November 6, 2000

Mr. Guy Campbell
Vice President - Nuclear
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE - NRC EXAMINATION REPORT 50-346/00-301(DRS)

Dear Mr. Campbell:

On October 6, 2000, the NRC completed initial operator licensing examinations at your Davis-Besse Nuclear Power Station. The enclosed report presents the results of the examination.

Your training department personnel administered the written examination on October 6, 2000. NRC examiners administered the operating examination during the same week. Four of your previously licensed applicants were administered senior reactor operator upgrade examinations. The license applicants' performance evaluations were finalized on October 20, 2000. All applicants passed all sections of their corresponding examinations and were issued senior reactor operator licenses.

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 (PARS) 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. 50-346
License No. NPF-3

Enclosures:
1. Operator Licensing Examination Report 50-346/00-301(DRS)
2. Facility Comments and NRC Resolutions
3. Simulation Facility Report
4. Written Examination and Answer Key (SRO)

cc w/encls 1, 2:
B. Saunders, President - FENOC
H. Bergendahl, Plant Manager
D. Lockwood, Manager, Regulatory Affairs
M. O'Reilly, FirstEnergy
State Liaison Officer, State of Ohio
R. Owen, Ohio Department of Health
A. Schriber, Chairman, Ohio Public Utilities Commission

cc w/encls 1, 2, 3: W. Mugge, Training Manager

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	50-346/00-301(DRS)
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449-9760
Dates:	October 2 - 6, 2000
Examiners:	H. Peterson, Chief Examiner D. McNeil, Examiner
Approved by:	David E. Hills Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000346-00-301(DRS), on 10/02-06/2000; FirstEnergy Nuclear Operating Company; Davis-Besse Nuclear Power Station. 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. No significant findings were identified.

<u>Examination Summary:</u>

- Four senior reactor operator applicants were administered the written examination and operating tests. All four applicants passed all portions of their respective examinations and were awarded senior reactor operator licenses (Section 4OA5.1).

Report Details

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Inspection Scope

The NRC examiners conducted announced operator licensing initial examinations during the week of October 2, 2000. The facility licensee's training staff used the guidance prescribed in NUREG-1021, Operator Licensing Examination Standards for Power Reactors (ES), Revision 8, dated April 1999, to prepare the outline, and develop the written examination and operating test. The examiners administered the operating test consisting of job performance measures (JPMs) and dynamic simulator scenarios, October 3 - 4, 2000. The facility licensee administered the written examination on October 6, 2000. Four senior reactor operator applicants received written examinations and operating tests.

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. Although the quality of the examination was acceptable, the examiners identified that the licensee's methodology to systematically and randomly select knowledge and abilities (K/As) for the written examination outline was not completely in accordance with examination development guidelines. Section D.1.b and Attachment 1 of ES-401, NUREG-1021, Revision 8, describes the guidance on the methodology in selecting specific K/As for the written examination outline. The licensee's method only selected K/As up to the general categories (i.e., K1, K2, A1, A2, etc.) and did not extend to individually select the specific K/A's (i.e., K1.01, K2.02, A1.02, A1.03, etc.). Examiners had previously made a verbal comment to the licensee regarding a similar observation during their September 1999 license examination. The examiners determined that the independent examination developer's influence in selecting the specific K/A's was minimized, and therefore, the examination was adequate. The examiners informed the licensee that Supplement 1 to NUREG-1021, Revision 8, additionally clarifies the required methodology and expectations for specific K/A selections. A condition report was generated by the licensee to resolve this issue, Condition Report No. 2000-2580.

The NRC examiners independently graded the written examination and concluded that all four applicants achieved or surpassed the passing criteria of 80.0 percent. On October 17, 2000, the licensee submitted four post-examination comments on the written examination involving correction of referenced technical information. The licensee also submitted one administrative correction. All five proposed corrections were accepted and the examinations were graded accordingly. The comments and the NRC's resolutions are contained in Enclosure 2 of this report.

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.

All applicants demonstrated satisfactory performance in all three areas of the operating examination (administrative, control room and systems walkthrough, and integrated plant response).

.2 Examination Security

a. Inspection Scope

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

b. Findings

The NRC examiners determined that the licensee's overall 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. Bergendahl and other members of the licensee management on October 5, 2000. The licensee acknowledged the observations and findings presented, and did not identify any information as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

H. Bergendahl, Plant Manager
D. Bondy, Senior Nuclear Training Advisor
R. Coad, Plant Operations Manager
J. House, Supervisor Operations Training
D. Imlay, Plant Operations Superintendent
D. Miller, Supervisor Regulatory Affairs Compliance
W. Mugge, Nuclear Training Manager
G. Skeel, Security Manager (Acting)
B. Young, Qualification Instructor

NRC

T. Kozak, Chief, Reactor Projects Branch 4
K. Zellers, Senior Resident Inspector, Davis-Besse
D. Simpkins, Resident Inspector, Davis-Besse
H. Peterson, Chief Examiner
D. McNeil, Examiner

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS

CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
ES	Examination Standards
JPM	Job Performance Measure
K/A	Knowledge and Abilities
NRC	Nuclear Regulatory Commission

Facility Comments and NRC Resolutions

The licensee submitted four (4) post written examination comments (questions 22, 41, 49, and 60) and one administrative correction (question 36) that could affect the final grading of the written examinations. The four comments concerned correction of referenced technical information.

Question No. 22
The following plant conditions exist:

- Reactor is tripped.
- Subcooling margin meters indicate 0°F.
- EDG1 has tripped on overspeed and cannot be reset.
- D1 bus has experienced a lockout.
- SBODG has failed to start.

Which one of the following is the correct response to this event?

a. Maintain plant conditions and continue effort to restore C1 bus.
b. Begin a cooldown at < 100°F/hr. to Mode 5.
c. Fully open both Atmospheric Vent Valves.
d. Open the PORV until the Core Flood Tanks start to discharge.

Answer: c

<u>Comment</u>:
"The stem of the question fails to state that a loss of off-site power has occurred which is required to qualify choice 'c' as the correct answer. Without assuming a loss of off-site power then, C1 bus is still available which would make the correct answer choice 'b' according to DB-OP-02000. The routing through DB-OP-02000 is to perform section 4 until step 4.11 is reached, which routes to section 5 if a lack of Subcooling Margin exists. In section 5, steps are performed until step 5.15 is reached, which routes to section 11, RCS Saturated Steam Generator Removing Heat. Section 11 then directs an RCS cooldown at step 11.11, according to DB-OP-06903, Plant Shutdown and Cooldown. DB-OP-06903 will direct a cooldown at 100°F per hour to Mode 5. The original answer of choice 'c' would only be correct if there was a loss of off-site power."

<u>NRC Resolution:</u>
Recommendation was accepted. The question stem as originally written indicated conditions from a loss of off-site power (i.e., operation of emergency and station blackout diesel generators). Both the licensee and the NRC reviewed the question inherently assuming the condition of a loss of off-site power, and no comments were made. The statement of the condition of "loss of off-site power" in the stem of the question would have clarified the question more. Based on this lack of clarifying statement, the question could be answered assuming off-site power was not completely lost and that bus C1 was still available. Review of plant procedures technically noted that cooldown rate can exceed 100°F per hour; however, maintaining cooldown rate less than 100°F per hour as in answer choice 'b' was also acceptable. Subsequently, the licensee's recommendation was accepted and the correct answer choice was changed from 'c' to 'b'.

Question No. 36
The following plant conditions exist:

- A steam generator tube rupture has occurred following a loss of off-site power.
- All systems performed as expected.
- The ruptured steam generator indicates 100 inches.

In accordance with DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture, what is the maximum cooldown rate allowed to reach the minimum RCS temperature to isolate the ruptured steam generator?

a. 50°F/hour to 500°F Thot
b. 100°F/hour to 520°F Thot
c. 50°F/hour to 520°F Thot
d. 100°F/hour to 500°F Thot

Answer: b.

<u>Comment:</u>
"The original submittal and final question indicated choice 'b' as the correct answer. The examination outline cross-reference explanation referenced step 8.14 of DB-OP-02000 which clearly indicates choice 'c' as the correct answer. The answer was incorrectly transposed on to the question and answer key. The correct answer is choice 'c'. The answer of choice 'b' on the key is incorrect because reactor coolant pumps would not be running due to the loss of off-site power."

<u>NRC Resolution:</u>
Recommendation was accepted. The review of the plant procedure indicates choice 'c' as the correct answer. However, the question cross-reference indicated that the correct choice was 'b.' This was an administrative error on part of the licensee. The recommendation was accepted and the correct answer choice was changed from 'b' to 'c.'

Question No. 41
The following plant conditions exist:

- Reactor is at 100% power.
- Pressurizer level is 220 inches on the chart recorder.

The pressurizer level on the chart recorder has dropped to 70 inches indicated over 5 minutes.

Which one of the following statements is correct concerning the decrease in pressurizer level?

a. Pressurizer level reference leg pressure has decreased.
b. Pressurizer level sensing leg pressure has decreased.
c. Pressurizer level temperature compensation has failed low.
d. Pressurizer level temperature compensation has failed high.

Answer: a

Comment:
"The original submittal had choice 'b' as the correct answer. During the exam review process, the answer was incorrectly changed to choice 'a.' After reviewing the fundamental training material associated with a wet reference leg level transmitter, choice 'b' is the correct answer. The answer choice 'a' on the key is not correct."

NRC Resolution:
Recommendation was accepted. During the exam review process, the NRC questioned the inconsistency between the answer choice and the cross-reference justification. The original answer choice 'b' pertained to a sensing leg problem, but the cross-reference justification noted that the answer was due to a reference leg problem. The licensee reviewed the question and the answer, and concurred that the correct answer choice should have been 'a' and not 'b.' The licensee's cross-reference information was incorrect. After reviewing the new reference material submitted by the licensee, the fundamental training material supports the sensing leg problem as being the correct answer. The licensee's recommendation was accepted and the correct answer choice was changed from 'a' to 'b.'

Question No. 49
The plant is at 100% power when Alarm 5-6-E CRD SEQ FAULT alarms.

Which one of the following would be the correct action to take for the failed rod position indication?

a. Reduce power to ≤ 45% and restore failed indication.
b. Verify control rod position on the absolute position indicators and maintain current power.
c. Verify control rod position on the relative position indicator and reduce power to 60%.
d. Verify control rod position on the zone reference lights and maintain current power level.

Answer: d

Comment:
"Within the stem of the question, it is stated that alarm 5-6-E has actuated. According to the associated alarm procedure, DB-OP-02005 step 3.3, the positions of the control rods are verified using the Absolute Position Indication. Using DB-OP-02005 makes choice 'b' a correct answer and using Technical Specification 3.1.3.3 makes choice 'd' a correct answer."

NRC Resolution:
Recommendation was accepted. The licensee only referenced the Technical Specification 3.1.3.3 as justification for choice 'd' as being the correct answer. The technical specification information was correct and very specific which made choice 'd' as the correct answer. The NRC reviewed the licensee's procedure DB-OP-02005 and determined that it also justified choice 'b' as a correct answer. This was a technical error on part of the licensee, where two separate references supported conflicting answers. After further review, the NRC found that the actions supported by the annunciator response procedure, DB-OP-02005, were the actions the operator would be expected to take in a short period of time following the identification of the problem. The verification actions required by the technical specifications are action items required to be taken within eight hours to support continued plant operation. Therefore, both answers were found to be correct depending upon the response time associated with the corrective actions. The licensee's recommendation was accepted and the correct answer was changed to accept either 'b' or 'd.'

Question No. 60
A miscellaneous waste monitor tank release needs to be performed. RE 1878A and B have been declared inoperable.

Which of the following is required in order to perform the release?

1. Must recirc two tank volumes.
2. Must have two independent samples analyzed.
3. Must have two independent flow rate calculations.
4. Must have two independent verifications of the discharge flowpath valving.
5. Must reprocess the monitor tank prior to release.

a. 2, 3, 4
b. 1, 3, 5
c. 3, 4, 5
d. 1, 2, 4

Answer: a

Comment:
"The stem of the question fails to clearly tie the conditions of the stem to the question asked. If the stem of the question were to have stated, "Because the radiation elements are inoperable, which of the following are additional requirements that would have to be performed prior to starting the release," then the only correct answer would be choice 'a.' As the question is currently stated, according to DB-OP-03012, Radioactive Liquid Batch Release, steps 2.1.4 and 4.1.8, choice 'a' and choice 'd' are both correct answers since condition 1, recirculating two tank volumes, is performed for all releases regardless of the status of RE 1878 A and B."

NRC Resolution:
Recommendation was accepted. The licensee's comment was correct that the generic condition 1 for recirculating two tank volumes are required prior to performing any release regardless of the status of radiation monitors RE 1878 A and B. While the condition of the question stem clearly stated that both radiation monitors were declared inoperable, the wording of the question did not specify that the three actions chosen had be specific to the inoperable monitors. The licensee's recommendation was accepted and the correct answer was changed to accept either 'a' or 'd'.

Enclosure 3

SIMULATION FACILITY REPORT

Facility Licensee: Davis-Besse Nuclear Power Station

Facility Licensee Docket No: 50-346

Operating Tests Administered: October 3-4, 2000

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 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. None

Enclosure 4

WRITTEN EXAMINATION AND ANSWER KEY (SRO)