EXAMINATION REPORT

Facility Licensee: Kansas Gas and Electric Company
Wolf Creek Generating Station
P. O. Box 208
Wichita, Kansas 67201

Facility Docket No.: 50-482

Facility License No.:

Initial operator examinations at Wolf Creek Generating Station near Burlington, Kansas

Chief Examiner: John L. Pellet
John L. Pellet — 7-2-84
Date Signed

Approved by: R. A. Cooley
R. A. Cooley, Section Chief — 7-2-84
Date Signed

Summary

Initial operator license examinations at Wolf Creek in May 1984

Initial operator license examinations were conducted at Wolf Creek during the weeks of May 14 and 21. Twenty-seven persons were examined for SRO and RO licenses. Eighteen of the twenty-three SRO candidates passed the written, oral, and simulator examinations. All four RO candidates passed the written, oral, and simulator examinations.

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WCGS EXAMINATION REPORT

Report Details

1. Persons Examined

   SRO Licensees:
   Twenty-three candidates were examined for an SRO license.

   RO Licensees:
   Four candidates were examined for an RO license.

2. Examiners - May 14-17, 1984

   R. Cooley, NRC
   J. Pellet, NRC (Chief Examiner)
   S. McCrory, NRC
   L. Brooks, EG&G Idaho, Inc.
   G. Jeffries, EG&G Idaho, Inc.

3. Examiners - May 21-24, 1984

   J. Pellet, NRC
   R. Gruel, Battelle-Pacific Northwest Labs
   L. Defferding, Battelle-Pacific Northwest Labs

This Examination Report is composed of the sections listed below.

A. Examination Review Meeting Comment Resolution
B. Meeting/Visit Summaries
   1. May 14-17 Site Visit Exit Meeting Summary
   2. May 21-24 Site Visit Exit Meeting Summary
C. Generic Comments
D. Examination Master Copy (SRO/RO Questions and Answers)

Performance results for individual candidates are not included in this report because, as noted in the transmittal letter attached, examination reports are placed in NRC's Public Document Room as a matter of course.

A. Examination Review Meeting Comment Resolution

In general, editorial comments or changes made during the exam, the exam review, or subsequent grading reviews are not addressed by this resolution section. This section reflects resolution of substantive comments made during the exam review. The modifications discussed below are included in the master exam key which is provided elsewhere in this report as are all other changes mentioned above but not discussed herein. The following personnel were present for the exam review:

NRC	UTILITY
L. Brooks	F. Scheimann
G. Jeffries	M. Herrell
	L. Baker
	W. Hunter

1. Q.2.04a Accept 'low flow' or 'minimum flow' as well as shutoff head.
   Resp.: ACCEPT.

2. Q.2.05a Accept 80 psig +/- 10 psig - variable press. system.
   Resp.: ACCEPT.

3. Q.2.07c Allow +/- 5 v band for inverter voltages due to design/instrument variation.
   Resp.: ACCEPT.

4. Q.3.02.a Add SIS signal to list which will initiate per SNP-AF-4 figure, NPS 223, chapter 5.
   Resp.: ACCEPT.

5. Q.4.02 May see 'immediate' for 'emergency' boration.
   Resp.: ACCEPT.

6. Q.4.04.a May also answer 300,1000,2000 with approval.
   Resp.: ACCEPT.

7. Q.4.04.c May also see answer 'blanket RWP.'
   Resp.: ACCEPT.

8. Q.4.06a May see 'ECCS' for 'CCP,SI.'
   Resp.: ACCEPT.

9. Q.4.06.b Add 'Figure 1' after 'RCS subcooling LT req'd.'
   Resp.: ACCEPT.

10. Q.4.06.b May not get 330# RHR press since not part of SI act.
    Resp.: No. Question specifically asks for it.

11. Q.4.08.c Correct ans. is EOF per ADM-12-6.0, sec. n 6.2, p. 1.
    Resp.: ACCEPT.

12. Q.5.08.b For upper core DNBR increases.
    Resp.: ACCEPT. Only if answer states upper core.

13. Q.5.08.d For uniform crud buildup DNBR decreases (crud slows HT) but for roughened surface causes turbulent flow which increases DNBR.
    Resp.: ACCEPT. Only if answer states assumption.

14. Q.5.08 Answer may be in terms of 'subcooling.'
    Resp.: ACCEPT. if complete and accurate.

15. Q.6.02 Answers may assume failures that challenge heat sink such as failure of steam dumps as part of answer
    Resp.: ACCEPT. Only if assumptions are explicitly stated and description correct for stated assumption. If assumption not stated then deduct 0.5 and grade rest accordingly.

16. Q.6.07.b Change 17.2% to 32.3% per T.S. p. 2-6.
    Resp.: ACCEPT.

17. Q.6.07.c Add SG Lo-Lo Level per dwg 7250D64, SH-13,15.
    Resp.: ACCEPT.

18. Q.6.08 Accept Generator L/O relay or differential relay since diff. relay is the only DG L/O in the emergency mode per NPS Chapter 3, p. 3-1 and Ch. 4, p. 4-22.
    Resp.: ACCEPT.

19. Q.7.01 Accept CEA failure to insert or SDM limit not met as requiring emergency boration.
    Resp.: ACCEPT. For one answer only - CEA for item 1, SDM for item 2.

20. Q.7.01 Accept flowpath through the BIT per proc. FR-51.
    Resp.: ACCEPT.

21. Q.7.02 1,3, & 6 are correct for a & b.
    Resp.: ACCEPT. Now 9 ans. @ 0.33 ea. For add'l incorrect ans. divide by total #.

22. Q.7.05.a Add as correct answer 'Withdraw SD banks (0.5) to provide SD reactivity (if criticality achieved during dilution) (0.5)'
    Resp.: ACCEPT.

23. Q.7.05.d Accept SDM related answers.
    Resp.: ACCEPT. For 1/2 credit only.

24. Q.7.05.d Add to d.1, 'except init. crit./orig. & after refuel.'
    Resp.: ACCEPT. Accept 'none' as ans. w/ above - not req'd w/ original answer.

25. Q.7.08 Accept as correct answers: man. crane hoist cutout on overload, man. crane interlock prevents movement unless gripper is engaged or disengaged, man. crane interlock prevents movement unless properly positioned over core or xfer. sys., gripper remains engaged on loss of air, man. crane lateral movement restricted by cutouts, crane cannot be moved laterally unless gripper fully raised, upender cannot be lowered unless man. crane is positioned away from upender or gripper raised, loss of power to hoist motor engages brake, gripper interlocked to prevent release unless lowered in vessel or xfer. sys., cable overhauling hoist interlock with brake, long/short fuel handling tool in SFP area positive locking.
Resp.: ACCEPT. Per NPS-219 Chapter 4 & Bechtel doc. 10466-M-OOKE (Q), Fuel Storage, Fuel Handling, and Reactor Servicing System SNUPPS, p. 29-30.

26. Q.7.08.c Add ans 'place items in xfer. in a safe condition'.
Resp.: ACCEPT.

27. Q.8.01 Accept 'comply w/ Admin T.S. 6.7.1' as correct ans.
Resp.: ACCEPT.

28. Q.8.04 Accept '8 gpm/RCP controlled leakage' for '32 gpm'.
Resp.: ACCEPT.

29. Q.8.07.a Add 'or his designee' per ADM 10-007, p.1.
Resp.: ACCEPT.

30. Q.8.07.b Add 'SS or his designee' per ADM 10-007, p. 1 & 'Ops. Super.' per ADM 01-004, p. 2.
Resp.: ACCEPT.

31. Q.8.07.d Add horizontal support provided by transfer car during movement from FB to cont. per NPS-219, p. 4-31/33.
Resp.: ACCEPT.

32. Q.8.09 Accept 'Shop Building' for 'OSC' per EP p. 4.1-3.
Resp.: ACCEPT.

B. MEETING/VISIT SUMMARIES

B.1. May 14-17 Site Visit Exit Meeting Summary

At the conclusion of the site visit examiners met with representatives of the plant staff to discuss the results of the examinations. The following personnel were present for the exit interview:

NRC	UTILITY
R. A. Cooley	W. T. Hensley
S. L. McCrory	R. L. Hoyt
J. L. Pellet	C. W. Russo
	O. Maynard
	P. E. Turner
	R. Trevillian
	S. Hatch
	A. S. Mah
	M. G. Williams
	G. L. Koester
	C. Steinert
	F. T. Rhodes

Mr. Cooley started the discussion by noting that the examiners as a group, had encountered a positive, helpful attitude in everyone concerned. However, he noted that the individuals who had provided a plant tour to the examiners were not as knowledgeable as desired about detailed facility design or operations. This needs to be corrected for future visits. The following general topics were discussed:

1. For the 12 candidates examined this week, 1 is not a clear pass on the oral/simulator (the final 4 orals are in progress and are not included in this total).

2. The following areas of weakness were observed on more than one candidate and are presented for the use of the facility.

   a. General knowledge about portable radiation monitoring instruments was weak. This may be due to state of the plant.

   b. There was some confusion about power supplies to radiation monitoring systems

   c. There was some tunnel vision in the simulator by the SO with respect to SS responsibilities.

   d. Several candidates had difficulty in deciding when to trip the reactor.

   e. Several candidates had difficulty detecting and responding to multiple radiation alarms.

   f. Several candidates were slow to respond to plant transients, especially as the SO/SS.

3. A candidate who is not a clear pass or who has an area of weakness does not imply unacceptable performance. Such a candidate is one who cannot be judged a clear pass during initial evaluation. A weak area is simply one where knowledge or skill is less completely developed than in other areas.

4. The general state of the procedures will need to be considerably improved before the next examinations are given.

5. We will return our current copies of WCGS procedures after this examination grading is completed.

6. We will reserve the weeks of September 24 and October 1 for the next set of examinations.

7. Current estimates are for the next examinations to be for 15-20 persons.

8. WCGS response to generic letter 84-10 is sufficient for us.

B.2 May 21-24 Site Visit Exit Meeting Summary

At the conclusion of the site visit, examiners met with representatives of the plant staff to discuss the results of the examinations. The following personnel were present for the exit interview:

NRC	UTILITY
J. L. Pellet	S. F. Hatch
L. Defferding	A. S. Mah
R. Gruel	O. Maynard
	G. L. Koester
	C. W. Russo
	P. E. Turner
	M. G. Williams
	F. T. Rhodes
	J. Zell
	J. McKinstray
	J. Burns
	R. Trevillian

Mr. Pellet started the discussion by stating that everyone involved appreciated the attitude of the candidates and staff. The comments in B.1 above apply with the following additions:

1. Areas of weakness added this week were discussed briefly.

   a. Several candidates displayed hesitation at the panels.

   b. Several candidates had difficulty finding local manual valves in the plant (e.g., bypass valves in boric acid filter line).

   c. Candidates were noticeably stronger in plant knowledge than in theory

2. Preliminary results on the candidates e ,uated this week are that of the 15 candidates, 6 are <u>not</u> clear passes on the simulator/oral portions of the examination.

3. NRC will attempt to return formal results within 30 days of leaving the site (June 25, 1984).

C. Generic Comments

The generic comments provided below were generated during grading of the written examinations and review of the oral and simulator examinations.

1. Because of the stage of plant construction and the consequent substantial physical plant changes taking place, NRC does not at this point intend to waive the oral examination portion of the retake examination for those candidates failing the simulator examination.

2. For the RO written examinations, candidate performance on category 4, procedures, was noticeably poorer than the other categor es.

3. For the SRO written examinations, candidate performance on category 6, plant systems design, was noticeably poorer than the other categories.

4. Overall candidate performance, especially on the written examinations, was very good.

D. EXAMINATION MASTER COPY (SRO/RO QUESTIONS AND ANSWERS)

The RO and SRO master examinations, consisting of RO and SRO examinations and grading keys with answers and references, are enclosed with this report.

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: WOLF CREEK

REACTOR TYPE: PWR

DATE ADMINISTERED: 84/05/15

EXAMINER: BROOKS, L.

APPLICANT: ______________________

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	25.00	________	________	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
25.00	25.00	________	________	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.00	25.00	________	________	3. INSTRUMENTS AND CONTROLS
25.00	25.00	________	________	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
100.00	100.00	________	________	TOTALS

Utility Reviewers 5/15/84
Fred Schelmann
Max Herrell
Larry Baker
Wesley Hunter

FINAL GRADE ________________%

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

____
APPLICANT'S SIGNATURE

MASTER COPY

With Facility Exam Review Comments

QUESTION 1.01 (1.00)

Assume that you have established a 1 DPM SUR on your reactor. After 30 seconds would you have increased power by 1/2 decade (i.e., by a factor of 5)? EXPLAIN your answer. (1.0)

QUESTION 1.02 (3.00)

A. Provide TWO removal and TWO production schemes for Xenon 135 in a critical reactor. (Equation form is acceptable, but not required.) (1.5)

B. Approximately how many hours does it take for Xenon to reach equilibrium concentration after the reactor is brought to full power from a Xenon free condition? (0.5)

C. What is the approximate reactivity, in PCM, for:

1. 100% equilibrium Xenon concentration? (0.5)
2. 100% peak Xenon concentration? (0.5)

QUESTION 1.03 (2.00)

Will the Departure from Nuclear Boiling Ratio (DNBR) increase, decrease or remain the same if the following plant parameters increase during power operation? Consider each parameter independently.

A. Reactor Coolant System (RCS) Pressure.
B. RCS Temperature.
C. RCS Flow.
D. Reactor Power. [0.5 ea.] (2.0)

QUESTION 1.04 (3.00)

The ratio of the Pu239 and Pu 240 atoms to the U235 atoms increases over core life. Explain the effect this ratio change has on the following:

A. Delayed neutron fraction. (1.0)
B. Reactor period. (1.0)
C. Doppler Temperature Coefficient. (1.0)

QUESTION 1.05 (1.00)

A. What effect (increase, decrease, or none) does fast neutron irradiation of the reactor vessel (RV) wall have on the RV Reference Transition Temperature (RTT)? (0.5)

B. Indicate whether a ductile failure OR a brittle failure would be the most TYPICAL during RV overpressurization at temperatures <RTT. (0.5)

QUESTION 1.06 (4.40)

Assume:
1. Reactor power = 50 %.
2. Rod control: Manual.
3. Turbine control: Manual.
4. No operator action, no control or protective actions other than those listed.

Compare the final values (higher than, lower than, or the same as), with the initial values, of the parameters listed below for the following two transients. Explain your answers.

Parameters:
1. Reactor power (actual).
2. T cold.
3. T avg.
4. T fuel.

Transients. (Consider each independently.)

A. Boron dilution of 20 ppm. (2.2)

B. Turbine power decrease of 10 %. (2.2)

QUESTION 1.07 (2.50)

A. Explain the effect on Shutdown Margin of a 25 ppm boron addition while operating at 50% power and all control systems in automatic. (1.0)

B. List three (3) factors, other than RCS boron concentration, which effect Shutdown Margin (SDM) and are used in the SDM calculation. (1.5)

QUESTION 1.08 (3.00)

Below is a T-S diagram which closely approximates the steam cycle for your plant. Using the diagram, answer the following questions.

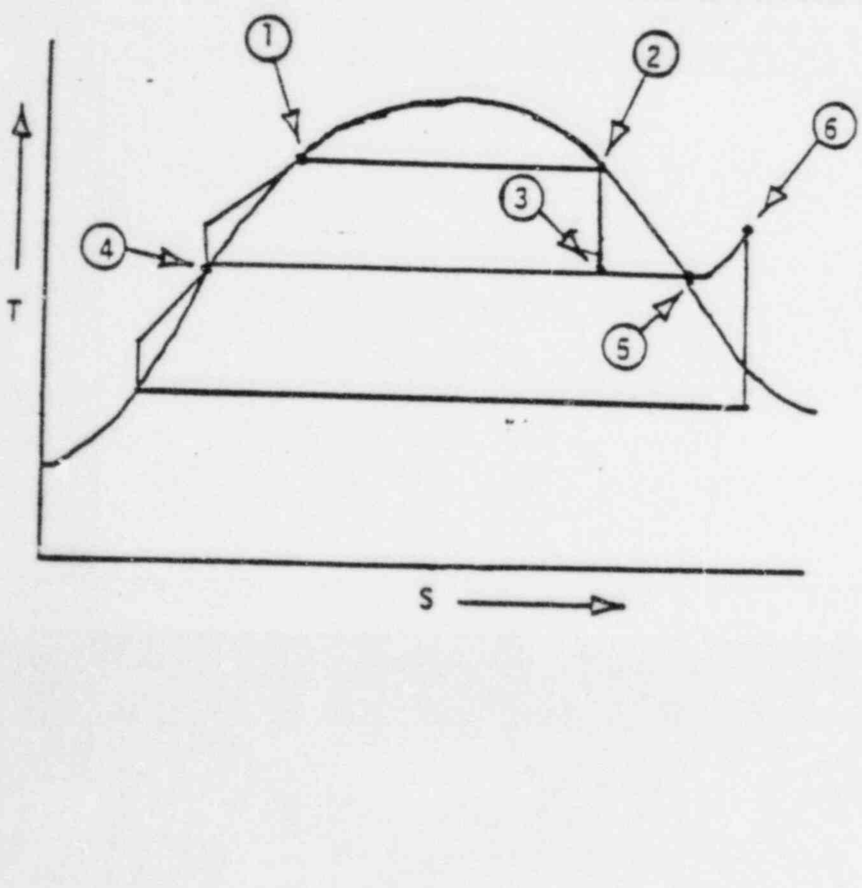
A. What name is given to the energy (enthalpy) GAINED between points 1 and 2? (0.6)

B. What plant process occurs between points 3 and 4? (0.6)

C. Why is there a GAIN in STEAM QUALITY between points 3 and 5? (0.6)

D. What does the line between points 5 and 6 represent? (0.6)

E. What state is the vapor in at point 6? (0.6)



QUESTION 1.09 (2.00)

Indicate on your answer sheet whether the following statements are TRUE or FALSE. (No explanation is required.)

A. The faster a centrifugal pump rotates, the greater the NPSH required to prevent cavitation. (0.5)

B. One of the pump laws for centrifugal pumps states that the volume flow rate is inversely proportional to the speed of the pump. (0.5)

C. Pump runout is the term used to describe the condition of a centrifugal pump running with no volume flow rate. (0.5)

D. The pressurizer level channels which are HOT calibrated will indicate higher than the actual pressurizer level at lower operating temperatures. (0.5)

QUESTION 1.10 (3.10)

A. If a steam leak goes through a throttling process, specifically as in a leak from the main steam header to atmosphere, will the following parameters increase, decrease or remain the same? (No explanation is required.)

1. Enthalpy (h).
2. Pressure.
3. Entropy (s).
4. Specific volume (v).
5. Temperature. (2.35)

B. State whether the steam will be SUBCOOLED, SATURATED or SUPERHEATED as it leaks to atmosphere. (0.75)

QUESTION 2.01 (2.00)

A. Where does the Reactor Vessel Head Vent System (RVHVS) vent to? (0.5)

B. The two RVHVS vent branch lines each have a flow limiting orifice installed. List the two reasons why the flow is limited. (1.0)

C. Is the manual vent valve located in the common piping upstream of the RVHVS branch vent lines open OR shut during normal power operation? (0.5)

QUESTION 2.02 (2.50)

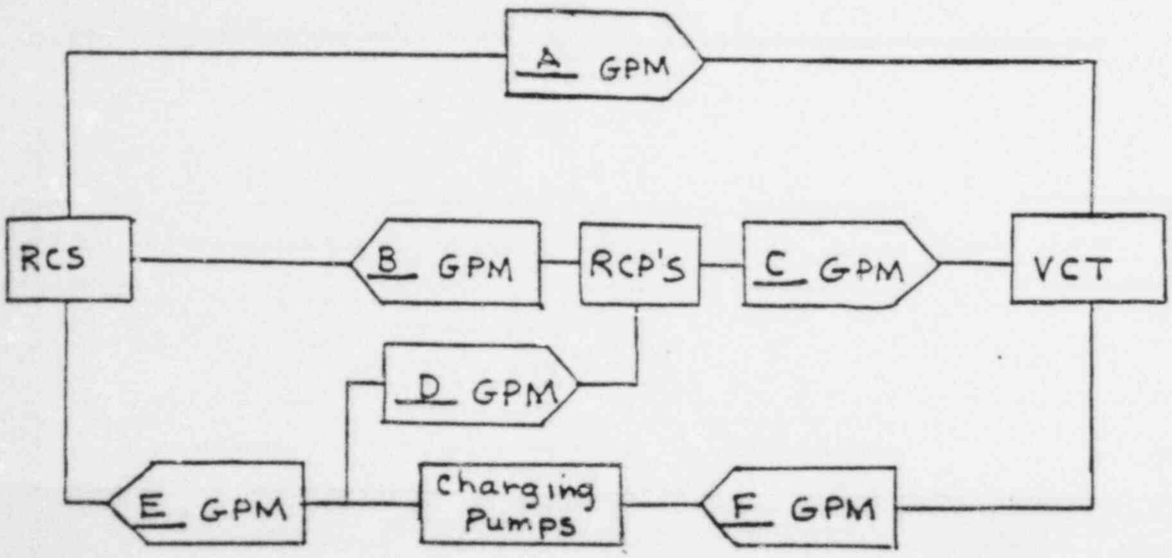
A. Which Reactor coolant System (RCS) loops supply the pressurizer spray valves? (1.0)

B. Which RCS loop does the pressurizer surge line connect to? (0.5)

C. List two reasons why a small continuous flow is maintained through the pressurizer spray lines. (1.0)

QUESTION 2.03 (3.00)

Using the figure below, fill in the blank (lettered) FLOW RATES for A through F to show the charging and letdown flow balance. Assume NORMAL LINEUP and CONDITIONS for power operations. (Put your answers on your answer sheet.)



QUESTION 2.04 (3.50)

A. What is the purpose of the Residual Heat Removal (RHR) pump miniflow valves? (1.0)

B. List the RHR pump flow setpoints which will automatically OPEN and SHUT the RHR miniflow valves. (FCV-610 & 611) (1.0)

C. List the THREE RHR system valves which must be closed in in order to satisfy the opening interlock for the RHR pump hot leg suction valves, PV-8701 & 8702 A or B. (Valve numbers not required.) (1.5)

QUESTION 2.05 (3.50)

The following questions pertain to the fire protection system at the Wolf Creek Facility.

A. What is the normal pressure maintained on the Fire Loop water supply AND how is this pressure maintained? (1.5)

B. List the fire pumps that would automatically start should the fire main pressure decrease due to a large load. (1.0)

C. Briefly explain the major difference between an Automatic Wetpipe Sprinkler System and an Automatic Preaction Sprinkler System. (1.0)

QUESTION 2.06 (3.50)

The following questions pertain to the Component Cooling Water (CCW) System.

A. What component(s) is/are automatically isolated from the CCW system when either CCW surge tank water level reaches its low-low level alarm setpoint? (0.5)

B. What automatic control signal(s) will shut the CCW surge tank vent valves (RV-9 or RV-10) and WHY is this action required? (1.0)

C. List four of the six CCW Safety Loops. (2.0)

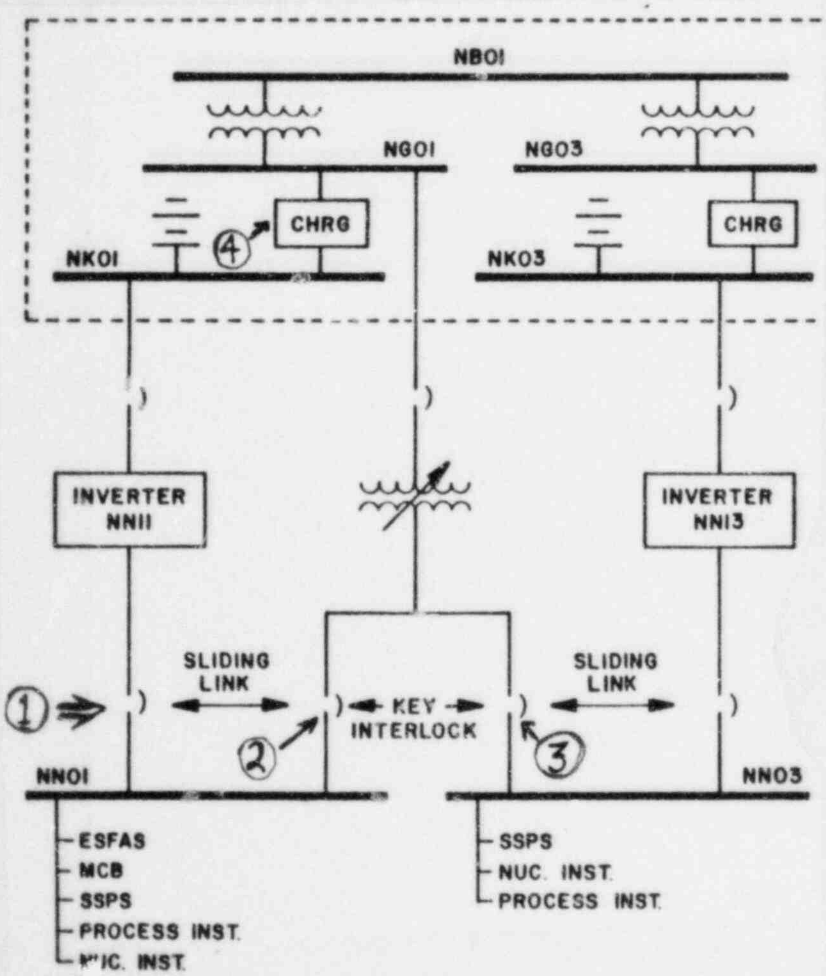
QUESTION 2.07 (3.00)

The following questions pertain to the safeguards power supply figure below. (Breaker positions are shown open, NOT in their normal lineup.)

A. Indicate whether the breakers labeled 1 AND 2 are OPEN or SHUT during normal system operation. Assume all other breakers in a normal operating lineup. (1.0)

B. WHAT function does the key operated interlock perform in regard to the breakers labeled 2 AND 3 and WHY is the interlock required? (1.0)

C. WHAT are the voltage outputs of the Charger at point 4 when in the Normal AND in the Equalizer Mode? (1.0)



QUESTION 2.08 (2.50)

A. There are more than one group of ventilation fans located within Containment. Which TWO (2) groups of Containment fans are automaticaly shifted to slow speed upon reciept of a Safety Injection Signal (SIS) AND WHY are they shifted to slow speed? (1.5)

B. List TWO unrelated signals other than a SIS or MANUAL signal that will automaticaly isolate the Control Room Ventilation System from outside air. (1.0)

QUESTION 2.09 (1.50)

List the three protective signals AND their setpoints which will cause automatic Main Steam Isolation. Do not include manual. (1.5)

QUESTION 3.01 (3.50)

A. What are the three plant parameter input signals used by the three element Steam Generator Feedwater Control System? (Do not include turbine impulse pressure or steam pressure.) (1.5)

B. What is the purpose of the turbine impulse pressure signal used in the S/G Level Control System? (0.5)

C. Considering only the Steam Generator Level Control System, indicate whether feedwater flow would initialy INCREASE, DECREASE, or NOT CHANGE if the controlling S/G pressure transmitter failed high during 50% power operation. Briefly explain your answer. (Assume turbine feed pump speed control is in MANUAL.) (1.5)

QUESTION 3.02 (2.50)

A. List the four signals (non-similiar/unique) which will initiate a motor driven Auxiliary Feedwater Actuation Signal (AFAS). (2.0)

B. With an AFAS signal initiated, what signal(s) will cause an automatic shift of the Auxiliary Feed pump water supply from the Condensate Storage Tank to the Essential Service Water System? Include logic. (0.5)

QUESTION 3.03 (4.50)

A. What input signal is used to provide the programmed pressurizer level for pressurizer level control? (0.5)

B. What is the normal programmed pressurizer level at no load AND at full load? (1.0)

C. The controlling pressurizer level channel fails high during 100% power operation. Assuming NO operator action is taken, which Reactor Protection Signal will cause the Reactor to trip? Explain your answer. (3.0)

QUESTION 3.04 (1.50)

One of the two excore Intermediate Range (I/R) channels compensating voltage is set abnormally high. Indicate whether each of the following statements is TRUE or FALSE.

A. Cancellation of both the gamma and neutron signals would occur causing the channel to indicate low. (0.5)

B. An addition of gamma and neutron signals would occur causing the channel to indicate high. (0.5)

C. Because of the P-6 input signal logic requirements, the I/R mismatch would prevent the manual block of the Excore Source Range Channels during a reactor startup. (0.5)

QUESTION 3.05 (2.50)

A. What signal is required to automatically arm the steam dump when in the Tavg mode. Include the setpoint? (0.5)

B. What plant parameter provides the T-reference signal for the load rejection controller? (0.5)

C. What permissive signal automatically isolates the Tavg mode load rejection controller and energizes the Tavg mode plant trip controller? (0.5)

D. What is the load rejection controllers dead band in degrees fahrenheit and why is that value used? (1.0)

QUESTION 3.06 (2.00)

A. What rod control system components supply the rod control 125/70 VDC hold cabinet AND what component(s) interrupt the power source in the event of a Reactor Trip? (1.0)

B. The reactor trip breakers and the bypass breakers have undervoltage and shunt trip devices. Indicate which of these devices (or both) are actuated by the following signals.
1. Automatic Reactor Protection Trip Signal.
2. Control Room initiated Manual Trip Signal. (1.0)

QUESTION 3.07 (3.50)

A. List the four plant parameter input signals to the Overtemperature Delta-T (OTdT) protection circuit. (2.0)

B. Which of the following core parameters does the OTdT protective circuit provide protection against exceeding?
1. DNB
2. Core Power Density. (0.5)

C. What two control functions other than reactor trip does the OPdT protection channel provide? (1.0)

QUESTION 3.08 (1.50)

A. What Process monitors will cause the Fuel/Auxiliary Building Emergency Exhaust Fans to automatically start and exhaust the Fuel Building? (0.5)

B. If the Fuel/Auxiliary Building Emergency Exhaust fans were started by the FPIS signal and a Safety Injection signal is initiated at a later time, WHAT happens to the Fuel/Auxiliary Building Emergency Exhaust System? (1.0)

QUESTION 3.09 (3.50)

A. List the five reactor trips (non similar/unique) that are blocked when the "at power " permissive P-7 is NOT satisfied. (2.5)

B. What conditions are required to actuate P-7? Include logic/coincidence. (1.0)

QUESTION 4.01 (3.50)

A. What is the maximum percent by volume Oxygen concentration allowed in the CVCS VCT during normal power operation and WHY is this limit established? (1.0)

B. By procedure, (CVCS Precautions and Limitations) WHAT is the maximum letdown FLOW in gpm AND maximum inlet TEMPERATURE allowed to the demineralizers? (1.0)

C. During Reactor Coolant System (RCS) boron concentration changes at power, WHAT is the maximum boron concentration difference in ppm allowed between the pressurizer and the RCS and WHAT procedural step in regard to the pressurizer is taken to promote proper mixing? (Briefly explain why this action is taken.) (1.5)

QUESTION 4.02 (3.00)

A. When transferring from manual rod control to automatic rod control at power, WHAT is the largest temperature error allowed between Tavg and Treference? (0.5)

B. List three reasons why the control banks must be operated above their respective insertion limits (low-low alarm) when critical. (1.0)

C. What operator action must be immediatly initiated in regard to reactivity control if the rod insertion limit low-low alarm is actuated when at power? (1.0)

QUESTION 4.03 (3.00)

The following refer to information found in GOP 00-006 "Hot Standby to Cold Shutdown."

A. What is the maximum cooldown rate in F/HR for the:
   1. Pressurizer?
   2. Reactor Coolant System (RCS)? (1.0)

B. In order to meet the administrative requirements during a forced RCS cooldown, WHAT is the maximum time interval between RCS and Pressurizer temperature plots? (0.5)

C. What is the maximum Delta-T allowed between the pressurizer and the spray line fluid? (0.5)

D. What is the maximum RCS pressure and temperature at which the Residual Heat Removal System may be placed in operation for system cooldown? (1.0)

QUESTION 4.04 (2.50)

A. Under the ALARA administrative whole body penetrating radiation guidelines, WHAT are the normal WEEKLY and QUARTERLY dose limits? (1.0)

B. In order for an area to be designated a Clean Area:
   1. What is the maximum removeable Beta/Gamma contamination in dpm/100sq.cm. as taken by a smear sample?
   2. What is the maximum whole body exposure rate in mr/hr from fixed background radiation? (1.0)

C. What administrative document controls access to Controlled Radiation Areas by operations personnel? (0.5)

QUESTION 4.05 (4.00)

The following questions concern the immediate actions required by ES-02 "Reactor Trip Response".

A. List the four items and their conditions which must be observed by the operator to VERIFY a Reactor Trip. (2.0)

B. List the items and conditions which must be observed by the operator to VERIFY a turbine AND generator trip. (2.0)

QUESTION 4.06 (3.50)

The following questions concern the information found in EMG E-0, "Safety Injection".

A. List the plant conditions which would require tripping ALL Reactor Coolant Pumps. Include adverse Containment requirements. (1.5)

B. Assuming that Safety Injection (SI) was reset and the RHR pumps were stopped by operations during performance of this procedure, WHAT plant conditions would require re-initiating SI and restarting the RHR pumps? Include adverse Containment requirements for re-initiating SI. (2.0)

QUESTION 4.07 (2.50)

Emergency Procedure EMG E-3 "Steam Generator Tube Rupture" lists five methods for identifying a steam generator with a ruptured tube. What are the five methods? (2.5)

QUESTION 4.08 (3.00)

The following refer to information found in the "Emergency Plan".

A. List the four Emergency Action Levels in ORDER of INCREASING severity. [2.0]

B. Assume the Wolf Creek Generating Station Emergency Response Plan has been implemented while you are NOT on shift. A call out has requested that all operations personnel report to the plant. Unless you are told otherwise, which assembly area do you report to? (1.0)

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Net work out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$ $\qquad A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$W = v\ \Delta P$

$A = \dfrac{\pi D^2}{4}$

$t_{1/2}eff = \dfrac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$

$\Delta E = 931\ \Delta m$

$\dot{m} = V_{av} A \rho$

$I = I_o e^{-\Sigma x}$

$\dot{Q} = \dot{m} Cp \Delta t$

$\dot{Q} = UA \Delta T$

$Pwr = W_f \Delta h$

$I = I_o e^{-\mu x}$

$I = I_o\ 10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

$P = P_o 10^{sur(t)}$

$P = P_o e^{t/T}$

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - k_{eff2})$

$T = (\ell^*/\rho) + [(\beta - \rho)/\bar{\lambda}\rho]$

$T = \ell/(\rho - \beta)$

$T = (\beta - \rho)/(\bar{\lambda}\rho)$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$M = 1/(1 - K_{eff}) = CR_1/CR_o$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 10^{-4}$ seconds

$\bar{\lambda} = 0.1\ \text{seconds}^{-1}$

$\rho = [(\ell^*/(T\ K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda}T)]$

$I_1 d_1 = I_2 d_2$

$I_1 d_1{}^2 = I_2 d_2{}^2$

$P = (\Sigma\phi V)/(3 \times 10^{10})$

$\Sigma = \sigma N$

$R/hr = (0.5\ CE)/d^2(\text{meters})$

$R/hr = 6\ CE/d^2\ (\text{feet})$

<u>Water Parameters</u>

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 lbf/in.

<u>Miscellaneous Conversions</u>

1 curie = 3.7×10^{10}dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

ANSWERS -- WOLF CREEK -84/05/15-BROOKS, L.

ANSWER 1.01 (1.00)

No. [0.50] SUR is an exponential increase [P=Po10expSUR(t)] [0.5] (1.0)

REFERENCE
W.C. Review Book, Chapter 41, p. 30

ANSWER 1.02 (3.00)

A. PRODUCTION Iodine decay.
Direct yield from fission.
REMOVAL Neutron absorption (burnout)
Xenon decay [0.375 ea]
[Correct equation also acceptable] (1.5)
B. 50 +/- 10 hours. (0.5)

C. 1. 2900 +/- 300.
2. 6600 +/- 300. (CAF) [0.5 ea.] (1.0)

REFERENCE
W.C. NPS-229, Chapter 3, pp. 3-50, 3-51.
Core Physics, figures A.27 & A.29.

ANSWER 1.03 (2.00)

A. Increase.
B. Decrease.
C. Increase.
D. Decrease. [0.5 ea.] (2.0)

REFERENCE
W.C. Review Book, Chapter 42, pp. 136,137.

ANSWERS -- WOLF CREEK -84/05/15-BROOKS, L.

ANSWER 1.04 (3.00)

A. Delayed neutron fraction decreases [0.5] because the Beta is less for Pu239 as compared to U235. [0.5] (1.0)

B. Shorter reactor period [0.5] because delayed neutron fraction decreases. [0.5] (1.0)

C. Doppler Coefficient is more negative [0.5] because Pu240 has a higher resonance cross section than U235. [0.5] (1.0)

REFERENCE
W.C. Chapter 2, Reactivity and Fuel Temperature Effects, pp. 2-23, 2-42, 2-43.

ANSWER 1.05 (1.00)

A. Increase.
B. Brittle failure. [0.5 ea.] (1.0)

REFERENCE
W.C. Review Book, Chapter 42, pp. 124, 128.

ANSWER 1.06 (4.40)

A. Boron decreases: = + reactivity.
Reactor power: increases.
T fuel: Increases = negative reactivity via FTC or Power defect.
T avg.: Increases = T fuel increased, = -reactivity via MTC.
Reactor power: Decreases, T cold increases (because same turbine power.) (1.0)

Final value comparison:
1. Reactor power: same as. [0.3]
2. T cold: higher than. [0.3]
3. T avg.: higher than. [0.3]
4. T fuel: higher than. [0.3] (1.2)

B. Turbine power: decreases, t cold increases, tavg increases - reactivity via MTC, Reactor power decreases, T fuel decreases, + reactivity via FTC, Reactor power levels off. (1.0)

Final Comparison value:
1. Reactor power: lower than. [0.3]
2. T cold: higher. [0.3]
3. T avg.: higher. [0.3]
4. T fuel: lower. [0.3] (1.2)

REFERENCE
W.C. NPS 229, Chapter 6.

ANSWER 1.07 (2.50)

A. SDM is increased [0.5], with power remaining constant, rod position will be higher (and boron concentration will increase) [0.5]. (Since SDM is the instantaneous amount of reactivity by which the reactor is, or would be subcritical from its present condition.) (1.0)

B. 1. Control rod position.
2. RCS average temperature.
3. Fuel burnup.
4. Xenon concentration. (Time since shutdown.)
5. Power level.
[3@0.5 ea.] (1.5)

REFERENCE
W.C. Review Book, Chapter 9.

ANSWER 1.08 (3.00)

A. Latent heat of vaporization.
B. The removal of EXTRACTION STEAM for feedwater pre-heating.
C. Water is removed, making THE REMAINING STEAM DRYER and/or results of MSR action and reheating taking place.
D. Steam reheaters (between the low and high pressure turbines which add energy to the steam).
E. Superheated steam. [5 @ 0.6 ea.] (3.0)

REFERENCE
W.C. Review Book, Chapter 42, pp. 81, 82, 83.

ANSWER 1.09 (2.00)

A. True.
B. False.
C. False.
D. True. [4 @ 0.5 ea.] (2.0)

REFERENCE
W.C. Review book, Chapter 21, 42, pp. 116-122.

ANSWER 1.10 (3.10)

A. 1. Remain the same.
2. Decrease.
3. Increase.
4. Increase.
5. Decrease. [5 @ 0.47 ea.] (2.35)

B. Superheated. (0.75)

REFERENCE
W.C. Review Book, Chapter 42.

ANSWER 2.01 (2.00)

A. Containment atmosphere. (0.5)

B. 1. Limit the flow of hydrogen from the RCS. (To prevent exceeding containment atmosphere combustable limits.)
2. Limit flow to within the capacity of one centrifugal charging pump. [2 @ 0.5 ea.] (1.0)

C. Open. (0.5)

REFERENCE
W.C. NPS-215, Volume 3, pp. 3-12,13.

ANSWER 2.02 (2.50)

A. Loops 1 & 2. (1.0)

B. Loop 4. (0.5)

C. 1. To reduce thermal stresses (shock) to the spray nozzle.
2. To maintain uniform water chemistry in the pressurizer.
3. To maintain temperature in the pressurizer.
[any 2 @ 0.5 ea.] (1.0)

REFERENCE
W.C. NPS 215, Chapter 3, pp. 3-32,33.

ANSWER 2.03 (3.00)

A. 75
B. 20
C. 12
D. 32
E. 55
F. 87 [6 @ 0.5 ea.] (3.0)

REFERENCE
W.C. NPS 217, Chapter 1, pp. 1-32.

ANSWER 2.04 (3.50)

A. Prevents RHR pump overheating (or vibration) [0.5]
when the RHR pump is operating at shutoff head. [0.5] OR will accept "when operating at low flow, or minimum flow conditions." DB (1.0)

B. Opens at 534 +/- 50 gpm.
Shuts at 1000 +/- 50 gpm. [2 @ 0.5 ea.] (1.0)

C. 1. RHR suction valve from containment sump. (HV-8811 A or B)
2. RHR suction valve from the RWST. (HV-8812 A or B)
3. RHR pump discharge valve to the centrifugal charging pump and Safety Injection pump suction (HV-8804 A & B)
[3 @ 0.5 ea.] (1.5)

REFERENCE
W.C. NPS 217, Chapter 4, pp. 4-12,15.

ANSWER 2.05 (3.50)

A. 80 psig ±10 psig DB [0.5] maintained by the Jockey Fire Pump [0.5] and the Service Water System. [0.5] (1.5)

B. One motor driven (fire pump 1A) [0.5] and one diesel driven (fire pump 1B) [0.5] start as required. (1.0)

C. An Automatic Wetpipe Sprinkler System is normaly pressurized with water [0.5]. An Automatic Preaction Sprinkler System is normally dry. [0.5] (A diluge valve opens on a fire detector signal to flood the Preaction System.) (1.0)

REFERENCE
W.C. NPS 229, Chapter 2, pp. 2-6, 2-7, 2-10, 2-12.

ANSWER 2.06 (3.50)

A. The Post Accident Sampling Station. [0.25]
The Rad Waste Building. [0.25] (0.5)

B. CCW Radiation Monitors (RE-9 & RE-10) [0.5]
To prevent releasing radioactivity to the Auxiliary Building. [0.5]
(1.0)

C. SI pump oil coolers.
CCP pump oil coolers.
Spent Fuel Pool heat exchangers.
RHR pump seal coolers.
RHR heat exchangers.
Post Accident sample coolers. [any 4 @ 0.5 ea.] (2.0)

REFERENCE
W.C. NPS 217, pp. 5-6, 5-9, 5-12.

ANSWER 2.07 (3.00)

A. 1 is shut, 2 is open. [2 @ 0.5 ea.] (1.0)

B. The interlock prevents shutting breakers 2 and 3 at the same time [0.5] and is required to maintain electrical separation of the instrument busses. [0.5] (1.0)

C. 130 VDC Normal. ± 5VDC UB
140 VDC Equalizer. [2 @ 0.5 ea.] (1.0)
± 5VDC. UB

REFERENCE
W.C. NPS 213, Chapter 5, pp. 5-4, 5-13.

ANSWER 2.08 (2.50)

A. The Hydrogen Mixing fans (four) [0.5] and the Containment Fan Cooler fans (four). [0.5]
They are shifted to slow speed to prevent overload in a high pressure (high humidity) environment. [0.5] (1.5)

B. High Radiation (at Control Room Supply).
High Chlorine (at Control Room Supply). [2 @ 0.5 ea.] (1.0)

REFERENCE
W.C. NPS 221, Chapter 4, pp. 4-6, 4-7, 4-35, 4-36.

ANSWER 2.09 (1.50)

Low steam line pressure [0.3] 585 psig [0.2]
High steam pressure rate [0.3] 110 psig/50 sec [0.2]
Containment pressure Hi-2 [0.3] 17 psig [0.2] (1.5)

REFERENCE
W.C. NPS 221, Chapter 1, pp. 1-20.

ANSWER 3.01 (3.50)

A. 1. Feedwater Flow.
   2. Steam Flow.
   3. S/G Water Level. (Error signal) [3 @ 0.5] (1.5)

B. To provide a constant level setpoint. (0.5)

C. Increase. [0.75] The failed high steam pressure transmitter causes the steam flow input to SGWLC to increase. [0.75] (1.5)

REFERENCE
W.C. NPS 223, Chapter 6, pp. 6-15, 6-16.

ANSWER 3.02 (2.50)

A. 1. Manual.
   2. S/G low-low level.
   3. Trip of both Main Feed Pumps.
   4. Undervoltage on both NB01 & NB02 busses. (Class 1E Busses) DB [4 @ 0.5 ea.] (2.0)
   5. SIS signal. Ref: figure SNP-AF-4. DB

B. Low suction pressure. [0.25] 2 of 3. [0.25] (0.5)

REFERENCE
W.C. NPS 223, Chapter 5, pp. 5-12, 5-13. Figure SNP-AF-4.

ANSWER 3.03 (4.50)

A. Tavg. (0.5)

B. 25 % [0.5] to 61.5 %. [0.5] (1.0)

C. High pressurizer level trip (92 %) [0.6]. Charging flow decreases [0.6]. Pressurizer level decreases [0.6]. Letdown isolates [0.6]. and pressurizer level increases [0.6]. (3.0)

REFERENCE
W.C. NPS 215, Chapter 6, pp. 6-14, 6-16.

ANSWER 3.04 (1.50)

A. True. (0.5)

B. False. (0.5)

C. False. (0.5)

REFERENCE
W.C. NPS 227, Chapter 2, pp. 2-11, 2-21.

ANSWER 3.05 (2.50)

A. Load rejection [0.25] > 10 % in 120 sec. [0.25] (0.5)
B. First stage turbine impulse pressure. (0.5)
C. P-4 (Reactor trip breaker open.) (0.5)
D. 5 degrees F. [0.5] To allow rods to control the transient.[0.5] (1.0)

REFERENCE
W.C. NPS 223, Chapter 4, pp. 4-6, 4-12, 4-18.

ANSWER 3.06 (2.00)

A. 1. The rod drive motor generators.
   2. The reactor trip breakers open. (1.0)

B. 1. Undervoltage trip device only.
   2. Both undervoltage and shunt trip devices. (1.0)
   (CAF)

REFERENCE
W.C. NPS 227, Chapter 4, pp. 4-23, 4-27.
Chapter 5, pp. 5-10.

ANSWER 3.07 (3.50)

A. 1. Tavg.
2. dT.
3. Pressure.
4. dI. [4 @ 0.5ea.] (2.0)

B. DNB. (0.5)

C. 1. Turbine runback. [0.5]
2. Blocks automatic rod withdrawel. [0.5] (1.0)

REFERENCE
W.C. NPS 227, Chapter 5, pp. 5-16, 5-18.

ANSWER 3.08 (1.50)

A. Fuel Building Process Radiation Monitors (RE-27 & 28). (0.5)
B. The exhaust dampers shift from Fuel Building suction to Auxiliary Building suction. (1.0)

REFERENCE
W.C. NPS 221, Chapter 4, pp. 4-29, 4-32.

ANSWER 3.09 (3.50)

A. 1. Pressurizer low pressure.
2. Pressurizer high level.
3. RCP UV.
4. RCP UF.
5. RCP low flow trip in more than one loop. [5 @ 0.5 ea.] (2.5)

B. First stage turbine pressure. [0.3] 1 of 2 [0.2]
Power Range level. [0.3] 2 of 4. [0.2] (1.0)

REFERENCE
W.C. NPS 227, Chapter 5, pp.5-24.

ANSWERS -- WOLF CREEK -84/05/15-BROOKS, L.

ANSWER 4.01 (3.50)

A. 5 % [0.5]
To prevent an explosive mixture in the VCT. [0.5] (1.0)
B. 120 gpm. [0.5]
130 F. [0.5] (1.0)
C. 50 ppm. [0.5]
Pressurizer backup heaters energized [0.5] to force pressurizer spray flow. [0.5] (1.5)

REFERENCE
W.C. Review Book, Chapter 1, pp. 54; SOP BG120, pp.1; SOP BG200, pp.1.

ANSWER 4.02 (3.00)

A. + or - 1 degree F. (0.5)
B. 1. To assure adequate SDM.
2. To assure ejected rod reactivity limits are not exceeded.
3. To assure acceptable core power limits are not exceeded.
[2@ 0.5] (1.5)
C. Emergency (Rapid) (Immediate) DB Boration MUST be initiated. (1.0)

REFERENCE
W.C. Review Book, Chapter 1, pp. 1, 2.

ANSWER 4.03 (3.00)

A. 200 F. [0.5]
100 F. [0.5] (1.0)
B. 30 minutes. (0.5)
C. 583 F. (0.5)
D. <425 psig. [0.5]
<350 F. [0.5] (1.0)

REFERENCE
W.C. GOP 00-006, pp. 1, 2.

ANSWERS -- WOLF CREEK -84/05/15-BROOKS, L.

ANSWER 4.04 (2.50)

A. 300 mr/week [0.5] 1000 mr/quarter. [0.5] (2000 with SS Approval) DB (1.0)
B. <1000dpm/100sq.cm. [0.5] 0.1 mr/hr. [0.5] (1.0)
C. RWP (Radiation Work Permit). (0.5)

REFERENCE
W.C. Radiation Protection Procedures, pp. 6-9, 6-15, 6-26.

ANSWER 4.05 (4.00)

A. All rod bottom lights lit.
Reactor trip breakers open.
Reactor bypass breakers open.
Neutron flux decreasing. [4 @ 0.5 ea.] (2.0)
B. All turbine stop valves closed.
All turbine control valves closed.
Main generator breaker open.
Exciter breaker open. [4 @ 0.5 ea.] (2.0)

REFERENCE
W.C. ES-02, Reactor Trip Response.

ANSWER 4.06 (3.50)

A. -RCS pressure <1400 psig [0.5] OR
-1700 psig with adverse Containment [0.5] AND
-CCPS or SI pumps running. [0.5] (1.5)
(ECCS) DB
B. RCS subcooling less than required (per figure 1) [0.5] OR
Pressurizer level cannot be maintained greater than
5% [0.5] OR 20% for adverse Containment. [0.5]
Start RHR pump if RCS pressure <330 psig. [0.5] (2.0)

REFERENCE
W.C. Emg E-0, pp. 13, 16.

ANSWERS -- WOLF CREEK -84/05/15-BROOKS, L.

ANSWER 4.07 (2.50)

1. Unexpected rise in S/G narrow range level.
2. High radiation from S/G sample.
3. High radiation from S/G steamline.
4. High radiation from S/G blowdown line.
5. Turbine driven AFW exhaust radiation. [5 @ 0.5 ea.] (2.5)

REFERENCE
W.C. EMG E-3, Steam Genertor Tube Rupture, p.2.

ANSWER 4.08 (3.00)

A. Unusual Event.
Alert.
Site Area Emergency.
General Emergency. [4 @ 0.5 ea.] (2.0)

B. ~~Operations Support Center (OSC).~~ (1.0)
Emergency Offsite Facility (EOF) (See adm 12-6.) DB

REFERENCE
W.C. Emergency Plan, 2.2-1; 4.1-3.
ADM 12-6.0, Section 6.2, page 1 of 5. DB

QUESTION	VALUE	REFERENCE
01.01	1.00	LOB0000038
01.02	3.00	LOB0000039
01.03	2.00	LOB0000040
01.04	3.00	LOB0000041
01.05	1.00	LOB0000042
01.06	4.40	LOB0000043
01.07	2.50	LOB0000044
01.08	3.00	LOB0000045
01.09	2.00	LOB0000046
01.10	3.10	LOB0000047

	25.00	
02.01	2.00	LOB0000048
02.02	2.50	LOB0000049
02.03	3.00	LOB0000050
02.04	3.50	LOB0000051
02.05	3.50	LOB0000052
02.06	3.50	LOB0000053
02.07	3.00	LOB0000054
02.08	2.50	LOB0000055
02.09	1.50	LOB0000056

	25.00	
03.01	3.50	LOB0000057
03.02	2.50	LOB0000058
03.03	4.50	LOB0000059
03.04	1.50	LOB0000060
03.05	2.50	LOB0000061
03.06	2.00	LOB0000062
03.07	3.50	LOB0000063
03.08	1.50	LOB0000064
03.09	3.50	LOB0000065

	25.00	
04.01	3.50	LOB0000066
04.02	3.00	LOB0000067
04.03	3.00	LOB0000068
04.04	2.50	LOB0000069
04.05	4.00	LOB0000070
04.06	3.50	LOB0000071
04.07	2.50	LOB0000072
04.08	3.00	LOB0000073

	25.00	

	100.00	

MASTER COPY

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: WOLF CREEK

REACTOR TYPE: PWR

DATE ADMINISTERED: 84/05/15

EXAMINER: JEFFERIES, G.

APPLICANT: MASTER COPY

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	25.00	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00	________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00	100.00	________	________	TOTALS

FINAL GRADE ________________%

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

Reviewed by the following Utility Training Instructors 5/15/84

____
APPLICANT'S SIGNATURE

Fred Scheimann
Larry Baker
Max Herrell
Wesley Hunter

QUESTION 5.01 (3.00)

0 a. Power defect changes over core life. Of the coefficients that contribute to power defect, which contributes most to this change over core life? EXPLAIN (1.0)

0 b. Explain why power defect is desireable for reactor operation at power. (1.0)

0 c. Which of the reactivity coefficients that contribute to power defect act first to affect reactivity on a sudden power change due to rod movement? EXPLAIN WHY. (1.0)

3 QUESTION 5.02 (1.00)

How does the period of time for neutron counts to level off after a 50 step rod bank pull change as criticality is approached? EXPLAIN WHY. (1.0)

QUESTION 5.03 (3.00)

Explain HOW AND WHY a control rod's worth is affected (increased, decreased, or no change), relative to its worth prior to the following changes:

53 a. Increase in moderator temperature. (1.0)

22 b. The insertion of an adjacent control rod. (1.0)

2 c. Uniform flux increase throughout the core as power increases on a stable period from criticality to the point of adding heat. (1.0)

QUESTION 5.04 (3.00)

19 The reactor is at 100% power with equilibrium xenon and all rods out when the boron concentration is reduced, causing a deep insertion of control rod bank D to maintain Tave constant. Describe how the axial core power distribution will change WITH TIME as a result of this action. Be complete in your answer. Assume no further rod motion. (3.0)

QUESTION 5.05 (3.00)

For a reactor power increase with a stable period of 90 sec:

0 a. How much will power increase in 90 sec? (0.5)

0 b. What is the startup rate, in decades per minute? (1.0)

2 c. How long will it take to raise power from 10MW to 1000MW on this period? (0.5)

2 d. If the reactor is at 10MW, what will power be after 3 minutes? (1.0)

QUESTION 5.06 (2.00)

37 a. For a constant boron concentration (in ppm) how does differential boron worth change (more negative, less negative) as moderator temperature is increased? EXPLAIN. (1.0)

0 b. For a constant moderator temperature, how does differential boron worth change (more negative, less negative) as boron concentration is increased? EXPLAIN. (1.0)

QUESTION 5.07 (3.40)

2 a. How do each of the following parameters change (increase, decrease or no change) if one main steam isolation valve closes with the plant at 50% load. Assume all controls are in automatic and that no trip occurs.

1. Affected loop steam generator level (INITIAL change only)
2. Affected loop steam generator pressure
3. Affected loop cold leg temperature
4. Unaffected loop steam generator level (INITIAL change only)
5. Unaffected loop steam generator pressure
6. Unaffected loop cold leg temperature (3.0)

48 b. Which of the reactor protection system signals could be expected to cause a reactor trip? (If more than one, list the one that would reach the trip point first.) (0.4)

QUESTION 5.08 (3.60)

9 For the following changes in plant status, indicate WHY DNBR will increase, decrease or stay the same in the core. Consider each independently from the others.

a. Increase in core inlet temperature
b. Control rod insertion (core power constant)
c. Increase in Reactor Coolant System (RCS) pressure
d. Crud buildup on fuel rod cladding
e. Decrease in Reactor Power
f. Increase in RCS flow (3.6)

QUESTION 5.09 (1.00)

4 Condensate depression could be considered to be an ADVANTAGE for for condensate pump operation but a DISADVANTAGE for plant thermal efficiency. EXPLAIN. (1.0)

QUESTION 5.10 (2.00)

11 a. The speed of a centrifugal pump is decreased to half its initial value. Given the following initial conditions, what are the final conditions.

1. Fluid Horsepower	25 HP	
2. Flow	45 gpm	
3. Head	250 psi	(1.5)

6 b. Sketch the effect of an INCREASE in pump speed on the system operating point on the pump's characteristic curves. (0.5)

QUESTION 6.01 (3.00)

23 a. What chemical is used to control Reactor Coolant System (RCS) pH AND WHY is pH control necessary? (1.0)

6 b. How is oxygen formed in the RCS AND how does the hydrogen blanket in the VCT prevent a buildup of this oxygen in the RCS? (1.0)

6 c. Why is hydrazine added to the RCS AND why should the CVCS demineralizers be removed from service during this addition? (1.0)

QUESTION 6.02 (2.50)

17 How does the pressurizer function during a REACTOR TRIP to minimize pressure changes? Include both inherent and automatic features (setpoints for automatic controls) that act to affect pressurizer pressure. Assume all pressurizer controls are in automatic. (2.5)

QUESTION 6.03 (2.45)

20 Utilizing the attached CVCS drawing, (Figure 6.1), answer the following questions.

a. Where does this valve divert to? (0.25)

b. What are the TWO purposes of this valve? (0.5)

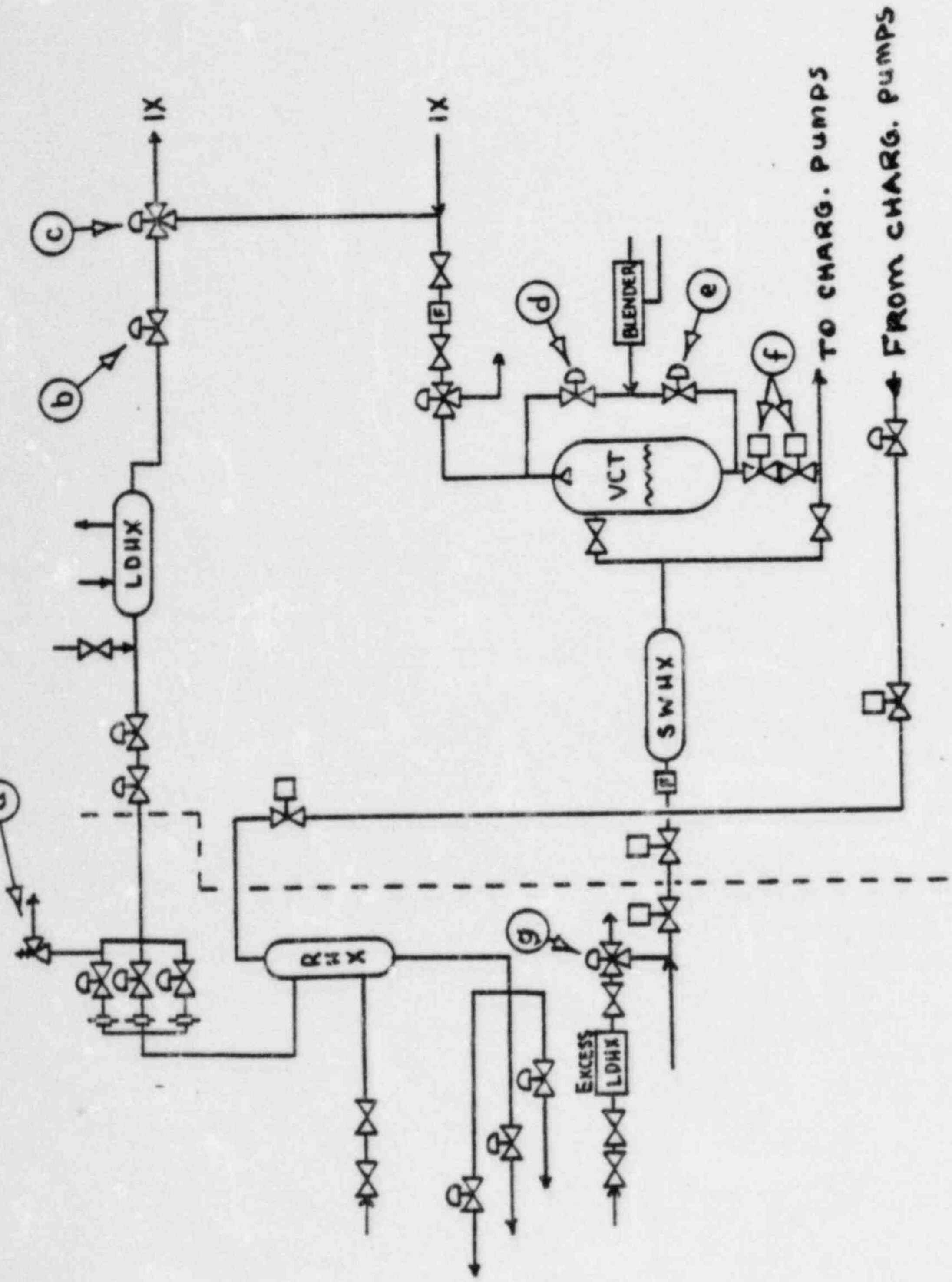
c. To which position will this valve fail on loss of control air? (0.25)

d. In which positions on the Reactor Makeup Control System Mode selector switch is this valve (FCV-111B) enabled to open? (0.3)

e. In which positions on the Reactor Makeup Control System Mode selector switch is this valve (FCV-110B) enabled to open? (0.4)

f. What are TWO signals that will result in AUTOMATIC closure of these valves (LCV-112B and C)? (No setpoints required). (0.5)

g. Where does this valve divert to? (0.25)



IX
IX
LDHX
SWHX
VCT
BLENDER
RHX
EXCESS LDHX
TO CHARG. PUMPS
FROM CHARG. PUMPS
a
b
c
d
e
f
g

Figure 6.1

QUESTION 6.04 (2.50)

36 a. What two conditions will initiate Phase B Containment Isolation. Include logic and setpoints. (1.0)

20 b. How does Phase B Containment Isolation affect the Component Cooling Water System? Be specific. (1.5)

QUESTION 6.05 (2.50)

26 a. Describe the flowpath of seal injection water through the Reactor Coolant Pump (RCP). Include supply, approximate flowrates through each path and discharge collection points. (Consider normal seal injection only at normal operating conditions.) (2.0)

19 b. What would be your concern if the VCT pressure was less than 15 psig during RCP operation? (0.5)

QUESTION 6.06 (2.00)

The reactor is operating at 100% power with all control systems in automatic. For the following failures, what reactor protection signal will cause the reactor to trip? Assume no operator action and consider each failure independently. BRIEFLY EXPLAIN.

1 a. CVCS charging flow rate drops to minimum of 47 gpm. (1.0)

12 b. A cold leg temperature detector (controlling) fails high. (1.0)

QUESTION 6.07 (3.85)

34 a. Describe the Main Feed Pump speed control circuit. Include parameter inputs and speed control program in the description. (1.8)

10 b. Assume the controlling feedwater flow channel for a steam generator fails high. With no operator action, explain why a reactor trip could occur. Include COINCIDENCE AND SETPOINTS. (1.0)

19 C. What are three protective actions that will over ride the automatic steam generator level control system signals? (No setpoint, coincidences required) (1.05)

QUESTION 6.08 (2.70)

a. What are the TWO conditions that will cause automatic startup of The Emergency Diesel Generators? (0.6)

b. What are FIVE conditions that will cause automatic trip of the Emergency Diesel ENGINES when running in the Emergency Mode? (1.5)

c. What are the TWO conditions that will cause automatic trip of the Emergency Diesel Generator BREAKERS when running in the Emergency Mode? (0.6)

QUESTION 6.09 (1.50)

a. List TWO of the three reasons (Safety Design Bases) for the orifices located in the auxiliary feedwater headers to the steam generators. (1.0)

b. There are four accumulators charged with 750 psig nitrogen which backs up the air supply to the turbine driven auxiliary feedwater pump discharge valves. What other major plant components are also supplied by each of the four accumulators. (0.5)

QUESTION 6.10 (2.00)

The following refer to the Reactor Vessel Level Instrumentation System (RVLIS). Match the statements in Column B with the appropriate instrument input to the RVLIS in Column A. Place your answer on answer sheet (e.g., e-5).

COLUMN A	COLUMN B
a. Wide Range RVLIS Delta-P	1. Used for RVLIS level compensation when superheat exists at the hot leg temperature measurement point.
b. Narrow Range RVLIS Delta-P	2. The most accurate indication of reactor vessel water level with reactor coolant pumps on.
c. Hot Leg Wide Range Temperature	3. Used for RVLIS level compensation when the system is subcooled.
c. Wide Range RCS Pressure	4. The most accurate indication of reactor vessel water level when reactor coolant pumps are NOT on.

(2.0)

QUESTION 7.01 (3.50)

18 a. List five conditions that require Immediate Boration. (1.5)

6 b. How is Immediate Boration flow initiated and verified normally? (1.5)

0 c. What is the difference in Immediate Boration initiation if the normal flowpath is plugged? (0.5)

QUESTION 7.02 (3.00)

10 Match the trends from Column B that would be indicative of conditions for Column A malfunctions prior to any protective function actuations. There may be more than one answer. Place answers on answer sheet (e.g., c-7,8,9)

COLUMN A

a. Small Break LOCA Inside Containment

b. Steam Leak Inside Containment

COLUMN B

1. Decreasing Pressurizer Level
2. Decreasing Steam Pressure
3. Increasing Containment Pressure
4. Decreasing Tave
5. Increasing Containment Radiation
6. Decreasing Pressurrizer Pressure

(3.0)

QUESTION 7.03 (1.50)

9 a. Consider two point gamma sources, each with 1 curie strength. Source A gamma energy is 2 MEV and Source B gamma energy is 1 MEV. If readings were taken at the same distance from each unshielded source with a Geiger Mueller (GM) type meter, how would the readings compare? Briefly explain. (1.0)

0 b. If a worker was exposed to a 1 R/hr NEUTRON radiation field, would the biological damage be less than, greater than, or the same as if the 1 R/hr field was due to GAMMA radiation? (0.5)

QUESTION 7.04 (2.00)

2 a. While performing "Natural Circulation Cooldown" (EMG ES-04), you are procedurally directed to depressurize the RCS.

1. What method will be used to depressurize the RCS if Letdown is in service? (0.5)

2. What method will be used if letdown is NOT in service? (0.5)

0 b. Why is the depressurization more restrictive if CRDM fans are not running? (1.0)

QUESTION 7.05 (3.50)

The following questions concern Precautions and Limitations in "Hot Standby to Minimum Load" (GEN-00-003).

4 a. What control action should be taken by the operator before adjusting RCS boron concentration? EXPLAIN WHY. (1.0)

2 b. While diluting to the Estimated Critical Boron Concentration, one source range channel changes from 8 counts to 17 counts and the other channel changes from 6 to 9 counts. What action is required? (1.0)

0 c. Tave is lowered to 548 F with the reactor at 2% power due to excessive steam removal. What action is required by Technical Specifications? (0.5)

9 d. What "Precautions and Limitations" are being challenged if you were performing the following operations?

1. Diluting to the Estimated Critical Boron Concentration with the reactor slightly subcritical? (0.5)

2. Increasing power from P-6 to P-10 with control bank D withdrawal while slowly diluting because control bank D rods are nearly all the way out. (0.5)

QUESTION 7.06 (3.00)

12 Match the description or phrase in Column B which BEST depicts or describes the item in Column A. Place answer on answer sheet (e.g., f-11).

COLUMN A	COLUMN B
a. Cobalt 60	1. Wolf Creek Notification Dose Level
b. Alpha radiation	2. Wolf Creek Administrative Exposure Limit
c. Nitrogen-16	3. Half Life of 7.3 days
d. 300 mrem/week	4. Causes fission in U-235
e. Beta radiation	5. Major source of radiation in containment while operating at power
	6. Maximum exposure limit (10CFR20)
	7. May dictate the need for eye protection
	8. Major source of radiation in containment while shutdown
	9. Significant level indicates fuel leak

(3.0)

QUESTION 7.07 (3.00)

29 Safety Injection termination criteria in "Loss of Reactor or Secondary Coolant" (EMG E-1) are specified for normal and ADVERSE containment conditions. Which criteria are affected by adverse containment conditions and EXPLAIN WHY they are different? (3.0)

QUESTION 7.08 (3.00)

0 a. A spent fuel assembly is being moved from the reactor canal to the upender when it drops to the bottom of the canal. What type of radiation hazard can be expected if the fuel assembly is severely damaged? (0.5)

1 b. What are three features of the fuel handling equipment that should prevent dropping a fuel element? (1.5)

0 c. What are <u>two</u> actions that the Refueling Supervisor should assure is taken if containment radiation alarms actuates during fuel handling operations? (1.0)

QUESTION 7.09 (1.00)

Indicate the lowest emergency event classification that requires activation of the following Emergency Centers:

a. Emergency Operation Facility (EOF) (0.5)

b. Technical Support Center (TSC) (0.5)

QUESTION 7.10 (1.50)

List the Critical Safety Functions (CSF) in the order that the Status Trees for the CSFs are to be checked to evaluate the safety state of the plant in accordance with "Critical Safety Function Status Trees (CSFST)", EMG F-0. (1.5)

QUESTION 8.01 (1.50)

20 What action must be taken if the Reactor Coolant System Pressure Safety Limit is exceeded, in accordance with Technical Specificatons. Consider all MODES. (1.5)

QUESTION 8.02 (1.50)

The Technical Specification for reactor trip system instrumentation channels specifies if one channel of Power Range Nuclear Instrumentation is inoperable, a Quadrant Power Tilt Ratio must be done at least once per 12 hours if power is at 100%.

22 a. How is the Quadrant Power Tilt Ratio determined in this case? (0.5)

50 b. If the Quadrant Power Tilt Ratio is not determined within the allowable time, what must be done? (1.0)

QUESTION 8.03 (3.00)

The concentration of the boric acid solution in the Boron Injection Tank (BIT) shall be verified once per 7 days in accordance with Technical Specifications. The chemist sampled the BIT under the following schedule. (All samples taken at 1200 hours).

January 1 -- January 8 -- January 16 -- January 24 -- January 31

0 a. EXPLAIN why or why not surveillance time interval requirements were exceeded on January 16. (1.5)

0 b. EXPLAIN why or why not surveillance time interval requirements were exceeded on January 24. (1.5)

QUESTION 6.04 (4.00)

For each of the following leak locations, give the maximum allowable leak rate AND the basis for each.

33 a. Unknown location

26 b. Through a pressurizer code safety valve to the Pressurizer Relief Tank.

14 c. Through the wall of the line between the pressurizer relief valves and the pressurizer.

29 d. Total flow to Reactor Coolant Pump seals.

8 e. TOTAL Steam Generator tube leakage. (4.0)

QUESTION 8.05 (2.75)

8 a. In accordance with Administrative Procedure ADM 02-001 "Operations", what are the minimum duty shift manning requirements for the following during power operations? Do not consider the two hour exceptions.

1. Licensed Senior Operator. Include the titles for positions held.

2. Reactor Operators. Include license requirements.

3. Nuclear Station Operators. (1.95)

4 b. What is the minimum number of Fire Brigade members required onsite at all times AND how many of the minimum Shift Crew are NOT to be included as members of the Fire Brigade? (0.8)

QUESTION 8.06 (3.00)

0 a. What provides the tagout function for a live breaker which must be opened and closed several times during trouble shooting under a "Clearance Without DNO Tag"? (0.75)

0 b. How many components can be controlled by a single "Clearance Without DNO Tag"? (0.75)

20 c. What is the maximum time period that a "Clearance Without DNO Tag" may remain open? (0.75)

4 d. When filling out the Clearance Order for a "Clearance Without DNO Tag", what information is supplied in the blank for the tag number? (0.75)

QUESTION 8.07 (3.00)

24 a. Who, by title, must authorize all fuel transfers and new fuel shipments? (0.5)

17 b. Who, by title, is responsible for carrying out all fuel movements after the authorization for fuel transfer has been given? (0.5)

16 c. List three different areas where lateral support of a fuel assembly can be applied if required. (1.5)

30 d. How is support provided for horizontal handling of a fuel assembly? (0.5)

QUESTION 8.08 (2.50)

In the event of a plant emergency requiring implementing the Emergency Plan, who, by title:

0 a. Initially assumes the duties of the Duty Emergency Director? (0.5)

4 b. Can relieve the Duty Emergency Director (Both Titles)? (1.0)

0 c. Initally assumes the responsibilities of the Operations Emergency Coordinator? (0.5)

0 d. Is the normal relief for the Operations Emergency Coordinator? (0.5)

QUESTION 8.09 (1.75)

Emergency Kits which are available for use during plant emergencies and scheduled drills are established at six different locations.
28 List FIVE of these locations. (1.75)

QUESTION 8.10 (2.00)

During refueling operations involving core alterations:

3 a. What are the minimum neutron flux monitoring requirements as required by Technical Specifications? Include the monitoring locations and method of monitoring. (0.8)

7 b. What are the minimum Technical Specification requirements in regard to:

1. Containment Building equipment door?

2. Containment Building airlocks?

3. Each Containment penetration providing access to the outside environment? (1.2)

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Net work out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

$A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$W = v\ \Delta P$

$A = \dfrac{\pi D^2}{4}$

$t_{1/2}eff = \dfrac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$

$\Delta E = 931\ \Delta m$

$\dot{m} = V_{av} A \rho$

$I = I_o e^{-\Sigma x}$

$\dot{Q} = \dot{m} Cp \Delta t$

$\dot{Q} = UA \Delta T$

$Pwr = W_f \Delta h$

$I = I_o e^{-\mu x}$

$I = I_o\ 10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

$P = P_o 10^{sur(t)}$

$P = P_o e^{t/T}$

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - k_{eff2})$

$T = (\ell^*/\rho) + [(\beta - \rho)/\bar{\lambda}\rho]$

$T = \ell/(\rho - \beta)$

$T = (\beta - \rho)/(\bar{\lambda}\rho)$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$M = 1/(1 - K_{eff}) = CR_1/CR_o$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 10^{-4}$ seconds

$\bar{\lambda} = 0.1$ seconds^{-1}

$\rho = [(\ell^*/(T\ K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda}T)]$

$I_1 d_1 = I_2 d_2$

$I_1 d_1{}^2 = I_2 d_2{}^2$

$P = (\Sigma\phi V)/(3 \times 10^{10})$

$\Sigma = \sigma N$

$R/hr = (0.5\ CE)/d^2(\text{meters})$

$R/hr = 6\ CE/d^2\ (\text{feet})$

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 lbf/in.

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

MASTER

ANSWERS -- WOLF CREEK -84/05/15-JEFFRIES, G.

ANSWER 5.01 (3.00)

a. Power defect increases (more negative) from BOL to EOL [0.5] due to increase in MTC as boron concentration is reduced over core life [0.5]. (1.0)

b. Power defect has a stabilizing influence on reactor operation because it resists power changes. (As power increases, power defect adds negative reactivity and as power decreases, power defect adds positive reactivity). (1.0)

c. Doppler [0.5]. Fuel temperature changes first [0.5]. (1.0)

REFERENCE
Review Book, Chapter 41, pp 56-66 GLJ 84

ANSWER 5.02 (1.00)

Increase [0.5]. Due to the increase in the number of generations that pass before neutron level stabilizes as Keff approaches 1 [0.5]. (1.0)

REFERENCE
Review Book, Chapter 41, p 21 GLJ 85

ANSWER 5.03 (3.00)

a. 1. Neutron migration length increases as moderator temperature increases, allowing the rod to see more neutrons, increasing rod worth [0.5].

2. Reduced competition from moderator neutron absorption at higher temperature increases rod worth [0.5]. (1.0)

b. The presence of an adjacent rod will reduce the control rod's worth compared to the case with no adjacent inserted rod because the relative flux is depressed with an adjacent control rod inserted. (Half credit given "if increased flux at adjacent rod position assumed and answer is increased rod worth based on this assumption) (1.0)

c. No affect on control rod worth since the relative flux is not changed. (1.0)

REFERENCE
Review Book, Chapter 41, pp 70, 76 GLJ 86

ANSWER 5.04 (3.00)

Rod insertion causes flux shift towards botton of core.[.3] With time, xenon buildup in top of core[.2] due to less burnout[.2] and xenon reduction in bottom of core[.2] due to increased burnout[.2] causes flux to shift towards the bottom of the core even more.[.4] Later, xenon buildup in bottom of the core[.2] due to increased production[.2] and xenon reduction in top of the core[.2] due to xenon decay[.2] causes a flux swing towards the top of the core.[.4] These feedback effects between xenon and power result in an axial power oscillation.[.3] (3.0)

REFERENCE
Review Book, Chapter 3 GLJ 87

ANSWER 5.05 (3.00)

a. By a factor of e or 2.718 (0.5)

b. SUR = 26/T [0.5]
= 26/90 = 0.289 DPM [0.5] (1.0)

c. 2 decades/0.289 DPM = 6.9 min (0.5)

d. P = Po10EXP(SUR)(t) [0.5]
= (10)10EXP(0.289)(3)
= (10)(7.36) =73.6MW [0.5] (1.0)

REFERENCE
Review Book, Chapter 3 GLJ 88

ANSWER 5.06 (2.00)

a. Boron differential worth is less negative [0.5] due to reduced boron density [0.5]. (1.0)

b. Boron differential worth is less negative [0.5] due to increased competition [0.5]. (1.0)

REFERENCE
Review Book, Chapter 41, p 77 GLJ 89

ANSWER 5.07 (3.40)

a. 1. Decrease
   2. Increase
   3. Increase
   4. Increase
   5. Decrease
   6. Decrease [0.5 each] (3.0)

b. Lo-Lo S/G Level (0.4)

REFERENCE
Review Book, Chapter 3 GLJ 90

ANSWER 5.08 (3.60)

a. Decrease - Heat flux at which DNB occurs is reduced
b. Decrease - Actual heat flux increases (Bottom) OR Increase - Actual heat flux decreases (Top)
c. Increase - Heat flux at which DNB occurs is increased
d. No change - Neither actual heat flux or heat flux at which DNB occurs has changed. OR Decrease if reduced flow or non uniform buildup, reducing subcooling, is assumed OR Increase slightly if increased turbulence considered due to rough buildup, increasing subcooling.
e. Increase - Actual heat flux decreases
f. Increase - Heat flux at which DNB occurs is increased [0.6 ea] (3.6)

NOTE: Explanations given in terms of subcooling also accepted

REFERENCE
Review Book, Chapter 43, p 136 GLJ 91

ANSWER 5.09 (1.00)

Advantage - Increases NPSH for condensate pump (decreases chance of pump cavitation) [0.5].

Disadvantage - Steam used to reheat condensate could be used for production of electricity (heat rejected from condensate is lost to the cycle) [0.5]. (1.0)

REFERENCE
Review Book, Chapter 3 GLJ 92

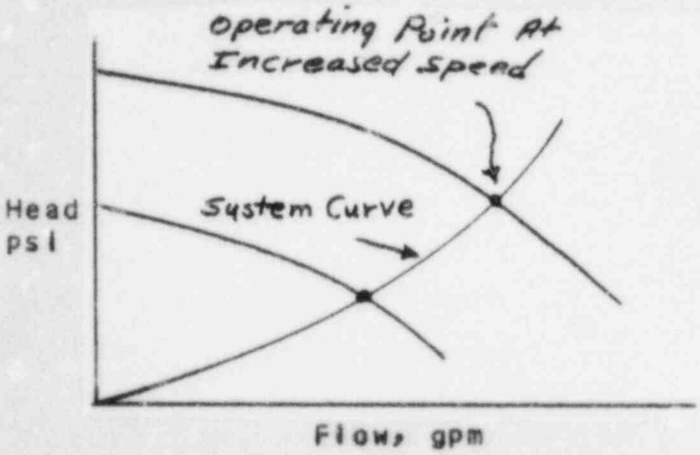
ANSWER 5.10 (2.00)

a. 1. (25)(0.5)(0.5)(0.5) =3.125 HP

2. (50)(0.5) = 22.5 gpm

3. (250)(0.5)(0.5) = 62.5 psi [0.5 each] (1.5)

b.



(0.5)

REFERENCE
Review Book, Chapter 42, pp 118 -120 GLJ 93

ANSWERS -- WOLF CREEK -84/05/15-JEFFRIES, G.

ANSWER 6.01 (3.00)

a. LiOH (enriched in Li-7) [0.5]. Reduced corrosion and crud deposition at higher pH [0.5]. (1.0)

b. Oxygen in the RCS is formed by radiolysis of water. Hydrogen gas from the VCT enters the RCS to force the back reaction of hydrolysis; i.e., oxygen and hydrogen combine to form water. (1.0)

c. To scavenge oxygen [0.5]. Demins taken out of service to reduce resin damage from chemical interaction (with ammonia) [0.5]. (1.0)

REFERENCE
NPS 219, Chapter 1, pp 1-8 thru 1-13 GLJ 94

ANSWER 6.02 (2.50)

INHERENT

Outsurge of water from pressurizer due to Tave drop from trip [0.5] causes bubble volume to increase, reducing pressure below P-sat, which results in water flashing in the pressurizer to reduce the extent of the pressure decrease from the outsurge [1.0].

AUTO

As pressure decreases, variable heater power increases to full power at 2220 psig (compensated error of -15 psig) [0.5] and backup heaters are energized at 2210 psig (compensated error of -25 psig) [0.5] (2.5)

REFERENCE
NPS 215, Chapter 3, p 3-29 & Chapter 6, p 6-12 GLJ 95

If insurge from Tave increase assumed without stating a valid cause of the increase (e.g. steam dumps not operating) 0.5 points deducted for error in assumption and the remaining answer corrected for assumption (CFA) made. If steam dump failure assumed, no deduction and remaining answer CFA made.

ANSWERS -- WOLF CREEK -84/05/15-JEFFRIES, G.

ANSWER 6.03 (2.45)

a. PRT (0.25)
b. -Control RCS pressure while in solid plant control [0.25]
   -Maintain backpressure to prevent flashing in orifices [0.25] (0.5)
c. To the VCT (0.25)
d. -Manual
   -Dilute
   -Alternate Dilute [0.1 each] (0.3)
e. -Auto
   -Manual
   -Borate
   -Alternate Dilute [0.1 each] (0.4)
f. -Lo-lo VCT level
   -SI Signal
   -Boron Dilution Protection Signal [2 required, 0.25 each] (0.5)
g. RCDT (0.25)

REFERENCE
NPS 217, Chapter 1, pp 1-6,-10,-13,-20 & Chapter 2 pp 2-21,-25,-29,-35
GLJ 98

ANSWER 6.04 (2.50)

a. -Manual [0.2] 2/2 switches per set 2/4 overall [0.2]
   -Hi-3 Containment Pressure [0..2] 27 psig [0.2] 2/4 [0.2] (1.0)

.3 .1 .1
b. -Shuts CCW supply to RCP's, RCDT HX & Letdown HX (CISB valves HV-58 & 71) [0.5]
   .5
   -Shuts CCW return to RCP thermal barriers (CISB valves HV-61 & 62) [0.5]
   .3
   -Shuts CCW return to RCP coolers (except thermal barriers) RCDT HX & Letdown HX (CISB valves HV-59 & 60) [0.5] (1.5)

REFERENCE
NPS 217, Chapter 5, p 5-16 and Technical Specifications, p3/43-15 GLJ 99

ANSWER 6.05 (2.50)

a. -CVCS seal injection ..pplies 8 gpm to the RCP [0.2]
-5 gpm flows past the radial bearing and thermal barrier and into the RCS [0.3]
-3 gpm flows through the No.1 seal and returns to the CVCS seal return [0.3]
-3 gph flows through the No. 2 seal and discharges to the RCDT [0.3]
-Reactor makeup water (standpipe) supplies 800 cc/hr to the No. 3 seal (double dam) [0.3]
-400 cc/hr flows through the No. 3 seal to the No. 2 seal leakoff line [0.3]
-400 cc/hr flows through the No. 3 seal to the normal containment sump [0.3] (2.0)

b. Insufficient lubrication/cooling flow through the No. 2 seal. (0.5)

REFERENCE
NPS 215, Chapter 4, pp4-10 thru 4-22 & NPS 217, Chapter 1, p1-18 GLJ 100

ANSWER 6.06 (2.00)

a. Pressurizer level decreases (slowly) [0.25]
Letdown isolates (heaters off) on low level [0.25]
Pressurizer level increases (slowly) (heaters re-energize [0.25]
Hi pressurizer level trip [0.25] (1.0)

b. Rods insert [0.25]
Tave decreases [0.25]
Pressurizer pressure (level) decreases OR steam pressure (temperature) decreases [0.25] (either answer acceptable)
Low Pzr pressure trip OR low steam line pressure SI [0.25] (either answer acceptable) (1.0)

REFERENCE
NPS 215, Chapter 5, p 5-13 & Chater 6, pp 5-16,-17 GLJ 101

ANSWER 6.07 (3.85)

a. Program dp, derived from total steam flow of all S/G, 45 to 195 psid/0-100% steam flow [0.5]. This signal goes through a lag circuit to give the system a slower response to large and rapid steam flow changes [0.3]. Actual dp, steam header pressure to feed header pressure [0.5]. Actual dp and programmed dp compared and error signal sent to the master speed controller [0.5]. (1.8)

b. Controlling flow channel failure high will cause actual flow to decrease so S/G level will decrease [0.5]

Rtr trip due to low-low S\G level, ~~17.2%~~ 32.3%, 2/4 channels on any one S/G [0.5] (1.0)

c. 1. Manual control by the operator
   2. P-14
   3. SI
   4. Rtr trip (P-4) coincident with Low Tave signal (564 F)
   5. S/G Lo-Lo Level [3 required, 0.35 each] (1.05)

REFERENCE
NPS 223, Chapter 6, pp 6-15 thru 6-21 and 6-25 & ~~Review Book, Chapter 1, p 17~~ Technical Specifications P 2-6, and ~~W~~ Dwg 7250D64, SH-13, -15 GLJ 102

ANSWER 6.08 (2.70)

a. 1. SI Signal
   2. Low voltage on the respective emergency bus [0.3 each] (0.6)

b. 1. Overspeed
   2. High jacket water temperature
   3. Low lube oil pressure
   4. Engine start failure
   5. Generator L/O relay energized OR Differential Relay
   6. High crank case pressure [5 required, 0.3 each] (1.5)

c. 1. Differential relay OR Generator L/O relay
   2. Diesel engine trip [0.3 each] (0.6)

REFERENCE
Review Book, Chapter 11 & NPS 213, Chapter 3, P 3-10 & Chapter 4, p 4-18, 4-22 GLJ 103

ANSWER 6.09 (1.50)

a. 1. Limit containment pressure during a steam break (by limiting the AFW flow to the faulted S/G).

2. Limit the pressure drop on the feedwater headers during a feed line break.

3. Limit damage to the AFW pumps due to runout. [2 req'd, 0.5 ea] (1.0)

b. Each accumulator also supplies a S/G PORV. (0.5)

REFERENCE
NPS-223, Chapter 2, p 2-9 & Chapter 5, pp 5-9,5-10 GLJ 122

ANSWER 6.10 (2.00)

a. 2
b. 4
c. 3
d. 1 [0.5 each] (2.0)

REFERENCE
Reactor Vessel Level Instrument System Description, pp1,2,7,13 GLJ 123

ANSWERS -- WOLF CREEK -84/05/15-JEFFRIES, G.

ANSWER 7.01 (3.50)

a. 1. Failure of any RCCA to fully insert following a reactor trip. (Also accept failure of 2 or more RCCA to fully insert... per ES-02, OR ATWT per FRS1)
2. Control rod height below the insertion limit. (OR SDM Limits not met)
3. Failure of the Reactor Makup Control System
4. Uncontrolled RCS cooldown
5. Unexplained or uncontrolled reactivity increase [0.3 each] (1.5)

b. 1. Open the immediate boration control valve (HV-8104)
2. Start both boric acid transfer pumps
3. Verify proper flow indication on the Immediate Boration Flow Meter. [0.5 each] (1.5)

c. Flowpath is through manually operated Alternate Immediate Boration valve (V-177) and the boric acid flow control valve (FCV-110A)

OR

Flowpath is through the RWST supply valves to the charging pump suction header (LCV-112D and E). [Either answer acceptable] (0.5)

OR

Flowpath is through the BIT

REFERENCE
NPS 217, Chapter 2, pp 2-38 thru 2-41 & BG-0-01, FR-S1 GLJ 104
ES-02, Technical Specifications pp 3/4 1-1, 1-3; 9-1.

ANSWER 7.02 (3.00)

a. 5
b. 2,4
a. & b. 1,3,6 [0.~~5~~ .33 each] [If additional (incorrect) answers given, divide points equally among all answers] (3.0)

REFERENCE
EMG E-0, EMG E-1, BB-0-04, & Transient Traces Book GLJ 105

ANSWERS -- WOLF CREEK -84/05/15-JEFFRIES, G.

ANSWER 7.03 (1.50)

a. The readings would be approximately the same [0.5]. GM meter readings are not dependent on energy level of source radiation (as each interaction results in complete ionization of the gas in the detector, giving a pulse) [0.5]. Also, if appropriate compensating filter assumed in GM tube, full credit will be given for Source A reading approximately twice Source B reading. (1.0)

b. Greater than. (0.5)

REFERENCE
Radiation Protection Manual pp 9-7, A-3 & 10CFR20 GLJ 106

ANSWER 7.04 (2.00)

a. 1. Auxiliary pzr spray (0.5)
2. Pzr PORV (0.5)

b. The potential for void formation in the reactor vessel head increases if the vessel head cooling provided by the CRDM fans is not available. (1.0)

REFERENCE
EMG ES-04, p 4-6 GLJ 107

ANSWER 7.05 (3.50)

a. Energize one group of pressurizer heaters [0.5] to utilize pzr spray flow to minimize boron concentration difference between the RCS and pzr [0.5] OR withdraw shutdown banks [0.5] to provide shutdown reactivity (if criticality should be achieved by dilution) [0.5] (1.0)

b. Immediately stop the dilution [0.75] until core reactivity has been evaluated [0.25]. 01 (1.0)

c. Restore Tave to 551 or greater (within 15 min) or be in Hot Standby (within the next 15 min). (0.5)

d. 1. Criticality shall not be achieved by boron dilution((0.5)
2. Cannot add positive reactivity by more than one controlled method at a time. (0.5)

For d.1 - will also accept "none" if state assumption of initial criticality or initial criticality after refueling.

REFERENCE
GEN-00-003, pp 1,2 GLJ 108

ANSWER 7.06 (3.00)

a. 8
b. 9
c. 5
d. 2
e. 7 [0.6 each] (3.0)

REFERENCE
Radiation Protection Procedures, Section 6.3 and HP Notes, Lesson 1, pp 1,4 and Lesson 4 pp 2 thru 4 GLJ 109

ANSWER 7.07 (3.00)

-RCS subcooling [0.5]. Required to be greater for adverse containment due to potential error in pressure indication [0.5].

-S/G level [0.5]. Required to be greater for adverse containment due to potential reference leg heatup [0.5].

-Pzr level [o.5]. Required to be greater for adverse containment due to potential reference leg heatup [0.5] (3.0)

REFERENCE
EMG E-1, pp 5 & 16 GLJ 110

ANSWER 7.08 (3.00)

a. Release of fission gas. (0.5)

b. 1. Mechanical finger redundancy
2. Actuating air supply cutout
3. Positive downward force to release 3 required [0.5 each] (1.5)
See attached page for additional correct answers

c. [2 required, 0.5 each]
1. Fuel Handling stopped
2. Area Evacuation
3. Verify Containment Purge Isolation
4. Health Physics Verification of Alarm (1.0)
1.a Place items in transfer in safe condition

REFERENCE
Review Book, Chapter 8 GLJ 111

Additional correct answers to 7.08 b

4. Man. crane hoist cutout on overload
5. " " interlock prevents movement unless gripper is engaged or disengaged
6. " " " " " " properly positioned over core or transfer sys.
7. Gripper remains engaged on loss of air (air pressure to actuate)
8. Man. crane lateral movement restricted by cutouts to prevent contact of fuel with other components
9. Crane cannot be moved laterally unless gripper is fully withdrawn
10. Upender cannot be lowered unless man. crane is positioned away from upender or the gripper is in the full raised position
11. Loss of power to hoist motor engages brake
12. Gripper interlocked to prevent release unless completely lowered in vessel or transfer system.
13. Cable overhauling hoist interlock with brake
14. Long/short fuel handling tool in SFP area are positive locking.

ANSWERS -- WOLF CREEK -84/05/15-JEFFRIES, G.

ANSWER 7.09 (1.00)

a. Site Area Emergency (0.5)

b. Alert (0.5)

REFERENCE
EPIP ADM 12-6.1 and 12-6.3c GLJ 112

ANSWER 7.10 (1.50)

1. Subcriticality
2. Core Cooling
3. Heat Sink
4. Integrity
5. Containment
6. Inventory [0.2 each and 0.3 for order] (1.5)

REFERENCE
EMG F-0 GLJ 124

ANSWER 8.01 (1.50)

-Modes 1,2 -- Be in HSB with pressure within its limit (in one hour) [0.5]

-Modes 3,4,5 -- Reduce pressure to within its limit (within 5 minutes) [0.5]

-All Modes -- Notify the NRC Operations Center immediately (within one hour) [0.5] (1.5)
OR
Comply with Admin Technical Specification 6.7.1 [0.5]

REFERENCE
Technical Specifications, pp 2-1,6-13 GLJ 113

ANSWER 8.02 (1.50)

a. The QPTR is determined by using the incore moveable detectors. (0.5)

b. Reactor power must be reduced (to less than 75%) [0.5] and the Power Range High neutron flux trip setpoint must be reduced (to less than 85% within 4 hours)[0.5]. (1.0)

REFERENCE
Technical Specifications, pp 3/4 2-13,3-5 GLJ 114

ANSWER 8.03 (3.00)

a. Interval requirement not exceeded [0.5]. Eight days does not exceed 1.25 times the specified interval [1.0]. (1.5)

b. Interval requirement exceeded [0.5]. The last 3 consecutive intervals exceed 3.25 times the specified interval [1.0]. (1.5)

REFERENCE
Technical Specifications, pp 3/4 0-2, 5-10 GLJ 115

ANSWER 8.04 (4.00)

a. 1 gpm [0.3]--it is sufficiently low to allow for early detection of additional leakage [0.5]

b. 10 gpm [0.3] -- allowance for leakage from known sources which would not interfere with detection of unidentified leakage [0.5]

c. 0 gpm [0.3] -- may be indicative of an impending gross failure [0.5]

d. 32 gpm or 8 gpm per RC pump controlled leakage [0.3] (at 2235 ±20 psig) -- that SI flow will not be less than assumed in accident analysis in event of a LOCA [0.5]

e. 1 gpm [0.3] -- ensures the dosage contribution from tube leakage will be a small fraction of 10CFR100 limits in event of a Steam Generator Tube Rupture or Steam Line Break [0.5] (4.0)

REFERENCE
Technical Specifications, p B3/4 4-4,3/4 4-19 GLJ 116

ANSWER 8.05 (2.75)

a. 1. Two SRO's [0.4]. One Shift Supervisor [0.25] and one Supervising Operator [0.25].

2. Two licensed Reactor Operators [0.4]. They may hold SRO licenses [0.25].

3. Four Nuclear Station Operators [0.4] (1.95)

b. 5 members [0.4], of which, 3 members of the minimum shift crew may not be included [0.4] (0.8)

REFERENCE
ADM 02-001, pp 3&4 GLJ 117

ANSWERS -- WOLF CREEK -84/05/15-JEFFRIES, G.

ANSWER 8.06 (3.00)

a. A human DNO Tag (0.75)

b. Only a single component (0.75)

c. For the duration of the shift for the Shift Supervisor who authorized the Clearance. (0.75)

d. The name of the individual performing the Human DNO Tag function. (0.75)

REFERENCE
ADM 02-100, p 12 GLJ 118

ANSWER 8.07 (3.00)

a. The Reactor Engineering Supervisor, or his designee. (0.5)

b. The Operations Supervisor. OR The Operations Superintendant OR the Operations Shift Supervisor or his designee (any of these acceptable) (0.5)

c. 1. Grid Locations
2. Top Nozzle
3. Bottom Nozzle [0.5 each] (1.5)

d. The shipping container support frame OR upender/transfer cart container (either acceptable for full credit) (0.5)

REFERENCE
FHP 01-001, pp 2 & 3, ADM 01-004, P. 2, ADM 10-007, P. GLJ 119
NPS 219, Chapter 4, PP 4-31, 4-33

ANSWER 8.08 (2.50)

a. The Shift Supervisor (0.5)

b. 1. The Plant Manager
2. The Call Superintendant [0.5 each] (1.0)

c. The Supervising Operator (0.5)

d. The Superintendant of Operations (or his designee) (0.5)

REFERENCE
EPIP, ADM 12-1.2, pp 1,3 GLJ 120

ANSWER 8.09 (1.75)

1. Control Room
2. TSC
3. OSC (or the Shop Building)
4. EOF
5. Ambulance
6. Ransom Memorial Hospital [5 required, 0.35 each] (1.75)

REFERENCE
EPIP, ADM 12-17.0, p 1, Emergency Plan P 4.1-3. GLJ 121

ANSWER 8.10 (2.00)

a. Two source range instruments [0.2], each with continuous visual indication in the control room [0.2] and one with audible indication in the containment [0.2] and in the control room [0.2]. (0.8)

b. 1. Held in place by a minimum of 4 bolts [0.2]
2. One door in each air lock closed [0.2]
3. Closed by an isolation valve [0.2], flange [0.2], or manual valve [0.2] or be capable of being closed by an automatic containment purge isolation valve [0.2]. (1.2)

REFERENCE
Technical Specifications, pp 3/4 9-2,9-4 GLJ 125