RO & SRO)

cc w/encls 1, 2, 3:
B. Saunders, President - FENOC
N. Bonner, Director, Nuclear Maintenance Department
G. Dunn, Manager, Regulatory Affairs
K. Ostrowski, Director, Nuclear Services Department
T. Rausch, Director, Nuclear Engineering Department
R. Schrauder, General Manager, Nuclear Power Plant Department
A. Schriber, Chairman, Ohio Public Utilities Commission
Ohio State Liaison Officer
R. Owen, Ohio Department of Health

cc w/encls 1, 2, 3, 4: R. Collings, Training Manager

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A. Schriber, Chairman, Ohio Public Utilities Commission
Ohio State Liaison Officer
R. Owen, Ohio Department of Health

cc w/encls 1, 2, 3, 4: R. Collings, Training Manager

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	50-440/01-301
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	P.O. Box 97 A200 Perry, OH 44081
Dates:	January 8 - 19, 2001
Inspectors:	A. M. Stone, Chief Examiner M. Bielby, Examiner R. Lantz, Examiner
Approved by:	David E. Hills Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000440-01-301, on 01/08-19/2001, FirstEnergy Nuclear Operating Company, Perry Nuclear Power Plant, Unit 1. Other Activities.

The announced operator licensing initial examination was conducted by regional examiners in accordance with the guidance of NUREG-1021, Operator Licensing Examination Standards for Power Reactors, Revision 8.

Examination Summary:

- Seven applicants were administered senior reactor operator license examinations and five applicants were administered reactor operator license examinations. Six applicants passed all sections of their respective examinations resulting in the issuance of one reactor operator license and five senior reactor operator licenses. A seventh applicant passed all sections of the examinations; however, will not be issued a senior reactor operator license until the conditions of a pre-approved eligibility waiver are complete. An eighth applicant passed all sections of the examinations; however, will not be issued a reactor operator license until potential examination appeals are resolved. Four applicants demonstrated unsatisfactory performance on the written examination and were not issued their respective operator licenses. (Section 4OA5.1)

- The number of post-examination comments was considered high for an examination authored by a facility. (Section 4OA5.1)

Report Details

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

The NRC examiners conducted announced operator licensing initial examinations during the weeks of January 8 and January 15, 2001. The facility's training staff used the guidance established in NUREG-1021, Operator Licensing Examination Standards for Power Reactors, Revision 8, to prepare the examination outline and to develop the written and operating examinations. The facility's training staff administered the written examination on January 19, 2001. The NRC examiners administered the operating examination January 8 through January 18, 2001. Five reactor operator applicants and seven senior reactor operator applicants were examined.

b. Findings

Written Examination

The NRC examiners determined that the written examination, as originally submitted by the licensee, was within the range of acceptability expected for a proposed examination. Examination changes, agreed upon between the NRC and the licensee, were made according to NUREG-1021. The licensee provided comments on seven written examination questions that were administered to the applicants. Three of these questions appeared on the reactor operator examination and six of the questions appeared on the senior reactor operator examination. The licensee’s specific comments and the NRC’s resolution of those comments are included in Enclosure 2 to this report. The number of post-examination comments represented greater than five percent of the written examination and was considered high. The licensee initiated a condition report to address this issue.

Operating Test

The NRC examiners determined that the operating test, as originally submitted by the licensee, was within the range of acceptability expected for a proposed examination. Examination changes, agreed upon between the NRC and the licensee, were made according to NUREG-1021.

Examination Results

Six applicants passed all sections of their respective examinations resulting in the issuance of one reactor operator license and five senior reactor operator licenses. A seventh applicant passed all sections of the examinations; however, will not be issued a senior reactor operator license until the conditions of a pre-approved eligibility waiver are complete. An eighth applicant passed all sections of the examinations; however, will not be issued a reactor operator license until potential appeals are resolved. Four applicants demonstrated unsatisfactory performance on the written examination and were not issued their respective operator licenses.

.2 Examination Security

a. Inspection Scope

The examiners reviewed and observed the licensee's implementation of examination security requirements during the examination preparation and administration.

b. Findings

The NRC examiners determined that the licensee's examination security practices associated with the development and administration of the operator license examinations were satisfactory.

4OA6 Management Meetings

Exit Meeting Summary

The chief examiner presented the examination team's preliminary observations and findings to Mr. J. Woods and other members of the licensee management on January 29, 2001. The licensee acknowledged the observations and findings presented and did not identify any proprietary information.

The chief examiner presented the final results of the examination to Mr. B. Boles on February 16, 2001. The licensee stated that a condition report was initiated to investigate the causes for the high applicant failure rate and number of post-examination comments.

Report Details

Licensee

J. Wood, Vice President-Nuclear
R. Schrauder, General Manager, Nuclear Power Plant Department
#* B. Boles, Operations Manager
#* R. Collings, Training Manager
D. Johnson, Training Instructor
J. McHugh, Training Supervisor
D. Pearson, Training Instructor
G. Rhoads, Quality Assurance
K. Russell, Compliance Engineer
#* R. Strohl, Operations Superintendent

NRC

* D. Hills, Chief, Operations Branch
# M. Bielby, Senior Operations Examiner
#* A.M. Stone, Chief Examiner
R. Vogt-Lowell, Resident Inspector, Perry

Attended a preliminary exit meeting on January 29, 2001.
* Participated in the examination results discussion on February 16, 2001.

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
NRC	Nuclear Regulatory Commission

Facility Comments and NRC Resolutions

Written Examination Comments:

Question #17 (Reactor Operator (RO))

Facility Comment: Change answer to A.

Section 2.3.2. of Integrated Operational Instruction (IOI)-09, “Refueling,” requires the operator to monitor flux level and reactor period during all core alterations which could result in adding positive reactivity. In addition, Perry Administrative Procedure (PAP)-0201, “Conduct of Operations,” step 6.5.3, clearly states that operation of mechanisms and apparatuses other than reactivity control, which may affect the reactivity or power level of the reactor shall only be accomplished with the knowledge and consent of the licensed operator at the controls. Therefore, the correct answer is A.

NRC Resolution: Agreed with facility. Changed answer to A.

The question asks for the reason why communication is established between personnel performing core alterations and the control room per IOI-9, step 4.1.8. Distractor C, the original answer, states that continuous communication allows the operator at the control to monitor for inadvertent criticality and inform the refuel floor personnel of such an event. This answer is consistent with section 2.3.2 of IOI-9. The licensee also supported this answer using section 6.5.3, “Communications” of PDB-R001, “Plant Data Book,” which states that communications capability ensures that refueling floor personnel can be promptly informed of significant changes (...) in core reactivity condition during movement of fuel. Based on this, distractor C was accepted as the correct answer.

The NRC agrees that PAP-0201 states that the operator at the controls needs to be knowledgeable of core alterations and that his/her consent is required; therefore, distractor A is a true statement and is a correct answer.

History:

No changes were made during question validation. The facility originally stated that unit supervisors authorizes core alterations, not the operator at the controls. Perry Administrative Procedure PAP-0201 was not listed as a technical reference.

Question #65 (Senior Reactor Operator (SRO)); #65 (RO)

Facility Comment: Accept answers C and D.

The annunciator response procedure subsequent action #1 directs the operator to depress the LEAD button and to confirm the LEAD light is illuminated. According to Operations Administrative Instruction-0502, control panel controls will be written in capital letters thus leading the candidate to a second correct answer of C. The candidates assumed the LEAD pushbutton needs to be backlit to be in lead to meet the intent of subsequent action #1 of the annunciator response procedure.

<u>NRC Resolution:</u> Agreed with the facility comments in part. Accepted answers C and D.

The question starts with initial conditions - Subloop 1 of reactor recirculation hydraulic power unit B is in LEAD and subloop 2 is READY. The question then asserts that alarm "FCV B HPU NEEDS MAINTENANCE" is received based on high oil temperature. The stem question asks, "Which one of the following describes the operational status of HPU "B"? Distractor C states that subloop 2 is in operation but <u>not</u> in LEAD and distractor D states the subloop 2 is in operation but <u>is</u> in LEAD.

The automatic action section of the annunciator response procedure states, "Standby sub-loop will shift to LEAD on low oil level or high oil temperature." In this case, the term LEAD refers to the mode of operation, not the status of the LEAD light. Subsequent Operator Action #1 directs the operator to depress its LEAD pushbutton and confirm the LEAD backlight comes on. The question did not state the operator took any actions once the alarm came in; thus only automatic actions are assumed. Therefore, HPU "B" would be running in LEAD but the LEAD light would not be backlight. The candidates were incorrect in assuming the LEAD pushbutton needed to be backlight for HPU to be in the lead mode of operation.

However, the NRC agrees that the capitalization of the word LEAD when referring to the mode of operation, though consistent with the annunciator response procedure, was confusing. The normal lineup for HPUs is to have one subloop operating in lead and the other in standby unless a system problem occurs. Both distractors C and D correctly state that subloop 1 swapped to the maintenance mode and subloop 2 swapped from standby to operation. If one assumes LEAD refers to the status of the lights, then the only correct answer is C. If one assumes LEAD refers to the operational mode, then the only correct answer is D. During the examination, one candidate asked whether LEAD referred to the light or system function. Because the term LEAD could refer to the status of the light, the NRC will accept answers C and D.

<u>History:</u>

No changes were made during question validation.

<u>Question #71 (SRO); #71 (RO)</u>

<u>Facility Comment:</u> Delete the question; no correct answers.

The question asks for the Average Power Range Monitor (APRM) Upscale Thermal Power Trip Setpoint based on a given set of plant parameters. Original answer B incorrectly used the APRM Upscale Thermal Power <u>Allowable</u> Value equation (0.628W+63.8%). The correct equation is (0.628W+60.9%). This yields an answer of 101.6% which is not included in the distractors.

<u>NRC Resolution:</u> Agreed with the facility; deleted the question.

In their original submittal, the facility provided the Technical Specification 3.3.1.1 APRM Upscale Thermal Power Allowable Value equation (0.628W+63.8%) and did not indicate a different trip setpoint equation existed. In their post-exam comments, the facility included PDB-R001, “Plant Data Book”, Attachment 2, “Table 1-RPS Trip Setpoint Table,” which lists the trip setpoint formula as (0.628W+60.9%). Because there are no correct answers using this new formula, this question was deleted.

History:

No changes were made during question validation.

Question #6 (SRO)

Facility Comment: Accept C and D as correct answers.

PEI-bases clearly states at levels > -42.5 inches adequate core cooling is assured. The covered portion of the core generates sufficient steam to preclude the peak clad temperature of the hottest fuel rod from exceeding 1800°F. This results in a second correct answer (D).

NRC Resolution: Agreed with facility in part. Accepted C and D as correct answers.

The question asked the candidate to determine which set of conditions would assure adequate core cooling given the initial conditions. In distractor C, the given water level, -40 inches, was above the minimum zero-injection reactor pressure vessel water level (defined as -42.5 inches) and also stated that no safety relief valves were open. In distractor D, the given water level was also -40 inches; however, five safety relief valves were assumed opened.

The PEI basis document stated that opening the safety relief valves prior to the water level decreasing to the minimum zero-injection level results in less effective steam cooling due to lower delta temperatures to drive the heat transfer and reduced period of time during which the core remained adequately cooled with no injection. During question validation, distractor D was ruled incorrect because adequate core cooling could not be assured since five safety relief valves were opened prior to reaching -42.5 inches. Therefore distractor C was considered the only correct answer.

However, because the question stem did not directly ask for the most effective conditions or the conditions which would result in a longer period of adequate core cooling, distractor D will be considered a correct answer.

History:

No changes were made during question validation.

<u>Question #13 (SRO)</u>

<u>Facility Comment:</u> Accept C and D as correct answers.

Perry Administrative Procedure (PAP)-0201, “Conduct of Operations,” requires the at-the-controls operator to shutdown the reactor when a scram setpoint is exceeded. This requires ONI-C71-1, “Reactor Scram” entry which results in a second correct answer of C. The PEI bases discusses and allows use of other instructions with the PEIs. In this case, ONI-C71-1 entry and use is taught and re-enforced in the student’s simulator sessions. Distractors C and D are correct.

<u>NRC Resolution:</u> Agreed with the facility in part. Accepted answers C and D.

The question states that the reactor is at 100 percent power when annunciators associated with high reactor vessel water level are received. The question further states that level is at 225 inches and the reactor remained at 100 percent power. The question stem asks which direction should be given to the at-the-controls operator. Distractors C and D address scramming the reactor using the ONI-C71-1 and PEI-B13, respectively.

In this question, reactor water level exceeded the level 8 automatic scram setpoint; however, the reactor remained critical. This condition is an entry point into PEI-B13 and corresponds to the original answer D. Distractor C does not address the required entry into PEI-B13; therefore, it is incomplete with respect to the required unit supervisor actions. If the question stem asked for the required unit supervisor actions, distractor C would be incorrect since entry into PEI-B13 is required.

However, the PEI bases document states that “the PEIs are the higher tier documents and shall direct the primary activities to ensure safe operation when the entry conditions are met during an emergency. The decision to utilize other approved procedures during PEI execution rests with the Unit Supervisor.” The unit supervisor may <u>direct</u> the operator-at-the-controls to carry out the immediate actions of ONI-C71-1 based on this discretion while the unit supervisor enters the appropriate PEI.

Although distractor C is incomplete by not discussing entry into PEI-B13, the direction to manually scram the reactor and carryout the immediate actions of ONI-C71-1 is not contrary to the actions within PEI-B13 or it’s bases.

<u>History:</u>

Eliminated “the main turbine did NOT trip and the reactor did NOT automatically scram” from the stem as this statement provided a cue that a reactor scram should have occurred thus eliminated two distractors. During validation week, there was discussion on whether the operator would enter ONI-C71-1 or PEI-B13. The facility provided assurance that the PEI-B13 should be entered; however, this concern was not resolved completely and resulted in a post-examination comment.

Question #18 (SRO)

Facility Comment: Change correct answer to B.

Perry Administrative Procedure (PAP)-0205, “Operability of Plant Systems,” allows the unit supervisor to waive independent verification for ALARA reasons. This is answer B.

NRC Resolution: Accepted facility’s comment. Changed answer key to “B.”

At the time of examination validation, a standing order disallowed waiving independent verification due to a conflict in the quality assurance program. On December 21, 2000, the standing order was canceled. Therefore, the correct answer at the time of the exam was “B.”

History:

The original question was categorized an unsatisfactory because distractors a and d were opposite statements; thus making the question a 50/50 choice. The distractors were modified. These modifications were independent of the post-exam concerns.

Question #23 (SRO)

Facility Comment: Accept C and D as correct answers.

The PEI-B13 flowchart gives -25 to +100 inches on conditions greater than four percent power. With level at 115 inches and reactor power at 10 percent level has not been lowered at this point. The last level band given would have been -25 to +100 inches thus leaving C as a second correct answer. The students assumed by the information given that he has not met the conditions of the hold step which lowers level and gives as his answer the last possible level band. C and D are both correct.

NRC Resolution: Delete question.

The question asks what is the level band required by PEI-B13 based on initial conditions. The initial conditions did not state the trend for the parameters cited and did not give a time frame on when the event occurred.

The intent of the question was to test the candidate’s knowledge of emergency operating procedure strategies, specifically, level control during an anticipated transient without a scram. A continuous decision block in the Level Control leg of PEI-B13 requires an operator to monitor power, level, suppression pool temperature, safety relief valve status and drywell pressure. Once predetermined values are met, the operator would terminate and prevent injection with the intention of deliberately lowering reactor water level until a new set of conditions were met. (For example, when power decreased to below four percent.) When this new set of conditions are met, a level band of -25 to (the point level was lowered to) would be required. This level band was described in distractor D.

The initial conditions in the question met the predetermined values; therefore, the operator would begin lowering level. However, the parameters given in the question stem did not meet the new set of conditions; therefore, the level band of -25 to (the point level was lowered to) would not be applicable. Therefore, distractor D is not a correct answer.

The facility stated that the candidates assumed that when the continuous decision block was reached the first time, the predetermined values were not met. The candidate then proceeded to the next continuous decision block which would lead to the level band of -25 to +100 inches. The facility contends that distractor C is correct. However, because the initial conditions did not state a time period, one cannot assume what the parameters were at the time the continuous decision block was first encountered. Therefore, distractor C is not a correct answer. Using the same reasoning, distractors A and B are also incorrect.

Because there are no correct answers, this question was deleted.

<u>History:</u>

Distractor D (the answer) contained a psychometric flaw, in that, it contained more information such as power level than the other distractors. The reference to power was removed which inadvertently resulted in no correct answer.

Operating Test Comments:

The licensee did not submit post-examination comments pertaining to the operating test.

Enclosure 3

SIMULATION FACILITY REPORT

Facility Licensee: Perry Nuclear Power Plant, Unit 1

Facility Docket No.: 50-440

Operating Tests Administered: January 8 - 18, 2001

The following documents observations made by the NRC examination team during the initial operator license examination. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR Part 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

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

ITEM	DESCRIPTION
1. HPCS diesel governor and voltage regulator status lights	During the performance of JPM B.1.b, “Parallel and load the HPCS diesel,” the status lights for the governor and voltage regulator did not light up as expected when the switches were taken out of normal. This did not affect the performance of the JPM.
2. Fuses sizes	The fuses used in the simulator for the scram pilot and SDV vent and drains are the same size (5 amp); however, the fuses sizes are different in the plant (5 and 20 amp).
3. Charging system response	The system experienced perturbations when a single control rod scrammed during a scenario. The system response distracted two crews.

Enclosure 4

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)