December 2, 2002

Mr. M. Bezilla
Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, Pennsylvania 15077

SUBJECT: BEAVER VALLEY UNIT 1 REACTOR OPERATOR AND SENIOR REACTOR OPERATOR INITIAL EXAMINATION REPORT NO. 50-334/02-301

Dear Mr. Bezilla:

This report transmits the results of the reactor operator (RO) and senior reactor operator (SRO) licensing examinations conducted by the NRC during the period of September 30 to October 8, 2002. This examination addressed areas important to public health and safety and was developed and administered using the guidelines of the "Examination Standards for Power Reactors" (NUREG-1021, Revision 8, Supplement 1).

Based on the results of the examination, all fourteen applicants passed all portions of the examination. The applicants included 3 ROs and 11 instant SROs. Examination results indicated that generally the applicants were well prepared for the examination. On October 31, 2002, final examination results, including individual license numbers, were given during a telephone call between Mr. T. Fish and Mr. C. Hynes and others of your staff. The NRC also noted your staff improved their implementation of the exam process as compared to the effort associated with the previous Unit 1 exam. No findings of significance were identified.

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). These records include the final examination and are available in ADAMS (SRO/RO Written-Accession Number ML023020109; SRO/RO Operating Section A-Accession Number ML023020126; SRO/RO Operating Section B-Accession Number ML023020135; and SRO/RO Operating Section C-Accession Number ML023020177). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/ADAMS.html (the Public Electronic Reading Room).

Should you have any questions regarding this examination, please contact me at (610) 337-5183, or by E-mail at RJC@NRC.GOV.

Sincerely,

/RA/

Richard J. Conte, Chief
Operational Safety Branch
Division of Reactor Safety

Docket No. 50-334
License No. DPR-66

Enclosure: Initial Examination Report No. 50-334/02-301

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-334
License No:	DPR- 66
Report No:	02-301
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Beaver Valley Power Station, Unit 1
Dates:	September 30 - October 8, 2002 (Operating Test Administration) October 11, 2002 (Written Examination Administration) October 21 - 25, 2002 (Examination Grading)
Examiners:	T. Fish, Senior Operations Engineer (Chief Examiner) J. D'Antonio, Operations Engineer R. Walton, Operations Engineer
Approved by:	Richard J. Conte, Chief Operational Safety Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000334/02-301; September 30 - October 8, 2002; Beaver Valley Power Station, Unit 1; Initial Operator Licensing Examination Report. Fourteen of fourteen applicants (3 ROs and 11 SRO instants) passed the examination.

The written examinations were administered by the facility and the operating tests were administered by three NRC region-based examiners.

A. Inspector Identified Findings

No findings of significance were identified.

B. Licensee Identified Findings

No findings of significance were identified.

Report Details

1. REACTOR SAFETY

Mitigating Systems - Reactor Operator (RO) and Senior Reactor Operator (SRO) Initial License Examinations

a. Scope of Review

The Beaver Valley examination team developed the written and operating initial examinations and together with NRC personnel, verified or ensured, as applicable, the following:

- The examination was prepared and developed in accordance with the guidelines of Revision 8, Supplement 1 of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors" and it met the overall quality goals (range of acceptability) of these standards. The review was conducted both in the Region I office and at the Beaver Valley power plant and training facility. Final resolution of comments and incorporation of test revisions was conducted during and following the onsite preparation week.

- Simulation facility operation was proper.

- Facility licensee completed a test item analysis on the written examination for feedback into the systems approach to training program.

- Examination security requirements were met.

The NRC examiners administered the operating portion of the examination to all applicants from September 30 - October 8, 2002. Beaver Valley training staff administered the written examination on October 11, 2002.

b. Findings

Grading and Results

All applicants (3 ROs and 11 instant SROs) passed all portions of the initial licensing examination.

The licensee submitted six post-examination comments for the written exams. The comments affected the answers to four common questions and two SRO-only questions and is included as Attachment 2 to this report. NRC resolution of the licensee's comments is included as Attachment 3. NRC staff accepted four comments, (three common questions and one SRO-only question) and denied two comments (one common question; one SRO-only question.)

Overall, the number of changes was 3% for the RO exam and 4% for the SRO exam. These percentages did not exceed the guidelines of ES-501, Section C.2.c, which limits post-examination changes on each written exam to less than 5% without licensee response. However, the licensee issued Condition Report 02-09326, which tasked Unit

1 training staff with determining why post-written exam comments were necessary and to develop appropriate corrective actions to prevent recurrence.

Examination Preparation and Quality

The quality of the draft examination was within acceptable range. The NRC also noted that the licensee improved their implementation of the exam process as compared to their effort associated with the previous Unit 1 exam.

Examination Administration and Performance

NRC examiners did not note generic performance errors by the applicants during examination administration.

4. OTHER ACTIVITIES

4OA6 Meetings, including Exit

On October 31, 2002, the NRC provided conclusions and examination results to the Superintendent of Operations Training, Mr. C. Hynes, via telephone. License numbers for the applicants were also provided during this call.

The NRC also expressed appreciation for the cooperation and assistance the licensee's training staff provided during the preparation and administration of the examination.

ATTACHMENT 1

KEY POINTS OF CONTACT

C. Hynes	Superintendent, Operations Training
T. Gaydosik	Supervisor, Licensed Operator Training
T. Wooley	Lead Instructor, Licensed Operator Training
G. Pelka	Instructor, Licensed Operator Training
E. Ernfeld	Simulator Instructor

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

None.

ATTACHMENT 2

LICENSEE COMMENTS

Common Question #2

Recommendation: Change the Answer Key to accept ‘B’ or ‘C’ as a correct answer.

Basis: The question asks for the response of the CVCS system following a dropped control rod event from 100% power. According to the Answer Key, the correct answer is ‘C’ [Charging flow is increased]. However, distractor ‘B’ [Letdown flow is decreased] is also correct. Both of these responses were validated post-exam on the BVPS Unit 1 Simulator.

The original answer - “Charging flow is increased” is correct due to the effects on RCS temperature and Pressurizer pressure when a single control rod is dropped into the core with the unit remaining on-line. As RCS Tavg lowers, Pressurizer pressure also decreases. As RCS pressure lowers, charging flow increases due to the characteristics of centrifugal pump flow. Charging flow continues to increase until the pump’s flow control valve has a chance to respond to the change in the Pressurizer level control error signal.

In addition to the response of charging flow to a drop in RCS pressure, the effects of a dropped control rod also impact letdown flow. The drop in RCS pressure causes the differential pressure across the CVCS letdown orifice to also decrease, which in turn causes a reduction in letdown flow.

Given the effect that a dropped control rod has on RCS pressure and letdown differential pressure which affect both charging and letdown flow, then both ‘B’ and ‘C’ are considered to be acceptable answers to this question.

Common Question #3

Recommendation: Change the Answer Key to accept ‘C’ as the correct answer.

Basis: The question asks for the required actions related to a low Reactor Coolant Pump (RCP) bearing oil reservoir condition. The revision of the alarm response procedure (REACT COOL PP BRG OIL RESERVOIR LEVEL LOW) used to develop this question directed actions to trip the reactor based upon the plant mode of operation. In Mode 3, per the stem’s initial conditions, a reactor trip was not warranted. Since the time of the question development and validation, the alarm response procedure has been revised and now refers the operator to take actions in accordance with abnormal operating procedure 1OM-53C.4.1.6.8, Abnormal RCP Operation.

The AOP addresses actions to be taken for various RCP problems and directs the operator to trip the reactor in any situation, regardless of plant mode, where conditions require the RCP to be stopped.

For conditions related to low bearing oil reservoir levels, the AOP directs the operator to consult with plant management to determine if a plant shutdown is necessary. This option was not specified as a choice in any of the distractors.

Distractor 'D' included the option to secure the RCP, but was incorrect due to an implausible pump bearing temperature limit. The other three distractors all included the option to trip the RCP; however, distractor 'A' did not include tripping the reactor, which is no longer correct per the new AOP guidance. Only distractor 'C' contained the combination of actions (trip the reactor, enter EOP E-0, and trip the RCP) that correctly answers the question.

The AOP guidance for securing a RCP directs the operator to trip the reactor, perform the immediate actions of E-0, Reactor Trip or Safety Injection and then trip the RCP. This is the appropriate action to take with the plant in Modes 1 - 3 and is correct based upon the training received by the license candidates.

<u>Common Question #17</u>

Recommendation: Change the Answer Key to accept two correct answers, 'A' or 'C'.

Basis: This question asked the candidates to evaluate the response of the core exit thermocouple (CET) temperatures following a reactor trip and trip of all RCPs (natural circulation conditions) given that all systems operate as designed. A validation of the plant response to the question conditions was analyzed using the BVPS Unit 1 Simulator.

Two assumptions are necessary to correctly answer the question.

The first is the amount of time that passes after the RCPs are tripped. If a short time frame is assumed, i.e., less than 30 minutes, then a second assumption regarding the mode of operation of the condenser steam dump valves is also necessary. If the steam dump valves operate in the "Tavg" mode, then plant response is adversely affected due to their undesired operation resulting from a lag in RCS temperature response during natural circulation conditions. However, if the steam dump valves operate in the "Steam Pressure" mode, then plant response differs as the steam dump valves are able to more effectively control RCS temperature.

At BVPS, the operators are required to take pre-emptive action to place the steam dump valves in the "Steam Pressure" mode in the event of a loss of forced flow (all RCPs are stopped) in order to more effectively control RCS temperature. This is documented in the BVPS - EOP Executive Volume User's Guide, which lists the actions that are allowed to be performed early in order to stabilize plant parameters.

In the "Steam Pressure" mode the expected plant response is that CET temperatures will initially rise and then stabilize at a value corresponding to the steam dump valve controlling pressure. This would lead to selecting 'C' as the correct answer.

In the "Tavg" mode, the expected response is that CET temperatures will initially rise and then drop as stated in answer 'A'.

However, if a long time frame, i.e., greater than 1 hour is assumed, regardless of the status of the condenser steam dump mode of operation, then as natural circulation is established temperature will drop as the core decay heat load decreases. This also would lead to selecting 'A' as the correct answer.

Due to the absence of information related to time frame and the condenser steam dump valve mode of operation, this question can be correctly answered by either 'A' or 'C'.

<u>Common Question #35</u>

Recommendation: Change the Answer Key to accept two correct answers, 'A' or 'B'.

Basis: The question evaluated the ability to calculate the amount of boric acid needed to reduce reactor power from 100% to 50% without inserting control rods and required a determination of the MINIMUM amount of boric acid in gallons. Depending on the accuracy and interpolation applied in reading the plant curves, the calculated answer will vary. When the value determined by performing the calculation is transposed to a corresponding value on the boron addition nomograph (plant curve CB-31), it is near the distinct increment of 900 gallons. Based on readability and accuracy of the curves, a value slightly higher or lower than 900 gallons could be obtained.

The nomograph used for the exam is incremented between 1000 and 1500 ppm for RCS boron concentration and between 100 and 150 ppm for the boron addition change. This requires interpolating between two sets of incremental numbers.

Using the information provided in the question and references, the calculated boron concentration is 128 ppm. This number must then be interpolated between the increments of 100 and 150 ppm. Depending on the accuracy of this interpolated point and the point of initial boron concentration (1100 ppm), the amount of boric acid addition may be determined as slightly above or below 900 ppm.

If the amount of boric acid is determined to be less than 900 gallons, then distractor 'A' is the most correct. If greater than 900 gallons, then distractor 'B' (1100 gallons) is the most correct answer for the MINIMUM amount of boric acid needed.

The calculation did not result in a clearly correct answer between these two distractors in part because the question asked for the "MINIMUM" amount.

Due to the acceptable tolerances in locating interpolated values on the nomograph curve, both 'A' and 'B' are considered to be correct answers.

<u>SRO Question #57</u>

Recommendation: Change the Answer Key to accept two correct answers, ‘B’ or ‘C’.

Basis: This question asked the candidate to evaluate a set of conditions to determine which would cause a control rod to be declared inoperable per Technical Specifications LCO 3.1.3.1. This was not an open reference question relying upon memory knowledge of the Technical Specification and operation of the Rod Control System.

At BVPS, Technical Specification LCO 3.1.3.1 states that the control rods are considered operable when positioned within ± 12 steps of the rods in a group (indicated position vs. demand position). Further, the action statements in the LCO address conditions where control rod(s) are also considered inoperable if found to be immovable due to friction or mechanical interference, untrippable, or trippable but inoperable due to other causes.

The original answer ‘C’ is correct since it satisfies the action statement condition in which a control rod is trippable, but inoperable due to other causes (cannot be moved electrically).

Distractor ‘B’ states that the rod bottom light for a single rod was not lit following a reactor trip. The alarm response procedure (ROD BOTTOM ROD DROP) that applies to the condition of a control rod with a rod bottom light extinguished lists the setpoint as 20 steps off the bottom. If the rod bottom light is not lit after a reactor trip, then it must be assumed that the rod is at least 20 steps from the bottom. There is no other information in the question or distractors that indicate otherwise. As such, in accordance with the guidance contained in NUREG-1021, Appendix E, the indication must be considered valid. Using this assumption, that the control rod is greater than 20 steps off the bottom, then it can also be assumed to be greater than 12 steps from the rest of the control rods in its group. Although in the current mode the Technical Specification is not applicable, the rod would be declared inoperable in accordance with station practices and tracked as such in the LCO section of the Shift Turnover List.

Given that the original answer, ‘C’ and distractor ‘B’ are valid conditions for declaring a control rod inoperable, both are considered to be correct answers.

<u>SRO Question #72</u>

Recommendation: Change the Answer Key to accept two correct answers, ‘A’ or ‘C’.

Basis: This question tested the candidate’s ability to evaluate plant conditions related to implementing the Loss of Secondary Heat Sink procedure. In particular this question tested the knowledge requirement of the method of re-establishing auxiliary feedwater flow to the steam generators (SG’s) after performing a feed and bleed of the RCS.

Given the information contained in the question stem and applying the guidance of Step 28 of EOP FR-H.1, “Response to Loss of Secondary Heat Sink”, the actions specified in answer ‘A’ and distractor ‘C’ would both be performed.

This is demonstrated by following the step through its performance. The original answer ‘C’ is arrived at by beginning at substep ‘a’ of Step 28 and working through to substep ‘c’ using the information provided in the question. By satisfying the criteria for RCS temperatures greater than 520F (588F and rising) and all SG wide range levels less than 13% (all offscale low) the next action is contained in substep ‘c’ to establish flow to one SG not to exceed 100 gpm. This is the endpoint considered as the correct answer when the question was developed and validated.

However, since RCS feed and bleed was previously established per the question conditions, then the fact that RCS temperature is given as 588F and rising slowly implies that the feed and bleed was not effective in lowering RCS temperature.

Because RCS temperature is not decreasing as would be expected through the addition of 100 gpm of auxiliary feedwater flow to one SG, then this also is considered as ineffective. Substep ‘d’ of Step 28 checks that core exit temperatures are stable or dropping. If not, then the next action is to feed one SG at the maximum feed flow available according to the Response Not Obtained column of substep ‘d’.

In following the procedure flowpath to a logical end in accordance with the information provided in the question, then both the original answer ‘C’ and distractor ‘A’ can be considered as correct answers.

ATTACHMENT 3

NRC RESOLUTION OF LICENSEE COMMENTS

Common Question #2

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment not accepted. The question specified *automatic* system response. Although a dropped control rod will indeed affect letdown flow, the change in flow is not due to *automatic* system response because the letdown system has no automatic flow controller. Rather, the change in flow is merely passive system response to a lower driving force across a fixed-sized restricting orifice in the letdown system. In contrast, the charging system does respond automatically because that system has an automatic flow controller. A dropped rod will cause a drop in Tavg, which causes PZR level to drop, which causes the charging system flow controller to *automatically* increase charging flow to restore PZR level. Consequently, only "C" is correct.

Common Question #3

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The proposed, original answer, "A", was based on conditions that required operators to trip the coolant pump and but not the reactor. However, facility staff revised the alarm response procedure (after the original question was developed and approved) such that the correct response now includes tripping the reactor. Since the applicants were taught this revision, the correct answer is changed to "C".

Common Question #17

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The training staff provided validated simulator information which showed the plant's response to this situation depends on whether a short time (less than 30 minutes) or a longer time (greater than one hour) is assumed following a reactor trip. The question stem does not restrict or define the time frame for which this situation applies. Therefore, depending on what post-trip time interval the applicants assumed, either "A" or "C" is correct.

Common Question #35

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The distractors did not adequately account for inevitable interpolations that result from reading these curves. Consequently, either "A" or "B" is correct.

SRO Question #57

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment accepted. The condition described in distractor “B” may be reasonably interpreted as an indication that the rod is stuck, and therefore INOPERABLE per Technical Specifications. Therefore, the correct answer is either “B” or “C”.

<u>SRO Question #72</u>

Refer to Attachment 2 for details of the question and basis for facility recommendation.

Comment not accepted. The question asked the applicants to evaluate the preferred method of *initiating* auxiliary feedwater (AFW) flow, and not what to do if RCS temperatures continued to rise once AFW had been initiated. Thus, for the conditions given in the question stem, the correct answer is to *initiate* feed to one S/G at 100 gpm. In order to choose distractor “A”, the applicant would need to reevaluate whether this initial action was effective. Such a reevaluation is beyond the bounds of the question and in any event talks about *increasing* auxiliary feedwater flow, not *initiating* auxiliary feedwater flow. Therefore, the only correct answer is “C”.