UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

October 18, 2011

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 05000482/2011301

Dear Mr. Sunseri:

On September 2, 2011, 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 September 1, 2011, with yourself and other members of your staff. A telephonic exit meeting was conducted on September 22, 2011, with Ms. Mona Guyer, who was provided the NRC licensing decisions.

The examination included the evaluation of two applicants for reactor operator licenses, five applicants for instant senior reactor operator licenses and three applicants for upgrade senior reactor operator licenses. The license examiners determined that nine of the ten 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 and summarizes post examination comment resolution.

During the validation of this examination, the NRC identified one finding that was evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has determined that a violation is associated with this finding. Because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy. Additionally, one licensee-identified violation, which was determined to be of very low safety significance, is listed in this report.

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Sincerely,

/RA/

Mark S. Haire, Chief
Operations Branch
Division of Reactor Safety

Docket: 50-482
License: NPF-42

Enclosure:
NRC Examination Report 05000482/2011301

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Resident Inspector (Charles.Peabody@nrc.gov)
WC Administrative Assistant (Shirley.Allen@nrc.gov)
Branch Chief, DRP/B (Geoffrey.Miller@nrc.gov)
Senior Project Engineer, DRP/B (Rick.Deese@nrc.gov)
Project Engineer, DRP/B (Nestor.Makris@nrc.gov)
Public Affairs Officer (Victor.Dricks@nrc.gov)
Public Affairs Officer (Lara.Uselding@nrc.gov)
Project Manager (Randy.Hall@nrc.gov)
Branch Chief, DRS/TSB (Dale.Powers@nrc.gov)
RITS Coordinator (Marisa.Herrera@nrc.gov)
Regional Counsel (Karla.Fuller@nrc.gov)
Congressional Affairs Officer (Jenny.Weil@nrc.gov)
OEMail Resource
RIV/ETA: OEDO (Mark.Franke@nrc.gov)
DRS/TSB STA (Dale.Powers@nrc.gov)
Branch Chief, DRS/OB (Mark.Haire@nrc.gov)

SUNSI Review Completed: __CCO__ ADAMS: ☒ Yes ☐ No Initials: __CCO__
☒ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☒ Non-Sensitive

SOE:OB	OE:OB	OE:OB	C:PBC	C:OB
COsterholtz	DStrickland	TFarina	GMiller	MHaire
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10/03/11	10/11/11	10/07/11	10/11/11	10/18/11

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-482
License:	NPF-42
Report:	05000482/2011301
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, Kansas
Dates:	August 26, 2011 - September 22, 2011
Inspectors:	C. Osterholtz, Senior Operations Engineer D. Strickland, Operations Engineer T. Farina, Operations Engineer
Approved By:	Mark Haire, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER05000482/2011301; August 26, 2011 - September 22, 2011; Wolf Creek Generating Station; Initial Operator Licensing Examination Report; Examination Development

NRC examiners evaluated the competency of two applicants for reactor operator licenses, five applicants for instant senior reactor operator licenses and three applicants for upgrade senior reactor operator licenses 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 August 26, 2011. NRC examiners administered the operating tests on the week of August 29, 2011.

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

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a Green noncited violation of Technical Specification 5.4.1.a, “Procedures,” due to insufficient procedural direction to operations personnel to perform a subcooled recovery of a steam generator tube rupture if the ruptured steam generator cannot be isolated from any of the intact steam generators. On August 2, 2011, inspectors identified during simulator scenario validation that step 9 of Emergency Mitigation Guideline 3, “Steam Generator Tube Rupture,” did not give adequate direction to operations personnel to mitigate a steam generator tube rupture event that required a subcooled recovery. The licensee entered the issue into their corrective action program as condition report 43515.

The finding is more than minor because the performance deficiency is associated with the procedure quality attribute of the mitigating systems cornerstone, and adversely affected the cornerstone’s attribute to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheets, the finding was determined to have very low safety significance because the finding is a deficiency confirmed not to result in a loss of operability or functionality of the overall ability to mitigate an unisolable steam generator tube rupture, if Emergency Mitigation Guideline 3 is used correctly as written. The finding does not have a crosscutting aspect because the deficiency was incorporated into the procedure in May 2000 and was not considered indicative of current licensee performance (Section 4OA5.2).

B. Licensee-Identified Violations

A violation of very low safety significance that was identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. This violation and the corrective action tracking number are listed in Section 4OA7 of this report.

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.

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

b. Findings

Introduction. The inspectors identified a Green noncited violation of Technical Specification 5.4.1.a, "Procedures," due to insufficient procedural direction to operations personnel to perform a subcooled recovery of a steam generator tube rupture (SGTR) if the ruptured steam generator cannot be isolated from any of the intact steam generators.

Description. On August 2, 2011, validation of dynamic simulator scenarios was being conducted at the Wolf Creek simulator facility in preparation for an initial licensing exam that was scheduled to be conducted the week of August 29, 2011. The scenario validation included representatives from the Wolf Creek training and operations departments as well as the NRC examiners assigned to administer the examination.

During validation of scenario 4 (a scenario slated to be set aside as a spare), the Wolf Creek operations personnel performing the scenario incorrectly interpreted a procedural step that resulted in the inspectors identifying a deficiency in Emergency Mitigating Guideline (EMG)-3, "Steam Generator Tube Rupture." The scenario's major transient was a steam generator tube rupture that was complicated with a failure of all main steam isolation valves to close, automatically or manually. The correct procedural flow path would have been to transition out of EMG-3 at step 5 with the direction, "If any ruptured steam generator can NOT be isolated from at least one intact steam generator, THEN go to EMG C-31, SGTR WITH LOSS OF REACTOR COOLANT – SUBCOOLED RECOVERY DESIRED." The Wolf Creek operations personnel incorrectly interpreted this step and subsequently incorrectly continued on in Procedure EMG-3. EMG-3, step 9, then provided operators a "second chance" to determine if a ruptured steam generator could be isolated from any intact steam generator by directing transition to EMG C-31 only if main steam isolation valves could not be closed AND Attachment A, "Main Steam Header Isolation – Control Room," could not be completed. The Wolf Creek operators had successfully completed Attachment A, so they believed that transition to EMG C-31 was still unnecessary. The scenario validation was then halted.

The NRC examiners determined that EMG-3, step 9, provided erroneous procedural transition instructions, as the successful performance of Attachment A only isolates the ruptured steam generator from the main steam header and does not isolate the ruptured steam generator from any of the remaining intact steam generators. The misdirection provided in step 9 of EMG-3 prevents operators from correctly transitioning to a procedure (EMG C-31) that would have provided for a subcooled recovery.

The EMG background document describes the necessity of the transition to EMG C-31 from EMG-3 when a ruptured steam generator cannot be isolated from any intact steam generator:

> Isolation of the ruptured steam generator effectively minimizes release of radioactivity from this generator. In addition, isolation is necessary to establish a pressure differential between the ruptured and non-ruptured steam generators in order to cool the reactor coolant system and stop primary-to-secondary leakage. Two important conditions are desired for stopping primary-to-secondary leakage. First, in order to remove heat generated in the primary system, the ruptured steam generator pressure and reactor coolant system pressure must be maintained greater than the non-ruptured steam generator pressures. Secondly, as this pressure differential is increased, so is the subcooling in the primary system. If sufficient pressure differential cannot be maintained, leakage from the reactor coolant system will continue since reactor coolant system pressure will remain greater than the ruptured steam generator pressure in order to remove decay heat. In that case, the operator is directed to EMG C-31, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired, to minimize this leakage.

The licensee acknowledged the procedural deficiency in EMG-3, step 9, and entered the issue into their corrective action program as Condition Report (CR) 43515. The licensee indicated that operator training would be performed on the correct usage of EMG-3, that EMG-3 would be revised to correct the deficiency, and that an extent of condition review would be performed on all EMG's to identify any similar procedural transition deficiencies. The examiners did note that the scenario could have been successfully mitigated had the Wolf Creek operators interpreted EMG-3 correctly and properly transitioned to EMG C-31 during performance of EMG-3, step 5.

Analysis. The failure of EMG-3, step 9, to direct operations personnel to perform a subcooled recovery of a steam generator tube rupture if the ruptured steam generator cannot be isolated from any of the intact steam generators was a performance deficiency. The performance deficiency is more than minor and, therefore, a finding because the performance deficiency is associated with the procedure quality attribute of the mitigating systems cornerstone, and adversely affected the cornerstone's attribute to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the finding was determined to have very low safety significance because the finding is a deficiency confirmed not to result in a loss of operability or functionality of the overall ability to mitigate an unisolable steam generator tube rupture if EMG-3 is used correctly as written. The finding does not have a crosscutting aspect because the deficiency was incorporated in May 2000 and was not considered indicative of current licensee performance.

Enforcement. Technical Specification 5.4.1.a, "Procedures," requires that written procedures be established and implemented covering activities specified in Appendix A, "Typical Procedures for Pressurized Water Reactors," of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," February 1978. Regulatory Guide 1.33, Appendix A, Section 6, requires, in part, that the licensee should have procedures in place to combat emergencies such as significant steam generator tube leaks. Contrary to the above, on August 2, 2011, inspectors identified that the licensee had failed to adequately establish procedures covering activities related to significant steam generator tube leaks. Specifically, inspectors identified during simulator scenario validation that step 9 of EMG-3 would not successfully direct operations personnel to mitigate a steam generator tube rupture event that required a subcooled recovery. Because of the very low safety significance of this finding and because the licensee entered this issue into the corrective action program as Condition Report 43515, this violation is being treated as a noncited violation in accordance with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000482/2011301-01, "Inadequate Emergency Operating Procedure for Steam Generator Tube Rupture."

.3 Operator Knowledge and Performance

a. Scope

On August 26, 2011, the licensee proctored the administration of the written examinations to all ten applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis and post examination comments to the NRC on September 12, 2011.

The NRC examination team administered the various portions of the operating tests to all ten applicants on the week of August 29, 2011.

b. Findings

No findings were identified.

Nine applicants passed all parts of the operating test. One of the applicants failed the administrative portion and the walk-through portion 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 section 4OA5.6 of this report.

The examination team noted a generic weakness in the ability of the majority of operators to take prompt action when warranted. During performance of the simulator job performance measures, a majority of applicants examined failed to secure reactor coolant pumps on a loss of component cooling water even after bearing temperature limits had been substantially exceeded. During the simulator scenarios, there were several circumstances where applicants delayed taking manual actions that should have occurred automatically until directed to do so by procedure. The delays unnecessarily complicated scenario event recovery. The licensee indicated they would evaluate this issue and take the appropriate corrective actions. The licensee entered this issue into their corrective action program as CR 43514.

.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 examiners with two post administration comments on the written examination (Question 37 and Question 85). The following are the respective questions, the licensee's comments, and the examiners' evaluation of the licensee's comments:

Question 37

Given the following plant conditions:

- The unit is in MODE 6

- "A" train RHR is in service
- “B” train RHR is in standby
- Refueling Pool level is 23' above the fuel on BB LI-53A, RCS LEVEL WR COLD CAL LOOP 4
- Control Rod Shaft Unlatching is the next evolution
- All Refueling personnel are on station

Which ONE of the following describes whether or not unlatching can begin and why or why not?

A. Yes, level and RHR are adequate.

B. Yes, ONLY proper level is required.

C. No, RHR loops are inadequate.

D. No, level is inadequate.

Facility Comment

Question 37 is a new question. This question was missed by 7 out of 10, with all choosing distracter D. An applicant submitted a question during the exam’s administration asking if level was exactly 23 feet above the fuel. In assessing a response to the question, it was discussed that although TRM requirement is ≥ 23 feet, there are procedural requirements that exceed the TRM values. It was further discussed whether a value should be provided that met all requirements. It was decided that no additional information would be provided during the exam. Validation results showed that the question was missed by 2 out of 6. Validators provided no comments on the question.

TRM 3.9.7 requires water level ≥ 23 feet above the fuel during unlatching to meet accident analysis for fuel handling accidents. This level is specifically marked on BB LI-53A/B as 23 feet above fuel at ~ 240 inches. Control rod latching requirements are different from fuel movement, since all of the fuel assemblies are seated in the vessel. For fuel movement, TS 3.9.7 requires ≥ 23 feet above the vessel flange instead of the top of fuel. Both requirements are reflected in Operator surveillance during refueling STS CR-002, SHIFT LOG FOR MODES 4, 5, AND 6, which requires 241” above the fuel assemblies. The original goal of this part of the question was to differentiate between the TRM and Technical Specification requirements. The second part of the question was to test TS 3.9.6 requiring two RHR loops to be operable with one loop in operation.

From a procedural standpoint in addition to the TRM, Answer D is correct. In order to assure adequate level the procedural requirements in GEN 00-009, REFUELING, and FHP 02-012, CONTROL ROD UNLATCHING/LATCHING TOOL OPERATING INSTRUCTIONS, are set higher than TRM requirement to provide diverse visual indication of pool level with water at or above control rod drive shaft button height. GEN 00-009 step 6.1.15 contains a note stating: “when above button height (267” ~25 feet above fuel), unlatching may proceed,” and is tied to a reference in FHP 02-012. FHP 02-012 step 5.13 states: “Refueling Pool level must be at or above button height for latching, unlatching and weight verification activities. Button height (2035 ft 6.5 inches, 267" on BB LI-462) is 27.2 inches higher than the TR 3.9.7, Refueling Pool Water Level requirement for core alterations of 23 ft above the top of irradiated fuel assemblies seated in the Reactor Vessel (2033 ft 3.25 inches, 239.8" on BB LI-53A). The purpose of the additional water is to shield the workers from radiation exposure from the upper internals structure and the Control Rod drive shafts."

Answer A meets the strict TRM requirement; however, it fails to meet the procedural requirement. Question 37 asks: "Which ONE of the following describes whether or not unlatching can begin and why or why not?" Although the TRM answer was originally selected as the correct choice upon further review it was found to be incorrect per the operating procedures. Answer A would have been correct had the question been "Can unlatching begin based solely on the stated conditions meeting TRM and Technical Specifications requirements and why?" As asked, the question is too broad and therefore includes <u>all</u> requirements, procedural, TS and TRM.

Based on the question as written the correct Answer is D because answer A did not consider the additional level requirements in the procedures. Per NUREG 1021 ES-403 this is considered newly discovered technical information supporting a change in the proposed answer key. The question stem should be reworded to specify that the GEN 00-009 and FHP 02-012 procedural requirements are to be considered in the determination.

<u>NRC Evaluation</u>

The examiners concurred with the licensee's assessment. Procedures GEN 00-009, "Refueling," and FHP 02-012, "Control Rod Unlatching/Latching Tool Operating Instructions," provide the most restrictive limits regarding minimum spent fuel pool level, and should be the overriding guidance when determining spent fuel pool level requirements for the evolution described in the question stem.

The final answer key has been revised to indicate that answer "D" is the one and only correct answer.

<u>Question 85</u>

Given the following plant conditions:

- The unit is operating at 80%
- Generator power is 1050 Megawatts
- Annunciator 00-130E, GEN AUX TROUBLE, is in alarm
- ALR 408-06A, MACH GAS PRESSURE HIGH-LOW, is in alarm
- CC PI-1, MACHINE GAS PRESSURE, reads 41 psig
- Reactive Load is outgoing at 200 MVA
- Minor leakage from the Main Generator housing is reported

Which ONE of the following is the MINIMUM hydrogen pressure needed to ensure the Generator has enough cooling (assume hydrogen cooling aligned normal) and what actions should the CRS direct?

(REFERENCE PROVIDED)

A. 75 psig for MVARs up to 750

If leakage cannot be isolated, trip the unit and emergency depressurize the Main Generator using SYS CC-321, GENERATOR HYDROGEN EMERGENCY DEPRESSURIZATION

B. 60 psig for MVARs up to 600

Isolate the leak in accordance with OFN AF-025, UNIT LIMITATIONS

C. 48 psig for MVARs up to 650

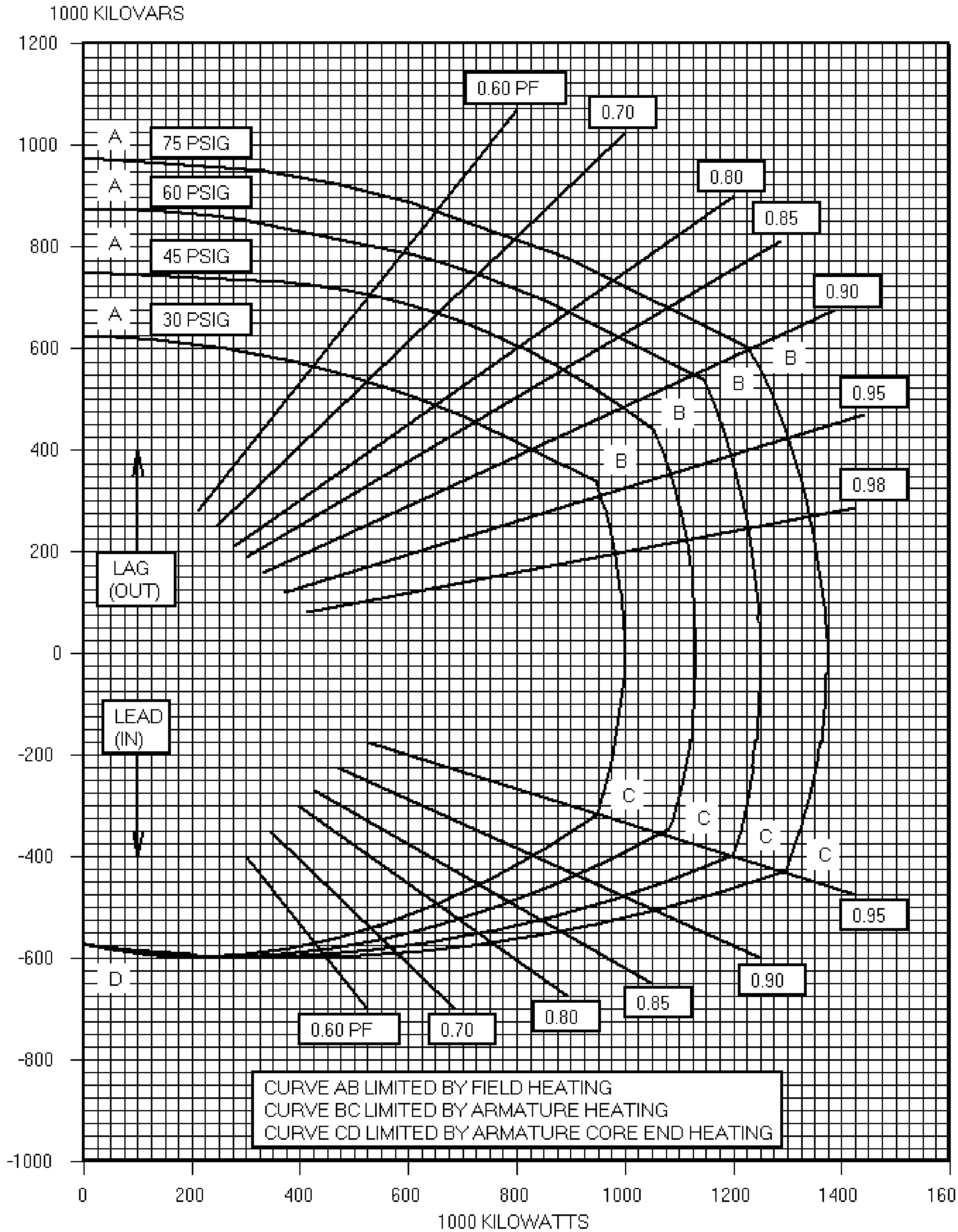
Isolate the leak in accordance with OFN AF-025, UNIT LIMITATIONS

D. 45 psig for MVARs up to 500

If leakage cannot be isolated, trip the unit and emergency depressurize the Main Generator using SYS CC-321, GENERATOR HYDROGEN EMERGENCY DEPRESSURIZATION

Correct Answer: D

(Provided reference on next page)



1000 KILOVARS
1200
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800
600
400
200
0
-200
-400
-600
-800
-1000
A
75 PSIG
A
60 PSIG
A
45 PSIG
A
30 PSIG
0.60 PF
0.70
0.80
0.85
0.90
0.95
0.98
B
B
B
B
LAG
(OUT)
LEAD
(IN)
C
C
C
C
0.95
0.90
0.85
0.80
0.70
0.60 PF
D
CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING
0
200
400
600
800
1000
1200
1400
1600
1000 KILOWATTS

Facility Comment

Question 85 is a new question. This question was missed by 7 of 8, with all choosing answer B. One post-exam comment received from an applicant asked why initial conditions existed in the stem when all answers had their own conditions. Another post-exam comment stated the answer was still higher than the line, but that they chose the one that was closest to the graph line. There were no misses during validation, but the question was changed in the final revision. The original draft answers were based on pressure and power factor. The NRC requested a change to MVAR loading since power factor is not a measured quantity on the control board indications. The question was redrafted, technically reviewed and revalidated by two licenses. They both missed it. The question was changed again based on their feedback and an error occurred in the final revision.

The question is two parts, the first part assesses main generator limitations and the second the required course of action to place the unit in a safe configuration. The applicant must find four points on the main generator spider curve and assess whether they are inside or outside of an acceptable pressure curve. This was first done with pressure versus power factor then later revised to pressure versus reactive load. The second part tests the applicant's knowledge of the required remedial actions contained in ALR vice OFN procedures.

Review of the associated points indicates there is no correct answer to this question since all points are outside their respective power curves on the figure. The question prior to the final version did have a correct answer with reactive load in Answer D at 400 MVAR. This was changed to 500 MVAR in the final version by a validator comment. This change was made in error without sufficient final review.

Answer B is closest to the curve but still outside. The procedural flowpath is incorrect since the OFN does not isolate the leak. The leak is isolated by the local ALR, which then sends the performer to a system operating procedure to emergency depressurize the generator.

The question requires repair. The original question may not have been discriminatory enough, but performed well in validation. If MVAR is to be used in the answer then the current MVAR loading statement needs to be removed from the stem. A correct answer must be provided.

This question is recommended for deletion from this exam because no correct answer is provided. For this question, an error occurred when changes were incorporated from validation without sufficient technical review.

NRC Evaluation

The examiners concurred with the licensee's assessment. When the question was converted to have the distracters read in MVARS rather than power factor, an error was incorporated into the question that rendered none of the distracters correct.

Since there is no correct answer, the final answer key has been revised to indicate that this question has been deleted from the examination.

4OA6 Meetings, Including Exit

The chief examiner presented the preliminary examination results to Mr. Matthew Sunseri and other members of the staff on September 1, 2011. A telephonic exit was conducted on September 22, 2011, between Mr. Clyde Osterholtz, Chief Examiner, and Ms. Mona Guyer, Operations Training Manager.

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

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section 2.3.2 of the NRC Enforcement Policy, for being dispositioned as a noncited violation:

> Criterion V of 10 CFR Part 50, Appendix B, requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings. Procedure OFN BB-005, "RCP Malfunctions," Revision 19 was discovered to be inadequate by the licensee during examination development. While attempting to validate a job performance measure that required securing a reactor coolant pump at less than 48% reactor power, the licensee discovered that OFN BB-005 directed placing feedwater control in automatic prior to stabilizing the affected loop's steam generator water level. This resulted in a level swell in the affected loop's steam generator which triggered an automatic turbine/reactor trip on high steam generator water level. The job performance measure was removed from the initial examination and replaced. The inadequacy of procedure OFN BB-005 was considered a performance deficiency. The finding is greater than minor because the performance deficiency could be reasonably viewed as a precursor to a significant event (reactor trip). The finding is of very low safety significance because the finding does not contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions will not be available. The issue was entered into the licensee's corrective action program as CR 43511.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
Louis Brosch, Procedure Writer
Dave Dees, Operations Manager
J. Lee Gorman, Exam Writer
Mona Guyer, Operations Training Manager
Jane Hartley, Examination Development
Lance Lane, Operations
Edward Ray, Operations Training
Lucille Rockers, Licensing
Brendan Ryan, Exam Development
Gautam Sen, Regulatory Affairs Manager
George Smith, Initial Licensing Supervising Instructor
Russell Smith, Plant Manager
Matt Sunseri, Chief Nuclear Officer

NRC Personnel
Charles Peabody, Resident Inspector

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000482/2011301-01	NCV	Inadequate Emergency Operating Procedure for Steam Generator Tube Rupture (Section 4OA5.2)

ADAMS DOCUMENTS REFERENCED

Accession No. ML 112700887 – FINAL WRITTEN EXAMS
Accession No. ML 112700890 – POST EXAM ANALYSIS
Accession No. ML 112700888 – FINAL OPERATING TEST