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

January 11, 2018

Mr. Adam C. Heflin, President,
Chief Executive Officer,
and Chief Nuclear Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - NRC EXAMINATION REPORT 05000482/2017301

Dear Mr. Heflin:

On November 15, 2017, 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 November 9, 2017, with Mr. C. Reasoner, Site Vice President, and other members of your staff. A telephonic exit meeting was conducted on December 19, 2017, with Mr. R. Meyer, Supervisor, Simulator and Examination Group, who was provided the NRC licensing decisions.

The examination included the evaluation of five applicants for reactor operator licenses, five applicants for an instant senior reactor operator license, and one applicant for an upgrade senior reactor operator license. The license examiners determined that all applicants satisfied the requirements of 10 CFR Part 55 and the appropriate licenses have been issued. There were two post examination comments submitted by your staff. Enclosure 1 contains details of this report, Enclosure 2 documents one observed simulator fidelity issue, and Enclosure 3 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 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 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/

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

Docket No. 50-482
License No. NPF-42

Enclosures:
1. Examination Report 05000482/2017301
   w/Attachment: Supplemental Information
2. Simulator Fidelity Report
3. NRC Resolution to the Wolf Creek Post
   Examination Comments

cc: Electronic Distribution

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000482
License:	NPF-42
Report:	05000482/2017301
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, Kansas
Dates:	November 6 through December 19, 2017
Inspectors:	C. Osterholtz, Senior Operations Engineer K. Clayton, Senior Operations Engineer T. Farina, Operations Engineer M. Hayes, Operations Engineer S. Hedger, Emergency Preparedness Inspector
Approved By:	Vincent G. Gaddy Chief, Operations Branch Division of Reactor Safety

Enclosure 1

SUMMARY

ER 05000482/2017301; 11/6/2017 – 12/19/2017; Wolf Creek Generating Station; Initial Operator Licensing Examination Report.

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

The NRC developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 11. The written examination was administered by the licensee on November 15, 2017. The NRC examiners administered the operating tests November 6-10, 2017.

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

A. <u>NRC-Identified and Self-Revealing Findings</u>

None

B. <u>Licensee-Identified Violations</u>

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 two 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 developed the written examination and operating tests in accordance with the requirements of NUREG-1021. The NRC examination team conducted an on-site validation of the operating test.

b. Findings

No findings were identified.

c. Other Observations

The examiners noted the following observations that were placed in the licensee's corrective action program for evaluation and resolution:

- The licensee currently administers job performance measures in the simulator with the annunciator audio turned off. This may not be the best practice, as it does not emulate the actual control room, which is desirable. Also, this practice limits the ability to use emergent annunciators to alert the operator to abnormal conditions, hindering the use of some alternate path job performance measures as training and evaluation tools. This issue was entered in the licensee's corrective action program as Condition Report 00117336.

- During examination validation, the licensee was inconsistent in its operability determination for a pressurizer power operated relief valve that failed open in automatic, but was able to be closed manually. This issue was entered in the licensee's corrective action program as Condition Report 00117333.

- During examination validation, the licensee was inconsistent in its expectation for procedural response to a ruptured steam generator concurrent with a single failed-open main steam isolation valve. This issue was entered in the licensee's corrective action program as Condition Report 00117334.

- Procedure SYS SP-121, "Operation of the G.A. Monitor System," needs enhancement to be able to be used to properly change setpoints for all of the equipment associated with it. This issue was entered in the licensee's corrective action program as Condition Report 00117335.

- Initial licensed operator training may need to be enhanced to more adequately address clearance order preparation and review. This issue was entered in the licensee's corrective action program as Condition Report 00117337.

.3 Operator Knowledge and Performance

a. Scope

On November 15, 2017, the licensee proctored the administration of the written examinations to all 11 applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis to the NRC on November 27, 2017.

The NRC examination team administered the various portions of the operating tests to all 11 applicants on November 6-10, 2017.

b. Findings

No findings were identified.

All 11 applicants passed all portions of the written examination operating test. The final written examinations and post-examination analysis may be accessed in the ADAMS system under the accession numbers noted in the attachment. There were two post-examination comments submitted by the licensee. The licensee requested and received approval by the NRC to withhold the written examinations from the public document room for 24 months after the administration date.

.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 the performance of one simulator scenario, NRC examiners observed a possible simulator modeling issue. While performing actions in response to a digital rod position indication (DRPI) channel failure, three out of the four crews examined selected “half accuracy” on DRPI, which inadvertently cleared the fault. This should not have happened had the failure occurred in the actual control room. This deficiency was not identified during scenario validation, as none of the validating crews opted to select “half accuracy” on DRPI. See Enclosure 2 for additional details.

.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.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The chief examiner presented the preliminary examination results to Mr. C. Reasoner, Site Vice President, and other members of his staff on November 9, 2017. A telephonic exit meeting was conducted on December 19, 2017, between Mr. C. Osterholtz, Chief Examiner, and Mr. R. Meyer, Supervisor, Simulator and Examination Group.

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

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

C. Reasoner, Site Vice President
D. Berry, Simulator Fidelity Coordinator
M. Blow, Shift Manager, Operations
J. Bouson, Operations Training
W. Brandt, Examination Licensing Representative
S. Duffy, Operations Training
J. Edwards, Manager, Operations
T. Farrow, Operations Training
J. Hudson, Operations Training
W. Isom, Licensed Supervisor Instructor, Initial
J. Knapp, Superintendent, Operations Training
P. Moore, Superintendent, Operations
W. Muilenburg, Supervisor, Licensing
E. Ray, STARs Representative
A. Servaes, Examination Team
R. Sims, Shift Manager, Operations
L. Stone, Licensing
W. Stucker, Superintendent, Operations

NRC Personnel

D. Dodson, Senior Resident Inspector
F. Thomas, Resident Inspector

ADAMS DOCUMENTS REFERENCED

Accession No. ML17355A650 – WC-2017-11-Final Written Exams (Do not release for 2 years)

Accession No. ML17355A648 – WC-2017-11-Final Operating Test (Do not release for 2 years)

Accession No. ML17355A651 – WC-2017-11-Post Exam Analysis (Do not release for 2 years)

Facility Licensee: __Wolf Creek Generating Station__

Facility Docket No.: __50-482__

Operating Tests Administered on: __November 6-10, 2017__

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and, without further verification and review in accordance with IP 71111.11, are not indicative of noncompliance with 10 CFR 55.46, "Simulation Facilities." No licensee action is required in response to these observations.

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

Item	**Description**
Scenario 4, Event 2: Dual Data Failure on one Shutdown Bank Rod (TS)	Three of four crews selected "half accuracy" on digital rod position indication (DRPI) in response to a DRPI failure, which inadvertently cleared the fault due to an apparent simulator modeling error. This made a successful DRPI surveillance test possible [+/- 12 inches when one channel is accurate to +10/-4 inches]. Therefore, this event had no technical specifications associated with it when off-normal procedure actions were taken coupled with the simulator modeling error. Technical specifications would have been applicable under the same circumstances in the actual control room. This simulator modeling deficiency was entered in the licensee's corrective action program as Condition Report 00117338.

NRC Resolution to the Wolf Creek Post Examination Comments

A complete text of the licensee's post examination analysis and comments can be found in ADAMS under Accession Number ML17355A651.

A Region IV examiner was assigned on November 27, 2017, to resolve three post examination comments submitted by Wolf Creek. The examiner was independent of the examination team. The following recommendations were submitted for branch chief review, and accepted by the branch chief on December 19, 2017:

Original Question

Question 67

According to GEN 00-009, REFUELING, when moving control rods in containment, the crew should verify refueling pool water level is at a MINIMUM of at least 23 feet ___1)___.

While monitoring the Main Control Board Wide Range Loop Level indicators, the channels should indicate within a MINIMUM of___2)___ of each other.

A. 1) Over the top of the reactor vessel flange
   2) 10 inches

B. 1) Over the top of the reactor vessel flange
   2) 3 inch

C. 1) Above the top of the fuel assemblies in the reactor vessel
   2) 10 inches

D. 1) Above the top of the fuel assemblies in the reactor vessel
   2) 3 inch

Answer: A

Explanation: GEN 00-009, Revision 39, in multiple locations states refueling pool water level must be at least 23 feet above the top of the Reactor Vessel flange Precaution and Limitation 4.7.2 specifies NR Loop Level channels should indicate within 3 inch of each other if using NPIS and 10 inches of each other if using Main Control Board indicators.

A. is correct. See explanation.

B. 1) is correct. 2) is wrong, but plausible in that 3 inches is required range if using NPIS.

C. 1) is wrong because refueling pool water level must be a minimum of at least 23 feet above the top of the flange when moving control rods. Plausible because GEN 00-009, paragraph G.6 specifies refueling pool water level must be a minimum of at least 23 feet over the top of the fuel assemblies in the reactor vessel for movement of control rods in the reactor vessel. 2) is correct.

D. 1) is wrong because refueling pool water level must be a minimum of at least 23 feet above the top of the flange when moving control rods. Plausible because GEN 00-009, paragraph G.6 specifies refueling pool water level must be a minimum of at least 23 feet over the top of the fuel assemblies in the reactor vessel for movement of control rods in the reactor vessel. 2) is wrong, but plausible in that 3 inches is required range is using NPIS.

Technical References:

GEN 00-009, REFUELING, Revision 39,

New information that supports question re-grade request:

For the question asked, movement of Control Rods requires at least 23 feet above the fuel assemblies in the reactor vessel and the MCB WR Loop Level Indicators must agree within 10 inches per GEN 00-009, REFUELING, Step G.6.1 and Step 4.7.2. This set of conditions corresponds with answer choice C while answer choice A was selected as the proposed correct answer.

G.6 Within 2 hours prior to movement of control rods or irradiated fuel, ENSURE refueling pool water level is greater than required. N/A unused step

G.6.1 At least 23 feet above the top of the fuel assemblies in the Reactor Vessel for movement of control rods in the Reactor Pressure Vessel as indicated on BBLI53A or BBLI54A greater than or equal to 239.8 inches (greater than or equal to 2033' 4" on tygon hose). ☐

G.6.2 At least 23 feet over the top of the Reactor Vessel Flange for movement of irradiated fuel assemblies within the containment as indicated on BB LI-462 greater than or equal to 376 inches (greater than or equal to 2044' 7½" on tygon hose). ☐

4.7 The following apply to loop level indication:

4.7.1 The following computer points may be monitored in place of level channels on RL002 or RL018, if NPIS is available.

* BBL0053A	*BBU0001
* BBL0053B	*BBU0002
* BBL0054A	*BBU0003
* BBL0054B	

4.7.2 Wide Range loop level channels should indicate within 3 inches of each other if using NPIS and 10 inches of each other if using Main Control Board Indicators. (3.1.4.1)

Recommendation:

Recommend changing the correct answer to C from A. It is expected that water level in the refueling pool be at least 23 feet above the top of the fuel assemblies during movement of control rods and the Main Control Board Wide Range loop level indicators should indicate within 10 inches of each other per GEN 00-009 requirements therefore answer choice C is correct. Wolf Creek wrote Condition Report 00117618 to document the exam writing error, which resulted in required regrade evaluation of the question by the NRC.

NRC RESPONSE

Question #67. The licensee requests changing the correct answer from A to C. I agree with the licensee that the correct answer is C.

This question is of a 2 x 2 format, and the first part is the only part that is in contention. The first part has to deal with the level of water in the refueling water pool when moving control rods in containment. The keyed correct answer is 23 feet over the top of the reactor vessel flange, and the licensee contends that the correct answer should be 23 feet above the top of fuel assemblies in the reactor vessel.

As reference, the licensee highlights GEN 00-009, REFUELING, Step G.6.1. This step states that the refueling pool water level should be "at least 23 feet above the top of the fuel assemblies in the Reactor Vessel for movement of control rods in the Reactor Pressure Vessel." However, Step G.6.2 states that the refueling pool water level should be "at least 23 feet over the top of the Reactor Vessel Flange for movement of irradiated fuel assemblies within the containment." Additionally, Step G.6 directs to "N/A unused step."

The main difference between the two steps is that Step G.6.1 deals with moving control rods, and Step G.6.2 deals with moving irradiated fuel assemblies.

The question stem clearly asks "According to GEN 00-009, REFUELING, when **moving control** rods in containment, the crew should verify refueling pool water level is at a MINIMUM … " thus Step G.6.1 applies, and C is the correct answer.

The correct answer will be changed for this question from A to C.

Original Question

Question 84

Per Technical Specification Bases for LCO 3.6.3 for Containment Isolation Valves, the single failure criterion imposed during the plant safety analysis for a loss of containment integrity is:

A. RCP seal injection valves

B. 36-inch shutdown purge valves

C. 18-inch purge isolation valves

D. Category 1 containment Isolation valves

Answer: C

Explanation:

A is wrong because...these are left open per a note in the LCO's.

B is wrong because these valves must be closed with a blind flange before entry to mode 4. They are so large that they are assumed to be unable to be shut against DBA pressure.

C is correct because this is the single failure criterion of this TS and these valves per the TS bases. They are opened intermittently at power for various reasons so the inboard and outboard isolation valves have separate power supplies, etc.

D is wrong because this is a group of valves listed in the table that have the shortest LCO completion time due to risk and other factors but these are not the single failure criterion for TS 3.6.3.

Technical References:

TS Bases Revision 75, page 3.6.3-3.

New Information that supports removal of question from the exam:

The question was written to test the applicants' knowledge of LCO 3.6.3 Bases. However, as noted in the Tech Spec Bases Background, single failure criterion applies to ALL containment Isolation valves.

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetration flow path not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, deactivated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration flow path so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

The applicants were therefore confused with regard to what the question was asking since single failure criterion is imposed on the design consideration for all four answer choices. One clarification request form was submitted for this question during the examination, but no clarification was provided. The specific discussion in LCO 3.6.3 background document about the single failure as it applies to the 18 inch containment mini-purge valves is nothing more than an explanation on why these valves are different from the others, and NOT that they are the ONLY set of containment isolation valves in which single failure criterion was imposed.

Licensing input was requested and obtained with the following results.

10 CFR 50 App. A General Design Criteria 55 and 56 provide the requirements for containment isolation valves:

GDC 55. Reactor coolant pressure boundary penetrating containment-

and

GDC 56. Primary Containment Isolation-

> ***Each line*** (that meets GDC 55 or 56) ***shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:***
>
> ***(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or***
>
> ***(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or***
>
> ***(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment***
>
> ***(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.***
>
> ***Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.***

Answer Choice A (RCP Seal Injection Valves) is correct. These valves meet single failure design criteria, which was modified NRC Letter dated November 3, 2010 transmitted License Amendment 190 and the accompanying Safety Evaluation to add a note allowing the RCP seal injection valves to be considered OPERABLE with the valves open and power removed.

Answer Choice B (36-inch purge valves) is correct. These valves meet single failure design criteria, per **Standard Review Plan (SRP) 6.2.4** (Rev 2, July 1981) Section II. Acceptance Criteria, 6 provides a number of acceptable alternatives for containment isolation, including:

f. ***Sealed closed barriers may be used in place of automatic isolation valves. Sealed closed barriers include blind flanges and sealed closed isolation valves which may be closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident. Sealed closed isolation valves should be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator.***

Answer Choice D (Category I Containment Isolation valves) is correct as listed in Table B 3.6.3-1 as they meet single design failure criteria per SRP 6.2.4 Section II. Acceptance Criteria, 6

e. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve will be acceptable if it can be shown that the system reliability is greater with only one isolation valve in the line, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containment

should be protected from missiles, designed to seismic Category I standards, classified Safety Class 2 (Ref. 9), and should have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak tested, unless it can be shown that the system integrity is being maintained during normal plant operations. For this type of isolation valve arrangement the valve is located outside containment, and the piping between the containment and the valve should be closed in a leak tight or controlled leakage housing. If, in lieu of a housing, conservative design of the piping and valve is assumed to preclude a breach of piping integrity, the design should conform to the requirements of SRP Section 3.6.2. Design of the valve and/or the piping compartment should provide the capability to detect leakage from the valve shaft and/or bonnet seals and terminate the leakage.

Recommendation,

Recommend removal of this question from the exam. Since Technical Specification Bases for LCO 3.6.3, Containment Isolation Valves specifies ALL containment isolation valves are designed with single failure design criteria considerations, then all four answer choices are correct and the question is invalid. Wolf Creek CR117619 was written to document the exam writing error, which caused the NRC to evaluate the need or remove the question from the exam.

NRC RESPONSE

Question #84. The licensee requests deleting this question from the examination because it has four correct answers. I agree with the licensee and believe the question should be deleted from the examination.

The questions asks what is the single failure criterion imposed during the plant safety analysis for a loss of containment integrity. The question is clearly attempting to the test knowledge of a statement in the Technical Specification LCO 3.6.3 Bases that states, “The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the 18 inch containment mini-purge valves.” My interpretation of this statement is simply a reiteration that the mini-purge valves had the single failure criterion imposed, not that they were the only valves that had the single failure criterion imposed. As the licensee pointed out, single failure criterion applies to ALL containment Isolation valves.

This question shall be deleted from the examination.

Original Question

Question 99

While performing EMG ES-03, SI TERMINATION the crew observes alarms are present on Fire Alarm Control Panel KC008. A few minutes later, an NSO calls the control room and reports there is a fire in MDAFW Pump Room B.

The CRS…

A. Is required to concurrently implement both EMG ES-03 and OFN KC-016, FIRE RESPONSE

B. Is required to complete the actions in EMG ES-03 and then transition to OFN KC-016, FIRE RESPONSE

C. Is allowed by procedure to implement OFN KC-016, FIRE RESPONSE as long as it does not interfere with performance of EMG ES-03

D. Is allowed by procedure to implement the actions in EMG ES-03 as long as they do not interfere with the actions of OFN KC-016, FIRE RESPONSE

Answer: C

Explanation: AP 15C-003, Section 6.2.3, says, "While performing EMGs, plant conditions may indicate the need to correct problems not directly related to the event mitigation strategy. The operator may perform OFNs and ALRs which address these problems as long as the actions do not interfere with performance of the EMGs." Therefore, the SRO may concurrently perform the Fire OFN as long as it does not interfere with performance of the EMGs.

A is wrong because AP 15C-003 says the SRO <u>may</u> perform OFN and EMG actions concurrently; not that the SRO is required to do so. Plausible because if the fire is not extinguished within 15 minutes, the crew will need to declare an EAL. Therefore, extinguishing the fire as soon as possible will be a priority for the crew; however, it is not required by AP 15C-003 to concurrently perform the OFN with the EMG, even where there is a fire.

B is wrong because AP 15C-003 allows the SRO to perform the OFN concurrently as long as it does not interfere with performance of the EMG. There is no requirement to wait to perform the OFN after performing the EMG actions. Plausible because performance of actions in EMGs are a higher priority than performance of actions directed by an OFN.

C is correct. See explanation.

D is wrong because the EMGs are a higher priority than OFNs. Plausible since concurrent implementation is allowed, and a candidate may think that addressing fire response is a higher priority because a fire has the potential to injure plant personnel, to spread and cause additional damage, which could complicate recovery with the EMGs, and to require entry into the Emergency Plan.

Technical References:

AP 15C-003, Revision 34, Page 19/60
Lesson Plan LO1733203, Revision 013, Page 16/33

New Information that supports question regrade:

Input was requested and obtained from Licensing, Fire Protection Engineering, and Operations Management with the following results:

10 CFR Part 50, Appendix R, states that (III. K.) administrative controls shall be established to minimize fire hazards in areas containing structures, systems, and components important to safety. These controls shall establish procedures to:

> ***10. Control actions to be taken by the control room operator to determine the need for brigade assistance upon report of a fire or receipt of alarm on control room annunciator panel, for example, announcing location of fire over PA system, sounding fire alarms, and notifying the shift supervisor and the fire brigade leader of the type, size, and location of the fire.***

WCNOC's Response to Appendix R III. K. Administrative Controls states (USAR App 9.5E):

> ***Administrative procedures define limitations to minimize fire Hazards in areas containing SSCs important to safety. Administrative procedures are also provided to promote prompt, appropriate action upon discovery of a fire.***

Section 2.C.(5)(a) of the Facility Operating License requires Wolf Creek to maintain in effect all provisions of the approved Fire Protection Program.

The Wolf Creek Fire Protection Program is based on the following defense in depth approach, which is a part of our license basis. (Reference USAR Section, 9.5.1.7.5; AP 10-100, Section 4.3; and Regulatory Guide 1.189, Section B):

- Prevent fires from starting.
- Detect rapidly, control, and extinguish promptly those fires that do occur.
- Protect SSCs important to safety, so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

For the given test question scenario, with the crew performing EMG ES-03, SI TERMINATION, a design basis accident has NOT occurred, which is an assumptions associated with the post-fire safe shutdown (PFSSD) analysis per XX-E-013, POST-FIRE SAFE SHUTDOWN ANALYSIS, Revision 4, Section 3-A-3, and NRC Generic Letter 86-10, Response to Question 7.2.

The analysis documented in E-1F9910 demonstrates that PFSSD can be achieved and maintained following a single fire located in any plant area. With the scenario presented in the test question, the plant is not in a normal configuration, the cause of the safety injection is not indicated, and the fire is in an area containing equipment that is credited for PFSSD. This presents a condition that is not directly analyzed regarding potential impact on PFSSD capability. Therefore, it is important for the Control Room to know and be prepared to complete the OFN KC-016 Attachment E actions that may be required to mitigate fire induced spurious equipment maloperation. Additionally, the fire brigade response, driven by OFN KC-016, is necessary to satisfy the second defense in depth element of the Fire Protection Program so that the final defense element is not challenged.

NRC Inspection Report 05000482/2009007 documents a non-cited violation (NCV) for failure to implement compensatory actions per Station Fire Protection Program requirements in a timely manner. On 8/19/2009 the plant experienced a momentary loss of offsite power (LOOP). Following the plant trip, multiple trouble alarms were annunciated in the Control Room on main fire alarm panel KC008. The trouble alarms were silenced by the Control Room Supervisor and a building watch was dispatched to walk down the affected areas. However, the fire watch compensatory measures required by AP 10-103, *Fire Protection Impairment Control*, were not established within one hour as required by the procedure. Page 26 of the Inspection Report contains the following statement:

> *"The control room supervisor was preoccupied with actions related to the reactor trip and did not perform the required action to initiate a fire protection impairment."*

This statement reflects a regulatory position that compliance with the Fire Protection Program is required even when addressing a plant transient. Therefore, it is reasonable to conclude that Control Room and Fire Brigade personnel would need to concurrently address an EMG response and a fire response in a safety related area. Ignoring or delaying fire response would be a violation of the Fire Protection Program, which is required to be maintained by our Operating License.

AP 15C-003, Step 6.2.2, specifies: **Under certain plant conditions, procedures of "lower priority" make take precedence over procedures of "higher priority."**

ALR KC-888, FIRE PROTECTION PANEL KC-008 ALARM RESPONSE, Step A.1.3.1, states: ***Implement OFN KC-016, Fire Response, for fire alarms.***

For the given scenario, Operations Management expects required concurrent performance of OFN KC-016 and EMG ES-03 with the following considerations:

- AP 15C-003 allows parallel performance and specifies there are specific conditions where general requirements do not apply universally.
- AP 10-100 REQUIRES response to fires, irrespective of other plant conditions.
- The KC-008 Alarm Panel response REQUIRES entry into OFN KC-016.
- AP 21-004, OPERATOR RESPONSE TIME PROGRAM specifies time critical actions associated with the plant Fire Protection Program Licensing Basis. The CRS cannot choose to ignore an active fire only because performance of an EMG procedure is in progress.
- Scenario placement in EMG ES-03 demonstrates that SI is NOT required and thus the accident in progress is something less than design basis.
- Scenario postulated fire area contains equipment vital to safety, which may well be required to respond to the accident in progress.

- Station Operating Experience includes a NCV for failure to write an impairment during emergency response. Writing an impairment is objectively less important than responding to an active fire in the MDAFW area.

Recommendation

Recommend changing the correct answer to A from C. The CRS is REQUIRED to enter and perform <u>BOTH</u> OFN KC-016 <u>AND</u> EMG ES-03 actions concurrently to comply with Operating License, 10 CFR 50, Appendix R, USAR, TECH SPECS, and AP10-100 requirements for the given scenario; therefore, answer choice A is correct. Wolf Creek Condition Report117620 is written to document the examination writing error, which resulted in required regrade evaluation of the question by the NRC.

<u>NRC RESPONSE</u>

Question #99. The licensee requests changing the correct answer from C to A. I disagree with the licensee, and believe that the only correct answer is C.

The question deals with how procedures should be implemented when, while performing, EMG ES-03, SI TERMINATION, a fire is reported. The keyed correct answer, “Is allowed by procedure to implement OFN KC-016, FIRE RESPONSE as long as it does not interfere with performance of EMG ES-03,” comes from AP-15C-003, Procedures User’s Guide for Abnormal Plant Conditions.” Step 6.2.3 of the procedure states, “While performing EMGs, plant conditions may indicate the need to correct problems not directly related to the event mitigation strategy. The operator may perform OFNs and ALRs which address these problems as long as the actions do not interfere with performance of the EMGs.”

The licensee points out Step 6.2.2 that states, “Under certain plant conditions, procedures of lower priority’ make take precedence over procedures of ‘higher priority’” and also makes a statement that Operations Management expects required concurrent performance of OFN KC-016 and EMG ES-03.

However, there is no procedural guidance **<u>requiring</u>** concurrent performance, nor does there appear to be any classroom training **<u>requiring</u>** concurrent performance.

Since C is the only answer that has procedural guidance, I believe that C is the only correct answer.

No changes shall be made to this question.

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