



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
REGION IV  
1600 EAST LAMAR BOULEVARD  
ARLINGTON, TEXAS 76011-4511

September 3, 2019

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

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2 - NRC  
EXAMINATION REPORT 05000445/2019301 AND 05000446/2019301

Dear Mr. Peters:

On July 29, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an initial operator license examination at 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 June 20, 2019, with Mr. T. McCool, Site Vice President, and other members of your staff. A telephonic exit meeting was conducted on July 29, 2019, with Ms. D. Christiansen, Training Director, who was provided the NRC licensing decisions.

The examination included the evaluation of four applicants for reactor operator licenses, five applicants for instant senior reactor operator licenses, and four applicants for upgrade senior reactor operator licenses. The license examiners determined that 9 of the 13 applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued. There were six post-examination comments submitted by your staff. Enclosure 1 contains details of this report and Enclosure 2 summarizes post-examination comment resolution.

No findings were identified during this examination.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at <http://www.nrc.gov/reading-rm/adams.html> and at the NRC Public Document

K. Peters

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Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

**/RA/**

Gregory E. Werner, 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 05000445/2019301 and 05000446/2019301  
w/Attachment: Supplemental Information
2. NRC Post-Examination Comment  
Resolution

cc w/enclosure: Electronic Distribution Comanche  
Peak Nuclear Power Plant

**U.S. NUCLEAR REGULATORY COMMISSION  
REGION IV**

Docket: 50-445 and 50-445

License: NPF-87 and NPF-89

Report: 05000445/2019301 and 05000446/2019301

Enterprise Identifier: L-2019-OLL-0006

Licensee: Vistra Operations Company, LLC

Facility: Comanche Peak Nuclear Power Plant, Units 1 and 2

Location: Glen Rose, TX

Dates: June 10, 2019, to July 29, 2019

Inspectors: J. Kirkland, Chief Examiner, Senior Operations Engineer  
T. Farina, Operations Engineer  
N. Hernandez, Operations Engineer  
J. Drake, Senior Reactor Inspector

Approved By: Gregory E. Werner, Chief  
Operations Branch  
Division of Reactor Safety

## SUMMARY

ER 05000445/2019301; 05000446/2019301; June 10 – July 29, 2019; Comanche Peak Nuclear Power Plant, Units 1 and 2; Initial Operator Licensing Examination Report

The NRC examiners evaluated the competency of four applicants for reactor operator licenses, five applicants for instant senior reactor operator licenses and four applicants for upgrade senior reactor operator licenses at 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 11. The written examination was administered by the licensee on June 20, 2019. The NRC examiners administered the operating tests on June 10 – 14, 2019.

The NRC examiners determined that 9 of the 13 applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

A. NRC-Identified and Self-Revealing Findings

None

B. Licensee-Identified Violations

None

## REPORT DETAILS

### OTHER ACTIVITIES – INITIAL LICENSE EXAM

#### .1 License Applications

##### a. Scope

The NRC examiners reviewed all license applications submitted to ensure each applicant satisfied relevant license eligibility requirements. The NRC examiners also audited three of the license applications in detail to confirm that 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

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

##### b. Findings

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

The NRC examiners determined the written examinations and operating tests initially submitted by the licensee were within the range of acceptability expected for a proposed examination.

#### .3 Operator Knowledge and Performance

##### a. Scope

On June 20, 2019, the licensee proctored the administration of the written examinations to all 13 applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis and post-examination comments to the NRC on July 11, 2019.

The NRC examination team administered the various portions of the operating tests to all applicants on June 10 – 14, 2019.

b. Findings

No findings were identified.

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

The examination team noted 12 generic weaknesses associated with applicant performance on the written examination. The licensee-initiated Tracking Report TR-2019-005720 to determine if changes to the training program are required. Specifically, the applicants displayed weaknesses associated with:

- the number of rupture disks in the pressurizer relief tank
- the target steam generator level during a natural circulation cooldown
- why excess letdown isolation valves close on a safety injection signal
- whether to initially place control rods in manual or automatic during an automatic transient without a scram
- the actions associated with closing main steam isolation valves during a loss of power
- the basis for stroke times associated with containment spray heat exchanger discharge valves
- the actions taken in response to a release of radioactive and/or toxic gas
- who can authorize trainees to manipulate the reactor controls
- starting residual heat removal in response to a residual heat removal system malfunction
- the actions associated with Technical Specification 3.7.1, "Main Steam Safety Valves"
- personnel air locks
- the amount of time to secure emergency core cooling following a spurious safety injection

Copies of all individual examination reports were sent to 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.

.5 Examination Security

a. Scope

The NRC examiners reviewed examination security for examination development during both the onsite preparation 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

No findings were identified.

**EXIT MEETINGS AND DEBRIEFS**

Exit Meeting Summary

The chief examiner presented the preliminary examination results to Mr. T. McCool, Site Vice President, and other members of the staff on June 13, 2019. A telephonic exit was conducted on July 29, 2019, between Mr. J. Kirkland, chief examiner, and Ms. D. Christiansen, Training Director. 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  
D. Christiansen, Training Director  
J. Ruby, Exam Developer

#### **NRC Personnel**

J. Josey, Senior Resident Inspector

### **ADAMS DOCUMENTS REFERENCED**

Accession No. ML19198A018 - Final Written Exams (Do Not Release for Two Years)  
Accession No. ML19198A017 - Final Operating Test (Do Not Release for Two Years)  
Accession No. ML19198A019 - Post Exam Comments (Do Not Release for Two Years)



## NRC Resolution to the Comanche Peak Nuclear Power Post-Examination Comments

A complete text of the licensee's post-examination analysis and comments can be found in ADAMS under Accession Number ML19198A019.

### **QUESTION 6**

An inadvertent SI Actuation has occurred on Unit 1.

Which of the following is an action performed in EOS-1.1A, Safety Injection Termination and the reason the action is performed?

- A. Secure SI Pumps to prevent motor heating due to running for extended time periods at minimum flow.
- B. Restore Instrument Air to containment to allow use of normal PRZR spray valves to control PRZR pressure.
- C. Secure CCPs running in Injection Mode to prevent PRZR bubble collapse and risk damaging PRZR safety valves.
- D. Secure RCPs to prevent running with inadequate pump seal cooling due to seal water return containment isolation.

Keyed correct answer is 'C.'

**COMMENT:** The licensee recommends accepting both answers 'B' and 'C' as correct answers. During the written examination review it was identified that Question 6 has two correct answers. Answers 'B' and 'C' are both correct as identified below:

- The question simply asks which of the answers is an action performed in EOS-1.1A and the reason the action is performed.
- Answers 'B' and 'C' are both actions performed in EOS-1.1A with correct reasons for the actions.
- Answer 'C' is identified as the correct answer on the key and it is correct per EOS-1.1A.
- Answer 'B' was identified during the exam review as a correct answer as well.
- Answer 'B' states: "Restore Instrument Air to containment to allow use of normal PRZR spray valves to control PRZR pressure."
- Per EOS-1.1A, Step 6 Bases: "The restoration of Instrument Air and Nitrogen is necessary to allow the operation of pneumatically operated valves in containment. While opening the containment isolation valves is sufficient to restore nitrogen to containment, it might also be necessary to start an air compressor to restore instrument air to containment."
- EOS-1.1A, Step 20 requires the use of normal PRZR spray as necessary to maintain PRZR pressure stable. Without instrument air restored to containment, the operation of the normal PRZR spray valves would not be possible.
- Question 6 was an exam question used from the 2014 NRC Exam (Q31). Originally the question read as follows: "Which of the following is an adverse effect of allowing Safety Injection to continue while performing EOS-1.1A, Safety Injection Termination?" The question was changed to read as follows: "Which of the following is an action performed in EOS-1.1A, Safety Injection Termination and the reason the action is performed?"

- Answer 'B' originally read as follows: "Loss of Instrument Air to Containment will not allow the use of the normal Pressurizer Spray Valves to control Pressurizer Pressure." Answer 'B' was changed to read as follows: "Restore Instrument Air to containment to allow use of normal PRZR spray valves to control PRZR pressure."
- Answers 'A' and 'D' are incorrect as identified in the question plausibility statements.

**NRC RESOLUTION:** The NRC agrees that 'B' is a correct answer but does not agree that the keyed answer 'C' is a correct answer. The licensee is correct that restoration of instrument air and nitrogen is necessary to allow operation of pneumatically operated valves in containment which would allow the use of normal pressurizer spray valves to control pressurizer pressure, which results in 'B' being a correct answer. Answer 'C' directs stopping the coolant charging pumps (CCPs) running in injection mode, yet EOS-1.1 directs stopping all but one CCP. The NRC reviewed the SI starting sequence and found that all CCP would be running in injection mode, therefore 'C' would have an applicant stop all CCPs. Since EOS-1.1 directs stopping all but one CCP, and answer 'C' directs stopping all CCPs, 'C' cannot be a correct answer. The NRC has changed the answer key for the written exam to accept 'B' as the only correct answer to Question 6.

## QUESTION # 7

Complete the statement below regarding the PRT.

The PRT has   (1)   rupture disk(s) that will rupture at a MAXIMUM PRT pressure of approximately   (2)   psig above containment pressure.

- A. (1) One  
  (2) 106
- B. (1) One  
  (2) 91
- C. (1) Two  
  (2) 106
- D. (1) Two  
  (2) 91

Keyed correct answer is 'D.'

**COMMENT:** The licensee recommends removal from the exam. During written examination review of the CPNPP 2019 NRC exam it was identified that Question 7 has multiple flaws resulting in no correct answer. In particular:

### Reason 1

- Part 1 of the question states: "The PRT has   (1)   rupture disk(s)..."
- The selections are either "One" or "Two."
- Both selections "One" and "Two" are correct because the statement in the stem of the question is not bounded.
- The applicant does not know if the question is asking for the MAXIMUM number of installed Rupture Disks, therefore, either answer selection of "One" or "Two" is correct because the PRT has "One" rupture disk and it also has "Two" rupture disks.

### Reason 2

- Part 2 of the question states: "...rupture disk(s) that will rupture at a MAXIMUM PRT pressure of approximately   (2)   psig above containment pressure."
- The selections are either "106" or "91."
- Per DBD-ME-250 the rupture disk release pressure is NOMINALLY set to 91 psig with a range of 86-100 psig.
- The correct answer identified in the question is "91 psig."
- "91 psig" cannot be correct because this value is not the MAXIMUM value in which the rupture disks could rupture. 100 psig would be the MAXIMUM value at which the rupture disk could rupture.
- The alternate selection of "106 psig" also cannot be correct because this is beyond the range of 86-100 psig.

**NRC RESOLUTION:** The NRC disagrees with the licensee's recommendation remove Question 7 from the exam, and answer 'D' remains the only correct answer. As to the

licensee's explanation for Reason 1, it is unnecessary to bound the statement. The question clearly asks how many rupture disks the pressurized relief tank (PRT) has, and the PRT has two rupture disks, not one.

As to the licensee's explanation for Reason 2, the word MAXIMUM is in the stem to prevent a subset issue with the two, Part 2 answers. The reference material clearly states, "The tank is constructed of austenitic stainless steel, and is protected against a discharge exceeding the design value of 100 psig by two rupture disks which discharge nominally at 91 psig into the reactor containment."

The licensee correctly points out that reference material discusses a range of 86-100 psig, thus answers 'A' and 'C' are clearly incorrect. Though the licensee argues that 91 psig is not a "maximum" value of the rupture disk release pressure, the question asks for an approximate maximum. Since 91 psig is in the range of 86-100 psig and 106 psig is not, 91 psig is an approximate maximum.

The NRC does agree that the stem wording could be improved, however, there did not appear to be any confusion of the applicants concerning maximum PRT pressure while answering the question. No questions were asked during administration of the examination, and all thirteen applicants selected either answer 'B' or 'D.' Thus, all applicants knew the correct rupture disk pressure was 91 psig.

## QUESTION 61

In accordance with the Bases for the RCS Pressure Safety Limit, IF the \_\_\_\_\_ were to fail to operate as designed, THEN the RCS would still be protected from overpressurization by operation of the RCS High Pressure Trip.

- A. Steam Dump System
- B. Pressurizer Safety Valves
- C. Main Steam Safety Valves
- D. Pressurizer Pressure Control System

Keyed correct answer is 'A.'

**COMMENT:** The licensee recommends deletion of Question 61 from the exam. During the written examination review it was identified that Question 61 was missed by all candidates and has multiple correct answers. Answers 'A, B and "D' are correct as identified below:

- The question asks which of the answers, if it failed to operate as designed, would still allow the RCS High Pressure trip to protect the RCS from a High RCS pressure condition, in accordance with the TS Bases.
- Answer 'A' – "Steam Dump System" is identified as the correct answer on the key.
- Answer 'A' is listed in TSB 2.1.2 as not being credited for operation, therefore, if it does not function, the RCS High Pressure trip would still provide protection. This is a correct answer.
- Answer 'B' – The bases for TSB 2.1.2 states "The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization. The safety analyses for both the high-pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices."
- Answer 'B' – The RCS pressurizer safety valves and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded. Therefore, if the RCS Pressurizer Safety Valves failed to operate the RCS would still be protected from overpressurization by the RCS High Pressure Trip. This is a correct answer.
- Answer 'C' – "Main Steam Safety Valve" TSB 2.1.2 states that both MSSVs and RCS High Pressure Trip must operate to provide overpressure protection (TS bases 2.1.2 - The Reactor Trip System Allowable Values in Table 3.3.1-1, together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs), therefore, Answer 'C' cannot be a correct answer.
- Answer 'D' - "Pressurizer Pressure Control System" is not specifically listed in TSB 2.1.2 as not being credited for operation, however, the active components used to reduce RCS pressure, (PORV/s and Pressurizer spray valves) are specifically listed.
- PORV's and Pressurizer spray valves are the pressure control devices for reducing RCS Pressure. Their operation is not credited so RCS High Pressure trip would still provide protection per the analysis.
- This makes answer 'D' correct as well.
- Revision 0 of this question more specifically asked about analysis assumption for RCS High Pressure Trip. Rev 1 of this question was an attempt to solicit the same information but did not exclude the other answers as correct answers.

**NRC RESOLUTION:** The NRC agrees with the licensee's recommendation to remove Question 61 from the exam. The change from Revision 0 to Revision 1 of the question inadvertently changed the intent of the question and resulted in three correct answers. The NRC has changed the answer key for the written exam to delete Question 61 from the exam.

## QUESTION 67

A non-licensed operator is enrolled in the ILOT Program and has successfully completed the Systems portion of classroom training.

- Trainee is under instruction of the Unit 2 RO
- Trainee requests to perform a required RCS dilution for training

What is the LOWEST level of authority that can approve the trainee performing the dilution?

- A. Shift Operations Manager
- B. Reactor Operator
- C. Unit Supervisor
- D. Shift Manager

Keyed correct answer is 'D.'

**COMMENT:** The licensee recommends changing the correct answer to 'C.' During the written examination review it was identified that Question 67 has the wrong answer identified as correct. Answer 'C' is correct as identified below:

- The question asks what is the LOWEST level of authority that can approve the trainee performing a dilution.
- Since the question asks "what is the LOWEST level of authority" there can be only one correct answer.
- The question does NOT specifically ask "per ODA-102" what is the LOWEST authority. It specifically asks what is the LOWEST authority.
- CPNPP does NOT require the Unit Supervisors to gain permission from the Shift Manager for each reactivity manipulation for under instruction watches.
- The Shift Managers have the overall responsibility, however, per ODA-102 the Shift Managers delegate this authority to the Unit Supervisors to allow or deny under instruction watches performing reactivity manipulations
- This is accomplished in two ways. First, the Under Instruction watch attends the Shift Turnover (conducted per OPGD-3, Attachment 3a) where he/she presents information for the upcoming shift related to Tests / Briefing Times / Support Needed / Other Items of Interest. Second, the under instruction watch leads the beginning of shift brief (conducted per OPGD 3, Attachment 1) with the on-shift reactor operator, Unit Supervisor, and Shift Manager as participants in the brief.
- During the beginning of shift brief the under instruction watch will discuss all aspects of reactivity management including the expected number of dilutions/borations to be conducted during the shift as well as the size and their effect on reactivity.
- The Shift Manager is allowed to delegate this authority per ODA-102 as it states: "The Shift Manager's duties and responsibilities are to maintain overall responsibility for the Control Room command function; however, supervision and operation of the individual units will normally be delegated to the assigned unit's US."
- This is how the use and approval of under instruction watches performing reactivity manipulations is accomplished, in practice, at the plant.

- What the question is asking and how these activities are performed, in practice, in the plant are the same and the question correct answer should therefore be changed to 'C.'
- Answers 'A', 'B', and 'D' cannot be correct as the question asks for the LOWEST level of authority.

**NRC RESOLUTION:** The NRC disagrees with the licensee's recommendation to change the correct answer for Question 67 to 'C', and answer 'D' remains the only correct answer. Although the Shift Manager can delegate (and usually does) to the Unit Supervisor, the lowest level of authority that can approve this evolution as listed in the stem of the question with no assumptions of delegation is the Shift Manager and therefore there is only one correct answer, 'D.' Appendix E to NUREG 1021 states, "When answering a question, do 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." Appendix E also states, "Finally, answer all questions based on actual plant operation, procedures, and references." ODA 102 states, "Manipulation of controls by a non-licensed person (Trainee), is permissible only if that person is currently enrolled in the replacement license program as described in TRA-203, has successfully completed classroom instructions for the given evolution, is directly supervised by a Licensed Operator, and approval is granted by the SM [Shift Manager]." Delegation of authority is discussed in another section of ODA 102, but it is NOT automatic. It is normally granted, but not always. Since there is no indication in the stem that the Shift Manager has delegated authority to the Unit Supervisor, the Shift Manager remains the lowest authority to approve the dilution.



## QUESTION 76

Unit 1 plant conditions:

- Mode 6
- RHR Train A in service
- RHR Train B in standby
- RCS level starts to lower
- ABN-104, Residual Heat Removal System Malfunction, Section 5.0, Mode 5 or 6  
Unexpected Decrease in RCS Level – RCS Not Filled is entered
- CCP 1-01 has just been started

Based on the above plant conditions, complete the following statements:

1. If RCS level continues to lower, ABN-104 directs securing the TRAIN A RHR pump and closing (1).
2. If RCS level still continues to lower, ABN-104 directs (2).
  - A. (1) 1/1-8701A, RHRP 1 HL RECIRC ISOL VLV  
(2) re-starting Train A RHR in accordance with Attachment 7, RHR System Vent/Re-Start Instructions
  - B. (1) 1/1-8701A, RHRP 1 HL RECIRC ISOL VLV  
(2) starting Train B RHR in accordance with Attachment 18, Standby RHR Train Startup Instruction
  - C. (1) 1/1-8809A, RHR TO CL 1 & 2 INJ ISOL VLV  
(2) re-starting Train A RHR in accordance with Attachment 7, RHR System Vent/Re-Start Instructions
  - D. (1) 1/1-8809A, RHR TO CL 1 & 2 INJ ISOL VLV  
(2) starting Train B RHR in accordance with Attachment 18, Standby RHR Train Startup Instruction

Keyed correct answer is 'B.'

**COMMENT:** The licensee recommends accepting both answers 'A' and 'B' as correct answers. During the written examination review it was identified that Question 76 has two correct answers. Answers 'A' and 'B' are both correct as identified below:

- Part 2 of the question simply asks: "If RCS level still continues to lower, ABN-104 directs (2)."
- Answers 'A' and 'B' are both correct because both actions are directed by ABN-104 when RCS level continues to lower after shutting the 1/1-8701A valve.
- Answer 'B' is identified as the correct answer on the key and it is correct per ABN-104.
- Answer 'A' was identified during the exam review as a correct answer as well.
- Answer 'A' states: "re-starting Train A RHR in accordance with Attachment 7, RHR System Vent/Re-Start Instructions."
- Per ABN-104, Section 5.0, Step 7: "Verify at least ONE RHR Train – IN SERVICE" Per Part 1 of the question, Train A RHR was secured as directed by ABN-104 due to RCS level continuing to lower, therefore, the RNO of Step 7 must be applied.

- Step 7 RNO directs both starting the Standby RHR Train per Attachment 18 AND re-starting the affected train per Attachment 7 if the Standby Train will not start.
- The Stem of the question does not ask what is the priority or which step must be performed first, it simply asks which is directed by ABN-104 and both are directed by ABN-104.
- Part 1 of the question has only one correct answer thereby eliminating Answers 'C' and 'D' as possible correct answers.

**NRC RESOLUTION:** The NRC disagrees with the licensee's recommendation to accept both 'A' and 'B' as correct answers for Question 76, and answer 'B' remains the only correct answer. To go in order of the steps in the procedure is basic knowledge and in accordance with their procedural requirements. There are no statements in the stem that the standby train did not start and restarting the train with the leak would not be good unless you had no cooling at all, hence the step in the response not obtained (RNO) column of the procedure to restart train A if the standby train won't start. Appendix E to NUREG 1021 states, "When answering a question, do 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." In the order of items given and with no information given in the stem that train B fails to start and because the leak is most likely in train A, restarting it is not a correct action and 'A' would not be correct.

## QUESTION 88

Unit 1 plant conditions:

- 1-ALB-6D, Window 1.3 – 1 OF 4 HI SETPT PR FLUX HI, is in alarm
- All PR NIs are 100% and stable
- OVERPOWER TRIP HIGH RANGE light for N-42 is LIT on NIS Panel due to failure of High Setpoint High Flux Trip Bistable
- ABN-703, Power Range Instrument Malfunction, in progress to place channel out-of-service for repairs

Which of the following identifies...

(1) the impact on RPS?

(2) the required action to comply with Technical Specifications and ABN-703?

- A. (1) An OP HI FLUX ROD STOP C-2 is generated that can be bypassed.  
(2) Reactor Trip Bistables for Loop 2 must be placed in TRIP within one hour.
- B. (1) A Power Range High Flux Trip is generated that can be bypassed.  
(2) If Reactor to remain at 100% RTP, QPTR must be determined using Core Power Distribution Measurement information.
- C. (1) A Power Range High Flux Trip is generated that can be bypassed.  
(2) Reactor Trip Bistables for Loop 2 must be placed in TRIP within one hour.
- D. (1) An OP HI FLUX ROD STOP C-2 is generated that can be bypassed.  
(2) If Reactor to remain at 100% RTP, QPTR must be determined using Core Power Distribution Measurement information.

Keyed correct answer is 'D.'

**COMMENT:** The licensee recommends accepting both answers 'B' and 'D' as correct. During the written examination review it was identified that Question 88 has two correct answers. Answers 'B' and 'D' are both correct as identified below:

- Part 1 of the question asks the impact on RPS given that Power Range Channel N-42 has a failed High Setpoint High Flux Trip Bistable.
- Answer 'D' is identified as the correct answer on the key and it is correct.
- Answer 'B' was identified during the exam review as a correct answer as well.
- Answer 'B' states: "A Power Range High Flux Trip is generated that can be bypassed."
- Channel N-42 has a failed High Setpoint High Flux Trip Bistable and 1-ALB-6D, Window 1.3 – 1 OF 4 HI SETPT PR FLUX HI alarm indicates the bistable is failed High.
- When the bistable for PR Channel N-42 failed high (as described above) a Power Range High Flux Trip was generated for Channel N-42.
- The Power Range High Flux Trip bistable can be bypassed by placing 1-NS-N42D, OVERPOWER TRIP-HIGH switch in the BYPASS position at the rear of the NI Cabinet (see INC-7375A procedure below).
- Question 88 was an exam question used from the 2013 NRC Exam (Q86). The question was changed slightly for the 2019 exam. Originally Part 1 of answer 'B' read as follows: "A

Power Range High Flux Trip will be generated but can be blocked.” AND Part 1 of answer ‘C’ read as follows: “A Power Range High Flux Trip will be generated but cannot be blocked.”

- Part 1 of answers ‘B’ and ‘C’ were changed to read as follows: “A Power Range High Flux Trip is generated that can be bypassed.”
- Answers ‘A’ and ‘C’ are incorrect due to Part 2 of the answers as identified in the question plausibility statements.

**NRC RESOLUTION:** The NRC agrees with the licensee's recommendation to accept both ‘B’ and ‘D’ as correct answers for Question 88. The editorial changes from the 2013 bank question inadvertently changed Answer ‘B’ such that it is also a correct answer. The NRC has changed the answer key for the written exam to accept both ‘B’ and ‘D’ as correct answers to Question 88.

COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2 - NRC EXAMINATION  
REPORT 05000445/2019301; 05000446/2019301 – SEPTEMBER 3, 2019

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