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

September 12, 2013

Matthew W. Sunseri, President and
Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - NRC EXAMINATION REPORT 05000482/2013301

Dear Mr. Sunseri:

On July 26, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an initial operator license examination at Wolf Creek Generating Station. The enclosed report documents the examination results and licensing decisions. The preliminary examination results were discussed on July 25, 2013, with yourself and other members of your staff. A telephonic exit meeting was conducted on August 8, 2013, with Mr. James Knapp, who was provided the NRC licensing decisions.

The examination included the evaluation of six applicants for reactor operator licenses and one applicant for an instant senior reactor operator license. The license examiners determined that all seven applicants satisfied the requirements of 10 CFR Part 55 and the appropriate licenses have been issued. There were ten 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.

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). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Vince Gaddy, Chief
Operations Branch
Division of Reactor Safety

Docket: 50-482
License: NPF-42

Enclosure:
1. NRC Examination Report 05000482/2013301
   w/Attachment: Supplemental Information
2. NRC Post Examination Comment Resolution

cc: Electronic Distribution for Wolf Creek
Nuclear Generating Station

ADAMS ACCESSION NUMBER: **ML13255A296**

ADAMS: □ No ☒ Yes	☒ SUNSI Review Complete	Reviewer Initials: DGS
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	□ Non-publicly Available	□ Sensitive

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-482
License:	NPF-42
Report:	05000482/2013301
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, Kansas
Dates:	June 17, 2013 – August 8, 2013
Inspectors:	D. Strickland, Senior Operations Engineer K. Clayton, Senior Operations Engineer S. Hedger, Operations Engineer
Approved By:	Vince Gaddy, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER05000482/2013301; June 17, 2013 – August 8, 2013; Wolf Creek Generating Station; Initial Operator Licensing Examination Report; Examination Development

NRC examiners evaluated the competency of six applicants for reactor operator licenses and one applicant for an instant senior reactor operator license at Wolf Creek Generating Station.

The licensee developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1. The written examination was administered by the licensee on July 19, 2013. NRC examiners administered the operating tests on the week of July 22, 2013.

The examiners determined that all seven of the 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

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. The 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

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 that the written examination initially submitted by the licensee was not within the range of acceptability expected for a proposed examination. There were 38 questions categorized as unsatisfactory per NUREG-1021 requirements, with the two most common issues being knowledge/ability mismatch or two or more distracters that were not credible. Additionally, there were 36 questions that needed editorial changes, and 26 questions that were satisfactory as originally submitted. Consistent with the corporate notification letter sent to licensees at the start of examination development for each examination, future unsatisfactory examination submittals may cause the examinations to be rescheduled or cancelled. The licensee wrote Condition Report 00073742 to address these exam development issues.

.3 Operator Knowledge and Performance

a. Scope

On July 19, 2013, the licensee proctored the administration of the written examinations to all seven applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis and post examination comments to the NRC on August 2, 2013.

The NRC examination team administered the various portions of the operating tests to all ten applicants on the week of July 22, 2013.

b. Findings

No findings were identified.

All seven applicants passed all portions of the operating test. All of the applicants passed the written examination. The final written examinations and post-examination analysis and comments may be accessed in the ADAMS system under the accession numbers noted in the attachment. Post examination comments formally submitted by the licensee are included in the Enclosure 2 of this report.

The examination team noted a generic weakness during the administration of the Job Performance Measure associated with the ability to complete a Containment Purge Isolation. Three out of seven applicants initially closed two incorrect dampers, prior to closing the correct dampers. There was also a generic weakness noted in the completion of the Job Performance Measure associated with a task on the Radiation Monitoring Panel. Two of seven operators had problems with the operation of the RM11R panel. The licensee wrote Condition Report 00072089 to address the Radiation Monitoring Panel issue.

.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 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.

.6 Facility Post Examination Comments

The facility provided the NRC chief examiner with ten post administration comments on the written examination. These comments are included in Enclosure 2 to this report and in ADAMS under as Accession Number ML 13248A545.

4OA6 Meetings, Including Exit

The chief examiner presented the preliminary examination results to Mr. Al Camp, Plant Manager, and other members of the staff on July 25, 2013. A telephonic exit was conducted on August 8, 2013, between Mr. Duane Strickland, Chief Examiner, and Mr. James Knapp, Supervisor, Operations Training.

All proprietary information and materials used during the examination were returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

Al Camp, Plant Manager
Steve Henry, Operations Manager
Shane Battenfield, Operations Instructor
Kenneth Fredrickson, Licensing
James Knapp, Supervisor, Operations Training
Edward Ray, Manager Training
Terry Damashek, Supervisor Operations Support Training
Brendan Ryan, Initial Licensing Supervising Instructor
Robert Acree, Simulator Fidelity Coordinator
George Smith, Initial Licensing Supervising Instructor

NRC Personnel

Charles Peabody, Senior Resident Inspector

ADAMS DOCUMENTS REFERENCED

Accession No. ML 13248A540– FINAL WRITTEN EXAMS (Delayed release until July 26, 2015)
Accession No. ML 13248A545 – POST EXAM ANALYSIS
Accession No. ML 13248A547 – FINAL OPERATING TEST

NRC Review of WC Written Post-Examination Comments

Note: A complete text of the licensee's post examination analysis and comments can be found in ADAMS under Accession Number ML13248A545.

QUESTION 2

Given the following plant conditions:

- The Unit is operating at 68% power
- The crew is responding to a control bank D rod inadvertent withdrawal
- The control rods are placed in manual and the withdrawal stops
- The reactor did not trip
- Control rod M4 indicates 6 steps above the rest of the bank

Which ONE of the following sets of parameters describes the effect of this malfunction?

A. QPTR goes down, Delta T goes up.
B. QPTR goes down, Delta T stays the same.
C. QPTR goes up, Delta T goes up.
D. QPTR goes up, Delta T stays the same.

Key Answer: D

Licensee Comments for Question 2:

Q2 The stem does not specify a time frame, thus the Applicant is left with two options, both an initial effect and a long term or steady state effect of the stem conditions. Therefore the Applicants could interpret 'D' as correct based on the textbook theory (GFES) or 'C' as correct based on actual plant response.

Request: Accept two correct answers for Q2. It is acceptable per NUREG 1021; section ES-403 D.1.c to accept two correct answers.

Context:
Answer 'D' states that QPTR goes up, and Delta T remains the same. This was the correct answer based on a reactor core power tilt from the misaligned rod, and that turbine load is not being changed.
Distractor 'C' states that QPTR goes up and Delta T goes up. This answer is correct based on actual plant operation. When plant power is stable, we operate the turbine in open-loop control, which results in a fixed turbine control valve position. The positive reactivity addition results in an increase in RCS temperature. The increase in RCS temperature increases SG pressure. The increase in SG pressure results in a higher steam flow and higher power. There is some negative offset due to the smaller Cp at higher temperatures affecting delta-T, but this is nearly insignificant. This effect is demonstrated by actual plant behavior during dilutions to maintain reactor power. Although this is very small and not visible on analog indications, our control band at full power is 3 MWt (0.1%). Therefore the Applicants witnessed this plant response multiple times during their time on shift. The effect would be much larger depending on the actual amount of rod withdrawal prior to placing rods in manual.

NRC Resolution of Question 2:

WC recommended that response C be accepted for this question based on plant response for a small positive reactivity addition they had experienced in the plant. In addition, answer D was the expected correct answer based on classroom theory taught to the applicants. The non-specified time frame, initial response vs. a longer term response, lead the applicants to choose a different answer. The NRC Region IV staff agrees with this position and therefore both answer C and answer D are accepted for this question in accordance with ES-403 D.1.c.

QUESTION 9

Given the following plant conditions:

- A large LOCA has occurred
- RCS is at saturation with system pressure matching Containment pressure
- RHR train flows are: 'A' flow is 3350gpm, 'B' flow is 3150 gpm
- RWST is 8% and dropping slowly

What is the expected configuration for the Residual Heat Removal Pumps under these conditions AND what parameter is used to determine subsequent RHR pump configuration?

A. Operating at rated flow of both pumps.
RCS Pressure is > 325 psig

B. Operating at rated flow of both pumps.
RCS Hot Leg Temperature > 360F and at least One RHR Pump Running

C. Operating at near run out conditions.
RCS Pressure is > 325 psig

D. Operating at near run out conditions.
RCS Hot Leg Temperature > 360F and at least One RHR Pump Running

Key Answer: A

Licensee Comments for Question 9:

Q9 The question should be deleted because it contains information in the stem that was unclear and confused the Applicants, which is listed as a reason to accept post-exam changes per ES-403 D.1.b.

Request: Delete Q9 from the exam. It is acceptable per NUREG 1021; section, ES-403 D.1.b to delete a question with an unclear stem that confused the Applicant or did not provide all the necessary information.

Context:
Stem: Post-exam feedback indicated that the Applicants thought RHR flows were low based on plant conditions, expecting a higher flow rate. They also commented that "near run out" is a vague term. Several expressed confusion since RCS pressure was already less than 325 psig and incapable of repressurization. Additionally, the Applicants were unsure as to plant lineup

status due to the “RWST dropping” slowly. This could be due to a leak on the tank, ECCS alignment, check valve problem, or containment spray alignment issue.

NRC Resolution of Question 9:

At WC, actual RHR flows in this configuration is not indicated on the RHR pump flow meters. Additional RHR flow from operating the SIP’s and CCP’s in the piggy-back mode must be added to the RHR pump flow meters to obtain actual flow. The RHR pump flows given in the stem was not realistic. In addition, due to the Large Break LOCA in progress, there was no expectation of RCS pressure increasing to >325 psig. Based on multiple confusing conditions given in the stem, the NRC Region IV staff agrees with this position and therefore this question was deleted from the written examination in accordance with ES-403 D.1.b.

QUESTION 10

Given the following plant conditions:

- PRT conditions are normal for 100% power
- The Unit was operating at 100% power when the turbine tripped
- The reactor failed to automatically trip but was manually tripped
- All other systems operated as expected
- The Emergency procedures have been performed and the plant stabilized
- It was noted that on the transient RCS pressure reached 2370 psig

Which ONE of the following represents the expected status of the PRT AND the actions that must be taken to restore it to normal limits?

A. PRT Pressure - 40 psig - High
   Vent excess pressure to the Gas Decay tanks.

B. PRT Pressure - 14 psig - High
   Recirc through the RCDT Hx and reduce pressure.

C. PRT Temperature - 220°F - High
   Recirc through the RCDT Hx and vent excess pressure to the Gas Decay tanks.

D. PRT Temperature - 220°F - High
   Add water to cool the tank and drain to the RCDT to maintain level.

Key Answer: A

Licensee Comments for Question 10:

Q10 The stem lacked some operational validity in that the PRT high temp would have been reached and the stem lacked info on the duration that the PORV was open. Because the stem did not provide all the necessary info, per NUREG-1021 Section ES-403 D.1.b, it should be deleted from the examination.

Request: Delete Q10 from the exam. It is acceptable per NUREG 1021; Section ES-403 D.1.b to delete a question with an unclear stem that confused the Applicant or did not provide all the necessary information.

Context:
Stem: Post-examination feedback indicated that the Applicants could not discern the amount of time that the PORV was open from the stem conditions and could not determine the amount of mass and energy added to the PRT. They expected the PRT to be in alarm for both pressure and temperature for any significant PORV operation. They stated that there was not enough information to answer the question.
The amount of PORV cycling is dependent on the timing of the manual reactor trip relative to the turbine trip. All that can be discerned from the stem is that the PORV cycled.

NRC Resolution of Question 10:

The stem of the question did not provide operational validity. Actual expected plant conditions should be given, not expect the applicants to determine plant conditions based on incomplete information. Based on incomplete and confusing conditions given in the stem, the NRC Region IV staff agrees with this position and, therefore, this question was deleted from the written examination in accordance with ES-403 D.1.b.

QUESTION 22

Given the following plant conditions:

- A Large Break LOCA has occurred
- All RCP's are stopped
- ECCS and ESF systems functioned as designed
- RCS pressure equals Containment pressure
- The operating crew is ready to transition out of EMG E-0, REACTOR TRIP OR SAFETY INJECTION

RCP #1 Seal Leakoff is . . .

A. NOT OCCURRING.

B. OCCURRING and flowing to the Pressurizer Relief Tank.

C. OCCURRING and flowing to the Reactor Coolant Drain Tank.

D. OCCURRING and flowing to the Volume Control Tank.

Key Answer: B

Licensee Comments for Question 22:

Q22 It is proposed to delete the question due to a stem that did not provide all the necessary information. If seal flows are normal, then distractor 'A' is correct. If seal flow is excessive, then answer 'B' is correct. Both 'C' & 'D' are incorrect under all conditions. Because answers 'A' & 'B' are mutually exclusive, deletion is recommended rather than accepting two answers.

Request: Delete Q22 from the exam. It is acceptable per NUREG 1021; section ES-403 D.1.b to delete a question with an unclear stem that confused the Applicant or did not provide all the necessary information. It is acceptable per NUREG 1021; section ES-403 D.1.c to delete a question with two answers that are mutually exclusive.

Context:
Answer 'B' is that seal return is occurring and flowing to the PRT. This is normally the case following CIS-A with the RCS pressurized. However, the stem indicates that a large break LOCA is in progress with RCS equal to Containment pressure. The relief valve to the PRT, BG-8121, lifts at 150 psig, which is unlikely unless seal injection flow is excessive.

Distractor 'A' is that seal return is not occurring. Based on plant conditions in the stem, #1 seal leakoff is unlikely to be occurring. Since the stem states that RCS and Containment pressure are matched, there is insufficient pressure to lift the relief. Seal injection continues, but the flow is diverted to the low pressure RCS, especially since the #1 seal faces are likely closed. The only way to provide enough backpressure to lift the relief is for seal injection flow to be so large that the thermal barrier labyrinth seals to provide the backpressure.

NRC Resolution of Question 22:

The amount of seal injection flow following a Large Break LOCA determines whether or not seal leakoff is occurring. This information was not given in the stem. The possible correct answers are mutually exclusive, based on the assumption the applicant uses. Based on not giving the value for seal injection in the stem, the NRC Region IV staff agrees with this position and, therefore, this question was deleted from the written examination in accordance with ES-403 D.1.c.

QUESTION 33

Why is it worse for an Anticipated Transient Without Scram event to occur at low power at the Beginning Of Life (BOL) conditions?

A. The additional burnable poisons provide LESS heat conduction at lower power; therefore, the fuel pin outer clad temperatures are HIGHER.

B. The Power Coefficient is LESS NEGATIVE at lower power; therefore, it provides MINIMUM negative feedback.

C. The higher SG pressures at lower power remove heat from the RCS at a LOWER RATE; therefore, RCS temperatures RISE to a higher level.

D. The high boron concentration causes the emergency boration to be LESS EFFECTIVE; therefore, taking a LONGER time to achieve adequate Shutdown Margin (SDM).

Key Answer: B

Licensee Comments for Question 33:

Q33 The stem is unclear. The stem doesn't ask for the "principal" reason why an ATWS is worse at low power at BOL vs. EOL, thus answer 'D' could also be argued as correct.

Request: Accept two correct answers for Q33. It is acceptable per NUREG 1021; Section ES-403 D.1.c to accept two correct answers.

Context:
Answer 'B' is correct due to the small value of the power coefficient. This is predominantly due to the moderator temperature coefficient (MTC) being only slightly negative or slightly positive under BOL conditions. Therefore, the RCS heatup following turbine trip will not significantly reduce reactor power.

Distractor 'D' is that the higher boron concentration makes emergency boration less effective. Although this is not the principal success path in EMG FR-S1, RESPONSE TO NUCLEAR PWR GENERATION/ATWS, it is a secondary action to add negative reactivity that is always performed.

NRC Resolution of Question 33:

There are multiple reasons why an ATWS is worse at low power BOL. The stem did not ask for the major reason, leaving more than 1 correct reason. NRC Region IV staff agrees that both B and D responses are correct and therefore both answer B and answer D are accepted for this question in accordance with ES-403 D.1.c.

QUESTION 38

Given the following plant conditions:

- The Unit has experienced a steamline break
- None of the main steam isolation valves can be closed
- EMG C-21, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, has been implemented

Which ONE of the following describes the reason for performing Attachment B, Main Steam Header Isolation - Local?

A. Isolation of both steam supply lines to the TDAFP is allowed, regardless of the status of the other sources of feed flow to the SGs, since no secondary heat sink is intact.

B. To restore integrity to at least one SG in order to transition to EMG E-2, FAULTED STEAM GENERATOR ISOLATION, via the foldout page.

C. Valves are closed in any order to restore integrity to all SGs as early as possible.

D. The Operator is allowed to place the Condenser Steam Dump valves to CLOSE only if the selected valve is NOT accessible for local isolation.

Key Answer: B

Licensee Comments for Question 38:

Q38 It is proposed to delete this question due to an unclear stem. EMG C-21 step 1 attempts to isolate one Steam Generator. Attachment B is entered in response to the inability to isolate one Steam Generator and directs the Operator to isolate the Main Steam Header.

Request: Delete Q38 from the exam. It is acceptable per NUREG 1021; section ES-403 D.1.b to delete a question with an unclear stem that confused the Applicant or did not provide all the necessary information.

Context:
Stem asks for an answer that is not supported by the background document. There is no information in the background document of EMG C-21 to support the answer provided. Performance of Attachment B will not result in one or more intact steam generators.

Other: TNA 2013-1345-1, Condition Report 00072233 and Procedure Change Request 58807 were written.

NRC Resolution of Question 38:

This question was written with the expectation that an attempt would be made in Attachment B to locally close the MSIV's. Due to the replacement of WC MSIV's with a different model valve, there is no procedural step or method to locally close the MSIV's. Based on there being no clear answer, the NRC Region IV staff agrees with this position and therefore this question was deleted from the written examination.

QUESTION 65

In addition to a Record Dose Dosimeter (RDD), which ONE of the following describes a MINIMUM acceptable monitoring method for personnel entering a 'Locked High Radiation Area'?

A. A radiation monitoring device that continuously monitors the radiation dose rate in the area AND alarms when the dose rate alarm setpoint is reached.

B. Radiation monitoring device that continuously transmits dose rate AND cumulative dose to a remote receiver.

C. Self-reading dosimeter AND continuous surveillance by radiation protection, via a closed circuit television with a means of communication with each person in the area.

D. Self-reading dosimeter AND an operable area radiation monitor for the general area.

Key Answer: C

Licensee Comments for Question 65:

Q65 Distractor 'B' could also be argued as correct based on an unclear stem not based on actual plant practices.

Request: Accept two correct answers for Q65. It is acceptable per NUREG 1021; Section ES-403 D.1.c to accept two correct answers.

Context:
It is proposed that distractor ‘B’ should be accepted in addition to answer ‘C’ based on a question with an unclear stem that confused the Applicants based on actual plant practices.

Answer ‘C’ is correct in that a self-reading dosimeter and continuous surveillance by RP with a means of communication.

The justification for distractor ‘B’ being incorrect was that it does not contain the reference to the device being monitored by Health Physics nor communications. However these conditions are implied since the only remote monitoring system at Wolf Creek is Teledose, which reads out at the RADs camera console at Access Control, with various options for local communication (radio, Gaitronics, etc.). In order to obtain Teledose dosimetry, the RADs console operator must check it out to the individual. The RADs console operator will not let the individual leave until it is verified that the device is communicating with the console and the means of field communication are established with the individual (radio, Gaitronics, or direct coverage by the HP lead in the field.) By simply obtaining the dosimetry, the field operator has met the requirements since the other requirements are part of that process.

NRC Resolution of Question 65:

Actual plant practices for entry into a ‘Locked High Radiation Area’ are similar to response B, while the proposed answer C is described in procedure AP 25A-200. Due to the confusion to the applicants based on actual practice compared to procedural description, NRC Region IV staff agrees that both B and C responses are correct and therefore both answer B and answer C are accepted for this question in accordance with ES-403 D.1.c.