UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

July 17, 1997

NOTE TO: NRC Document Control Desk
Mail Stop O-5-D-24

FROM: Laura Hurley, Licensing Assistant
Operations Branch, Region IV

SUBJECT: OPERATOR LICENSING EXAMINATIONS ADMINISTERED ON APRIL 15-16, 1997, AT WOLF CREEK NUCLEAR OPERATING STATION, UNIT 1

DOCKET #50-482

On April 15-16, 1997, Operator Licensing Examinations were administered at the referenced facility. Attached you will find the following information for processing through NUDOCS and distribution to the NRC staff, including the NRC PDR:

Item #1 - a) Facility submitted outline and initial exam submittal, designated for distribution under RIDS Code A070.

b) As given operating examination, designated for distribution under RIDS Code A070.

Item #2 - Examination Report with the as given written examination attached, designated for distribution under RIDS Code IE42.

If you have any questions, please contact Laura Hurley, Licensing Assistant, Operations Branch, Region IV at (817) 860-8253.

U. S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information	
Name:	Region: IV
Date: 4-14-97	Facility/Unit: Wolf Creek Generating Station
License Level: SRO	Reactor Type: W

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80 percent. Examination papers will be picked up 4 hours after the examination starts.

All work done on this examination is my own. I have neither given nor received aid.

_____
Applicant's Signature

Results	
Examination Value	100 points
Applicant's Score	_____ Points
Applicant's Grade	_____ Points

SRO EXAMINATION ANSWER SHEET

Wolf Creek Generating Station Upgrade Written Examination

PRINT NAME: _____ **DATE:** 04/14/97

FIRST M LAST

1.	A	B	C	D	26.	A	B	C	D	51.	A	B	C	D	76.	A	B	C	D
2.	A	B	C	D	27.	A	B	C	D	52.	A	B	C	D	77.	A	B	C	D
3.	A	B	C	D	28.	A	B	C	D	53.	A	B	C	D	78.	A	B	C	D
4.	A	B	C	D	29.	A	B	C	D	54.	A	B	C	D	79.	A	B	C	D
5.	A	B	C	D	30.	A	B	C	D	55.	A	B	C	D	80.	A	B	C	D
6.	A	B	C	D	31.	A	B	C	D	56.	A	B	C	D	81.	A	B	C	D
7.	A	B	C	D	32.	A	B	C	D	57.	A	B	C	D	82.	A	B	C	D
8.	A	B	C	D	33.	A	B	C	D	58.	A	B	C	D	83.	A	B	C	D
9.	A	B	C	D	34.	A	B	C	D	59.	A	B	C	D	84.	A	B	C	D
10.	A	B	C	D	35.	A	B	C	D	60.	A	B	C	D	85.	A	B	C	D
11.	A	B	C	D	36.	A	B	C	D	61.	A	B	C	D	86.	A	B	C	D
12.	A	B	C	D	37.	A	B	C	D	62.	A	B	C	D	87.	A	B	C	D
13.	A	B	C	D	38.	A	B	C	D	63.	A	B	C	D	88.	A	B	C	D
14.	A	B	C	D	39.	A	B	C	D	64.	A	B	C	D	89.	A	B	C	D
15.	A	B	C	D	40.	A	B	C	D	65.	A	B	C	D	90.	A	B	C	D
16.	A	B	C	D	41.	A	B	C	D	66.	A	B	C	D	91.	A	B	C	D
17.	A	B	C	D	42.	A	B	C	D	67.	A	B	C	D	92.	A	B	C	D
18.	A	B	C	D	43.	A	B	C	D	68.	A	B	C	D	93.	A	B	C	D
19.	A	B	C	D	44.	A	B	C	D	69.	A	B	C	D	94.	A	B	C	D
20.	A	B	C	D	45.	A	B	C	D	70.	A	B	C	D	95.	A	B	C	D
21.	A	B	C	D	46.	A	B	C	D	71.	A	B	C	D	96.	A	B	C	D
22.	A	B	C	D	47.	A	B	C	D	72.	A	B	C	D	97.	A	B	C	D
23.	A	B	C	D	48.	A	B	C	D	73.	A	B	C	D	98.	A	B	C	D
24.	A	B	C	D	49.	A	B	C	D	74.	A	B	C	D	99.	A	B	C	D
25.	A	B	C	D	50.	A	B	C	D	75.	A	B	C	D	100.	A	B	C	D

1. Cheating on the examination will result in a denial of your application and could result in more severe penalties.

2. After you complete the examination, sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination.

3. To pass the examination, you must achieve a grade of 80 percent or greater.

4. The point value for each question is indicated in parentheses after the question number.

5. There is a time limit of 4 hours for completing the examination.

6. Use only black ink or dark pencil to ensure legible copies.

7. Print your name in the blank provided on the examination cover sheet and the answer sheet.

8. Mark your answers on the answer sheet provided and do not leave any question blank.

9. If the intent of a question is unclear, ask questions of the examiner only.

10. Restroom trips are permitted, but only one applicant at a time will be allowed to leave. Avoid all contact with anyone outside the examination room to eliminate even the appearance or possibility of cheating.

11. When you complete the examination, assemble a package including the examination questions, examination aids, and answer sheets and give it to the examiner or proctor. Remember to sign the statement on the examination cover sheet.

12. After you have turned in your examination, leave the examination area as defined by the examiner.

Question 1
One Point

The plant is operating at full power when the supply breaker to 480 VAC Load Center PG19 opens. Which one of the following describes the effect that this will have on the rod control system?

a. No effect other than annunciator alarm(s).

b. Two reactor trip breakers will open but the reactor will not trip.

c. A Power Cabinet Urgent Failure alarm and loss of the ability to move rods in Auto or Manual Sequential modes.

d. The affected rod control power cabinets will swap to their backup power source and continue to operate normally.

Question 2
One Point

A Rod Control Power Cabinet Urgent Failure has occurred and has not been reset. Which one of the following describes the effect that this event will have on the ability to step control rods?

a. Only the banks associated with the unaffected cabinets may be moved manually. Automatic rod motion will occur but only for unaffected cabinets.

b. Manual motion is unaffected. Auto motion is possible for rods associated with the unaffected cabinets only.

c. Manual motion is unaffected. No auto motion is possible.

d. Only the banks associated with the unaffected cabinets may be moved individually. No other manual or auto motion is possible.

Question 3
One Point

The plant is operating at 50% power when the operator notes that the rods step inward a few steps and then stop with no operator action. Which one of the following instrument failures could cause this?

a. Power Range NI41 fails high.

b. Power Range NI43 fails low

c. Turbine Impulse Pressure Channel PT-506 fails low.

d. One loop T_h RTD fails low.

Question 4
One Point

During normal full power operation the operator notes that VCT pressure is 10 psig. Which one of the following is the most likely result of this condition?

a. Backflow from the RCS through the #1 RCP seal.

b. Insufficient flow through the #1 RCP seal.

c. Insufficient flow through the #2 RCP seal.

d. Flashing in the seal return cooler.

Question 5
One Point

Which one of the following describes the response of the Main Generator Output Breakers to a reactor trip from full power?

a. Open immediately as a result of the turbine trip.

b. Open almost immediately on reverse power.

c. Opens after a 30 second time delay.

d. Opens on reverse power after about 30 seconds.

Question 6
One Point

The seal water return filter becomes clogged. The indications in the control room of this event are:

a. Increasing Reactor Coolant Drain Tank level.

b. Increasing Pressurizer Relief Tank level.

c. Increasing Recycle Holdup Tank level.

d. Increasing Volume Control Tank level.

Question 7
One Point

Which one of the following operations places the greatest heat load on the seal water heat exchanger?

a. CCP recirc operation.

b. Excess letdown operation.

c. RCP operation without CCW flow to the thermal barrier.

d. Loss of normal letdown without loss of charging.

Question 8
One Point

A normal plant shutdown is in progress with the following conditions:
→ Pressurizer Pressure is 2000 psig
→ Steam line pressure is 800 psig
→ All other conditions are normal for this point in a plant shutdown.

At this point a large steamline rupture occurs upstream of the MSIV resulting in the complete depressurization of the affected steam generator in 1 minute. Which one of the following describes an expected response of the ESFAS to these events?

a. MSLIS only due to high rate of steamline pressure decrease.

b. SI and MSLIS due to high rate of steamline pressure decrease.

c. SI only due to low steam line pressure.

d. SI and MSLIS due to low steam line pressure.

Question 9
One Point

An SIS has occurred but reactor trip breaker B did not open. Which one of the following will occur when the operator attempts to reset SIS?

a. Both trains of SIS will reset.

b. Neither train of SIS will reset.

c. Only the A train of SIS will reset.

d. Only the B train of SIS will reset.

Question 10
One Point

Which one of the following describes the actual position of a rod whose indicated DRPI position is 100 steps with Data A failure present?

a. 96-104 steps.

b. 90-104 steps.

c. 96-110 steps.

d. 90-110 steps.

Question 11
One Point

Given the following data concerning power range nuclear instruments:

Channel	N41	N42	N43	N44
Upper Detector	51mA	49mA	57mA	56mA
Lower Detector	53mA	54mA	56mA	54mA

Which one of the following is the Quadrant Power Tilt Ratio (QPTR)?

a. 1.019

b. 1.032

c. 1.051

d. 1.070

Question 12
One Point

The plant is operating at 100% power as verified by calorimetric data. All channels of Power Range Nuclear Instruments likewise read 100%. A load rejection then occurs, reducing power to 50% as read on the Power Range Nuclear Instruments. Which one of the following describes plant power conditions after the load rejection?

a. Actual power is greater than 50%.

b. Actual power is less than 50%.

c. Actual power is equal to 50%.

d. At BOL actual power will be greater than 50% and at EOL it will be less than 50%.

Question 13
One Point

Which one of the following combinations of incore thermocouple temperatures and RCS pressures indicate that the core is uncovered?

a. Temperature 506°F, pressure 485 psig.

b. Temperature 538°F, pressure 935 psig.

c. Temperature 538°F, pressure 1385 psig.

d. Temperature 629°F, pressure 1935 psig.

Question 14
One Point

The Containment Fan Coolers are running in fast speed when an SIS occurs. Which one of the following describes the expected response of the Fan Coolers?

a. They continue to run unless containment spray is initiated.

b. They trip and are restarted in slow speed by the LOCA sequencer.

c. They shift to slow speed.

d. They trip but will restart in slow speed if containment temperature reaches 160°F

Question 15
One Point

has been operating at 40% power for the past two days when a feed line shears off one steam Level in the affected steam generator rapidly blows down to zero. Levels in the other three S/Gs to a minimum of 25% NR. Which one of the following describes the response of AMSAC to this ev .

a. AMSAC will not respond to this event.

b. Immediately trips the turbine and initiates auxiliary feed to the steam generators.

c. Immediately trips the turbine and initiates auxiliary feed, after a 25 second delay, to the steam generators.

d. After a 25 second delay it trips the turbine and initiates auxiliary feedwater to the steam generators.

Question 16
One Point

The following plant conditions exist:

→ All three AFW pumps are running due to low levels in all S/Gs.
→ The source of water for all three pumps is the CST.
→ A low CST level causes suction pressure to drop to 0 psig.
→ An instrument failure in Train B results in actuation of only Train A LSP protection.

Which one of the following describes the expected response of the AFW System water sources to these events?

a. There will be no change in water source alignment.

b. The suctions of all three AFW pumps will realign to the ESW System.

c. The 'A' MDAFW pump will realign to the ESW. The other pumps will be unaffected.

d. The 'A' MDAFW pump and the TDAFW pump will realign to the ESW System. The other pump will be unaffected.

Question 17
One Point

The plant had been operating at full power when a ground fault resulted in bus NB01 being deenergized. 60 seconds have elapsed and it is still deenergized. All steam generator levels are within their normal operating range. Which one of the following describes the expected status of the AFW pumps?

a. No pumps are running.

b. All three are running.

c. The TD and the B MD AFW pumps are running.

d. Only the TD AFW pump is running.

Question 18
One Point

A small LOCA has occurred and the following plant conditions exist:
→ RCS pressure is 1500 psig.
→ Containment pressure is 0.3 psig.
→ Maximum measured containment radiation level 50 mr/hr.

Which one of the following flowpaths associated with the Reactor Coolant Drain Tank (RCDT) will receive an isolation signal as a result of these conditions?

a. RCDT flowpaths should be unaffected.

b. RCDT H_2/Vent line.

c. RCDT recirculation flow path.

d. Refueling Pool drain path to RCDT.

Question 19
One Point

The CHARMs (Containment High Area Radiation Monitor - GT RE 59) reaches the Alert level. This will result in which one of the following alarms on the Main Control Board?

a. None.

b. Process Rad Hi.

c. Process Rad Hi Hi.

d. Area Rad Hi

Question 20
One Point

The top of the core corresponds to which one of the following RVLIS indications?

a. 45% Natural Circ range.

b. 68% Natural Circ range.

c. 45% Forced Flow Range.

d. 68% Forced Flow Range.

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Question 21
One Point

At midnight (0000) the RCS Leakage STS shows that RCS leakage is in excess of the Tech Spec limit. At 0200 a reevaluation of the RCS Leakage STS performed at noon (1200) of the previous day shows that a math error was made. Correcting this error shows that the out of spec RCS leakage has existed since noon of the previous day. Which one of the following states the proper time to start the Action Statement clock?

a. Noon of the previous day.

b. Midnight, the time of the initial discovery.

c. 2am, the time of verification

d. 4am, four hours past the time of discovery.

Question 22
One Point

Following an automatic SIS actuation and subsequent reset, another automatic ESF actuation is prevented by which one of the following?

a. Placing affected pump controls in Pull-To-Lock, until the auto-actuate signal is no longer present.

b. Placing both trains of the affected ESF subsystems in manual.

c. The P-4 interlock, actuated by the opening of the reactor trip breakers.

d. The original SIS signal maintains a block signal on followup ESF signals.

Question 23
One Point

Which one of the following describes the operation of the High Head SI (CCPs) pumps during the Hot Leg Recirc phase of a LOCA recovery?

a. They are automatically secured.

b. They also discharge to the hot legs.

c. They continue to discharge to the cold legs.

d. They recirculate back to the sump.

Question 24
One Point

The plant is shutdown with one RCP ('C') running. RCS temperature is 400°F. The operators discover that RCS pressure does not respond to Pressurizer Spray Valve PCV-455C. Which one of the following is the MOST LIKELY cause of this?

a. RCP 'C' does not generate sufficient driving head for spray flow.

b. PCV-455C is the wrong valve, the correct choice is PCV-455B.

c. RCS pressure does not respond to spray flow below 450°F.

d. A nitrogen bubble in the pressurizer.

Question 25
One Point

A LOCA occurs which results in a high containment temperature. Which one of the following describes the effect that this event will have upon INDICATED pressurizer level?

a. INDICATED level accuracy should not be affected.

b. INDICATED will be lower than actual.

c. INDICATED will be higher than actual.

d. INDICATED level will trend higher until the reference leg is fully warmed then indicate normally.

Question 26
One Point

The unit is at full power with one Reactor Trip Bypass Breaker closed for SSPS testing. Which one of the following will be the consequence of racking in and closing the other Bypass Breaker?

a. The Reactor Trip and Bypass Breakers will open causing a reactor trip.

b. It is not possible to close a Bypass Breaker if the other one is already closed.

c. The Reactor Trip Breaker associated with the Bypass Breaker that you are closing, will open but the reactor will not trip.

d. Both Bypass Breakers will trip open. The Reactor Trip Breakers will be unaffected.

Question 27
One Point

The plant had been operating at 30% power when the following symptoms were observed:
→ Level in the 'A' steam generator increased rapidly then slowly returned to normal.
→ Level in the other steam generators followed the same pattern but their levels increased much less than 'A'.
→ The feed reg valves are more closed than before the transient started.
→ Feed pump speed is faster than before the transient.

Which one of the following is the most likely cause of this transient?

a. S/G 'A' Steam Flow failed low.

b. S/G 'A' Steam Flow failed high.

c. S/G 'A' Feed Flow failed low.

d. S/G 'A' Feed Flow failed high.

Question 28
One Point

The plant is operating at full power when a Loop 1 Wide Range T_c RTD fails upscale. This event will have which one of the following effects on indicated subcooling on the 'B' Train Core Subcooling Monitor?

a. No effect since only core exit thermocouples provide input.

b. No effect since the Subcooling Monitor is bypassed unless the reactor trip breakers are open.

c. Small decrease since the Subcooling Monitor uses the average of the Wide Range RTDs and the core exit thermocouples.

d. Large decrease since the Subcooling Monitor uses the highest reading Wide Range RTD or core exit thermocouple.

Question 29
One Point

The Spent Fuel Cooling System is in operation when a Loss of Offsite Power occurs. Assuming no operator action, which one of the following describes the expected status of Spent Fuel Cooling five minutes after the diesel generators have repowered their buses?

a. The pump is running with CCW flow to the Spent Fuel heat exchanger.

b. The pump is running but CCW to the Spent Fuel heat exchanger has been isolated.

c. No Spent Fuel Cooling pumps are running but there is CCW flow to the Spent Fuel heat exchanger.

d. No Spent Fuel Cooling Pumps are running and there is no CCW flow to the Spent Fuel heat exchanger.

Question 30
One Point

Refueling operations are in progress and the operators are preparing to raise a fuel assembly with an RCCA insert. Unnecessary Refueling Machine overload trips due to the extra weight of the RCCA, are avoided by which one of the following methods?

a. The RCCA is removed prior to raising the fuel assembly.

b. The overload trip is manually bypassed using a key switch.

c. The operator can manually select the setpoint to reflect the type of insert.

d. There is one setpoint but it is set high enough to allow for RCCA inserts.

Question 31
One Point

The unit is operating at power with blowdown in progress from all four steam generators. Which one of the following describes how the blowdown system will respond to a high radiation condition resulting from a primary to secondary leak in the 'B' steam generator?

a. Blowdown from the 'B' S/G will divert to radwaste.

b. Blowdown from only the 'B' S/G will be isolated.

c. Blowdown from the 'B' & 'C' S/Gs will be isolated.

d. Blowdown from all four S/Gs will be isolated.

Question 32
One Point

Which of the following are entry conditions for Technical Specifications while in Mode 1.

a. Plant inlet water temperature for the Ultimate Heat Sink is 85°F.

b. One Steam Generator Atmospheric Relief Valve has excessive seat leakage.

c. An open main steam line isolation valve is determined to be inoperable and cannot be closed.

d. Containment average air temperature has increased to 110°F.

Question 33
One Point

During a prolonged station blackout core damage is most likely to occur due to which one of the following?

a. Loss of heat sink due to exhaustion of feedwater inventory.

b. Loss of plant control due to loss of DC control power.

c. Loss of heat sink due to lack of ability to feed S/Gs.

d. Loss of RCS inventory due to RCP seal failure.

Question 34
One Point

The following conditions exist:
→ A loss of offsite power to NB01 has occurred.
→ The Shutdown Sequencer is currently re-energizing its loads following the start of NE01.

Which one of the following loads will not be started on the shutdown sequencer.

a. 'A' Essential Service Water Pump

b. 'A' Component Cooling Water Pump

c. 'A' Centrifugal Charging Pump

d. 'A' Safety Injection Pump

Question 35
One Point

A manually initiated CRVIS (CONTROL ROOM VENTILATION ISOLATION SIGNAL) will affect the operability of the Control Room Air Supply monitors (GK RE 04 & 05) in which one of the following ways?

a. Their operability is unaffected.

b. GK RE 04 becomes inoperable, GK RE 05 remains operable.

c. GK RE 04 remains operable, GK RE 05 becomes inoperable.

d. Both become inoperable.

Question 36
One Point

Loss of Service Water pressure will have which one of the following effects on the Fire Protection System?

a. The Jockey Pump will trip.

b. The Electric Fire Pump will auto start.

c. The Diesel Fire Pump will auto start.

d. Both the Electric and the Diesel Fire Pumps will auto start.

Question 37
One Point

To allow a reactor startup to continue the crew has written an OTSC to GEN 00-003. If the Shift Supervisor acts as the qualified reviewer, the SS cannot also _____ for an immediate use OTSC

a. have assisted in the drafting of the change

b. have directed the procedure change be made

c. be one of the background information reviewers

d. be one of the approvers.

Question 38
One Point

Which one of the following describes how CCW flow to the RHR heat exchangers is initiated following a large LOCA?

a. Automatically by the SIS.

b. Manually, as soon as possible following an SIS.

c. Automatically as part of the auto swap over to sump recirc.

d. Manually after swap over to sump recirc has occurred.

Question 39
One Point

A full load rejection has occurred and Steam Dump Demand (ABUI500) reads 65%. Which one of the following describes the expected steam dump valve configuration that exists at this time?

Group	1	2	3	4
a.	open	open	open	throttled
b.	open	open	throttled	closed
c.	open	throttled	closed	closed
d.	throttled	closed	closed	closed

Question 40
One Point

A rapid plant shutdown is in progress due to a loss of instrument air pressure. Reactor power level is 25% and decreasing. A reactor trip, if it occurs, will be most likely due to which one of the following?

a. Low Pressurizer Pressure.

b. High Pressurizer Pressure.

c. High Pressurizer Level.

d. High S/G Level Turbine Trip.

Question 41
One Point

The following conditions exist:

→ Power level is 80% and increasing.
→ Pressurizer level is following its program level.
→ T_{ave} is increasing above its program value.
→ Containment pressure and temperature are normal.
→ Pressurizer pressure is normal.
→ All systems are in their normal mode.
→ No operator actions have been taken.
→ Core life is near EOL.

Which one of the following is the most likely cause of these conditions?

a. Partial loss of feedwater heating.

b. Continuous rod withdrawal.

c. Steam leak outside containment.

d. Closure of 1 MSIV.

Question 42
One Point

With the plant operating at 50% power which one of the following describes the effect that a continuous rod withdrawal accident will have on rod insertion limits and shutdown margin?

a. The rod insertion limit remains the same and shutdown margin increases.

b. The rod insertion limit increases and shutdown margin remains the same.

c. The rod insertion limit remains the same and the shutdown margin remains the same.

d. The rod insertion limit increases and the shutdown margin decreases.

Question 43
One Point

During a startup, the connection between the drive mechanism for a high worth control rod and its RCCA becomes uncoupled. As a result, the drive mechanism withdraws normally but the RCCA associated with it remains fully inserted in the core. Which one of the following describes a symptom that the operators will see as a result of this event?

a. The affected rod bottom light will remain on.

b. The computer will show no rod motion for that rod.

c. Criticality will occur below the Insertion Limit.

d. The ECP will predict criticality at a lower than actual critical rod position.

Question 44
One Point

The following plant conditions exist:
→ The plant is at full power.
→ Rod control is in auto.
→ The selected turbine impulse pressure instrument fails downscale.
→ No control rod motion occurs.

Which one of the following would explain these events?

a. The rods should have stepped inward which suggests an auto rod control failure (in addition to the impulse pressure instrument).

b. The rods should not have moved since the low impulse pressure will also generate a block of auto rod motion.

c. The rods should not have moved since T_{ref} uses auctioneered high impulse pressure.

d. Auto motion demand depends upon which impulse pressure instrument failed since T_{ref} is selectable but the rod block always comes from PT-505.

Question 45
One Point

A Loss of All AC Power has occurred and the operators are conducting EMG C-0 (Loss of All AC Power) when Red Path conditions on Core Cooling arise. Which one of the following describes the proper response to this event?

a. Remain in C-0 since no other emergency procedures may be considered valid at this time.

b. Remain in C-0 and perform the necessary steps of the FR Series procedure concurrently.

c. Immediately transition to FR-C.1 since red paths always have priority under these conditions.

d. Remain in C-0 and perform an evaluation of other CSFST's before transitioning to FR-C.1.

Question 46
One Point

Many of the level setpoints found in the EOPs give one value followed by another in brackets[ ]. Under which one of the following conditions would you use the values inside the brackets?

a. If the observed value cannot be independently verified.

b. If containment radiation is more than twice normal full power background.

c. If containment pressure is presently 4.6 psig following a peak of 12 psig.

d. If containment pressure is presently 5.5 psig and increasing.

Question 47
One Point

The following conditions exist:
→ A large LOCA has occurred.
→ Containment pressure peaked at 35 psig.
→ Present containment pressure is 15 psig.
→ All safeguards equipment is functioning per design.

Which one of the following describes the response of the Containment Spray Pumps when the RWST reaches its lo-lo alarm point?

a. The pumps will continue to run and their suction remains aligned to the RWST.

b. The pumps will continue to run but their suction will auto swap to the RHR pump discharge.

c. The pumps will continue to run but their suction will auto swap to the containment sump.

d. The pumps will trip and their suction will auto swap to the containment sump.

Question 48
One Point

Partial failure of an RCP impeller has resulted in a decrease in flow in one loop. Which one of the following is the most likely result of this event?

a. Decreased pressure in affected loop S/G.

b. Decreased steam flow from affected loop S/G.

c. Increased temperature in affected loop S/G.

d. Decreased T_c in the affected loop.

Question 49
One Point

Which one of the following describes the expected response of the CVCS to emergency boration?

a. VCT level will rise.

b. Seal injection is isolated.

c. Seal return is diverted to the PRT.

d. Letdown is diverted to the RCDT.

Question 50
One Point

The plant is operating at full power when the operators note that CCW Surge Tank level is decreasing. A leak in which one of the following could be the cause of this?

a. RCP thermal barrier heat exchanger.

b. Seal water heat exchanger.

c. BTR3 Moderating heat exchanger.

d. Letdown heat exchanger.

Question 51
One Point

The reactor is critical at 10^{-8} Amps power when an inadvertent Emergency Boration occurs. RCS boron concentration is raised by 10 ppm before it is stopped. Which one of the following describes the effect that this event will have on T_{ave}?

a. It will decrease the most at BOL.

b. It will decrease the most at EOL.

c. It will decrease the same at EOL as at BOL.

d. It will not be significantly affected.

Question 52
One Point

A reactor trip has just occurred and the operators are conducting Step 1 of EMG E-0. Which one of the following conditions would require transition to EMG FR-S1 (RESPONSE TO NUCLEAR POWER GENERATION/ATWT)? For each case assume all other indications are as expected for a reactor trip.

a. All DRPI lights are dark.

b. Power range 8% and decreasing slowly.

c. Intermediate range stable at 10^{-10} Amps.

d. Reactor trip breakers closed.

Question 53
One Point

Which one of the following CSF Status Tree parameters requires entry into EMG FR-S1 (RESPONSE TO NUCLEAR POWER GENERATION/ATWT)?

a. Six control rods not fully inserting following a reactor trip.

b. The 'A' Reactor Trip Breaker fails to open.

c. Nuclear power 15% and decreasing slowly following a reactor trip.

d. Intermediate Range indicated power 10E-8 amps and decreasing following a reactor trip.

Question 54
One Point

During the conduct of EMG C-21 (UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS) the operators are cautioned that if any S/G narrow range level is less than 6%, a minimum of 30,000 lbm/hr must be maintained to that S/G. Which one of the following describes why this caution is necessary?

a. Assure uniform cooldown of all loops.

b. Prevent waterhammer in the feed line.

c. Maintain adequate steam supply to TD AFW pump.

d. Maintain pressure in the S/G to reduce tube sheet stress.

Question 55
One Point

The following plant conditions exist:
→ The plant has tripped on turbine trip/reactor trip from 100% power.
→ The turbine trip was caused by low vacuum.
Decay heat is being removed using the S/G ARV's set at 1125 psig.

Under these conditions the Steam Dumps will not function because of which the following signals?

a. P-4 signal is actuated.

b. P-12 permissive is energized.

c. C-9 control signal is deenergized.

d. C-7 control signal is deenergized.

Question 56
One Point

The operators are conducting EMG C-0 (LOSS OF ALL AC POWER) and no AC power has been restored. An SIS is then received. Which one of the following describes the actions required concerning the SIS?

a. Allow it to remain in effect to ensure rapid initiation when power is restored.

b. Allow it to remain in effect to provide an auto start signal for the diesel when it becomes available.

c. Reset the SIS to provide diesel engine cooling.

d. Reset the SIS to allow manual loading of equipment on an ESF bus when it becomes available.

Question 57
One Point

The following plant conditions exist:
→ RCS pressure is 1000 psig.
→ RCS temperature is 400°F.
→ All rods are fully inserted.
→ 24 hours ago the power source for Vital AC bus NN01 was swapped to MCC NG01A.

Assuming no changes to this lineup, which one of the following actions is required by Tech Specs?

a. Cooldown to 350°F within 6 hours

b. Cooldown to 200°F within 36 hours.

c. Cooldown to 200°F within 30 hours.

d. Cooldown to 350°F within 6 hours and to 200°F within the next 30 hours.

Question 58
One Point

The fire and smoke detection system has detected a fire in a diesel room. Which one of the following describes the effect that this will have on the operation of that diesel engine and its support systems?

a. The engine is prevented from starting in any mode.

b. The engine can only be started in its safeguards mode.

c. The fuel oil transfer pump will be shutdown and cannot be run until the fire signal is clear.

d. The fuel oil transfer pump is shutdown unless the engine is running.

Question 59
One Point

You are the shift supervisor when a fire in the control room makes control room evacuation necessary. Which one of the following actions can you expect the Balance Of Plant Operator to take concerning the pressurizer PORVs?

a. Remove DC control power from both PORVs without awaiting your direction to do so.

b. Standby to remove DC control power from both PORVs and do so at your direction.

c. Standby to close the block valves for both PORVs and do so at your direction.

d. Close the block valves for both PORVs without awaiting your direction to do so.

Question 60
One Point

In procedure EMG FR-Z1 (HIGH CONTAINMENT PRESSURE), when the Containment Spray pumps are verified running, the next step is to secure the RCPs. Which one of the following describes the reason it is necessary to stop the RCPs at this time?

a. Containment spray could damage a running RCP.

b. Prevent RCP seal failure due to loss of seal water flow.

c. Prevent RCP damage due to overheating.

d. To reduce the heat load on CCW.

Question 61
One Point

The following plant conditions exist:
→ RCS temperature is 180°F
→ An RCP motor is being removed from containment through the containment equipment hatch.
→ The 'A' Train of CCW is out of service for retubing of the CCW heat exchanger.
→ RCS cooling is provided by 'B' Train RHR.

The 'B' Train RHR pump then becomes inoperable and RCS temperature is seen to be rising at 20°F per hour. Which one of the following actions is required under these circumstances.

a. Cross connect 'B' Train CCW to the 'A' Train RHR heat exchanger and provide RCS cooling with 'A' Train RHR.

b. Connect Fire Water to the 'A' Train RHR heat exchanger and provide RCS cooling with the 'A' Train RHR.

c. Set containment closure before RCS temperature reaches 200°F.

d. Set containment closure before RCS temperature reaches 350°F.

Question 62
One Point

An Inadequate Core Cooling condition exists and core exit T/Cs are reading in excess of 1200°F. Which one of the following is the preferred action to reestablish core cooling under these conditions?

a. Rapid S/G depressurization.

b. Establish Safety Injection flow to the RCS.

c. Start an RCP.

d. Open Pzr PORVs.

Question 63
One Point

In response to an Inadequate Core Cooling Condition the operators are preparing to depressurize the S/Gs to atmospheric pressure in accordance with EMG FR-C.1. Prior to commencing this operation, they first secure all RCPs. Which one of the following describes why it is necessary to first secure the RCPs?

a. Minimize inventory loss through any primary break.

b. Avoid RCP damage due to cavitation.

c. Avoid RCP runout due to low mass steam water mixture.

d. Avoid RCP seal failure due to loss of seal D/P.

Question 64
One Point

Which one of the following is the basis for reducing T_{ave} to less than 500°F when the specific activity of the RCS is greater than 100/E-bar μCi/gm of gross radioactivity?

a. Elimination of CRUD bursts.

b. Elimination of the iodine spiking phenomenon.

c. Allows faster cleanup by the letdown mixed bed demineralizer.

d. Limits the release of activity should a steam generator tube rupture.

Question 65
One Point

Immediately following a reactor trip generated by the RPS, the operators note that there are Urgent Failures present in all power cabinets. There are no Non-Urgent Failures present. Which one of the following is the most likely reason that these conditions exist?

a. This is a normal indication resulting from the reactor trip.

b. One or more rods have failed to fully insert.

c. The Rod Drive MG Sets have failed to trip.

d. The Reactor Trip Breakers have failed to open.

Question 66
One Point

In response to a Reactor Trip and Safety Injection the operators are conducting EMG E-0. Step 21 directs the operators to 'Establish S/G Pressure Control'. Which one of the following describes how and why this must be done?

a. Place Steam Dumps in Tave Mode to prevent excessive RCS cooldown if the RCPs are lost.

b. Place Steam Dumps in Tave Mode since the steam pressure controller is not active until P-4 is reset.

c. Place Steam Dumps in Steam Pressure Mode since they will not open in Tave Mode under these conditions.

d. Place Steam Dumps in Steam Pressure Mode to prevent excessive cooldown if RCPs are lost.

Question 67
One Point

The crew is conducting EMG E-1, (LOSS OF REACTOR OR SECONDARY COOLANT) when the following conditions are observed.

→ RCS Pressure - 1700 psig
→ Containment Humidity - 100%
→ RCS Temperature - 547°F
→ Containment Rad Monitors - Alarming
→ Pressurizer Level - 100%

Which one of the following events is the most likely cause of these conditions?

a. Large break LOCA in cold leg

b. Steam line break inside containment

c. Pressurizer steam space leak

d. Stuck open pressurizer spray valve

Question 68
One Point

The operators are conducting EMG FR-C 1, (RESPONSE TO INADEQUATE CORE COOLING) and the following conditions are observed:

→ A LOCA due to a cold leg break has occurred.
→ Average of 5 highest Core Exit T/Cs - 610°F and stable
→ Hot leg Temp - 550°F
→ Steam generator pressure - 1005 psig and stable
→ RCS pressure - 1100 psig
→ Cold leg temp - 300°F
→ RVLIS Natural Circ Level - 10%

Which one of the following describes the core cooling mechanism?

a. Forced cooling

b. Natural circulation

c. Reflux boiling

d. Break flow

Question 69
One Point

Which one of the following parameters is the most effective in discriminating between a Small Break LOCA and a Steam Line Break Inside Containment?

a. Containment Pressure

b. RCS Pressure

c. Containment Temperature

d. Steam Generator Pressure

Question 70
One Point

The plant is operating at steady state power level of 50% when VCT level sensor LT-149F fails high. This will eventually result in which one of the following if no operator action is taken?

a. The charging pumps will trip on low suction pressure.

b. The VCT will overfill.

c. Control Rods will start stepping outward.

d. The operators will note an increase in VCT makeup rate.

Question 71
One Point

The following plant conditions exist:

→ The plant has been shutdown for 10 days after setting the world record for the longest full power run by a PWR.
→ No fuel has been replaced as yet.
→ The RCS is drained to mid loop.
→ The RCS is vented to atmosphere.
→ Heat removal is being provided by RHR.

Which one of the following is closest to the time it will take for the active fuel to become completely uncovered if RHR is lost.

a. 12 minutes

b. 31 minutes

c. 125 minutes

d. 260 minutes

Question 72
One Point

The following conditions exist:

→ The plant is in HOT STANDBY.
→ T_{ave} is 557°F.
→ One pressurizer PORV is open.
→ The backup heaters are on.

Which one of the following is the most likely cause of these conditions?

a. Controlling pzr level has failed low.

b. Controlling pzr level has failed high.

c. Controlling pzr pressure has failed low.

d. Controlling pzr pressure has failed high.

Question 73
One Point

The plant is at full power when a source range channel inadvertently energizes. All of the source range fuses are immediately removed. If they are not reinserted before the next shutdown, which one of the following problems will occur?

a. The reactor will trip when two power range instruments decrease below 10%.

b. The reactor will trip when one intermediate range instrument decreases below 10^{-10} Amps.

c. The reactor will trip when both intermediate range instruments decrease below 10^{-10} Amps.

d. It will not be possible to reenergize the source range unless at least two power range drawers are above 10%

Question 74
One Point

Twenty minutes after a reactor trip the following conditions are observed:
→ N35 - $1.0X10^{-11}$ AMPS.
→ N36 - $3.5X10^{-9}$ AMPS.

Which one of the following is the correct operator response?

a. Allow N36 to decrease below P6 and verify source ranges N31 and N32 energize.

b. Verify N31 energized and manually energize N32.

c. Manually energize N31 and N32 using both source range reset switches.

d. Verify both N31 and N32 energized below P10.

Question 75
One Point

The following steam generator leakrates are reported by the chemist.
→ S/G 'A' .35 GPM
→ S/G 'B' .10 GPM
→ S/G 'C' .20 GPM

Which one of the following describes the status of primary to secondary leakage relative to its Tech Spec limit?

a. The limits are already exceeded.

b. Not exceeded now but would be if S/G 'D' leakage was .15 GPM

c. Not exceeded now but would be if S/G 'D' leakage was .25 GPM

d. Not exceeded now but would be if S/G 'D' leakage was .45 GPM.

Question 76
One Point

The following conditions exist:
→ A S/G tube rupture has occurred.
→ Pressurizer level is 33%.
→ The operators are conducting step 45 of EMG E-3 (STEAM GENERATOR TUBE RUPTURE).
→ Neither letdown nor excess letdown can be established.

Which one of the following is the next step that they should perform?

a. Stop execution of the procedure until either letdown or excess letdown can be established.

b. Align charging pump suction to the VCT.

c. Return to step 1 of EMG E-3.

d. Return to EMG E-0 (REACTOR TRIP OR SAFETY INJECTION)

Question 77
One Point

Which one of the following will cause the auto start of both motor driven AFW pumps but will not start the turbine driven AFW pump?

a. Trip of both MFW pumps.

b. Lo-lo level on any two S/Gs

c. AMSAC

d. Loss of power on buss NB01.

Question 78
One Point

The following conditions exist:
→ A large feedwater leak has occurred.
→ The crew is conducting EMG E-1 (LOSS OF REACTOR OR SECONDARY COOLANT).
→ Conditions for a CSF GREEN Path on Subcriticality are satisfied.
→ Conditions for a CSF ORANGE Path on Core Cooling exist.

Which one of the following actions must be taken in response to this situation?

a. Immediately enter the appropriate FR-C procedure.

b. Concurrently complete E-1 and refer to the appropriate FR-C procedure.

c. Monitor the rest of the CSF Trees. If no red paths exist, exit E-1 and enter the appropriate FR-C procedure.

d. Monitor the rest of the CSF Trees while completing E-1. When E-1 is complete enter the FR procedure dictated by the highest priority status tree.

Question 79
One Point

Loss of Vital DC bus NK01 coupled with a trip of NB01 normal feeder breaker will affect Diesel Generator NE01 in which one of the following ways?

a. The diesel generator will start and load, but the fuel oil transfer pump will be deenergized and the diesel will stop when the day tank goes dry.

b. The diesel generator will not start because the loss of power to the air start solenoids.

c. The diesel generator will start but not load because NB01 DC control power has been lost.

d. The diesel generator will not start because the shutdown sequencer has lost DC power and will not send a start signal.

Question 80
One Point

Excess letdown is in service when instrument air pressure is lost. Which one of the following describes the effect that this event will have on excess letdown flowpath?

a. It is unaffected.

b. Diverts to PRT.

c. Diverts to RCDT.

d. It is isolated.

Question 81
One Point

With the plant operating at 20% power with all controls in automatic, which one of the following will cause ACTUAL pressurizer level to increase?

a. Adverse containment conditions

b. Leak in the reference leg of the controlling instrument.

c. A loop T_h RTD fails high.

d. A loop T_c RTD fails low.

Question 82
One Point

In MODE 6 there must be a minimum of 23 feet of water above irradiated fuel in the reactor vessel. Which one of the following is the basis for this requirement?

a. To provide shielding for personnel on the refueling bridge.

b. To provide adequate iodine removal in the event of a cladding rupture.

c. To provide sufficient subcooling to assure adequate natural circulation cooling.

d. To provide adequate suction head for the RHR pumps.

Question 83
One Point

A loss of all AC power has occurred and the operators are conducting EMG C-0 (LOSS OF ALL AC POWER). If midway through the execution of C-0, AC power is restored, which one of the following pumps can be expected to start without operator action?

a. Charging

b. Safety Injection

c. Component Cooling Water

d. Essential Service Water

Question 84
One Point

Using which one of the following to verify the current revision of a procedure will result in a violation of procedure use regulations?

a. Computer to access Configuration Status Accounting Records (CSAR) System.

b. Document Services: Controlled Document Look-up on paperless environment.

c. The Control Room may use the Shift Supervisor's copy for verification.

d. Check that procedure is the same revision as the last copy used.

Question 85
One Point

During use of procedures, verification should be made that identification tags are in place and in good condition. Which one of the following actions should be taken if an identification tag is missing?

a. Initiate an engraving request.

b. Initiate an On-The-Spot-Change.

c. Initiate a Document Revision Request.

d. Obtain Shift Supervisor's approval before proceeding.

Question 86
One Point

Which one of the following violates the administrative limits on overtime?

a. Working 16 hours in 24

b. Working 24 hours in 48

c. Working 20 hours in 48 with 8 hours between shifts.

d. Working 80 hours in 5 consecutive days with 8 hour between shifts.

Question 87
One Point

The ENS phone system, FTS 2000 line, has failed. The NRC must be notified of this with a/an _____ using a backup line.

a. Four hour notification.

b. One hour notification.

c. Immediate notification.

d. Six hour notification.

Question 88
One Point

During a certain transient the SRO gives a complex direction to the RO. Before performing the necessary actions, the RO should first do which one of the following?

a. Offer a clear acknowledgment that the instruction was understood.

b. Repeat back the instruction.

c. Repeat back the instruction and wait for affirmation from the SRO.

d. Repeat back the instruction and make any appropriate logbook entry.

Question 89
One Point

A containment depressurization was initiated but a log entry was not made for several hours. Which one of the following is the correct way for this log entry to be made?

a. Insert the entry such that the sequence of events on the page is correct.

b. Make a normal log entry with the time the action was actually performed and marking it as a late entry.

c. Make a normal log entry and then draw an arrow to the page location where it should have been entered initially.

d. Make a normal log entry but include both the time of logbook entry and the time the action was actually performed.

Question 90
One Point

The following conditions exist:
→ It is discovered that a surveillance has been missed on a certain piece of equipment.
→ The discovery was made exactly one week after the expiration of the surveillance's late date.
→ The Tech Spec action statement associated with the equipment allows it to be out of service for 72 hours before requiring a plant shutdown.

In addition to declaring the equipment inoperable, the Shift Supervisor must direct which one of the following courses of action?

a. Start the Action Statement clock as of the time of discovery of the missed surveillance.

b. Start the Action Statement clock in 24 hours.

c. Direct an immediate orderly shutdown. The shutdown may be terminated or the plant restarted when the surveillance is completed.

d. Direct an immediate orderly shutdown. The plant may restarted after a formal review by the PSRC.

Question 91
One Point

The plant is operating at full power when Plant Chemistry reports the following concentrations in the RCS:

→ Dissolved Oxygen 0.12 ppm
→ Chloride 0.11 ppm
→ Fluoride 0.14 ppm

Assuming no changes in these concentrations, the plant must be taken to HOT STANDBY within which one of the following times?

a. None - Power operation may continue indefinitely.

b. 30 hours.

c. 24 hours.

d. 6 hours.

Question 92
One Point

A General Emergency has occurred and an offsite release of radioactive material seems likely. Which one of the following is authorized to initiate evacuation of the subzones identified by the dose assessment?

a. USNRC

b. Plant Manager

c. Duty Emergency Director

d. Coffey County

Question 93
One Point

If a locked valve must be operated during performance of a Clearance Order, which one of the following is required?

a. Shall be documented in APF 21G-001-01, Log of Locked Component Manipulations.

b. If the maintenance work group is authorized to do so by the Work Week Manager, they may manipulate the valve within the boundaries of the Clearance Order.

c. Need not be documented if it is clearly identified on the clearance order and restored to the locked position and is second verified.

d. Need not be documented if it is clearly identified on the clearance order and restored to the locked position, second verification is not required.

Question 94
One Point

During full power operation you note the following computer point readings:
→ SCU1118 (Rx Total Thermal Q LCK) is 3600 MWt.
→ RJU157MA (Rx Pwr 10 min. Mov. Avg) is 3595 MWt.
→ The cause of this is traced to a small decrease in grid frequency.

Which one of the following actions is required as a result of these conditions?

a. No action required, the readings are within the plus or minus 1% tolerance required for thermal power.

b. Adjust power to assure that computer point RJU158MA (moving hour average) does not exceed 3565 MWt.

c. Adjust power to assure that computer point RJU159MA (moving shift (8) hour) does not exceed 3565 MWt.

d. The terms of the Operating License have been violated. Commence an immediate shutdown and notify USNRC.

Question 95
One Point

Four individuals received the following dose equivalents for the year.

→ Individual	A	B	C	D
→ Eff. Dose Equiv. (TEDE)	4R	6R	3R	2R
→ Total Organ Dose Equiv. (TODE)	30R	13R	15R	60R
→ Lens Dose Equivalent (LDE)	12R	18R	20R	10R

Which one of these individual's dose equivalent is within allowable limits per 10CFR20?

a. A

b. B

c. C

d. D

Question 96
One Point

A certain task must be performed under the following conditions:
→ Weighted DAC-hours (Derived Air Concentration) = 3
→ External Doserate = 90 mrem/hr
→ Time to complete the task while wearing a respirator = 3.5 hrs.
→ Time to complete the task without wearing a respirator = 3 hrs.
→ 1 DAC-hour = 2.5 mrem
→ The respirator may be considered 100% efficient.

Which of the following is the correct statement regarding the wearing of a respirator while performing this task?

a. There is no choice, a respirator must be worn anytime that airborne radiation is detected.

b. Wearing a respirator will decrease total exposure.

c. Wearing a respirator will increase total exposure.

d. Wearing a respirator will make no difference to total exposure.

Question 97
One Point

Containment is posted LHRA at the entrance and a Security Officer has been stationed at the personnel hatch. Which one of the following containment entry rules is in effect?

a. No containment entry is allowed.

b. All containment entry is limited to 30 minutes.

c. The only containment entry allowed is for HP survey.

d. All persons entering containment must be accompanied by an HP Technician.

Question 98
One Point

A continuous fire watch shall be established when any boundary to the ESF Switchgear Rooms or Rod Drive Motor/Generator Sets Rooms is breached. Which one of the following is the reason for this procedural requirement?

a. To maintain security control over all vital areas.

b. Is an IEEE requirement.

c. The ability to concentrate halon on a potential fire may have been reduced.

d. The ability to concentrate CO_2 on a potential fire may have been reduced.

Question 99
One Point

You are Emergency Rescue qualified and have been stationed as the attendant outside a confined space while another person makes an entry. Which one of the following describes when you may enter the confined space to effect a rescue should that become necessary?

a. When replaced by another qualified attendant.

b. After notifying the control room.

c. After the Emergency Rescue Team arrives.

d. Immediately if necessary to prevent death or serious injury.

Question 100
One Point

Which one of the following is permitted to be a member of the Fire Brigade?

a. Control Room RO.

b. Shift Supervisor.

c. Shift Chemist.

d. Security Personnel.

Question	Answer	Question	Answer	Question	Answer
1	A	35	D	68	C
2	D	36	B	69	D
3	A	37	D	70	C
4	C	38	D	71	D
5	C	39	B	72	C
6	B	40	4/24/97 BC	73	C
7	B	41	B	74	C
[illegible]	D	42	B	75	A
9	C	43	D	76	B
10	C	44	A	77	A
11	D	45	A	78	C
12	A	46	D	79	B
13	A	47	A	80	D
14	B	48	B	81	C
15	A	49	A	82	B
16	D	50	B	83	D
17	D	51	D	84	D
18	B	52	B	85	A
19	B	53	C	86	D
4/22/97 [illegible] A		54	B	87	B
21	A	55	C	88	C
22	C	56	D	89	B
23	C	57	C	90	A
24	A	58	D	91	B
25	C	59	A	92	D
26	A	60	C	93	C
27	B	61	C	94	C
28	D	62	B	95	A
29	C	63	D	96	C
30	C	64	D	97	D
31	D	65	A	98	C
32	C	66	D	99	A
33	D	67	C	100	C
34	D				

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE SUITE 400
ARLINGTON, TEXAS 76011-8064

MAY - 2 1997

Otto L. Maynard, President and
Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas 66839

SUBJECT: NRC INSPECTION REPORT 50-482/97-03

Dear Mr. Maynard:

An NRC inspection was conducted April 14-17, 1997, at your Wolf Creek Generating Station reactor facility. The enclosed report presents the scope and results of that inspection.

The inspection included an evaluation of four applicants for senior reactor operator licenses. We determined that all four applicants satisfied the requirements and the licenses have been issued.

The operating test portion of the examination submitted by your staff was inadequate for administration prior to incorporation of numerous changes suggested by the chief examiner. These changes involved revision or replacement of all eight questions in the administrative topics portion and nine of ten questions related to the control room systems and facility walkthrough portion of the examination to achieve a level of difficulty which would discriminate at the required knowledge and skill level. These revisions were necessary to satisfy the NRC's individual test item quality expectations, consistent with the NRC's Examination Standards, which had been recently clarified to the regional staff following an internal audit. We regret the impact this imposed on your staff, but at the same time, we commend them for their prompt and effective efforts to enhance the quality of the examinations.

In order to more clearly communicate NRC's expectations, which are articulated in Generic Letter 95-06, "Changes in the Operator Licensing Program," and NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," Interim Revision 8, we will be hosting a workshop on facility licensee-developed examinations on May 29-30, 1997. The details of this workshop were provided in separate correspondence.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR).

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IE42

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Arthur T. Howell III, Director
Division of Reactor Safety

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

Enclosure:
NRC Inspection Report
50-482/97-03

cc w/enclosure and Attachments 1-2:
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E-Mail report to T. Boyce (THB)
E-Mail report to NRR Event Tracking System (IPAS)
E-Mail report to Document Control Desk (DOCDESK)

bcc to DCD (IE01)

bcc distrib. by RIV w/Enclosure and Attachments 1-2:

Regional Administrator	Resident Inspector
DRP Director	SRI (Callaway, RIV)
Branch Chief (DRP/B)	DRS-PSB
Project Engineer (DRP/B)	MIS System
Branch Chief (DRP/TSS)	RIV File

bcc distrib. by RIV w/Enclosure and Attachments 1-3:
S. Richards (HOLB/NRR)
L. Hurley

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-482
License No.:	NPF-42
Report No.:	50-482/97-03
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility:	Wolf Creek Generating Station
Location:	1550 Oxen Lane, NE Burlington, Kansas
Dates:	April 14-17, 1997
Inspectors:	H. Bundy, Chief Examiner M. Murphy, Examiner
Approved By:	J. L. Pellet, Chief, Operations Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1:	Supplemental Information
Attachment 2:	Simulation Facility Report
Attachment 3:	Final Written Examination and Answer Key

EXECUTIVE SUMMARY

Wolf Creek Generating Station
NRC Inspection Report 50-482/97-03

NRC examiners evaluated the competency of four senior reactor operator upgrade license applicants for issuance of operating licenses at the Wolf Creek Generating Station facility. The licensee developed the initial license examinations using the pilot process program guidance contained in Generic Letter 95-06 and NUREG-1021, Supplement 1, "Operating Licensing Examiners Standards." NRC examiners reviewed, approved, and administered the examinations. The initial written examinations were administered to all four applicants on April 14, 1997, by facility proctors in accordance with instructions provided by the chief examiner. The NRC examiners administered the operating tests on April 15-16, 1997. All of the applicants displayed the requisite knowledge and skills to satisfy the requirements of 10 CFR 55 and were issued senior reactor operator licenses.

Operations

- All four applicants passed the senior reactor operator written examination. No broad knowledge or training weaknesses were identified as a result of evaluation of the graded examinations.

- All four applicants passed the operating test. A strength was identified in the area of event classification and protective action recommendations.

- The licensee submitted an examination outline which was adequate for examination development.

- The written examination, dynamic scenarios, and administrative and system and facility walkthrough tasks were acceptable for administration as submitted. The administrative and system walkthrough tasks were of high quality. It was necessary for the licensee to upgrade the administrative portion questions and the system walkthrough followup questions in accordance with the chief examiner's recommendations to meet minimum requirements.

- The simulation facility supported the examination administration well.

Report Details

Summary of Plant Status

The plant operated at essentially 100 percent power for the duration of this inspection.

I. Operations

03 Operations Procedures and Documentation

An apparent deficiency in Procedure OFN EJ-015, "Loss of RHR Cooling," Revision 4, is discussed in Section O5.1.2b.

04 Operator Knowledge and Performance

O4.1 Initial Written Examination

a. Inspection Scope

On April 14, 1997, the facility licensee proctored the administration of the written examination approved by the chief examiner and NRC Region IV supervision to four individuals who had applied for initial upgrade senior reactor operator licenses. The licensee graded the written examinations and the staff reviewed its results. The licensee also performed a post-examination question analysis which was reviewed by the examiners.

b. Observations and Findings

The minimum passing score was 80 percent. The scores for the applicants ranged from 83.8 to 91.9 percent, with an average score of 85.8 percent. More than half the applicants missed Questions 7, 9, 20, 21, 30, 40, 42, 51, and 79. The licensee initiated Performance Improvement Requests 971145 and 971142 to investigate the causes for excessive applicant error rate on these questions and any associated training or operational issues. The chief examiner reviewed the preliminary results of these investigations and as a result determined that Question 20 had no correct answer and that the correct answer on Question 40 was "C" instead of "B". The examinations were regraded based on this determination. The licensee was performing additional analysis on Questions 7, 9, and 79 with regard to question construction and the range of assumptions the applicants may have used in answering the questions. However, because the applicants had no clarification questions on these test items and the answer key provided the correct answers for reasonable assumptions, the chief examiner concluded that these questions were valid test items. The licensee determined that the four remaining questions (21, 30, 42, and 51) were technically accurate and valid job linked test items and that the answer key provided the correct answers. No broad training or knowledge weaknesses were identified. Reasons for missing the valid questions

appeared to be related to question difficulty and isolated training weaknesses. Overall, the licensee concluded that the questions on the written examination had the requisite job applicability and technical correctness and were covered in the training program. The licensee found the examination valid as administered and the chief examiner concurred with this determination.

c. Conclusions

All four applicants passed the senior reactor operator written examination. No broad knowledge or training weaknesses were identified as a result of evaluation of the graded examinations.

O4.2 Initial Operating Test

a. Inspection Scope

The examination team administered the various portions of the operating examination to the four applicants on April 15-16, 1997. Each applicant participated in three dynamic simulator scenarios. Each also received a walkthrough test which consisted of five system tasks together with two followup questions for each system and five subjects in four administrative areas. Three of the administrative areas were covered by questions and the remaining administrative area was covered by an administrative task.

b. Observations and Findings

All applicants passed all portions of the operating test. During the dynamic simulator scenarios, there was one instance in which the applicant in the supervising operator position directed an unnecessary plant trip as a result of a failed steam generator level instrument because the applicant in the balance of plant operator position reported an excessive steam demand. After the trip, the balance of plant operator discovered and reported that he had misread increasing feedwater flow as increasing steam flow and that there was no excessive steam demand. The chief examiner determined that the position of these two indicators had been reversed in the plant in March 1996 and in the simulation facility in September 1996. The applicant described this modification in a post-scenario interview and stated that he had remembered the previous configuration when confronted with the instrument failure. Although this trip challenged the reactor protection system, the plant was not placed in an unsafe condition and the overall safety significance was minimal. Except for this one error, the applicants performed well in the dynamic scenarios. Communications were clear and observed to be effective and did not deteriorate when problems were encountered.

All the applicants performed well on the walkthrough portion of the test. There were no unsatisfactory grades for any applicant on any part of the test. One administrative task required the applicants to observe a series of events involving a

steam generator tube rupture followed by a main steam line break on the dynamic simulator and then classify the event level and provide protective action recommendations. Because all applicants performed this task flawlessly, it was considered an applicant strength.

c. Conclusions

All four applicants passed the operating test. A strength was identified in the area of event classification and protective action recommendations. A performance error in one of the dynamic simulator scenarios not sufficient for license denial was identified for licensee and applicant consideration and corrective action as appropriate.

O5 Operator Training and Qualification

O5.1 Initial Licensing Examination Development

The facility licensee developed the initial licensing examination in accordance with guidance provided in Generic Letter 95-06, "Changes in the Operator Licensing Program."

O5.1.1 Examination Outline

a. Inspection Scope

The facility licensee submitted the initial examination outline on February 18, 1997. The chief examiner reviewed the submittal against the requirements of NUREG-1021, "Licensed Operator Examiner Standards," Revision 7, Supplement 1, and NUREG/BR-0122, "Examiner's Handbook for Developing Operator Licensing Written Examinations," Revision 5.

b. Observations and Findings

The initial examination outline was satisfactory as a guide for development of the examination. However, the chief examiner provided several enhancement suggestions for licensee consideration. Some of the suggestions related to the scope and depth of the administrative portion of the examination. The licensee made some revisions on the final submittal in response to these suggestions. It appeared that Job Performance Measures 1 and 2 were both closely related to indirect radioactivity release control, although Job Performance Measure 1 was referenced to reactor coolant system inventory control. The standards require that all job performance measures relate to different safety functions for this class of applicants. The licensee replaced Job Performance Measure 2 in response to this comment. The chief examiner also provided several enhancement suggestions on the scenario outline, which were addressed by the licensee on the final submittal.

c. Conclusions

The licensee submitted an examination outline which was adequate for examination development. Several enhancement suggestions provided by the chief examiner were incorporated in the final submittal.

O5.1.2 Examination Package

a. Inspection Scope

The facility licensee submitted the completed examination package on March 17, 1997. The chief examiner reviewed the submittal against the requirements of NUREG-1021, "Licensed Operator Examiner Standards," Revision 7, Supplement 1, and NUREG/BR-0122, "Examiner's Handbook for Developing Operator Licensing Written Examinations," Revision 5.

b. Observations and Findings

The draft-written examination contained 100 new questions. The draft examination was responsive to the knowledge and abilities sample plan submitted on February 18, 1997, technically valid, and discriminated at the proper level. It was considered adequate for administration. The chief examiner provided comments on construction for 25 questions. Another 11 comments related to question references. The comments generally related to the suitability of question distractors, construction of the question stem, or accuracy of the reference to the regulation item. In response to the chief examiner's comments, the licensee revised or replaced 19 questions. Several of the revisions were minor editorial changes. Three of the questions commented on by the chief examiner were replaced because of similarity to a question on the audit examination. One other question was also replaced because of similarity to another question on the licensee's final audit examination to assure applicant readiness. The chief examiner concurred with resolution of the comments and the final product. As discussed in Section O4.1b, as a result of the licensee's post examination analysis and further review, the chief examiner deleted Question 20 and changed the key answer on Question 40.

The licensee submitted five dynamic scenarios, including one backup scenario which was not used during the examination. The submitted scenarios were considered adequate for administration. The licensee subsequently developed detailed expected operator action lists in accordance with the standards as requested by the chief examiner. Also, Scenario 4 was enhanced as suggested by the chief examiner to make it more realistic and more challenging. During onsite preparation, the chief examiner observed that the normal events in Scenario Three and the backup scenario were not challenging enough to permit reliable evaluation of the applicants. They were essentially surveillance checks of control room indications. In response to these observations, the licensee replaced them with more performance-oriented evolutions.

To support the systems walkthrough section of the operating test, the facility licensee provided job performance measures developed to evaluate selected operator tasks that contained well-written task elements, performance standards, and comprehensive evaluator cues which were acceptable for administration as submitted. To facilitate grading, the licensee designated two additional steps as critical in Job Performance Measure 4, as suggested by the chief examiner.

Two followup questions associated with each task were also submitted by the licensee. The followup questions were generally unacceptable for administration as submitted. Most of the questions were written at the fundamental knowledge level and could not be expected to discriminate between a competent and less than competent senior reactor operator. All except Question 2 associated with Job Performance Measure 3 were either replaced or upgraded by the licensee prior to administration in accordance with the chief examiner's recommendations to meet minimum requirements. For example, Job Performance Measures 3, 4, and 5, Question 1, were direct lookup questions. Question 4-1 asked the applicant when, in the emergency procedures, it was allowed to stop immediate boration of the reactor coolant system. Since the answer is explicitly listed in the procedure that would be used, properly answering this question required only reading the in-use procedure. This was considered a direct lookup question, which did not require higher cognitive skills, and to discriminate at too low a level. During administration it was discovered that the answer key was wrong for Job Performance Measure 1, Question 1. After research at the request of the chief examiner, the licensee discovered that the formula contained in Procedure OFN EJ-015, "Loss of RHR Cooling," Revision 4, for calculating the final accumulator pressure following a safety injection was not clearly written. The licensee initiated Performance Improvement Request 97-1135 to address this problem. Because all applicants correctly performed this calculation, it appeared that training had compensated for a lack of procedure clarity.

The licensee submitted a job performance measure and questions to cover the administrative section of the walkthrough test. The job performance measure, relating to plant emergency classification and protective action recommendations, was high quality. However, none of the questions were suitable for administration as submitted. A wrong answer was given for one question. The chief examiner evaluated the remaining questions as discriminating at too low a level for senior reactor operator applicants. Several were direct lookup questions, which were contrary to the guidance specified in the examiner standards. For example, administrative questions A1-1, A1-2, A1-3, and A1-4 were either direct lookup questions or discriminated at too low a level. Question A1-1 asked the applicant to describe how one would determine if members of a fuel movement team were qualified. This could be answered by reviewing the licensees administrative procedures for the one titled, "Fuel Handling Equipment Qualifications," which describes personnel qualification. Properly answering this question required reading the index of administrative procedures. This was considered a direct lookup question, which did not require higher cognitive skills, and to discriminate at too low

a level. All questions were replaced or upgraded by the licensee prior to administration in accordance with the chief examiner's recommendations to meet minimum requirements.

c. Conclusions

The written examination, dynamic scenarios, and administrative and system and facility walkthrough tasks were acceptable for administration as submitted. The administrative and system walkthrough were of high quality. However, it was necessary for the licensee to upgrade the questions associated with the administrative portion and upgrade the system walkthrough followup questions in accordance with the chief examiner's recommendations to meet minimum requirements.

O5.2 Simulation Facility Performance

a. Inspection Scope

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

b. Observations and Findings

The simulation facility supported examination administration well. No deficiencies affecting examination administration were identified. One deficiency involving the main generator load rate circuit was identified during preparation week and is discussed in Attachment 2.

c. Conclusions

The simulation facility supported examination administration well.

V. Management Meetings

X1 Exit Meeting Summary

The examiners presented the inspection results to members of the licensee management at the conclusion of the inspection on April 17, 1997. The licensee acknowledged the findings presented.

The licensee did not identify as proprietary any information or materials examined during the inspection.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Damashek, Supervisor, Licensing
R. Flannigan, Manager, Nuclear Engineering, Safety, and Licensing
R. Guyer, Superintendent, Operations Training
S. Hatch, Instructor, Training
R. Hubberd, Superintendent, Operations
O. Maynard, President and Chief Executive Officer
B. McKinney, Plant Manager
T. Morrill, Assistant to Vice President, Engineering
D. Parks, Training Supervisor
J. Pippin, Manager, Training
G. Smith, Senior Instructor, Training
C. Warren, Vice President Operations, Chief Operating Officer
C. Younie, Manager, Operations

NRC

F. Ringwald, Senior Resident Inspector

ATTACHMENT 2

SIMULATION FACILITY REPORT

Facility Licensee: Wolf Creek Nuclear Operating Corporation

Facility Docket: 50-482

Operating Examinations Administered at: Wolf Creek Generating Station

Operating Examinations Administered on: April 15-16, 1997

These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility, other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

No simulator performance problems affected examination administration. One simulator fidelity deficiency was identified during preparation week. While performing a load decrease using the load rate circuit, the average rate decrease was 0.5 percent every 2.5 minutes instead of the set 0.5 percent per minute. Simulator Deficiency Report 97-045 was issued for followup of this problem.

ATTACHMENT 3

FINAL WRITTEN EXAMINATION AND ANSWER KEY