UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD
ARLINGTON TX 76011-4511

September 23, 2016

Ken J. Peters, Senior Vice President
and Chief Nuclear Officer
Attention: Regulatory Affairs
Luminant Generation Company LLC
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2 – NRC EXAMINATION REPORT 05000445/2016301; 05000446/2016301

Dear Mr. Peters:

On July 18, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an initial operator license examination at the Comanche Peak Nuclear Power Plant, Units 1 and 2. The enclosed report documents the examination results and licensing decisions. The preliminary examination results were discussed on July 13, 2016, with Mr. T. McCool, Site Vice President, and other members of your staff. A telephonic exit meeting was conducted on August 10, 2016, with Mr. G. Struble, Simulator and Licensing Examination Manager, who was provided with the NRC licensing decisions.

The examination included the evaluation of three applicants for reactor operator licenses and three applicants for upgrade senior reactor operator licenses. The license examiners determined that five of six applicants satisfied the requirements of 10 CFR Part 55 and the appropriate licenses have been issued. There were four post-examination comments submitted by your staff. Enclosure 1 contains details of this report, Enclosure 2 summarizes post-examination comment resolution, and Enclosure 3 documents two minor simulator deficiencies.

No findings were identified during this examination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice and Procedure," 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

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Sincerely,

/RA/

Vincent G. Gaddy, Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50-445 and 50-446
License Nos. NPF-87 and NPF-89

Enclosures:
1. Examination Report 05000445/2016301; 05000446/2016301, w/Attachment: Supplemental Information
2. NRC Review of Comanche Peak Nuclear Power Plant Written Post-Examination Comments
3. Simulator Fidelity Report

cc w/encl: Electronic Distribution

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ADAMS ACCESSION NUMBER: ML16267A105

☐ SUNSI Review By: BLarson		ADAMS ☒ Yes ☐ No	☒ Publicly Available ☐ Non-Publicly Available		☒ Non-Sensitive ☐ Sensitive		Keyword: NRC-002
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SIGNATURE	VGG for	/RA/	/RA/	/RA/	RDA for	/RA/	
DATE	9/22/16	9/21/16	9/9/16	9/23/16	9/20/16	9/23/16	

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Letter to Ken Peters from Vincent G. Gaddy, dated September 23, 2016

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2 – NRC EXAMINATION REPORT 05000445/2016301; 05000446/2016301

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	05000445, 05000446
Licenses:	NPF-87, NPF-89
Report:	05000445/2016301; 05000446/2016301
Licensee:	Luminant Generation Company, LLC
Facility:	Comanche Peak Nuclear Power Plant, Units 1 and 2
Location:	6322 N. FM-56, Glen Rose, Texas
Dates:	July 11 through August 10, 2016
Inspectors:	B. Larson, Chief Examiner, Senior Operations Engineer J. Kirkland, Senior Operations Engineer M. Hayes, Operations Engineer
Approved By:	Vincent G. Gaddy Chief, Operations Branch Division of Reactor Safety

SUMMARY

ER05000445/2016301; 05000446/2016301; 07/11/2016 – 08/10/2016; Comanche Peak Nuclear Power Plant, Units 1 and 2; Initial Operator Licensing Examination Report.

NRC examiners evaluated the competency of three applicants for reactor operator licenses and three applicants for upgrade senior reactor operator licenses at the Comanche Peak Nuclear Power Plant, Units 1 and 2.

The licensee developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 10. The written examination was administered by the licensee on July 18, 2016. NRC examiners administered the operating tests the week of July 11, 2016.

The license examiners determined that two of three applicants for reactor operator licenses and all three applicants for upgrade senior reactor operator licenses satisfied the requirements of 10 CFR Part 55. One reactor operator license applicant failed the written portion of the examination. The appropriate licenses have been issued.

A. NRC-Identified and Self-Revealing Findings

None

B. Licensee-Identified Violations

None

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other Activities (Initial Operator License Examination)

.1 License Applications

a. Scope

NRC examiners reviewed all license applications submitted to ensure each applicant satisfied relevant license eligibility requirements. Examiners also audited three of the license applications in detail to confirm they accurately reflected the subject applicant's qualifications. This audit focused on the applicant's experience and on-the-job training, including control manipulations that provided significant reactivity changes.

b. Findings

No findings were identified.

.2 Examination Development

a. Scope

NRC examiners reviewed integrated examination outlines and draft examinations submitted by the licensee against the requirements of NUREG-1021. The NRC examination team conducted an onsite validation of the operating tests.

b. Findings

No findings were identified.

NRC examiners provided outline, draft examination, and post-validation comments to the licensee. The licensee satisfactorily completed comment resolution prior to examination administration.

NRC examiners determined the senior reactor operator (SRO)-only written examination initially submitted by the licensee was not within the range of acceptability expected for a proposed examination. The NUREG-1021 standard for an unsatisfactory submittal requires more than 20 percent of the written examination questions (reactor operator (RO)/SRO written examinations assessed separately) must be classified as unsatisfactory based on criteria in Section ES-401. The proposed SRO-only written examination was determined to be unsatisfactory as a result of a facility licensee requested post-examination change to Question 79. Section ES-501, C.2.c, states the regional office will also consider post-examination deletions and changes when evaluating the quality of the facility licensee's proposed examination for documentation in the examination report. The licensee indicated they would write a condition report to address the issue of an unsatisfactory SRO-only written examination submittal after this examination report is issued. As required by NUREG-1021, it is noted that future examination submittals should incorporate any lessons learned from the joint NRC and facility licensee examination review process.

NRC examiners determined the RO written examination initially submitted by the licensee was within the range of acceptability expected for a proposed examination. It is noted that the combined number of unsatisfactory questions and editorial revisions (76 percent) across both written examinations caused a substantial amount of work by NRC examiners and licensee staff to obtain an approved written examinations for administration.

The statistics for the proposed written examination were as follows:

Reactor Operator (RO) written examination (75 total questions)
1. Eleven questions were unsatisfactory (15 percent)
2. Forty-eight questions required editorial changes (64 percent)

Senior Reactor Operator (SRO) written examination (25 total questions)
1. Six questions were unsatisfactory (24 percent)
    - Question to K/A Mismatch - 2
    - Stem Focus Flaws – 3
    - Not SRO-only – 1
2. Eleven questions required editorial changes (44 percent)

Combined written examinations (100 total questions)
1. Seventeen questions were unsatisfactory (17 percent)
2. Fifty-nine questions required editorial changes (59 percent)

During the initial facility contact, the licensee was reminded of the option to submit some sample test items (up to ten written questions, one scenario, and one to two job performance measures) for preliminary review and comment by the chief examiner in accordance with the guidance contained in NUREG-1021, ES-201. The chief examiner described the intended approach of the test item review to be early in the examination development process as it was designed to give the licensee's examination writer(s) an understanding of what constitutes a satisfactory test item. It also provided an opportunity for the licensee to submit questions without them being assessed as unsatisfactory on the proposed examination submittal (if appropriate changes were made prior to the proposed examination submittal). The licensee did submit 10 questions for review, but chose to submit the questions after most of the written examination questions were completed. The facility expressed that their use of the sample test item review was more for correcting format type issues and did not expect to uncover significant errors that would have had any major impact on the question's content (also true for the sample operating test item review).

NRC examiners determined the operating test initially submitted by the licensee were within the range of acceptability expected for a proposed examination. One observation regarding operating test outlines was the submittal of two scenario outlines that were exactly the same and one scenario that was almost the same as scenarios used on a previous NRC initial license examination (third previous NRC examination). While the outlines did meet the criteria of not having been used on either of the last two NRC examinations, it was discussed with the Comanche Peak Nuclear Power Plant facility representative that the submittal did not meet the expectation for discrimination validity as discussed in NUREG-1021, Appendix A. The licensee re-submitted one entirely new scenario and significantly modified the other scenarios.

.3 Operator Knowledge and Performance

a. Scope

On July 18, 2016, the licensee proctored the administration of the written examinations to all six applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis to the NRC on July 22, 2016.

The NRC examination team administered the various portions of the operating test to all applicants during the week of July 11, 2016.

b. Findings

No findings were identified.

Five of the six applicants passed the written examination and all six applicants passed all parts of the operating test. The final written examinations, the operating test, and post-examination analysis and comments may be accessed in the ADAMS system under the accession numbers noted in the attachment.

The licensee noted the following generic weaknesses during the written examination reviews:

1. Containment spray chemical add tank discharge valve configuration and power supplies
2. Component cooling water train isolation valve arrangements
3. Reasons for performing additional load shedding per ECA-0.0, Attachment 2.C
4. The use of standing orders and their relationship to approved procedures
5. Tech Spec 3.1.6 Bases
6. Tech Spec 3.3.3 Bases
7. Performance of Operability Assessments

The licensee wrote Condition Report CR-2016-006876 to address these knowledge weaknesses.

The examination team noted the following generic weaknesses during the operating test:

1. Time critical actions associated with the containment spray system
2. ABN-712, Section 3.0, Step 15.h - "Verify Rod Control Urgent Failure Alarm - clear" and associated logic diagram from ALM
3. Response to Letdown temperature control valve controller failure, 1-TK-130
4. Main feedwater control during abnormal conditions
5. Post reactor trip auxiliary feedwater control
6. Procedure use and adherence

Additionally, the licensee submitted four post-examination comments (Q43, Q48, Q73, Q79) that required review and disposition by the chief examiner. The Region IV Operations Branch Chief assigned a panel of three examiners to review each question submitted in the post-examination comments. None of the examiners assigned to the panel had any involvement with examination development, validation, or administration of the written examination. The panel results were provided to the chief examiner for use in developing the NRC resolution for each question. Enclosure 2 of this report contains the Comanche Peak Nuclear Power Plant post-examination comments and NRC resolution for each of the four questions. The entire licensee's post-examination comments and analysis can be found in ADAMS using Accession Number ML16262A036.

Copies of all individual examination reports were provided the facility training manager for evaluation and determination of appropriate remedial training.

.4 Simulation Facility Performance

a. Scope

The NRC examiners observed simulator performance with regard to plant fidelity during examination validation and administration.

b. Findings

No findings were identified.

During administration of the operating tests, two simulator deficiencies were identified. See Enclosure 3, Simulator Fidelity Report, for details.

.5 Examination Security

a. Scope

The NRC examiners reviewed examination security for examination development during both the on-site validation week and examination administration week for compliance with 10 CFR 55.49 and NUREG-1021. Plans for simulator security and applicant control were reviewed and discussed with licensee personnel.

b. Findings/Observations

No findings were identified.

4OA6 Meetings, Including Exit

Exit Meeting

The chief examiner presented the preliminary examination results to Mr. T. McCool, Site Vice President, and other members of the staff on July 13, 2016. A telephonic exit was conducted on August 10, 2016, between Mr. B. Larson, Chief Examiner, and Mr. G. Struble, Operations Training Manager, where final licensing decisions were provided.

The licensee did not identify any information or materials used during the examination as proprietary.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

T. McCool, Site Vice President
J. Dreyfuss, Plant Manager
T. Hope, Regulatory Affairs Manager
G. Struble, Operations Training Manager
A. Glass, Operations Training Supervisor
D. Cox, Initial Operations Training Supervisor
S. Dixon, Licensing Analyst
R. Blankenship, Consulting Engineer
J. Ruby, Exam development staff

NRC Personnel

J. Josey, Senior Resident Inspector
R. Kumana, Resident Inspector

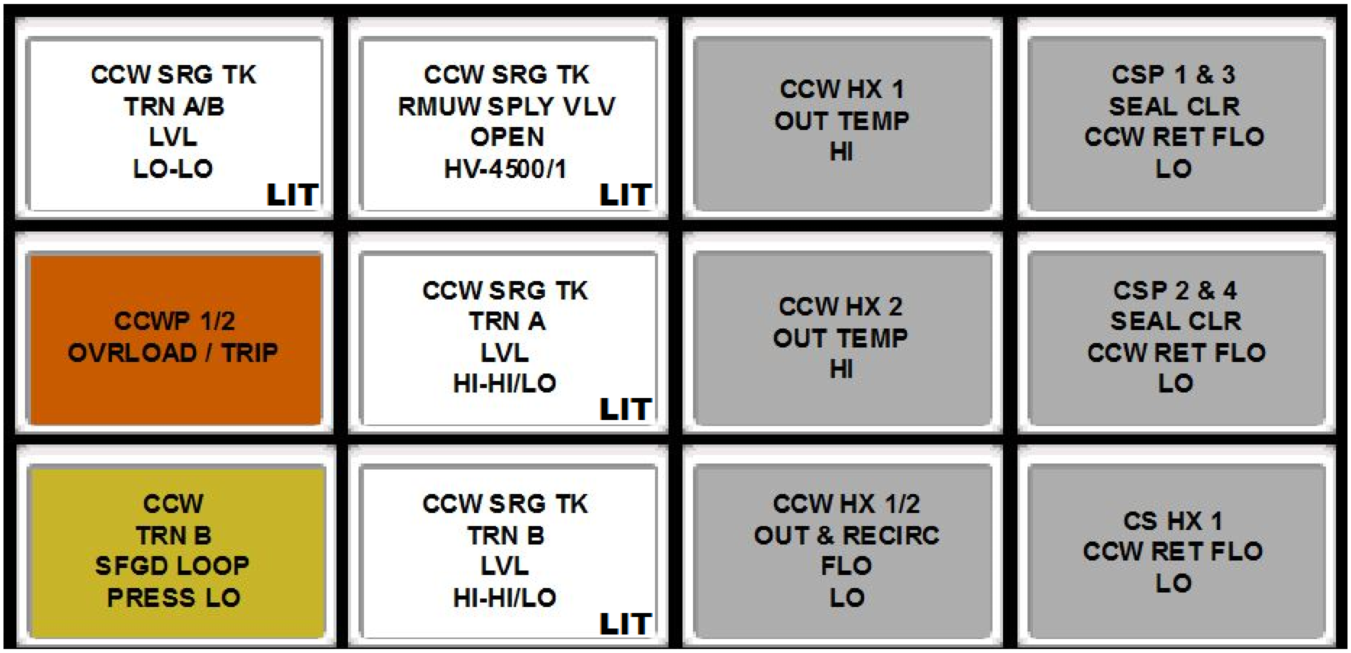
ADAMS DOCUMENTS REFERENCED

Accession No. ML16262A038 - FINAL WRITTEN EXAMS (delayed release July 18, 2018)

Accession No. ML16262A037 - FINAL OPERATING TEST (delayed release July 18, 2018)

Accession No. ML16262A036 - POST EXAM COMMENTS AND ANALYSIS (delayed release July 18, 2018)

Question 43



The above Unit 1, LIT alarms were received and subsequently

- 1-HS-4512, SFGD LOOP CCW RET VLV and 1-HS-4514, SGFD LOOP CCW SPLY VLV were closed on affected train
- The leakage was NOT stopped

Per ABN-502, Component Cooling Water System Malfunction, what additional valve must be closed to isolate affected train and restore normal operation in unaffected train?

A. 1CC-0021, CCW SRG TK 1-01 TRN A OUT VLV

B. 1CC-0023, CCW PUMP 1-01 SUCT ISOL VLV

C. 11CC-0071, CCW SRG TK 1-01 TRN B OUT VLV

D. 1CC-0067, CCW PUMP 1-02 SUCT ISOL VLV

Answer: A

Explanation:

A. Correct. Per ABN-502, the affected train surge tank outlet valve must also be closed in order to allow the surge tank to be refilled for the operating train.

B. Incorrect. Plausible as this is the manual pump suction valve. This valve is plausible if thought that the Train is isolated with the exception of the small portion of the system from the surge tank to the suction of the pump. However, this isolation would not isolate the section of piping from the surge tank to the suction piping which may be the location of the leak. Further, the procedure does not call out for this valve to be isolated as it does the surge tank isolation valve.

C. Incorrect. Plausible as this is the same valve as the correct Train A valve for Train B. The applicant must recognize the configuration and determine the train which must be isolated.

D. Incorrect. Plausible for Train B as described in 'B' above for the affected Train A.

Licensee Comments for Question 43

During the written examination review of the CPNPP 2016 NRC exam it was identified that Question 43 lacks the necessary information in the stem in accordance with NUREG-1021 to determine the correct answer.

- Based on the information provided in the stem the applicants were unable to ascertain which train was the affected train. This resulted in all applicants incorrectly determining that Train B was the affected train based on the valve numbers being 'even' numbers.
- The Train A valves, which are designated as Train A in the procedure, do not have Train A in the valve nomenclature provided in the stem.
- The originally validated question, prior to proposed exam submittal, did NOT include a picture and provided ONLY that the CCW Surge Tank Train A HI-HI/LO alarm was LIT. However, upon revision of the question a picture was inserted that indicated both the Train A AND Train B CCW Surge Tank HI-HI/LO Alarms were LIT, which is correct for the malfunction in progress. With both of these alarms LIT, it is not reasonable to believe that an applicant can determine which Train is the affected Train based on valves that are operated in the Control Room that have no indication in the nomenclature of which Train they belong to.
- Every applicant chose the distractor that would have been the correct answer if the affected Train were Train B, which is Distractor 'C'. Based on applicant selection, CPNPP believes that the applicant's selections on the exam DO NOT display a knowledge weakness but that the question is lacking the necessary information for the applicants to make the correct decision.

Solution: CPNPP recommends removing the question from the Reactor Operator written examination. Without adequate information in the stem it is not reasonable to expect the applicants to successfully answer the question.

NUREG-1021 justification: The proposed revision is in agreement with NUREG-1021 Rev. 10 ES-403.D.1.b 1st bullet which states "a question with an unclear stem that confused the applicants or did not provide all the necessary information." As well as Appendix B paragraph C.1.e which states, "Avoid questions that are unnecessarily difficult or irrelevant."

NRC Resolution of Question 43

NRC concludes that the question is acceptable as written, and does not accept the facility licensee's solution to remove the question from the RO exam. Knowledge of which valves are associated with which train is not considered to be minutia for the following reasons:

- Operators should be expected to know the train association for valves that receive automatic actuation signals during a lowering CCW Surge Tank level event

- Applicants were trained on the association of valve numbers and nomenclature to designated trains at least three times (systems, alarm response and abnormal operating procedures)

NRC believes there was sufficient information provided in the question stem for the applicant to ascertain which train was the affected train. Specifically, the valve numbers and nomenclature was sufficient if the applicant had recalled from memory the even number valves were train A. NRC also believes it was reasonable to expect that an applicant could have determined which train was affected based on both train A and train B CCW Surge Tank HI-HI/LO alarms being lit as given in the question stem. For the event in progress, it would be expected that both these annunciator alarms to be lit as CCW Surge Tank level lowered. The indications provided in the stem are consistent with the design of the facility's CCW system.

The facility licensee's reference to NUREG-1021, ES-403, D.1.b, and Appendix B, paragraph C.1.e, are not applicable to this question. NUREG-1021, ES-403, D.1.b, second paragraph, 2nd bullet provides more relevant guidance for questions where it is determined that a reference might have been needed for the applicants to determine the correct answer. The second paragraph states, "Given that both the NRC and the facility licensee agreed that the examination met the requirements of NUREG-1021 before the examination was administered, these types of question errors, identified after examination administration, are less likely to result in examination changes." During the proposed examination review conducted in the RIV offices, it was openly discussed and acknowledged that to obtain the correct answer to this question, the applicant would be required to recall from memory that the SFGD LOOP CCW RET and SPLY VLVs with even numbers (1-HS-4512/4514) were train A valves, not train B. Facility licensee exam team members stated this was not the standard labeling practice at CP, with these valves being possibly the only situation in the plant where even number valves were associated with train A, not train B. Knowing this, the applicant would be able to diagnose the leak in train A. With knowledge of ABN-502, they would recall the additional valve that must be closed to isolate the affected train was the CCW Surge Tank outlet valve (Answer A, 1CC-0021, CCW SRG TK 1-01 TRN A OUT VLV). Comanche Peak examination team members also stated it was required knowledge for operators to know this unique difference and that the applicants had been trained on this knowledge.

The licensee wrote CR-2016-006876 to address, in part, CCW train isolation valve arrangements.

Question 48

During a Station Blackout additional load shedding is performed when safeguards battery voltage is less than 110 volts to allow for __________________ and _______________.

A. battery charger restoration with portable generator
plant monitoring and control until AC power restored

B. battery charger restoration with portable generator
Safeguards Bus supply breaker closure

C. Diesel Generator field flashing
plant monitoring and control until AC power restored

D. Diesel Generator field flashing
Safeguards Bus supply breaker closure

Answer: D

Explanation:

A. Incorrect. Plausible because Attachments 2.A and 2.B when performed ensure sufficient time to restore battery chargers using a portable generator, however this is not what is accomplished by Attachment 2.C. The second part is incorrect but plausible (See C below).

B. Incorrect. First part is incorrect but plausible (See A above). Second part is correct (See D below).

C. Incorrect. First part is correct (See D below). The second part is incorrect but plausible because load shedding does provide for plant monitoring and control until AC power is restored during initial load shedding not the load shedding performed per Attachment 2.C.

D. Correct. If battery voltage lowers to less than 110 volts the associated bus is further load shed to ensure adequate voltage remains for flashing the diesel generator field or closing safeguards bus supply breakers for power restoration.

Licensee Comments for Question 48

Problem: During the written examination review of the CPNPP 2016 NRC exam it was identified that Question 48 lacks sufficient detail in accordance with NUREG-1021 to eliminate a distractor as a correct answer. In particular:

- The correct answer as approved on the worksheet and answer key is ‘D’.
- Answer ‘C’ is a statement which is true during initial load shedding of Attachment 2.A, it specifically states: “plant monitoring and control until AC power restored”
- However, based on information from the bases of ECA-0.0A for all Attachment 2’s (including Attachments 2.A, 2.B, 2.C, and 2.D) which states, “These attachments provide instructions for shedding of DC safeguards bus loads in order to conserve capacity to assist in future actions to restore AC power, while maintaining that minimum instrumentation necessary to monitor plant conditions.”, the applicants were unable to discern whether answer ‘D’ was a specific subset of answer ‘C’

- Therefore, the applicants were unable to eliminate answer ‘C’ as an incorrect answer.

Therefore, CPNPP ascertains that Question 48 answer ‘D’ and distractor ‘C’ should both be accepted as correct answers.

The first part of answer ‘D’ and distractor ‘C’ was reviewed and deemed to be accurate, thus making both ‘C’ and ‘D’ correct answers.

Solution: CPNPP recommends accepting distractor ‘C’ and answer ‘D’ as correct answers. With the inability of the applicants to eliminate distractor ‘C’ as it appears to include the specific subset called out in answer ‘D’ it is not reasonable to expect the applicants to successfully eliminate distractor ‘C’ as a possible answer to the question.

NUREG-1021 justification: The proposed revision is in agreement with NUREG-1021 Rev. 10 ES-403 D.1.c which states, “If it is determined that there are two correct answers, both answers will be accepted as correct.”

<u>NRC Resolution of Question 48</u>

NRC concludes that the question is acceptable as written, and does not accept the facility licensee’s position that the question stem lacks sufficient detail to support eliminating distractor ‘C’ and that choices ‘C’ and ‘D’ should both be considered correct answers.

The licensee argues that distractor ‘C’ represents a statement that is true during initial load shedding activities, and that choice ‘D’ is ultimately a subset of distractor ‘C.’ Essentially the licensee believes that the general basis for ECA-0.0A, Revision 9, step 18, “battery power must be conserved to permit monitoring and control of the plant until AC power is restored” and that the additional load shedding that occurs when battery voltage drops below 110 volts is part of the bigger effort to allow for control of the plant. However, the NRC finds the question stem provides sufficient distinction as to the purpose of the “additional load shedding” that occurs “when safeguards battery voltage is less than 110 volts.” This important distinction successfully frames the question in a manner that requires the applicant to exhibit the specific knowledge of the basis for these additional actions. The reasons for the additional actions are clearly laid out in the basis for ECA -0.0A, Revision 9, step 18, and are to reserve sufficient battery power to flash the diesel generator field or close safeguards supply breakers once AC power is restored. Distractor ‘C’ presents an answer choice that suggests the reason for the additional load shedding is to be able to flash the diesel generator field and to support plant monitoring and control. Plant monitoring is not part of the basis for the need to perform the additional load shedding; therefore, choice ‘D’ cannot be considered a subset of distractor ‘C.’

NRC believes that the specific understanding of the basis for additional load shedding that is performed once battery voltage drops below 110 volts is important RO-level knowledge. Further, the NRC believes the question stem provided sufficient distinction to support testing an applicant’s knowledge of the specific reasons for the additional load shedding. Therefore, the question remains as-is, and choice ‘D’ is the only correct answer.

The licensee wrote CR-2016-006876 to address, in part, reasons for performing additional load shedding per ECA-0.0, Attachment 2.C.

Question 73

- Crew composition:
  - Shift Manager
  - 3 other Senior Reactor Operators
  - Field Support Supervisor position is NOT filled
  - 5 Reactor Operators
  - 8 Nuclear Equipment Operators
  - Non Operations Staffing is at MINIMUM Shift Crew Composition

- An ALERT has been declared on Unit 1

- After initial notifications were completed, NRC requested ENS line be manned by a dedicated individual

Per ODA-102, Conduct of Operations ______________________ should be assigned to the ENS line, and that individual may also __________________________.

A. Unit 2 Balance of Plant Operator
operate common system equipment

B. Relief Reactor Operator
operate common system equipment

C. Unit 2 Balance of Plant Operator
keep OSC Manager informed of NEO activities

D. Relief Reactor Operator
keep OSC Manager informed of NEO activities

Answer: B

Explanation:

A. Incorrect. The first part is incorrect but plausible as ODA-102, states that the BOP on the unaffected unit is to perform the duties of the Relief Reactor Operator if the Relief Reactor Operator is unavailable. Knowledge of minimum shift staffing shows that the Relief Reactor Operator position is available. The second part is correct as described in 'B' below.

B. Correct. In accordance with ODA-102, the Relief Reactor Operator should be assigned these duties as the SROs are at minimum crew staffing and cannot be assigned to the ENS. ODA-102, lists other responsibilities that the Relief Reactor Operator can be asked to do during emergencies and states that they may operate the common equipment as directed.

C. Incorrect. The first part is incorrect but plausible as described in 'A' above. The second part is incorrect but plausible as described in 'D' below.

D. Incorrect. The first part is correct as described in 'B' above. The second part is incorrect but plausible if thought that since the Field Support Supervisor position is unmanned, the Relief Reactor Operator should assume the duty of keeping the OSC Manager informed of NEO activities in lieu of the Field Support Supervisor. This is not one of the duties assigned to the Relief Reactor Operator.

Licensee Comments for Question 73

Problem: During the written examination review of the CPNPP 2016 NRC exam it was identified that Question 73 lacks sufficient detail in accordance with NUREG-1021 to discriminate between the approved answer and one of the distractors. In particular:

- The correct answer as approved on the worksheet and answer key is 'B'.
- Answer 'B' is a specifically delineated responsibility of the Relief Reactor Operator in accordance with ODA-102, specifically, "Operate the common system equipment as directed."
- In contrast, distractor 'D' stated, "keep the OSC Manager informed of NEO activities"
- The action in distractor 'D' is considered incorrect as it was an action assigned to the Field Support Supervisor per ODA-102. However, the stem states the "Field Support Supervisor position is NOT filled"
- Therefore, the applicants were unable to discriminate this action as being incorrect in that the duties and responsibilities of the Relief Reactor Operator also includes, "Assist the Emergency Coordinator." "Assist the Emergency Coordinator" duty is further justified as being included in the discrimination process by the use of the word "may" in the second fill in the blank statement.
- Based on this lack of discrimination in what the Relief Reactor Operator can and cannot do in assisting the Emergency Coordinator, keeping the OSC Manager informed of NEO activities could not be ruled out as incorrect.

Therefore, CPNPP ascertains that Question 73 answer 'B' and distractor 'D' should both be accepted as correct answers.

The first part of answer 'B' and distractor 'D' were reviewed and deemed to be accurate, thus making 'B ' and 'D' correct.

Solution: CPNPP recommends accepting answer 'B' and distractor 'D' as correct answers. With the inability of the applicants to eliminate distractor 'D' as it is inclusive of 'Assist the Emergency Coordinator' it is not reasonable to expect the applicants to successfully eliminate distractor 'D' as a possible answer to the question.

NUREG-1021 justification: The proposed revision is in agreement with NUREG-1021 Rev. 10 ES-403 D.1.c which states, "If it is determined that there are two correct answers, both answers will be accepted as correct."

NRC Resolution of Question 73

NRC does not concur with the facility licensee's recommendation to change the Answer Key to accept two correct answers. NRC does agree that the term "may" in the question stem allows the applicant the opportunity to extend a generic "Assist the Emergency Coordinator" role of the Relief Reactor Operator to mean assignment of a task specifically identified by procedure to be the responsibility of the Field Support Supervisor, a Senior Reactor Operator, and when other crew members are available. Therefore, NRC resolution is to make no changes to the Answer Key (Answer "B" is the only correct answer).

The question was designed to test the applicant's knowledge of ODA-102, Conduct of Operations, for the specific assignment of duties a Relief Reactor Operator (RRO) should perform during emergencies. The first part of the question asked who should be assigned to the ENS line (U2 BOP or RRO) and is not challenged by this post-examination comment. With the selection of the individual in the first part, the second part of the question asked what another duty that person should perform. Answer B, operate common system equipment, is a duty taken directly from the list in ODA-102, Section 6.11, Duties/Responsibilities of the Relief Reactor Operator (RRO). Distractor D, keep OSC Manager informed of NEO activities, was taken directly from the list in ODA-102, Section 6, Duties/Responsibilities of the Field Support Supervisor (FSS). As identified on the final revision of the question worksheet, Distractor D was incorrect, but plausible if thought that since the FSS position was unmanned, the RRO should assume the duty of keeping the OSC Manager informed of NEO activities in lieu of the FSS. During question validation, it was known that the FSS position was presented in the stem as not being filled, and Distractor D was still incorrect, but plausible.

The facility licensee's comment that the applicant could not discriminate between the two duties for the following reasons:

- FSS position was not filled
- wording in the stem of "may also"
- list of RRO duties during emergencies includes "Assist the Emergency Coordinator"

As discussed above, the FSS position not being filled was included in the plausibility statement for Distractor D. Given the vacancy, there were two factors that NRC reviewed to determine its impact. It was assessed that during an Alert, the need for keeping the OSC Manager informed of Nuclear Equipment Operators (NEOs) activities was a valid need that should be accomplished. To determine who should perform the role, the defined crew composition at stated in the stem was reviewed. It was provided that three other SRO's were assigned in addition to the Shift Manager. For a dual unit site, one SRO would be required to serve as Unit Supervisor, leaving one SRO not directly assigned. Since the FSS position is an SRO position, it was determined that the Shift Manager (Emergency Coordinator) would more likely assign the extra SRO to perform the role of keeping the OSC Manager informed of NEO activities. In addition, the RRO assigned to the ENS line as their primary function would not have time available to also serve as the conduit between the Control Room and OSC for eight NEOs. It has been consistently demonstrated through Emergency Plan exercises that the individual serving as the ENS communicator is fully engaged.

The stipulation that the term "may also" when combined with "Assist the Emergency Coordinator" creates a condition where any conceivable emergency duty becomes acceptable to assign the RRO, while they are serving as ENS Communicator, is not acceptable. As stated above, the question was testing the applicant's knowledge of ODA-102, Conduct of Operations, for the specific assignment of duties an RRO should perform during emergencies. As used in the stem, the term "may also" was synonymous with the term "should." If the applicant were familiar with duties to be performed by an RO during an emergency, as specified in ODA-102, they would have been able to differentiate between a role performed by an RO and a role performed by an SRO. The question, as written, is an acceptable testing instrument to discriminate between the two choices and thereby ascertain whether or not the applicant possessed sufficient knowledge of ODA-102.

Question 79

- Unit 2 Steam Line Break OUTSIDE containment
- Fault occurred on 2ST, STATION SERVICE TRANSFORMER 2ST
- 2EA2, Safeguards 6.9KV received 86-1 lockout when Reactor tripped
- EOP-2.0B, Faulted Steam Generator Isolation, Step 8 'Check if ECCS Flow Should Be Reduced' in progress with the following parameters:
  - SG 2-01 NR level 0%
  - All other SG NR levels 5% to 8% and increasing
  - AFW total flow 470 gpm and stable
  - RCS subcooling 52°F and stable
  - RCS pressure 2240 psig and decreasing
  - Pressurizer Level 70% and increasing

TDAFWP ________ be placed in Pull Out.

Unit Supervisor has announced transition to _________________________.

A. should
EOP-1.0B, Loss of Reactor or Secondary Coolant

B. should
EOS-1.1B, Safety Injection Termination

C. should NOT
EOP-1.0B, Loss of Reactor or Secondary Coolant

D. should NOT
EOS-1.1B, Safety Injection Termination

Answer: D

Explanation:

A. Incorrect. First part is incorrect but plausible (See B below). Second part is incorrect but plausible (See C below).

B. Incorrect. First part is incorrect but plausible if the applicant does not recognize that 2 of the 3 intact SGs are being fed by the TDAFWP even though a single MDAFWP can supply greater than the minimum 460 gpm for heat sink maintenance. Second part is correct (See D below).

C. Incorrect. First part is correct (See D below). Second part is incorrect but plausible if the applicant does not identify RCS pressure decreasing is due to PORV cycling and transition to EOP-1.0B is not appropriate because transition to EOS-1.1B is needed to stop ECCS injection which is causing the PORV to cycle and PRZR level to rise.

D. Correct. First part is correct as the TDAFWP is feeding 2 of the 3 intact SGs and should not be secured. Second part is correct as SI termination criteria per EOP-2.0B are met even though RCS pressure is decreasing the applicant must determine that is due to a PORV cycling due to RCS pressure and PRZR level rising.

Licensee Comments for Question 79

Problem: During the written examination review of the CPNPP 2016 NRC exam it was identified that Question 79 lacks sufficient detail in accordance with NUREG-1021 to obtain the answer previously approved as correct. In particular:

- The correct answer as approved on the worksheet and answer key is 'D'.
- Answer 'D' included a transition to EOS-1.1B, Safety Injection Termination vice a transition to EOP-1.0B, Loss of Reactor or Secondary Coolant.
- In accordance with EOP-2.0B, Step 8, in order for the transition to be made to EOS-1.1B, the RCS pressure is required to be "Stable or Increasing."
- In contrast with the above statement, the stem of Question 79 states that "RCS pressure is 2240 psig and decreasing"
- The provided justification for the original answer was that this was indication following the opening of a PORV on high RCS pressure; however, no other information was included in the stem which indicated that a PORV had opened. In particular, the inclusion of Tailpipe outlet temperature, PORV position indication, PORV Not Closed alarm or PRT parameters would have been required to determine that a PORV had indeed opened.
- During the exam review it was determined that two of the three SRO applicants had interpreted the indication as the existence of a possible steam space leak or other anomaly which would have resulted in decreasing pressure with increasing pressurizer level. Based on this interpretation, the selection of a transition to EOP-1.0B is correct.

Therefore, CPNPP ascertains that Question 79 has information in the stem that indicates the correct answer is distractor 'C' [based on RCS pressure is 2240 psig and decreasing] vice answer 'D' therefore it is not reasonable to expect the applicants to eliminate distractor 'C' as the correct answer and it is reasonable to expect the applicants to eliminate answer 'D' as a correct answer.

The first part of distractor 'C' was reviewed and deemed to be accurate, thus making distractor 'C' correct.

Solution: CPNPP recommends accepting distractor 'C' as the correct answer and not accepting answer 'D' as a correct answer.

NUREG-1021 justification: The proposed revision is in agreement with NUREG-1021 Rev. 10 ES-403 D.1.b 1st bullet which states "a question with an unclear stem that confused the applicants or did not provide all the necessary information."

NRC Resolution of Question 79

NRC does not concur with the facility licensee's recommendation to change the Answer Key from "D" to "C." NRC does agree that the question stem did not provide all the necessary information, and could have led to applicant confusion in which answer was correct. NRC resolution is to delete this question from the examination.

Guidance in NUREG-1021, ES-403, Grading Initial Site-Specific Written Examinations, D.1.b, states that despite extensive reviews performed by both the NRC and the facility licensee before examination administration, it is possible that a few errors may be discovered only after an examination has been administered. The NUREG identifies several types of errors, if identified and adequately justified by the facility licensee, will likely result in post-examination changes agreeable to the NRC. One such type of error is a question with an unclear stem that confused the applicants or did not provide all the necessary information.

NRC agrees with the facility licensee that the stem of the question lacked sufficient information for an applicant to positively determine that the given parameter "RCS pressure 2240 psig and decreasing" was a direct result of a PZR PORV cycling due to high RCS pressure. Specifically, the stem did not provide any indications of a cycling PORV such as higher than normal PRT parameters (temperature and/or pressure), annunciators associated with an open or cycling PORV, or rising PORV Tailpipe outlet temperatures. Without indication of a cycling PORV, the applicant could have been confused on how to proceed at Step 8 of EOP-2.0B, Faulted Steam Generator Isolation. Step 8 was to check if ECCS flow should be reduced by checking SI termination criteria.

- If RCS pressure was evaluated to be stable or increasing, the applicant's knowledge would have him continue in Step 8, eventually transitioning to EOS-1.1B, Safety Injection Termination. (Answer D)
- If RCS pressure was not determined to be stable or increasing, the applicant's knowledge of the procedure's response not obtained action would have him transitioning to EOP-1.0B, Loss of Reactor or Secondary Coolant. (Answer C)

With the exception noted above, all other stem information was consistent with a terminated Steam Line Break outside of Containment due to a fault in Steam Generator 2-01. The question stem does not support a diagnosis of PZR vapor space leak (RCS pressure lowering, rising PZR level) or other anomaly as no Containment parameters (temperature, pressure, humidity) were given. NUREG-1021, Appendix E, Section B.7, contains guidance to not make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question. It was confirmed by the Chief Examiner during examination validation that all applicants had been provided a copy of Appendix E. Therefore, the facility licensee's contention that it was not reasonable to expect an applicant to eliminate distractor "C" is not accepted.

In post-examination discussions with the facility licensee, a question was raised regarding performance of EOP-2.0B, Step 8.c: Does the current procedure provide sufficient guidance such that operators are allowed to interpret RCS pressure as "stable" during the period of time where a PZR PORV is cycling? The facility agreed to engage Operations Department in order to determine if a procedure enhancement opportunity exists to add clarification, Note, Caution or other remedy.

The licensee wrote CR-2016-007280 to address a procedure enhancement opportunity with regard to EOP-2.0A/B to evaluate if the bases for step 8 should include wording describing the possibility of RCS pressure lowering due to PORV cycling to ensure proper transition to EOS-1.1A/B if the only reason for lowering RCS pressure is due to the PORV cycling.

Facility Licensee: Comanche Peak Nuclear Power Plant

Facility Docket No.: 05000445, 05000446

Operating Tests Administered on: Week of July 11, 2016

While conducting the simulator portion of the operating tests, examiners observed the following items:

Item	Description
Non-Safeguards Bus Volt Meter (V-1A) on CB-11	Meter reads approximately 6700 volts during normal conditions. Identified on SAR# - 14SA13.
Train A SSII panel acknowledge button	Button requires several attempts to acknowledge alarms. Identified on SAR# - 16SA69.