March 12, 2009

Mr. Mark Bezilla
Site Vice President
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P. O. Box 97, 10 Center Road, A-PY-A290
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT 1
NRC INITIAL LICENSE EXAMINATION REPORT 05000440/2009301(DRS);

Dear Mr. Bezilla:

On February 27, 2009, the Nuclear Regulatory Commission (NRC) examiners completed initial operator licensing examinations at your Perry Nuclear Power Plant. The enclosed report documents the results of the examination which were discussed on January 16, 2009, with Mr. A. Mueller Jr. and other members of your staff. An exit meeting was conducted by telephone on March 4, 2009, between Mr. A. Mueller Jr. of your staff and Mr. Walton, of Operator Licensing, to review the resolution of the station's post examination comments and the proposed final grading of the written examination for the license applicants.

The NRC examiners administered an initial license examination operating test during the week of January 12, 2009. The written examination was administered by Perry Nuclear Power Plant training department personnel on January 21, 2009. Eight Senior Reactor Operators and one Reactor Operator applicant were administered license examinations. The results of the examinations were finalized on February 27, 2009. All applicants passed all sections of their respective examinations and seven were issued senior operator licenses and one was issued an operator license. One senior operator license was withheld until the individual met experience requirements.

In accordance with 10 CFR 2.390 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), accessible from the NRC Web site at **http://www.nrc.gov/reading-rm/adams.html** (the Public Electronic Reading Room).

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

Sincerely,

***/RA/***

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50-440
License Nos. NPF-58

Enclosures:
1. Operator Licensing Examination Report 05000440/2009301(DRS) w/Attachment: Supplemental Information
2. Simulation Facility Report
3. Post Examination Comments w/NRC Resolution
4. Written Examinations and Answer Keys (SRO)

cc w/encls 1 & 2:
J. Hagan, President and Chief Nuclear Officer - FENOC
J. Lash, Senior Vice President of Operations and Chief Operating Officer - FENOC
D. Pace, Senior Vice President, Fleet Engineering - FENOC
K. Fili, Vice President, Fleet Oversight - FENOC
P. Harden, Vice President, Nuclear Support
Director, Fleet Regulatory Affairs - FENOC
Manager, Fleet Licensing - FENOC
Manager, Site Regulatory Compliance - FENOC
D. Jenkins, Attorney, FirstEnergy Corp.
Public Utilities Commission of Ohio
C. O'Claire, State Liaison Officer, Ohio Emergency Management Agency
R. Owen, Ohio Department of Health

cc w/encls 1, 2, 3, & 4: A. Mueller, Jr. Training Director, Perry Power Plant

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Public Utilities Commission of Ohio
C. O'Claire, State Liaison Officer, Ohio Emergency Management Agency
R. Owen, Ohio Department of Health

cc w/encls 1, 2, 3, & 4: A. Mueller, Jr. Training Director, Perry Power Plant

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

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|---|---|
| Docket Nos: | 50-440 |
| License Nos: | NPF-58 |
| Report No: | 05000440/2009301(DRS) |
| Licensee: | First Energy Corporation |
| Facility: | Perry Nuclear Power Plant, Unit 1 |
| Location: | Perry, Ohio |
| Dates: | January 12 – January 21, 2009 |
| Examiners: | R. Walton, Senior Operations Engineer<br>D. Reeser, Operations Engineer<br>C. Zoia, Operations Engineer |
| Approved by: | Hironori Peterson, Chief<br>Operations Branch<br>Division of Reactor Safety |

Enclosure 1

## SUMMARY OF FINDINGS

ER 05000440/2009301(DRS); 1/12/2009 - 1/21/2009; First Energy Corp., Perry Station Initial License Examination Report.

The announced initial operator licensing examination was conducted by regional Nuclear Regulatory Commission examiners in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9.

Examination Summary:

- Nine of nine applicants passed all sections of their respective examinations. Seven applicants were issued senior operator licenses and one applicant was issued an operator license. One senior operator will be issued a license after experience conditions have been met (Section 4OA5.1).

- The examiners identified that the licensee used software that incorporated two-phase fluid flow for modeling feedwater in the Perry simulator. This software has been used in some but not all BWR simulators. This condition is an unresolved item pending further review by the NRC (Enclosure 2).

## REPORT DETAILS

**4. OTHER ACTIVITIES (OA)**

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

The Nuclear Regulatory Commission's examiners prepared the examination outline and developed the written examination and operating test. The NRC examiners validated the proposed examination during the week of December 15, 2008, at the Perry Nuclear Power Station Training Building with the assistance of members of the licensee training staff. During the on-site validation week on December 15, 2008, the examiners audited two license applications for accuracy. The NRC examiners conducted the operating portion of the initial license examination during the week of January 12, 2009. The NRC examiners and members of the Perry Nuclear Power Station training department staff administered the written examination on January 21, 2009. The NRC examiners used the guidance established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, to prepare, validate, revise, administer, and grade the examination.

b. Findings

Written Examination

During the validation of the written examination several questions were modified or replaced. Changes made to the written examination were documented on Form ES-401-9, "Written Examination Review Worksheet" which is available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). The licensee submitted four written examination post-examination comments for consideration by the NRC examiners when grading the written examination. The post-examination comments and the NRC resolution for the post-examination comments are contained in Enclosure 3, "Post Examination Comments and Resolutions." The NRC examiners graded the written examination on February 19, 2009, and conducted a review of each missed question to determine the accuracy and validity of the examination questions.

Operating Test

During the validation of the operating test, two Job Performance Measures (JPMs) were modified and changes were made to the dynamic simulator scenarios. The JPMs were replaced since the JPM's were determined to be too simplistic in nature (inadequate difficulty level). Changes made to the operating test were documented in a document titled, "Operating Test Comments," which is available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). The NRC examiners completed operating test grading on February 19, 2009.

Examination Results

Eight applicants at the Senior Reactor Operator (SRO) level and one applicant at the Reactor Operator (RO) level were administered written and operating tests. Two of the SRO applicants were previously licensed as RO's at Perry Power Station. Nine applicants passed all portions of their examinations and eight applicants were issued operating licenses. One applicant's license was withheld until experience requirements had been met.

.2 Examination Security

a. Scope

The NRC examiners reviewed and observed the licensee's implementation of examination security requirements during the examination validation and administration to assure compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The examiners used the guidelines provided in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors" to determine acceptability of the licensee's examination security activities.

b. Findings

No Findings

4OA6 Meetings

Debrief

The chief examiner presented the examination team's preliminary observations and findings on January 16, 2009, to A. Mueller, Jr., and other members of the Perry Operations and Training Department staff.

Exit Meeting

The chief examiner conducted an exit meeting on March 4, 2009, with Mr. A. Mueller, Jr., Perry Station Training Director by telephone. The NRC's final disposition of the station's post-examination comments were disclosed and revised preliminary written examination results were provided to A. Mueller, Jr., during the telephone discussion. The examiners asked the licensee whether any of the material used to develop or administer the examination should be considered proprietary. No proprietary or sensitive information was identified during the examination or debrief/exit meetings.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

### KEY POINTS OF CONTACT

Licensee
K. Krueger, Plant General Manager
D. Evans, Manager Operations
A. Cayia, Director Performance Improvement
R. Coad, Manager – Regulatory Compliance
A. Mueller, Jr., Manager Training
J. Pelcic, Nuclear Compliance
D. Zielinsky, Training Department
R. Torres, Training Department
J. Kelley, Training Department
D. Richmond, Training Department

NRC
R. Walton, Chief Examiner

### ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

UNR (00050/440-2009-301-01), Two-Phase Fluid Flow Modeling for Feedwater

Closed

None

Discussed

None

### LIST OF DOCUMENTS REVIEWED

None

### LIST OF ACRONYMS USED

| | |
|---|---|
| ADS | Automatic Depressurization System |
| ADAMS | Agency-Wide Document Access and Management System |
| BWR | Boiling Water Reactor |
| DRS | Division of Reactor Safety |
| NRC | Nuclear Regulatory Commission |
| IR | Inspection Report |
| SPDS | Safety Parameters Display System |

# SIMULATION FACILITY REPORT

Facility Licensee: Perry Nuclear Power Station

Facility Docket No: 50-440

Operating Tests Administered: 1/12/2009 – 1/16/2009

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:

Incorporation of Two-Phase Fluid Flow Modeling for Feedwater

During onsite validation week of the initial license exam at the Perry station, the inspectors noted that reactor vessel water level appeared to increase with no high pressure injection or operator intervention after emergency depressurization. With reactor pressure lowering, the reactor vessel water level swelled about 100 inches. The vessel water level then lowered due to the ADS valves being opened.

The licensee informed the examiners that previously they had loaded computer software that changed the high pressure feedwater injection from a single phase fluid flow model to a two-phase fluid flow model. This resulted in a “flashing” of high temperature feedwater in feedwater heaters #6. This condition produced flow into the reactor vessel after the vessel pressure lowered to the high pressure feedwater heater #6 saturation pressure.

This simulator computer model was taught to the initial license class and requalification classes. The initial license exam was administered during the week of January 12, 2009, with this simulator software included.

The examiners determined that this simulator modeling had been included at two other facilities in the industry. The examiners were uncertain of the pedigree and approval status of this computer software modeling since it had not yet been approved by the BWR owners group, and had not been widely accepted by other BWR utilities. This issue was considered an Unresolved Item (50-440/2009301-01) pending further review by NRC Headquarters Operations staff.

Change in Simulator Modeling between On-Site Validation and Exam Administration

During the week of December 15, 2008, the examiners validated the Perry operating exam with an operating crew. The operating crew used the Safety Parameters Display System (SPDS) display screens in the overhead of the simulator to track and trend various parameters important to equipment operation and plant monitoring.  The SPDS computer received inputs from the simulator computer.  On January 9, 2009, the licensee implemented a change to the SPDS process computer that was believed to be a graphics change – a change that would not alter computer modeling.

The following week, on Monday, January 12, 2009, during the administration of the initial license operating test, the examiners, examinees and licensee simulator operators noted that the SPDS computer display panel did not accurately display reactor vessel wide range water level.

The following day, after running the first scenario, and seeing that the SPDS computer had rejected wide range reactor vessel water level input from the simulator computer, the licensee's staff determined that a change to the SPDS computer software had occurred since onsite validation.  Specifically, a change to the SPDS computer included a file that inhibited the SPDS computer from receiving wide range input from the simulator computer.  As a result, the SPDS displays for wide range level indication were erroneous.  The scenarios were continued with this software change included until the file was removed on Tuesday night, January 13, 2009.

NUREG 1021, ES-301-4, item 8 required that computer modeling not be changed between onsite validation and exam administration.  Since the SPDS computer software was changed that affected important monitored parameters, the examiners believed there was a potential for invalidating the Perry Initial operating exam for January 12 and 13, 2009.

The Operator Licensing Branch in Headquarters reviewed this event and determined that the exam was not invalidated.  The erroneous indications on the SPDS panel were clearly identified by their color, the applicants had access to accurate wide range level indications on the main control boards and that all other functions worked normally.  There was no reason to treat this any different than any other instrumentation malfunction or to invalidate the affected scenarios.  This event described illustrated the risk of making even simple changes that were not expected to alter the simulator's response.

**POST EXAMINATION COMMENTS AND RESOLUTIONS**

RO Question Number 17

A plant startup is in progress per IOI-0001 Cold Startup. The following plant conditions exist:

Reactor Pressure 200 psig
Main Condenser Vacuum 5.0” HgA
Mechanical Vacuum Pumps are being cycled to maintain vacuum
Main Turbine Warming is in progress
Motor Feed Pump is providing Reactor Level Control
TBCC Pumps A and B operating

The following alarm is received on 1H13-P870, TBCC PUMP SUCTION FLOW LOW. The operator checks TBCC Parameters at 1H13-P870 with the following indications:

TBCC A Pump red and green light off, no discharge pressure indicated.
TBCC B Pump red light on, green light off, no discharge pressure indicated.
TBCC C Pump red light off, green light on, no discharge pressure indicated.

Per ONI-P44 Loss of Turbine Building Closed Cooling, an ____.

A. immediate scram may not be necessary because the Main Turbine is not in operation

B. immediate scram may not be necessary because reactor pressure control is on the Bypass Valves

C. immediate scram is required because the Motor Feed Pump is providing level control

D. immediate scram is required because the Mechanical Vacuum Pumps can not be cycled

Answer: A

Reference: ONI-P44, “Loss of Turbine Building Closed Cooling,” Revision 7, Page 5

## POST EXAMINATION COMMENTS AND RESOLUTIONS

Applicant Comment:

An applicant asserted that the answer key should be changed so that distractor "C" was the only correct answer.

The applicant provided the following justification:

1. The question asks what actions are required for a loss of Turbine Building Closed Cooling (TBCC) per ONI-P44, Loss of Turbine Building Closed Cooling. The stem of the question contains the status of the TBCC pumps with indication of NO discharge pressure for any of the 3 TBCC pumps. Lack of discharge pressure is symptomatic of a system leak and a total loss of TBCC.

2. Plant TBCC pump discharge pressure indicates 16-17 psig when in standby due to the height of water from the expansion tank. The bottom of surge tank is at elevation 660'10" and pump suction is at elevation 625'9.250."

3. The procedure directs that for a total loss of TBCC, the reactor be scrammed. (ONI-P44 immediate action 3.4).

4. Core flow is < 58 mlbm during plant start-up at 200 psig. (No core flow reduction required.)

The applicant provided the following justifications for the distractors:

A. Incorrect answer – Based on a total loss of TBCC an immediate scram is required no standby TBCC pump is available - no discharge pressure indicated which signifies a leak in the system.

B. Incorrect answer - Based on a total loss of TBCC an immediate scram is required no standby TBCC pump is available - no discharge pressure indicated which signifies a leak in the system. Bypass valve HPU's require shutdown at 150 degrees in sump.

C. Correct answer – Based on a total loss of TBCC an immediate scram is required and the Feed and Condensate system will be shut down when temperature limits are reached.

D. Incorrect answer – Based on a total loss of TBCC an immediate scram is required however the Mechanical Vacuum Pumps CAN be cycled. The shutdown limit at 102 degrees F can be exceeded (reference ONI-P44 Attachment 1 limits).

Reference: ONI-P44**, "**Loss of Turbine Building Closed Cooling," pages 5, 8, 10, 11.

## POST EXAMINATION COMMENTS AND RESOLUTIONS

Facility Proposed Resolution:

The facility agreed with the applicant and stated that the answer key should be changed so that distractor "C" was the only correct answer. The facility also stated that the question asked what actions were required for a loss of Turbine Building Closed Cooling (TBCC) per ONI-P44. The stem of the question contained the status of the TBCC pumps with indication of no discharge pressure for any of the 3 TBCC pumps. The procedure directed that for a total loss of TBCC the reactor be scrammed. The facility referred to the applicant's comments for details.

Reference: ONI-P44**, "**Loss of Turbine Building Closed Cooling," pages 5, 8, 10, 11.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was resolved to delete the question from the examination.

The question asked what actions were required for a loss of Turbine Building Closed Cooling (TBCC) per ONI-P44, "Loss of Turbine Building Closed Cooling." The stem of the question contained the status of the TBCC pumps with indication of no discharge pressure for any of the 3 TBCC pumps. Lack of discharge pressure would be symptomatic of a system leak and a complete loss of TBCC.

Facility procedure ONI-P44, "Loss of Turbine Building Closed Cooling," Step 3.4, required that for a complete loss of TBCC, the reactor be scrammed. However, a NOTE preceding this step qualified this step by stating:

> "The Reactor is shutdown in anticipation of loss of cooling to various loads, e.g., Generator Stator. An immediate shutdown may NOT be necessary if the main turbine is NOT in operation."

Step 4.3.8 of ONI-P44 also required shutdown of TBCC components that reached their temperature limit. Attachment 1 of ONI-P44, "TBCC Served Component Limitations," provided the temperature limits for the Motor Feedwater Pump and the Mechanical Vacuum Pumps. In Attachment 1, the temperature limitation of 102°F for the Mechanical Vacuum Pumps had an asterisk that stated:

> "This limit for vacuum considerations only and may be exceeded."

However, a NOTE preceding Step 4.3.8 stated:

> "The Mechanical Vacuum Pumps should NOT be used due to the loss of cooling water to the seal water coolers."

Based on the above information, the following distractor evaluation was performed:

## POST EXAMINATION COMMENTS AND RESOLUTIONS

Distractor A was considered a correct answer based on the NOTE preceding Step 3.4 of ONI-P44, because it stated that an immediate scram may NOT be necessary if the main turbine is not in operation. For the conditions stated in the question stem, the main turbine was not in operation, and thus the NOTE was applicable. Thus, distractor A, "immediate scram may not be necessary because the Main Turbine is not in operation," was a correct answer.

Distractor B was an incorrect answer based on the following:

- ONI-P44, Step 3.4, required that for a complete loss of TBCC, the reactor be scrammed.
- With the temperature limitation of 150°F provided in Attachment 1 of ONI-P44, for the Steam Bypass Hydraulic Power Unit (HPU) Reservoir, the Bypass Valves could not be used for any significant time period before its temperature limit was reached and a scram was required.

Thus, distractor B, "immediate scram may not be necessary because reactor pressure control is on the Bypass Valves," remained an incorrect answer.

Distractor C is a correct answer based on the following:

- ONI-P44, Step 3.4 required that for a complete loss of TBCC the reactor be scrammed.
- The question stem stated that the Motor Feedwater Pump was providing Reactor Level Control. With the temperature limitations provided in Attachment 1 of ONI-P44, for the Motor Feedwater Pump, the Motor Feedwater Pump could not be run for any significant time period before its temperature limit was reached and a scram was required.

Thus, distractor C, "immediate scram is required because the Motor Feed Pump is providing level control," was a correct answer.

Distractor D is a correct answer based on the following:

- ONI-P44, Step 3.4, required that for a complete loss of TBCC, the reactor be scrammed.
- The NOTE preceding Step 4.3.8 stated that the Mechanical Vacuum Pumps should NOT be used due to the loss of cooling water to the seal water coolers."
- With the temperature limitations provided in Attachment 1 of ONI-P44, for the Mechanical Vacuum Pumps, the Mechanical Vacuum Pumps could not be run for any significant time period before its temperature limit was reached and a scram was required.

Thus, distractor D, "immediate scram is required because the Mechanical Vacuum Pumps can not be cycled," was a correct answer.

Finally, the Examiner's Standard, ES-403, part D.1.c stated:

> "If it is determined that there are two correct answers, both answers will be accepted as correct. If, however, both answers contain conflicting information, the question will likely be deleted. For example, if part of one answer states that operators are required to insert a manual reactor scram, and part of another answer states that a manual scram is not required, then it is unlikely that both answers will be accepted as correct, and the question will probably be

> deleted.
> If three or more answers could be considered correct or there is no correct answer, the question shall be deleted."

Since there were three correct answers (A, C, and D) identified for the question, and two combinations of these answers could not logically be true at the same time (A and C or D), it was resolved to delete the question from the examination.

## POST EXAMINATION COMMENTS AND RESOLUTIONS

SRO Question Number 12

The plant was operating at 100% reactor power when a grid disturbance caused a generator load rejection. This resulted in a reactor scram. All plant equipment responded as designed.

Per RPS Instrumentation Tech Spec Bases, the primary scram signal analyzed to provide protection from a generator load rejection event is __(1)__.

As the Unit Supervisor you direct a reactor level band of __(2)__ per EOP-1 RPV Control.

| | 1 | 2 |
|---|---|---|
| A. | reactor vessel steam dome pressure high | 130” to 219” |
| B. | reactor vessel steam dome pressure high | 178” to 219” |
| C. | turbine control valve fast closure, trip oil pressure low | 130” to 219” |
| D. | turbine control valve fast closure, trip oil pressure low | 178” to 219” |

Answer: A

References: Technical Specification (TS) 3.3.1.1 Bases
Updated Safety Analysis Report (USAR), Revision 12
EOP-1 Guideline, Revision 0

## POST EXAMINATION COMMENTS AND RESOLUTIONS

Applicant Comment:

An applicant asserted that the answer key should be changed so that distractor C should also be accepted as correct, in addition to distractor D.

The applicant provided the following basis for this reasoning:

1. The questions asks the bases for the generator load rejection scram and the level band the Unit Supervisor would direct following a grid disturbance that results in a generator load rejection.

2. Turbine Control Valve - Fast Closure, Trip Oil Pressure Low is the bases for the generator load rejection.

3. A level band of 130-219 inches is correctly given per the EOP bases for step RLC-4 in EOP-1, which states the wide RPV water level control band permitted by this step is sufficient to assure adequate core cooling yet avoid unwarranted demands on an operator's attention.

   a. If unnecessarily constrained within narrower limits, an operator may be less effective in performing concurrent duties.

4. The narrower band of 178-219 inches is suggested per Guideline 2 in EOP-1. Per the EOP bases guidelines provide supplemental information to the operator.

5.
   a. A guideline flag in the flowpath refers to the guideline text.
   b. The Guideline text provides supplemental bases information that the operator can call upon if need to help in decision making and performance of the flowcharts.

6. Since these steps are at the same relative location in the level leg of EOP-1 either band would be correct to order with the given conditions within the question and might be amended as plant conditions and Control Room work load changes.

7. The recommendation is that two answers (C and D) are correct because the two level bands given would be correct if directed.

The applicant provided the following distractor analysis:

A. Incorrect answer – Reactor vessel steam dome pressure high is not the bases for the trip.

B. Incorrect answer – Reactor vessel steam dome pressure high is not the bases for the trip.

C. Correct answer – Turbine Control Valve - Fast Closure, Trip Oil Pressure Low and a Level Band of 130-219 for the wider level band is also correct.

## POST EXAMINATION COMMENTS AND RESOLUTIONS

D. Correct answer – Turbine Control Valve - Fast Closure, Trip Oil Pressure Low and a Level Band of 178-219 for the narrow level band is still correct.

References: Technical Specification 3.3.1.1 Bases, pages 3.3-11 and 3.3-18
EOP Bases, page 22
EOP-1, RPV Control Bases, pages 25 and 26

Facility Proposed Resolution:

The facility agreed with the applicant and commented that the answer key should be changed so that distractor C should also be accepted as correct, in addition to distractor D. The facility stated that the question asked the basis for the generator load rejection scram and the level band the Unit Supervisor would direct following a grid disturbance that results in a generator load rejection. Turbine Control Valve Fast Closure Trip Oil Pressure Low is the bases for the generator load rejection, a level band of 130-219 inches is correctly given per the EOP basis and the narrower band of 178-219 inches is suggested per Guideline 2. The facility referred to the applicant's comments for details.

References: Technical Specification 3.3.1.1 Bases, pages 3.3-11 and 3.3-18
EOP Bases, page 22
EOP-1, RPV Control Bases, pages 25 and 26

NRC Resolution:
Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to accept the facility's comment and accept both distractors C and D as correct answers.

The question asked for the Technical Specification (TS) Bases (from the Reactor Protection System (RPS) Instrumentation section of the TS) for the generator load rejection event and the level band the Unit Supervisor would direct following a grid disturbance that resulted in a generator load rejection and resultant reactor scram.

From page B.3.3-18 of the RPS Instrumentation TS Bases, the Turbine Control Valve Fast Closure, Trip Oil Pressure Low function is the primary scram signal for the generator load rejection event.

From EOP-01, "RPV Control," Revision A, step RLC-4 stated to "Restore and Maintain RPV level between 130 inches and 219 inches." The EOP Bases document for this step stated:

> "The wide RPV water level control band permitted by this step is sufficient to assure adequate core cooling yet avoid unwarranted demands on an operator's attention. If unnecessarily constrained within narrower limits, an operator may be less effective in performing concurrent duties."

In EOP-01, prior to step RLC-4, there was a "guideline flag," which referred to "Guideline 2." In EOP-01, a list of "General Guidelines" was provided in a text box at the bottom of the flowchart.

## POST EXAMINATION COMMENTS AND RESOLUTIONS

General Guideline 2 stated:

"If other EOP actions have a higher priority allow HPCS or RCIC to operate in automatic between Level 2 and Level 8. Closely monitor system operation. The level band should be expanded above Level 8 and below Level 2 to allow the system to operate within the level band. Maintain level above 100 inches to 260 inches. RPV level should be maintained 178 to 219 inches whenever possible and shall be greater than 178 inches whenever shutdown cooling is in service."

The EOP Bases document associated with the Guideline text stated:

> "Guidelines provide supplemental information to the operator. A guideline flag in the flowpath refers to the guideline text. The Guideline text provides supplemental bases information that the operator can call upon if needed to help in decision making and performance of the flowcharts."

Based on the above information, the following distractor evaluation was performed:

Distractors A and B were incorrect in that the TS Bases for the generator load rejection event was "turbine control valve fast closure, trip oil pressure low," and not "reactor vessel steam dome pressure high." In addition, from page B.3.3-11 of the RPS Instrumentation TS Bases, no specific safety analysis took credit for the Reactor Vessel Steam Dome Pressure High function.

Distractor C was a correct answer based on the following:

- The TS Bases for the generator load rejection event was "turbine control valve fast closure, trip oil pressure low," and
- Step RLC-4 of EOP-01, which stated to restore and maintain RPV level between 130 inches and 219 inches. In addition, the EOP Bases document for this step stated that the wide RPV water level control band permitted by this step was sufficient to assure adequate core cooling yet avoid unwarranted demands on an operator's attention, and if unnecessarily constrained within narrower limits, an operator may be less effective in performing concurrent duties.

Distractor D is a correct answer based on the following:

- The TS Bases for the generator load rejection event is "turbine control valve fast closure, trip oil pressure low," and
- Step RLC-4 of EOP-01, which stated to restore and maintain RPV level between 130 inches and 219 inches. The level band of 178 to 219 inches was encompassed by the level band of 130 to 219 inches stated in step RLC-4. In addition, the narrower band of 178 to 219 inches was suggested per Guideline 2 in EOP-01. Per the EOP Bases, the guidelines provided supplemental information to the operator that the operator could call upon if needed to help in decision making and performance of the flowcharts.

Therefore, the answer key was modified to accept both distractors C and D as correct answers.

## POST EXAMINATION COMMENTS AND RESOLUTIONS

SRO Question Number 16

A plant startup is in progress with reactor power at 29%.

Number 1 Turbine Bypass Valve fails open.
Full Core Display, RPC MODE light is between GP1-4 Full Out and LO Power Set PT marks.

The __(1)__ and the Unit Supervisor would suspend control rod __(2)__.

| | (1) | (2) |
|---|---|---|
| A. | Rod Withdrawal Limiter is Inoperable | Withdrawal |
| B. | Rod Withdrawal Limiter is Inoperable | movement except by scram |
| C. | Rod Pattern Controller is Inoperable | Withdrawal |
| D. | Rod Pattern Controller is Inoperable | movement except by scram |

Answer: B

References: Technical Specification 3.3.2.1
ARI-H13-P680-0005-C9, Revision 11

<u>Applicant Comment</u>:

An applicant asserted that the question should be deleted from the examination since none of the distractors were correct.

The applicant provided the following justification:

1. The question asks for the operator to make a declaration of Operability in accordance with Technical Specifications for either the Rod Pattern Controller or the Rod Withdrawal limiter and then determine the required actions for control rod movement following a failed bypass valve with reactor power at 29%.

2. PNPP Technical Specifications do not require the Rod Withdrawal Limiter (RWL) to be Operable until greater than 33.3% RTP and the Rod Pattern Controller (RPC) is only required to be operable when less than or equal to 19% RTP.

3. The question asks what to do if at 29% RTP. In accordance with TS 3.3.2.1 APPLICABILITY: According to Table 3.3.2.1-1, there is none since current power is between

19% and 33.3%. Since there is no applicability there is no operability requirement and no required action until RTP is either greater than 33.3% or less than 19% RTP. This condition would be addressed by a 'Potential LCO' and administratively tracked.

4. Therefore, 'Answer A' (i.e. Rod Withdrawal Limiter is Inoperable and the Unit Supervisor would suspend control rod withdrawal) is incorrect.

The applicant provided the following distractor analysis:

A. Incorrect answer – RWL is NOT required to be operable therefore no required action.

B. Incorrect answer – RWL is NOT required to be operable therefore no required action.

C. Incorrect answer – RPC is NOT required to be operable therefore no required action.

D. Incorrect answer – RPC is NOT required to be operable therefore no required action.

References: Technical Specification 3.3.2.1, Control Rod Block Instrumentation and Table 3.3.2.1-1

Facility Proposed Resolution:

The facility agreed with the applicant and commented that the question should be deleted from the examination since none of the distractors were correct. The facility stated that the question asked for the operator to make a declaration of Operability for either the Rod Pattern Controller or the Rod Withdrawal Limiter and determine the required actions for control rod movement following a failed bypass valve with reactor power at 29%. The facility commented that the Perry Nuclear Power Plant Technical Specifications do not require the Rod Withdrawal Limiter to be Operable until greater than 33.3% reactor thermal power (RTP) and required the Rod Pattern Controller only operable at less than or equal to 19% RTP. The facility referred to the applicant's comments for details.

References: Technical Specifications 3.3.2.1, page 3.3-15 and Table 3.3.2.1-1

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to it was decided to delete the question from the examination.

The question asked for the Unit Supervisor to make a declaration of operability for either the Rod Pattern Controller or the Rod Withdrawal Limiter, and to determine the required actions for control rod movement following a failed open #1 turbine bypass valve with reactor power at 29% and a plant startup in progress.

Technical Specification (TS) 3.3.2.1, "Control Rod Block Instrumentation," defined the TS operability requirements for both the Rod Withdrawal Limiter (RWL) and the Rod Pattern Controller (RPC). The Applicability of the Control Rod Block Instrumentation is according to

Table 3.3.2.1-1 of the TS. In Table 3.3.2.1-1 of the TS, the "Applicable Modes or Other Specified Conditions," was given as:

- For the RWL, either Reactor Thermal Power (RTP) greater than 66.7% or between greater than 33.3% and less than or equal to 66.7%.
- For the RPC, RTP less than or equal to 19.0%, except during the reactor shutdown process if the coupling of each withdrawn control rod has been confirmed.

Based on the above information, the Technical Specifications did not require either the Rod Withdrawal Limiter to be operable until greater than 33.3% RTP, and the Rod Pattern Controller was only required to be operable when less than or equal to 19% RTP. Since the question asked for actions required with the plant at 29% reactor power, the TS did not apply, and there was no operability requirement and no required action for this plant condition. Thus, there were no correct answers, and it was decided to delete the question from the examination.

**POST EXAMINATION COMMENTS AND RESOLUTIONS**

SRO Question Number 25

Alert JA1 was declared.
Emergency Coordinator duties remain with the Shift Manager.

When the Shift Manager is ready to terminate from event, the Shift Manager is responsible to terminate the event ____.

Reference Provided: EPI-A1 Attachments 1 & 2

A. after consulting with the NRC, State and local counties

B. after consulting with the State and local counties

C. after consulting with the NRC

D. without consultation

Answer: D

References: EPI-A1, page 11
EPI-A2, pages 13 and 17

Applicant Comment:

An applicant asserted that the question should be deleted from the examination since none of the distractors were correct.

The applicant provided the following justification:

> The question asks for the required consultations when terminating from an Alert but does not specify a specific instruction. There are two references listed which include conflicting and in one case, vague guidance on correct execution.
>
> a. In accordance with EPI-A-2, Emergency Action Levels, Section 5.3.1.6 and on Event Termination Actions Checklist, item # A.4 – For events classified as an Alert or above, the NRC, State of Ohio, and local counties have been consulted regarding event termination.
>
> b. In accordance with EPI-A-1, Emergency Actions Based on Event Classification, Section 5.5.1.11 - Consult with NRC, State of Ohio, and local county officials regarding the decision to terminate the emergency. The intent of this action is to involve the NRC, State and local counties in event decision making; however, this action is not intended to

delay or hinder the Perry Plant's ability to terminate from an Unusual Event or Alert that no longer meets the criteria for any event at the time of declaration.

1) The last line of the step may given the correct circumstances contradict the first line in the step.
2) This issue has been identified and a Condition Report is to be written to resolve the differences between the above two procedures

The applicant provided the following distractor analysis:

A. Correct answer – IAW EPI A2 page 14 and 17, PNPP is to consult with the NRC, State and local county officials regarding event termination, 'Answer A' lists all three per the EPI.

B. Correct answer – IAW EPI A2 page 14 and 17, PNPP is to consult with the NRC, State and local county officials regarding event termination, Answer B just lists the State and local county officials however it does not say 'only', even though the NRC is also to be consulted, Answer B is not wrong, 'Answer B' is a subset of 'Answer A'.

C. Correct answer – IAW EPI A2 page 14 and 17, PNPP is to consult with the NRC, State and local county officials regarding event termination, 'Answer C' just lists the NRC however it does not say 'only', even though the State and local county officials are also to be consulted, 'Answer C' is not wrong, 'Answer C' is a subset of 'Answer A'.

D. Correct answer – EPI A1 page 11 (this is a different reference than for Answers A, B, C), PNPP is to consult with the NRC, State and local county officials regarding event termination however during an Unusual Event or an Alert this action is not intended to delay or hinder PNPP's ability to terminate from an Unusual Event or an Alert. Therefore given the correct circumstances consultation is not required.

References: EPI A1 page 11
EPI A2 pages 14 and 17

Facility Proposed Resolution:

The facility agreed with the applicant and commented that the question should be deleted from the examination since none of the distractors were correct. The facility stated that the question asked for the required consultations when terminating from an Alert in accordance with EPI-A-0001, Emergency Actions Based on Event Classification, Section 5.5.1.11, 'Answers A, B, C are correct'. In accordance with EPI-A-0002, Emergency Action Levels, Section 5.3.6, 'Answer D is correct'. The facility referred to the applicant's comments for details.

References: EPI A1 page 11
EPI A2 pages 14 and 17

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to it was decided to delete the question from the examination.

The question asked for the required consultations when terminating from an "Alert" condition in accordance with the emergency plan procedures.

In accordance with EPI-A2, step 5.3.1.6, it stated:

> "For events classified as an Alert or above, the NRC, State of Ohio, and local counties have been consulted regarding event termination."

In accordance with EPI-A2, Attachment 2, Section A, "Event Termination Actions, step 6, it stated:

> "[ALERT OR ABOVE ONLY] NRC, State of Ohio, and local counties consulted regarding the decision to terminate the emergency."

In accordance with EPI-A1, step 5.5.1.11, it stated:

> "Consult with NRC, State of Ohio, and local county officials regarding the decision to terminate the emergency. The intent of this action is to involve the NRC, State and local counties in event decision making; however, this action is not intended to delay or hinder the Perry Plant's ability to terminate from an Unusual Event or Alert that no longer meets the criteria for any event at the time of declaration."

Based on the above information, the following distractor evaluation was performed:

Distractor A was a correct answer based on EPI-A2, step 5.3.1.6, EPI-A1, step 5.5.1.11, and EPI-A2, Attachment 2, Section A, "Event Termination Actions, step 6, which stated that for events classified as an Alert or above, the NRC, State of Ohio, and local counties have been consulted regarding event termination.

Distractor B was a correct answer based on EPI-A1, step 5.5.1.11; EPI-A2, step 5.3.1.6; and EPI-A2, Attachment 2, Section A, "Event Termination Actions, step 6, which stated that for events classified as an Alert or above, the NRC, State of Ohio, and local counties have been consulted regarding event termination. Since distractor B required consulting with the State and local counties, and did not specifically state that these were the ONLY organizations to be consulted, this distractor was also a correct answer.

Distractor C was a correct answer based on EPI-A1, step 5.5.1.11; EPI-A2, step 5.3.1.6; and EPI-A2, Attachment 2, Section A, "Event Termination Actions, step 6, which stated that for events classified as an Alert or above, the NRC, State of Ohio, and local counties have been consulted regarding event termination. Since distractor C required consulting with the NRC, and did not specifically state that this was the ONLY organization to be consulted, this distractor was also a correct answer.

## POST EXAMINATION COMMENTS AND RESOLUTIONS

Distractor D was an incorrect answer based on EPI-A1, step 5.5.1.11; EPI-A2, step 5.3.1.6; and EPI-A2, Attachment 2, Section A, "Event Termination Actions, step 6, which stated that for events classified as an Alert or above, the NRC, State of Ohio, and local counties have been consulted regarding event termination. Per EPI-A1, the intent is to involve the NRC, state, and local counties in the decision making process. EPI-A1, step 5.5.1.11, goes on to state that this action is not intended to delay or hinder the plant's ability to terminate from an Unusual Event or an Alert that no longer meets the criteria for any event at the time of declaration. As stated in Section 5.3 of EPI-1A, an event that no longer meets the criteria for any event at time of declaration, need not even be classified.

The applicant asserted that, given the correct circumstances, consultation with any organizations was not required. However, applicants are instructed prior to the test per NUREG 1021, Appendix E not to make assumptions regarding conditions that were not specified in the question unless they occurred as a consequence of other conditions that were stated in the question. Nothing in the stem of the question supported an assumption that the event no longer met the EAL criteria at the time of the declaration. Therefore, the applicant's assertion is invalid and distractor D remained an incorrect answer.

Since three of the four distractors were correct answers, it was determined to delete the question from the examination.

# POST EXAMINATION COMMENTS AND RESOLUTIONS

## Post Examination Comments and Resolutions

## WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO/SRO Initial Examination ADAMS Accession # ML090650492