ANO 1 EXAMINATION REPORT

Report Number: 50-313/OL-85-01

Docket No: 50- License No.: DPR-51

Licensee: Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Examinations administered at Arkansas Nuclear One Unit 1 (ANO 1)

Chief Examiner: R. A. Cooley
R. A. Cooley 2-13-85
Date

Approved: R. A. Cooley
R. A. Cooley, Chief Section 2-13-85
Date

Summary

Examinations conducted on December 11, 1984.

Written and oral examinations were administered to thirteen Reactor Operators. Four Reactor Operator candidates failed these examinations.

ANO 1 EXAMINATION REPORT

Report Details

1. Examination Results

RO Candidates

Total	Pass	Fail	Passed
13	9	4	69

2. Examiners

R. Cooley, NRC, Chief Examiner
B. Gore, PNL
J. Huenefeld, PNL
J. Pellet, NRC

3. Examination Report

This Examination Report is composed of the sections listed below.

A. Examination Review Meeting

B. Facility Examination Comments and Resolutions

C. Exit Meeting Minutes

D. ANO 1 RO Examination

E. ANO 1 RO Examination Key

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

A. Examination Review Meeting

At the conclusion of the written examination, Jim Huenefeld met with A. E. Elliott, E. D. Wentz, K. W. Canitz, A. South, and E. Force of the training department and C. Zimmerman of the Operations department to review the exam and answer key. The facility provided Mr. Huenefeld and the Chief Examiner with written comments resulting from that review. Those comments and resolutions are included as part of this report. Mr. Huenefeld is in agreement with the Utility comments with the following exceptions:

Comment Resolution

1.1 Answer "a" is incorrect. Heat transfer occurring during a constant pressure process does not imply the existence of a phase change whereas heat transfer occurring during a constant temperature process does imply a phase change. It is the presence of a phase change that causes the lack of proportionality between the primary and secondary delta Ts. The concept is fundamental and important toward the understanding of heat transfer occurring between the secondary and primary system. The question stands on its own merit.

1.4 The question was designed to examine the RO candidates knowledge with respect to behavior after a trip. Indications in the control room include both linear and logarithmic displays of neutron power. An understanding of how the logarithmic display relates to reality is a reasonable expectation of an RO candidate. The question took a new and novel approach requiring some original thinking on behalf of the RO candidates.

1.5 The facility's concern stems from a misconception about NPSH held by the individual responsible for teaching the topic. The question stands on its own merit.

1.13 The question stands on its own merit. The candidates were told that explanations may be offered with True/False questions and Multiple Choice questions. One candidate did suggest that depending upon which group was chosen and which rod was chosen the statement may be true. Full credit was given for the explanation.

1.15 and
1.16 The questions are appropriate, and stand on their own merit.

1.18 The examiner agrees that it was poor judgement on his part to make the question worth 2.5 points. One-half credit was given for recognizing that the curve was integral. The question itself is regarded to be appropriate and relevant. A knowledge of the reactivity of all control rods, both integral and differential, is fundamental for understanding the instantaneous effects that rod motion may have on reactor conditions. This is particularly true of the APSRs that may have either positive or negative reactivity effects depending upon their position and direction of motion.

2.3 Percent power is the macroscopic parameter monitored by the operator during startups and shutdown, and it is important that the operator know approximately when the subject feedwater control transition will occur. The numbers were taken directly from facility training material as noted on the key.

2.15 The answer was taken directly from OP 1104.04, page 18. "b" is considered the only correct answer.

3.6 The facility training material also refers to "condenser dump valves" (see attachment from "Steam Systems Training Plan").

PLAN OF INSTRUCTION	INSTRUCTOR NOTES

QUESTION 3.6

b) Condenser dump valves interlocked through ICS to prevent opening with less than 20" vacuum.
c) EFW pump steam supply valves interlocked through ICS on loss of both feed pumps or less of all RCP's.

2) SLBIC Interlocks

a) SLBIC actuation signal causes:

(1) Both MSIV's shut
(2) EFW pump steam supply valve from affected OTSG opens.

b) SLBIC Exercise Logic signal causes:

(1) MSIV solenoid valve opens on time delay to:

(a) Bleed air off MSIV operating piston
(b) Allows valve to travel 20% shut
(c) Solenoid valve shuts
(d) Valve returns to open position

c) Bypass Logic

(1) SLBIC must be bypassed if steam pressure <600 psi to operate MSIV's.

d) Reset Logic

(1) Allows SLBIC closure signal to be overridden by use of valve hand-switches.

b. Reheat Steam

1) Ramp valves to second stage reheat and second stage reheat steam inlet valves interlocked through ICS.

a) Ramp valves open slowly to provide gradual heatup (~2 hours).
b) Inlet valves open 1/2 hour after ramp valve fully open.

2) Interceptor and stop valves interlocked through ICS to shut on load rejection.

a) Prevents turbine overspeed.

4.11 The question is worded:

The maximum allowable heatup rate during a normal heatup is interpreted as being: (Select one.)

The only reasonabl(;wer to this question was "d," no more than 1.67°F per minute.

4.14 The question stands on its own merit.

No questions were deleted or compromised as a result of the exam review.

B. Facility Examination Comments
and Resolutions

FORM 291
REV 11-70

ARKANSAS POWER & LIGHT COMPANY

INTRA COMPANY CORRESPONDENCE

Arkansas Nuclear One
Russellville, Arkansas
December 13, 1984

ANO-84-14580

MEMORANDUM

TO: Ralph Cooley

FROM: Ed A. Force Ed Force

SUBJECT: Arkansas Nuclear One
RO Examination

Attached are written comments on the RO License Examination administered on 12/11/84 at this facility.

EAF:rab

Attachment

cc: ANO-DCC

NRC RO EXAMINATION
GENERAL COMMENTS

SECTION 1

Section 1 of the exam was more Engineering oriented than operator oriented. Math concepts were over empathized. The Exam did not address the appropriate knowledge that a reactor operator must have. The safety aspects of reactor operations were not addressed. The test did not address operator oriented items such as MTC, FTC, Samarium, Rod worth hot and cold, EOL AND BOL Xenon worths or reactivity balances.

SECTIONS' 2, 3 and 4

These section were much better in that they did in most cases, check the candidate for operator oriented knowledge. Several questions were hard to understand and required an interpretation on the part of the candidate as to what was really being asked.

A majority of the true and false questions had circumstances where the answer was not a cut and dry true or false answer.

On at least one of the multiple choice questions, there were no correct answers (2.15)

NRC RO EXAMINATION
SECTION 1

1.4 (Q) Draw graphs showing the count rate after a reactor trip. Show as a linear function and as a log function.

(R) Conversion of linear to log and log to linear is strictly a math exercise. It bears no relationship to what reactor power is doing. The ANO reactor theory manual, from which the students were taught, discusses the power decrease after a trip in terms of power and not count rate. The question was not appropriate to determine an understanding of reactor theory, but was more a test of reading skills. A more appropriate way to ask may be: "Reproduce the traces drawn by the intermediate (log) and power range (linear) recorders after a reactor trip."

1.5 (Q) Given a tank full of water and several other initial condition the water was pumped from the tank. (A) What would be the final pressure? (B) Would the pump cavitate if the level went below 5 ft. with the vent closed? (C) Would the pump drain the tank with the vent open?

(R) There was a discussion between the review members and the examiner as to the definition of NPSH. The examiner contends that NPSH is an absolute value, yet the question gave a value of 5 ft with no units. The candidates should be given credit for the question if they used either absolute or gauge pressure to calculate NPSH.

In addition, the question was another example of a poorly written question where the candidate must first interpret what the question is asking then attempt to answer it.

~~1.9~~ 1.1 (Q) Heat transfer in the primary loop is proportional to the core ΔT (T_h - T_c), while the heat transfer in the secondary loop is not proportional to the ΔT across the OTSG, (TSTM - T_{feed}). Why?

(R) This was a multiple choice question and considering answer (a) and answer (d) it is the contention of the training staff that either answer could be correct for an OTSG, therefore, the candidates should be given credit for either.

1.13 (Q) Will the prompt drop be greater from dropping one rod as opposed to dropping a bank or group of rods.

(R) The question is not well defined. Depending on the location of a single rod, its worth might approach the worth of the group, thus the prompt drops would be about equal.

1.15 (Q) What would make power shift to the bottom of the core?

(R) Choice (d), the correct answer, was poorly stated. The answer ended with the statement....."therefore the rod index." This may lead the student to believe that part of the answer was left off.

1.16 (Q) Two identical cores have different rod heights a criticality. (Multiple choice)

(R) The question was poorly worded. Why not talk about our core under different conditions.

1.18 (Q) Given a graph of a APSR rod worth, is it an integral or differential curve, and explain your answer by drawing the other curve.

(R) The student should be able to identify an integral or differential curve. This is as far as it should go. To ask a candidate, especially an RO candidate, to derive a differential curve is beyond the scope of what is expected of an RO. In addition, the question was worth 2.5 points, or 10% of the section, which was the highest value of any question in Section 1.

NRC RO EXAMINATION
SECTION 2

2.2 (Q) Main Feedwater pump trips

(R) 1) Answer (i) no longer applies, (low flow trip)

2) No setpoints required for full credit

2.3 (Q) Main Feedwater Block Valves Setpoints

(R) Answers b and c should be in % ICS demand not % power.

2.6 (Q) Emergency Feedwater pumps auto start signals.

(R) In addition to those given on the key, candidates may give the new EFIC system auto starts which are:

Low OTSG level 14.5 inches
ESAS channels 3 or 4
Low OTSG pressure < 600 psig

2.8 (Q) Makeup System valves which operate on ESAS channel 1 and 2

(R) Candidates may also include RCP seal return directs to the Quench Tank.

2.15 (Q) Shutoff head of the LPI pump.

(R) Of the four possible answers, none were correct. The shutoff head is ~180 psi. The answer of 150 psi was the closest to correct, however, 150 psi represents full flow head. Students may choose the next higher number because of this. Credit should be given for either 150 or 290 psi.

2.17 (Q) Nuclear ICW expansion tank level increase.

(R) In addition to the key answer, the following are valid answers:

1) A leaking or failed automatic makeup valve.

2) A leaking cross connect valve between nuclear and non nuclear ICW.

NRC RO EXAMINATION
SECTION 3

3.1 (Q) OTSG level ranges

(R) Candidates may also include the EFIC level range of 6" to 500"

3.3 (Q) Decay heat suction valve interlocks (CV-1050 and CV-1410)

(R) Answer is incorrect; the < 290 psi should read > 290 psi for both valves.

3.4 (Q) List 5 of the 9 RCP starting interlocks

(R) Examiner agrees that setpoints are not required for full credit.

3.5 (Q) SLBIC setpoint and components actuated.

(R) 1) SLBIC Setpoint is 600 psig with no tolerance

2) Candidates may give components actuated by the new EFIC system which are:

"A OTSG"	"B OTSG
close "A" MSIV	Close "B" MSIV
close "A" MFWI	Close "B" MFWI
close "A" Steam to P7A	Close "B" Steam to P7B
open "A" ADV block valve	Open "B" ADV block valve

MSIV = Main Steam Isolation Valve
MFWI = Main Feedwater Isolation Valve
ADV = Atmospheric Dump Valve
P7A = Steam Driven Emergency Feed Pump

3.6 (Q) At what condenser pressure will condenser dump valves close

(R) Proper name is turbine bypass valves, not condenser dump valves.

NRC RO EXAMINATION
SECTION 4

4.1 (Q) List 5 things not associated with RPS which require a manual reactor trip.

(R) Candidates may substitute "EFIC" in the place of "SLBIC". This should be accepted.

4.4 (Q) List 4 indications that natural circulation cooling is occurring.

(R) Question does not specify the items listed in OP 1202.01. The following are also correct answers:

1. T_{cold} tracks OTSG T_{sat}.
2. T_{hot} tracks Incore thermocouples.

4.6 (Q) State the emergency cooldown rate limit and 2 conditions that must be present to allow this limit.

(R) OTSG tube to shell emergency rate or limit of 150°F tube to shell ΔT should also be acceptable as a condition limiting the cooldown rate.

4.10 (Q) If RCPs are secured because of loss of SCM, what additional actions are required of the operator because the RCPs were stopped.

(R) Some answers may reflect new EFIC requirements that allows operators to "monitor" OTSG fill rate, and respond by taking manual control as necessary.

4.11 (Q) What is the normal maximum heat up for the RCS?

(R) Since the question does not specify the procedural heat up limit or the Tech Spec limit the responses of 100°F/hour or 50°F/30 min are also correct and should be considered as such.

4.14 (Q) If a piece of equipment has a caution card on it that conflicts with instructions in a procedure, which takes precedence?

(R) This question calls for interpretation of a statement that does not clearly state a precedent. The procedure (1000.27) states the following.

4.14 (Cont) "When CAUTION Card instructions conflict with requirements specified in procedures, changes shall be made to the affected procedures."

A case can be made to support the key answer or to refute it, depending on how the above statement is interpreted.

C. Exit Meeting Summary

At the conclusion of the exam period, the NRC examiner met with representatives of the plant staff to discuss the results of the oral examinations. The following personnel were present for the exit interviews:

NRC	UTILITY
R. A. Cooley	J. Levine
J. L. Pellet	J. Vandergrift
	E. Force B. Baker E. Wentz A. Elliot

Mr. Cooley reported that nine candidates for Unit One were clear passes on the oral. However, areas of generic weakness were observed during the oral examinations. Some of these weaknesses can be attributed to the plant modifications in progress at this time. The following are some of the weak areas noted during the examinations for more than one candidate:

(1) Problems with procedure changes and use of the Environmental Technical Specifications.

(2) Confusion between the Administrative Procedures and the Technical Specification concerning the Source Range Nuclear Instrumentation.

(3) Use P&IDs and station procedures when discussing systems.

(4) The Reactor Coolant Pumps and their operation were not clearly understood.

(5) The Electrical Distribution System was a weak area.

(6) Understanding what happens to reactor power after a scram, and other reactivity effects.

(7) The use of portable radiation monitors and the different types of radiation.

The Shift Supervisors and the operations staff were very helpful in keeping the Control Room as quiet as possible during the exam. Overall the exams went very well, and most of the candidates did good on the oral examinations. This was passed along to the utility personnel at the exit meeting.

U. S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: ANO - 1

Reactor Type: Babcock and Wilcox

Date Administered: December 11, 1984

Examiner: JC Huenefeld

Candidate: ____________________

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheet. Points for each question are indicated in parenthesis 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	Candidate's Score	% of Cat. Value	Category
25	25	______	______	1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
25	25	______	______	2. Plant Design Including Safety and Emergency Systems
25	25	______	______	3. Instruments and Controls
25	25	______	______	4. Procedures: Normal, Abnormal, Emergency, and Radiological Control
100		______		TOTALS

Final Grade ______ %

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

____
Candidate's Signature

1.0 PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW (25 Points)

1.1 The energy being transferred at the steam generator is proportional to the primary Delta T (i.e., $T_h - T_c$), but not proportional to the secondary Delta T (i.e., $T_{steam} - T_{feed}$). The reason for this is because: (Select one). (1.0)

a) the energy transfer taking place within the steam generator is essentially a constant pressure process.

b) the secondary flow rate is greater than the primary flow rate.

c) the secondary flow rate is less than the primary flow rate.

d) the energy transfer taking place within the steam generator is essentially a constant temperature process.

1.2 In the condenser energy is being transferred to the circ-water. If the circ-water flowrate were reduced slightly while holding generated megawatts constant, the most probable result would be: (Select one). (1.0)

a) that the average temperature of the circ-water will increase slightly.

b) that the amount of energy transferred at the condenser will decrease slightly.

c) that the saturation pressure within the condenser will decrease slightly.

d) that the condenser delta T will decrease slightly.

1.3 Assume that ANO-1 is generating 700 MWe. Using typical parameters for steam temperature and pressure, estimate the steam flow rate. State your assumptions and show your work. (2.0)

1.4 Show the difference between a logarithmic plot of nuclear count rate (i.e., log count rate) versus time after a trip and a linear plot of count rate versus time after a trip by making a basic sketch of both cases. Assume that both plots start from the same initial power level. (2.0)

- Section 1 Continued on Next Page -

1.5 Given a large vented tank 30' in diameter and 60' high with a centrifugal pump taking a suction from its base. The pump is located at a vertical elevation corresponding to the bottom of the tank and it requires 5 ft of net positive suction head (NPSH) to prevent cavitation. The tank is almost entirely full of water and is maintained at 60°F by heaters. The tank is designed such that it could withstand 15 psi differential pressure in either direction. Assume the vent becomes totally clogged with ice while the pump is in operation. Further assume that the pump is of relatively low capacity such that equilibrium conditions are maintained inside the tank. Answer the following questions:

a) What is the lowest pressure that the tank will drop to as the pump continues to remove water from the tank? Explain. (1.5)

b) Will the pump lose NPSH and begin to cavitate prior to reaching a level of 5 ft in the tank? Explain. (State any assumptions.) (1.0)

c) Could the pump continue to pump water at a level below 5 ft without cavitation if the vent were open? Explain. (1.0)

1.6 The feedwater loop demand stations, the reactor demand station, and the S/G/Rx master are in "hand" with reactor power stable at 15% power. The operator commences a power escalation by bumping rods out and T_{ave} goes to 580°F. Which of the following describes the required operator course of action to bring T_{ave} back to 579°F? (Select one.) (1.0)

a) The operator must insert rods slightly, but not as far as they were withdrawn. He must request the I&C technicians to adjust the low level limits.

b) The operator must increase feed flow, restoring OTSG level, and lower the steam header setpoint slightly.

c) The operator must over-feed the OTSG slightly, increasing OTSG inventory, then stabilize at a higher feed flow and a higher OTSG level.

d) The operator does not need to take any action. Doppler feedback will return T_{ave} to 579°F.

- Section 1 Continued on Next Page -

1.7 Assume that a steam bubble has formed in the reactor vessel head during an RCS natural circulation cooldown. Is the collapsing of that bubble a relatively fast process or a relatively slow process? Explain. (2.0)

1.8 According to OP 1103.15, the reactivity balance calculation procedure, achieving a period of less than 5 seconds is a reportable event. What startup rate would this period correspond to? (Show your work) (1.5)

1.9 If 100% FP equilibrium Xe worth is -2.57 Delta K/K then 50% FP equilibrium Xe worth is: (Select one) (1.0)

a) -0.257 Delta K/K

b) -1.29 Delta K/K

c) -2.10 Delta K/K

d) -0.51 Delta K/K

1.10 The time to reach equilibrium Xe after a significant power change is approximately: (Select one) (1.0)

a) 4 - 8 hours

b) $\sqrt{\%\ \text{power}}$

c) 40 hours

d) 72 hours

1.11 TRUE or FALSE. The amount of reactivity needed to cause prompt criticality decreases over cycle life. (0.5)

1.12 Can reactor power be reduced at a rate greater than -1/3 DPM (-80 second period)? Explain. (1.5)

- Section 1 Continued on Next Page -

1.13 TRUE or FALSE. Dropping one rod results in the same sized prompt drop as dropping an entire group of rods. (0.5)

1.14 The reason that it takes longer and longer for startup rate (SUR) to reach zero as the reactor nears criticality is: (Select one) (1.0)

a) because the effect of the delayed neutrons is becoming more dominant than prompt neutrons.

b) because the decay of SUR is becoming dominated by the decay of the longest lived precursors.

c) because as K effective nears one (1) the effect of the previous generations of neutrons is becoming more dominant in the neutron population.

d) because as the rods are withdrawn from the core they are providing less shielding of the source range detectors.

1.15 Which of the following correctly describes a condition leading to shifting the power distribution to the lower regions of the core? (Select one) (1.0)

a) increasing flow through the core and therefore increasing the static pressure in the lower regions of the core.

b) lowering T_c, cooling the fuel in the lower regions of the core, and therefore suppressing doppler feedback.

c) increasing the boron concentration and therefore the rod index.

d) decreasing the boron concentration and therefore the rod index.

- Section 1 Continued on Next Page -

1.16 Assume two (2) reactor core configurations that are identical in every respect with the exception of one characteristic. Select the characteristic that would directly cause the critical rod height of one reactor to be different from the critical rod height of the other reactor. (Select one) (1.0)

a) Source strength

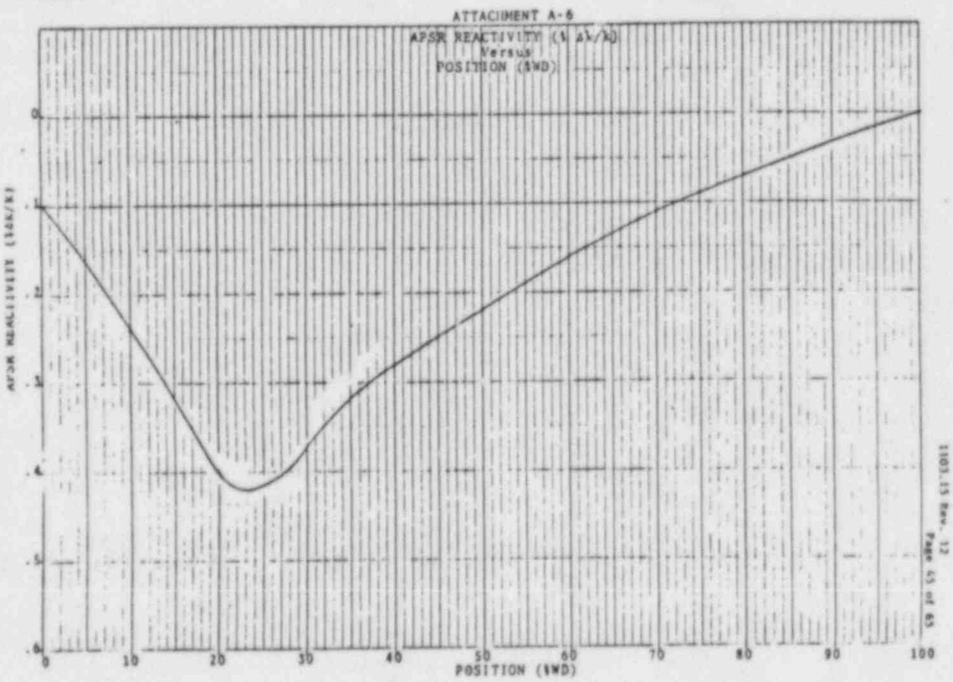
b) Rod speed

c) Delayed neutron fraction

d) Number of neutrons resulting per fission

1.17 Technical Specifications describe power peaking limits (approximately 20.1 KW/ft), but state that the peaking is not a directly observable quantity. What limit is observed to prevent exceeding the power peaking restrictions? (1.0)

1.18 Is the plot of APSR reactivity shown below a differential plot or an integral plot of reactivity? Explain your answer by drawing a sketch of what the opposite kind of plot would look like. (2.5)



- End of Section 1 -

2.0 PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS (25 Points)

2.1 The Auxiliary Feedwater Pump is capable of supplying about ______ of full load feedwater capacity. (Select one.) (0.5)

a) 0.3%
b) 3%
c) 30%

2.2 List five (5) of the ten (10) conditions that will cause a MFW pump to trip. (1.5)

2.3 a) At what position (i.e., % open) of the Startup Feedwater control valve will the associated low load block valve begin to open? (0.5)

b) At what power level (increasing) will the the Main Feedwater block valves open, shifting the Main Feedwater Pumps to speed control? (0.5)

c) At what power level (decreasing) will the Main Feedwater block valves go shut, shifting the Main Feedwater pumps to Delta P control? (0.5)

2.4 TRUE or FALSE. A level of 100% on the OTSG Operate Range is less than one-half of the distance to the upper tube sheet. (0.5)

2.5 The size of the condensate water storage tank is based upon the water inventory that will supply _____ hours of decay heat removal operation with the EFW system. Select the correct answer. (1.0)

a) 8

b) 24

c) 80

d) 800

2.6 List the conditions that will result in automatic starting of the turbine driven EFW pump. (2.0)

- Section 2 Continued on Next Page -

2.7 TRUE or FALSE. The Unit-one instrument air system may be interconnected with the Unit-2 instrument air system. (0.5)

2.8 List those Makeup and Purification System (CV) valves that automatically close and those that automatically open upon receiving an ES channel 1&2 (HPI) actuation signal. Common names are acceptable. (3.0)

2.9 Sketch the HPI system piping between the reactor coolant system and the motor operated loop A and B HPI isolation valves (CV-1219, 1220, 1227, and 1228) showing the cross connection piping. (2.0)

2.10 The letdown flow rate is based upon: (Select one) (1.0)

a) purifying one RCS volume/day.

b) purifying twenty four RCS volumes/day.

c) purifying one RCS volume/week.

d) purifying one RCS volume/month.

2.11 TRUE or FALSE. Because normal makeup demands are so low, the makeup pumps are designed to operate for prolonged periods at 40 gpm. (0.5)

2.12 TRUE or FALSE. One gallon per minute of makeup flow is maintained at all times to preclude thermal shock to the injection nozzle. (0.5)

2.13 Why does allowable makeup tank pressure decrease with decreasing makeup tank level? (2.0)

2.14 Show, using a basic sketch, how the low pressure injection headers are cross connected prior to entering the core flood nozzles. (2.0)

- Section 2 Continued on Next Page -

2.15 The shut-off head for the low pressure injection pumps is approximately: (Select one) (1.0)

a) 100 psi

b) 150 psig

c) 290 psig

d) 600 psig.

2.16 Which of the following components are a direct interface between RCS water and ICW water? (More than one answer may be correct.) (1.0)

a) Letdown coolers
b) RCP seal cooling heat exchangers
c) RCP seal return coolers
d) Control rod drive mechanisms
e) RCP motors
f) Reactor building coolers
g) Pressurizer sample cooler

2.17 The level in the intermediate cooling water surge tank (T37B) has been continuously increasing for several hours. Samples indicate no detectable activity or boron concentration. RCS leak rate determinations indicate no unaccountable leakage. What is a probable source of non-nuclear in-leakage into the nuclear ICW loop? (1.5)

2.18 Sketch the 6900/4160 V distribution system from the main generator and startup transformers to the 480 V load center transformers. Show which of the seven 480 volt load centers come off of each 4160 V bus. No other loads need be shown. Label each bus. Individual breaker designations are not required. (3.0)

- End of Section 2 -

3.0 INSTRUMENTS AND CONTROLS (25 Points)

3.1 Using the figure given on page 19 of this exam, sketch in the ranges for the three OTSG level indicators showing the overlap between the ranges. (3.0)

3.2 Which of the following statements is true regarding the function of the "cross limit" circuity of the ICS? (Select one.) (1.0)

a) When feedwater flow is greater than or less that feedwater demand by more than 5%, the reactor is feedwater limited.

b) When neutron power is greater than or less than feedwater demand by more than 5%, the feedwater is reactor limited.

c) When feedwater flow is less than feedwater demand by more than 5%, the reactor is feedwater limited.

d) When neutron power is less than feedwater demand by more than 5%, the feedwater is reactor limited.

3.3 Describe the interlock associated with the decay heat isolation valve (CV-1050) and the decay heat pump suction valve (CV-1410). (1.5)

3.4 List five (5) of the nine (9) interlocks that must be satisfied prior to starting a reactor coolant pump. (1.5)

3.5 a) What is the SLBIC actuation set-point? (0.5)

b) What actions result upon actuation of SLBIC? (1.5)

3.6 At what vacuum are condenser dump valves interlocked shut? (0.5)

3.7 TRUE or FALSE. The unit controls for ESAS actuated equipment allow taking manual control of a component even with an ES actuation signal present. (0.5)

- Section 3 Continued on Next Page -

3.8 TRUE or FALSE. Loss of power to a digital cabinet does not cause ES actuation. (0.5)

3.9 Which of the following is consistent with the Engineered Safeguards Actuation System logic? (Select one.) (1.0)

a) Analog subsystem #1 experiences both an RCS low pressure trip (1500 psig) and an RB high pressure trip (4 psig) resulting in the actuation of ESAS Channel 1, 3, and 5.

b) Analog subsystem #1 experiences an RCS low pressure trip (1500 psig) and Analog subsystem #3 experiences an RB high pressure trip (4 psig). This results in the actuation of ESAS Channels 1, 2, 3, and 4.

c) All three analog subsystems experience an RCS low pressure trip (1500 psig) resulting in the actuation of ESAS Channels 1 through 8.

d) All three analog subsystems experience an RCS low pressure trip (1500 psig) resulting in the actuation of ESAS Channels 1 through 6 only.

3.10 When less than 15% power, the "measured variable" position on the main feedwater pump hand-auto station: (Select one) (1.0)

a) has the same interpretation as it does when greater than 50% power.

b) indicates demanded position of the main feedwater pump turbine governor value.

c) indicates 0-100 psid across the startup and low load control valves.

d) indicates 0-100% demanded main feedwater pump speed.

3.11 When latching the main turbine and bringing it up to speed, a "NOTE" in the plant startup procedure (OP 1102.02) requires manual control of the steam dump valves. Why is this necessary? (1.5)

- Section 3 Continued on Next Page -

3.12 What happens to the low load and startup control valves as the main block valve starts coming open at 50% feedwater loop demand? (1.0)

3.13 Which of the following conditions most correctly defines a situation resulting in the 50 psi bias being applied to the steam dump valve circuitry? (Select one) (1.0)

a) The turbine is synchronized to the grid and the steam dump valves are partially open with unit load demand less than or equal to 15%

b) The turbine is synchronized to the grid and the steam dump valves are partially open with unit load demand greater than 15%.

c) All turbine by-pass valves are closed and unit load demand is at 13% with header pressure exceeding header pressure setpoint by 15 psi.

d) The turbine is in ICS auto with the steam dump valve hand - auto controllers in auto.

3.14 As the main feed block valve opens, what do you expect to see happen to the main feed pump speed? (Assume loop demand remains constant.) (Select one.) (0.5)

a) It remains constant
b) It slows down
c) It increases slightly.

3.15 The high level limit alarm is received. Which of the following is most correct? (Select one). (1.0)

a) OTSG level may or may not be at the high level limit.

b) Feedwater loop demand is frozen as is.

c) Main feed pump speed is frozen as is and the reactor is feedwater limited.

d) Main feed pump speed is frozen as is, but the reactor is not feedwater limited.

- Section 3 Continued on Next Page -

3.16 Using the figure given on page 20 of this exam, sketch in the ranges for the three nuclear instrumentation indicators showing the overlap between the ranges. (3.0)

3.17 The amount of time needed for the incore instruments to indicate a step change in actual incore power within 10% accuracy: (select one) (1.0)

a) is approximately 30 seconds
b) is approximately 3 minutes
c) is approximately 30 minutes
d) is very heavily dependent upon whether the power change was positive or negative.

3.18 TRUE or FALSE: The range of the source range SUR indication on the control board is sufficient to verify the stable negative SUR expected following a reactor shutdown. (0.5)

3.19 TRUE or FALSE: The reason that power range nuclear instrumentation is not compensated for gamma radiation is because the detector is shielded by 4 inches of lead. (0.5)

3.20 What value of source range and intermediate range instrumentation startup rate (SUR) will result in actuation of a SUR rod withdrawal interlock? (1.0)

3.21 TRUE or FALSE: The radiation detector on the line from the Reactor Building Coolers will cause the isolation of its respective coolers even if an ES actuation signal is present. (0.5)

- Section 3 Continued on Next Page -

3.22 Which of the following is true concerning the uncertainty associated with the RCS saturation margin monitoring instrumentation during an increase in RCS temperature? (Select one.) (1.0)

a) Uncertainty increases for both the temperature margin to saturation and pressure margin to saturation.

b) Uncertainty decreases for the temperature margin to saturation, but increases for the pressure margin to saturation.

c) Uncertainty decreases for the pressure margin to saturation but increases for the temperature margin to saturation.

d) Uncertainty decreases for both the temperature margin to saturation and the pressure margin to saturation.

- End of Section 3 -

4.0 PROCEDURES: NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL (25 Points)

4.1 A manual trip of the reactor is required following any automatic reactor trip or if the reactor protection system fails to function upon reaching any of its parameter setpoints. List 5 other specific conditions requiring a manual reactor trip as itemized in the reactor trip procedure (EP 1202.01). (2.5)

4.2 If RCPs are inadvertently left on for >2 minutes following a loss of subcooling margin: (Select one) (1.0)

a) Secure all but one of the RCPs.

b) Secure both RCPs in the loop with the highest subcooling margin.

c) Secure both RCPs in the loop with the lowest subcooling margin.

d) Leave only one pump running in each loop.

4.3 The auxiliary lube oil pump for a makeup pump: (Select one) (1.0)

a) should be run for at least three minutes prior to starting a makeup pump and should remain in operation until the makeup pump is stopped.

b) should be run for at least one minute and then secured prior to starting a makeup pump.

c) should be run until oil pressure is at least 10 psig prior to starting a makeup pump.

d) should be run for at least one minute prior to starting a makeup pump and then stopped after the makeup pump is running.

4.4 List the four (4) verifications for the presence of natural circulation decay heat removal. (2.0)

4.5 Should excessive reactor vessel thermal stress be experienced, a 3 hour soak period is required with one exception. What is that exception? (1.0)

- Section 4 Continued on Next Page -

4.6 During an OTSG tube rupture, what is the maximum allowable (emergency) cooldown rate, and what two (2) conditions must exist before it is allowed? (2.0)

4.7 Which of the following is true regarding the expected behavior of the RCS temperature following a reactor trip? (Select one.) (1.0)

a) RCS temperature should be at 532°F within five to ten minutes.

b) RCS temperature should be at 545°F within two to three minutes.

c) RCS temperature should be at 532°F within two to three minutes.

d) RCS temperature should be at 545°F within five to ten minutes.

4.8 What constitutes a loss of subcooling margin? (1.5)

4.9 What conditions determine whether the reactor vessel has been subjected to excessive thermal stresses? (2.0)

4.10 Assume that reactor coolant pumps are stopped because of a loss of subcooling margin. What must be done to prevent a resultant overcooling? (1.5)

4.11 The maximum allowable heatup rate during a normal heatup is interpreted as being: (Select one) (1.0)

a) 100°F in any one hour period
b) 50°F in any thirty minute period
c) 25°F in any fifteen minute period
d) No more than 1.67°F per minute.

4.12 TRUE or FALSE: During a heatup, deboration may continue while withdrawing one group of safety rods provided that shutdown margin is not lowered below 1.5% Delta K/K. (0.5)

- Section 4 Continued on Next Page -

4.13 TRUE or FALSE: Because of the uncertainty involved in the ECP calculation (+0.5% Delta K/K) it is possible to go critical in the restricted region of the rod withdrawal curve even though the ECP is performed according to procedure. (0.5)

4.14 TRUE or FALSE: When a CAUTION card's instructions conflict with requirements specified in procedures, the CAUTION card takes priority. (0.5)

4.15 TRUE or FALSE: Unless stated otherwise, valves on which a HOLD card is installed will be shut and breakers on which a HOLD card is installed will be open. (0.5)

4.16 Which of the following is the proper action to take upon discovering a fire? (Select one) (1.0)

a) Immediately attempt to control the fire.

b) Summon the fire brigade by using the plant paging system.

c) Evaluate the situation, and immediately report the fire to the control room.

d) Immediately evacuate the area then notify the control room.

4.17 a. What is the maximum background tolerable for using a "frisker"? (0.5)

b. While frisking, a count rate of greater than ________ indicates the possible presence of contamination. (0.5)

4.18 Match the four substances given below with their respective tenth thicknesses for gamma radiation: (1.0)

a) Lead	1) 24"
b) Steel	2) 2"
c) Water	3) 14"
d) Concrete	4) 4"

- Section 4 Continued on Next Page -

4.19 In the event of a remote shutdown AP-1203.29, four operators have responsibilities assigned by procedure. Which operators are these, and where do they station themselves? (Assume auxiliary feed is available.) (2.0)

4.20 What three (3) conditions in combination require the establishment of Reactor Building Integrity? (1.5)

- End of Section 4 -

EQUATION SHEET

Where $\dot{m}_1 = \dot{m}_2$

$(density)_1(velocity)_1(area)_1 = (density)_2(velocity)_2(area)_2$

$KE = \frac{mv^2}{2}$ $PE = mgh$ $PE_1+KE_1+P_1V_1 = PE_2+KE_2+P_2V_2$ where V = specific volume
P = Pressure

$Q = \dot{m}c_p(Tout-Tin)$ $Q = UA\ (Tave-Tstm)$ $Q = \dot{m}(h_1-h_2)$

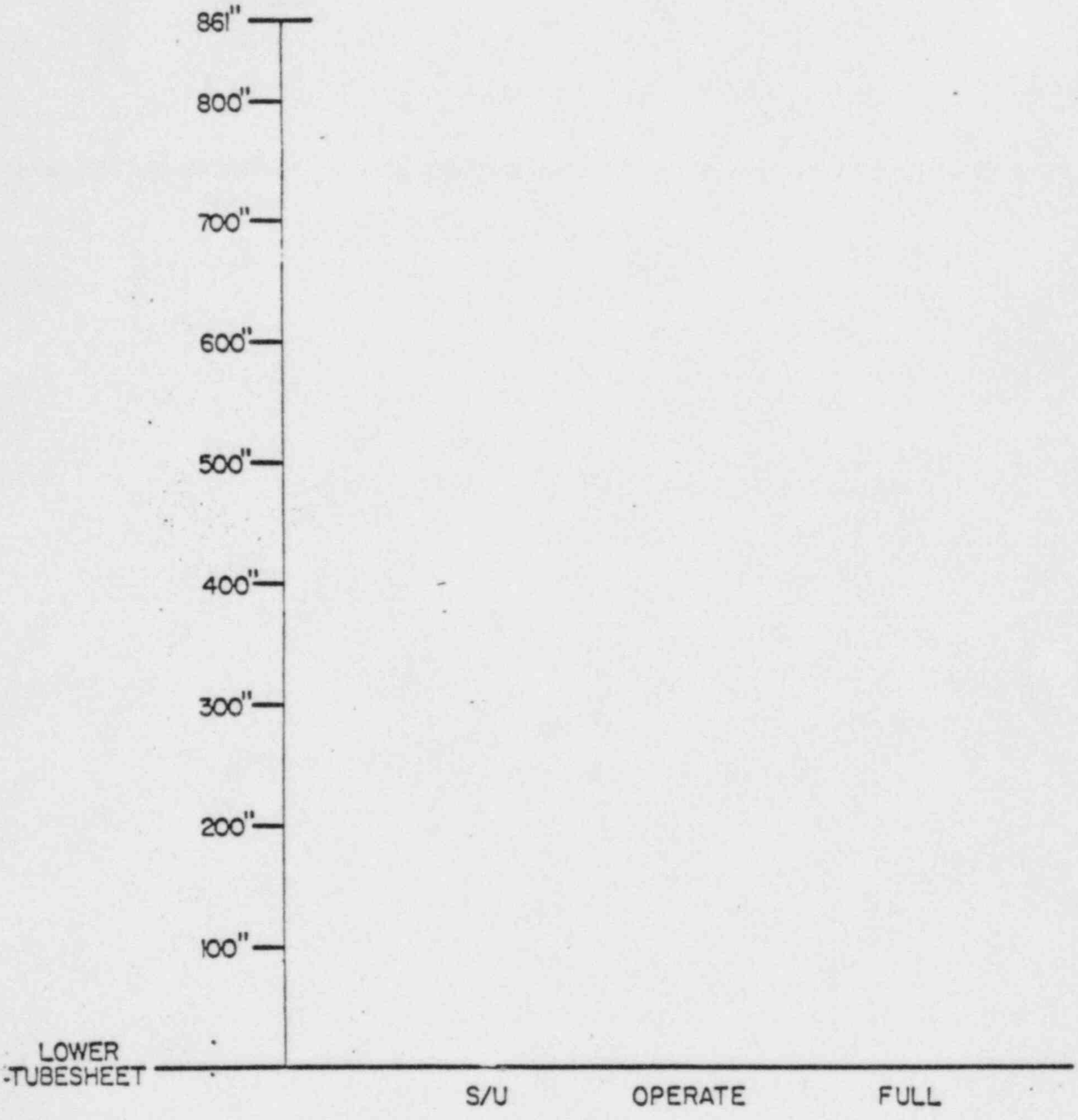
$P = P_o 10^{sur(t)}$ $P = P_o e^{t/T}$ $SUR = \frac{26.06}{T}$

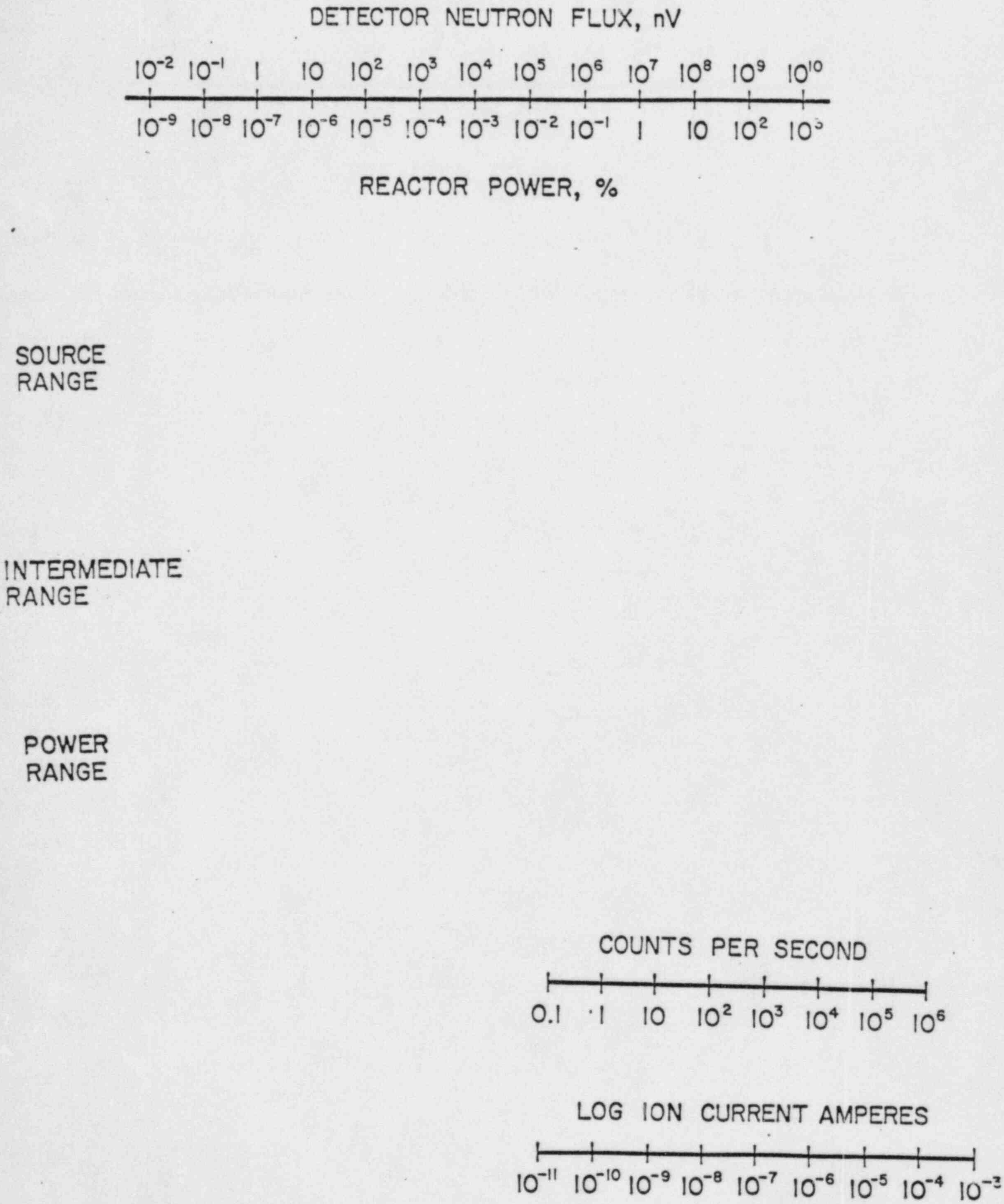
$\text{delta } K = (K_{eff}-1)/K_{eff}$ $CR_1(1-K_{eff1}) = CR_2(1-K_{eff2})$

$M = \frac{(1-K_{eff1})}{(1-K_{eff2})}$ $SDM = \frac{(1-K_{eff}) \times 100\%}{K_{eff}}$

$\text{decay constant} = \frac{\ln(2)}{t_{1/2}} = \frac{0.693}{t_{1/2}}$ $A = A_o e^{-(\text{decay constant}) \times (t)}$

Water Parameters	Miscellaneous Conversions
1 gallon = 8.345 lbs	1 Curie = 3.7×10^{10} dps
1 gallon = 3.78 liters	1 kg = 2.21 lbs
1 ft^3 = 7.48 gallons	1 hp = 2.54×10^3 Btu/hr
Density = 62.4 lbm/ft^3	1 Mw = 3.41×10^6 Btu/hr
Density = 1 gm/cm^3	1 inch = 2.54 centimeters
Heat of Vaporization = 970 Btu/lbm	Degrees F = (1.8) x (Degrees C) + 32
Heat of Fusion = 144 Btu/lbm	1 Btu = 778 ft-lbf
1 Atm = 14.7 psia = 29.9 in Hg	g = 32.174 ft-lbm/lbf-sec^2


861"
800"
700"
600"
500"
400"
300"
200"
100"
LOWER TUBESHEET
S/U
OPERATE
FULL


DETECTOR NEUTRON FLUX, nV
10^-2 10^-1 1 10 10^2 10^3 10^4 10^5 10^6 10^7 10^8 10^9 10^10
10^-9 10^-8 10^-7 10^-6 10^-5 10^-4 10^-3 10^-2 10^-1 1 10 10^2 10^3
REACTOR POWER, %
SOURCE RANGE
INTERMEDIATE RANGE
POWER RANGE
COUNTS PER SECOND
0.1 1 10 10^2 10^3 10^4 10^5 10^6
LOG ION CURRENT AMPERES
10^-11 10^-10 10^-9 10^-8 10^-7 10^-6 10^-5 10^-4 10^-3

U. S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: ANO-1

Reactor Type: Babcock & Wilcox

Date Administered: December 11, 1984

Examiner: J. C. Huenefeld

Candidate: KEY

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheet. Points for each question are indicated in parenthesis 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	Candidate's Score	% of Cat. Value	Category
25	25			1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
25	25			2. Plant Design Including Safety and Emergency Systems
25	25			3. Instruments and Controls
25	25			4. Procedures: Normal, Abnormal, Emergency, and Radiological Control
100				TOTALS

Final Grade ________ %

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

Candidate's Signature

Prepared By:	Date:	Project:
Title/Subject:		

1.0 PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW (25 Points)

1.1 The energy being transferred at the steam generator is proportional to the primary Delta T (i.e., $T_h - T_c$), but not proportional to the secondary Delta T (i.e., $T_{steam} - T_{feed}$). The reason for this is because: (Select one). (1.0)

a) the energy transfer taking place within the steam generator is essentially a constant pressure process.

b) the secondary flow rate is greater than the primary flow rate.

c) the secondary flow rate is less than the primary flow rate.

d) the energy transfer taking place within the steam generator is essentially a constant temperature process.

Answer: d Reference: ANO Heat Transfer, Chp 2.

1.2 In the condenser energy is being transferred to the circ-water. If the circ-water flowrate were reduced slightly while holding generated megawatts constant, the most probable result would be: (Select one). (1.0)

a) that the average temperature of the circ-water will increase slightly.

b) that the amount of energy transferred at the condenser will decrease slightly.

c) that the saturation pressure within the condenser will decrease slightly.

d) that the condenser delta T will decrease slightly.

Answer: a Reference: ANO Heat Transfer, Chp 7

1.3 Assume that ANO-1 is generating 700 MWe. Using typical parameters for steam temperature and pressure, estimate the steam flow rate. State your assumptions and show your work. (2.0)

Answer:

700 MWe = 700,000 KWe = 2.39×10^9 BTU/hr   3413 BTU/Kw hr
(using conversions in stm tables)

The inlet condition at the turbine are:

~600°F, ~~~1000~~ ~900 psia → h_i = ~~1249.3~~ 1260.6 BTU/lbm

The exit conditions in the condenser are:

28" vacuum → 13.75 psid   2.04 inHg/psi
(using conversions in stm tables)

$P_{sat} = 14.7 - 13.75 = 0.95$ psia

$h_o = 1106$

The total enthalpy change is ~~1249.3~~ 1260.6 − 1106 ~ ~~143~~ 154.6

Assume an ideal turbine:

$W_{Turb} = \dot{m}(\Delta h) \quad \dot{m} = \frac{W_{Turb}}{\Delta h} = \frac{2.39 \times 10^9}{\cancel{143}\ 154.6} = \cancel{1.7 \times 10^6}\ 1.5 \times 10^7$ lbm/hr

Reference: ANO Heat Transfer   Page # or Chap

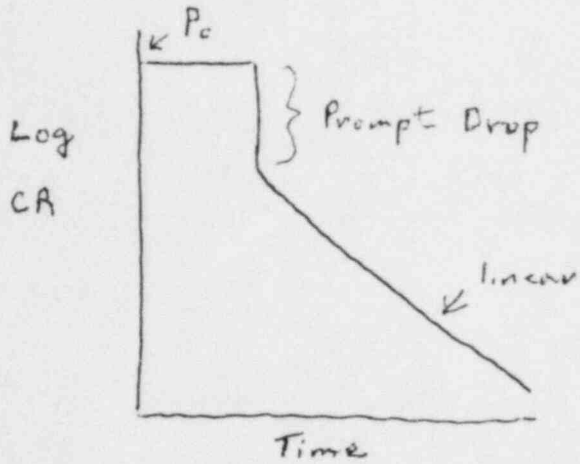
Chapter 7 Page 191
Chapter 4

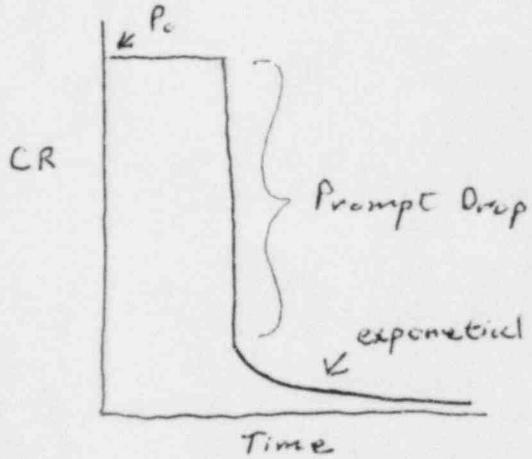
Prepared By: | Date: | Project:

Title/Subject:

1.4 Show the difference between a logarithmic plot of nuclear countrate (i.e., log countrate) versus time after a trip and a linear plot of countrate versus time after a trip by making a basic sketch of both cases. Assume that both plots start from the same initial power level. (2.0)

Answer:





Reference: Unit-1 P-5 Reactor Theory page# 138,139

1.5 Given a large vented tank 30' in diameter and 60' high with a centrifugal pump taking a suction from its base. The pump is located at a vertical elevation corresponding to the bottom of the tank and it requires 5 ft of net positive suction head (NPSH) to prevent cavitation. The tank is almost entirely full of water and is maintained at 60°F by heaters. The tank is designed such that it could withstand 15 psi differential pressure in either direction. Assume the vent becomes totally clogged with ice while the pump is in operation. Further assume that the pump is of relatively low capacity such that equilibrium conditions are maintained inside the tank. Answer the following questions:

a) What is the lowest pressure that the tank ~~would~~ will drop to ~~if~~ As the pump continues to remove water from the tank? Explain. (1.5)

b) ~~Would~~ Will the pump lose NPSH and begin to cavitate prior to reaching a level of 5 ft in the tank? Explain. (State any assumptions.) (1.0)

c) Could the pump continue to pump water at a level below 5 ft without cavitation if the vent were open? Explain. (Assume no vortexing.) (1.0)

Answer: (see next page)

Answer: (15 continued)

a) The lowest pressure that the tank could drop to would be the saturation pressure for 60°F which is 0.256 psia.

b) Assuming head loss due to flow is negligible, the answer is no. Cavitation would not begin until the level drops below 5 ft in the tank.

c) Yes. The added pressure of 14.7 psia at the pump suction would allow all of the water to be removed.

Reference: ANO Heat Transfer Page # 143 or Chap #6

1.6 The feedwater loop demand stations, the reactor demand station, and the S/G/Rx master are in "hand" with reactor power stable at 15% power. The operator commences a power escalation by bumping rods out and T_{ave} goes to 580°F. Which of the following describes the required operator course of action to bring T_{ave} back to 579°F? (Select one.) (1.0)

a) The operator must insert rods slightly, but not as far as they were withdrawn. He must request the I&C technicians to adjust the low level limits.

b) The operator must increase feed flow, restoring OTSG level, and lower the steam header setpoint slightly.

c) The operator must over-feed the OTSG slightly, increasing OTSG inventory, then stabilize at a higher feed flow and a higher OTSG level.

d) The operator does not need to take any action. Doppler feedback will return T_{ave} to 579°F.

Answer: c Reference: ICS Pg 9

1.7 Assume that a steam bubble has formed in the reactor vessel head during an RCS natural circulation cooldown. Is the collapsing of that bubble a relatively fast process or a relatively slow process? Explain. (2.0)

Answer: Reference - ANO-1 HT Chap 1 Pg 34-40 Chap 2, Chap 7

This process is a relatively slow one compared to how fast a bubble can form. Because there is no spray available in the vessel head, there is no fast way to cool the bubble. Rapid compression of the bubble results in it becoming superheated. (Reference above)

1.8 According to OP 1103.15, the reactivity bala ce calculation procedure, achieving a period of less than 5 seconds is a reportable event. What startup rate would this period correspond to? (SHOW YOUR WORK) (1.5)

Answer:

$$SUR = \frac{26.06}{T} = \frac{26.06}{5} = 5.2 \text{ DPM}$$

(equation on eq sheet)

Reference: OP 1103.15, Pg 4

1.9 If 100% FP equilibrium Xe worth is ~~2.57~~ -2.57 Delta K/K then 50% FP equilibrium Xe worth is: (Select one) (1.0)

a) - 0.257 Delta K/K

b) - 1.29 Delta K/K

c) - 2.10 Delta K/K

d) - 0.51 Delta K/K

Answer: c Reference: Unit 1 P-S Reactor Theory Pg 200

1.10 The time to reach Xe equilibrium after a significant power change is approximately: (Select one) (1.0)

a) 4 - 8 hours

b) ~~24 hours~~ $\sqrt{\%\ Power}$

c) ~~8 1/2 hours~~ 40 hours

d) ~~40 hours~~ 72 hours

Answer: ~~d~~ c Reference: Unit-1 P-S Reactor Theory Pg 200

1.11 TRUE or FALSE. The amount of reactivity needed to cause prompt criticality decreases over cycle life. (0.5)

Answer: True

Reference: Unit-1 P-S Reactor Theory Pg 126

1.12 Can reactor power be reduced at a rate greater than -1/3 DPM (-80 second period)? Explain. (1.5)

Answer:

Yes. Approximately 99% of the fission neutrons present in the core are the result of prompt fissions. These neutrons respond immediately to reactivity changes. ~~It is only at very low power levels that delayed neutr~~ It is only after the fast neutron flux has been totally suppressed that the reactor period is limited to -1/3 DPM.

Reference: Unit-1 P-S Rx Theory Pg 135-139

1.13 TRUE or FALSE. Dropping one rod results in the same sized prompt drop as dropping an entire group of rods. (0.5)

Answer:

False

Reference: Unit-1 P-S Reactor Theory Pg 139

1.14 The reason that it takes longer and longer for startup rate (SUR) to reach zero as the reactor nears criticality is: (Select one) (1.0)

a) because the effect of the delayed neutrons is becoming more dominant than prompt neutrons.

b) because the decay of SUR is becoming dominated by the decay of the longest lived precursors.

c) because as K effective nears one (1) the effect of the previous generations of neutrons is becoming more dominant in the ~~current generations of~~ neutron population.

d) because as the rods are withdrawn from the core they are providing less shielding of the source range detectors.

Answer: c Reference. Unit-1 P-S Reactor Theory Pg 145

1.15 Which of the following correctly describes a condition leading to shifting the power distribution to the lower regions of the core? (Select one) (1.0)

a) increasing flow through the core and therefore increasing the static pressure in the lower regions of the core.

b) lowering T_c, cooling the fuel in the lower regions of the core, and therefore suppressing doppler feedback.

c) increasing the boron concentration and therefore the rod index.

d) decreasing the boron concentration and therefore the rod index.

Answer: d Reference: Unit-1 P-S Reactor Theory Page# 161 characteristic characteristic

1.16 Assume two (2) reactor core configurations that are identical in every respect with the exception of one ~~attribute~~. Select the ~~attribute~~ that would cause the critical rod height of one reactor to be different from the critical rod height of the other reactor. (Select one) (1.0)

a) Source strength

b) Rod speed

c) Delayed neutron fraction

d) Number of neutrons resulting per fission

Answer: ~~Imbalance~~ d Reference: ANO-1
P-5 Reactor Theory
Pg 106

1.17 Technical Specifications describe power peaking limits (approximately 20.Ø1 KW/ft), but state that the peaking is not a directly observable quantity. What limit is observed to prevent exceeding the power peaking restrictions? (1.0)

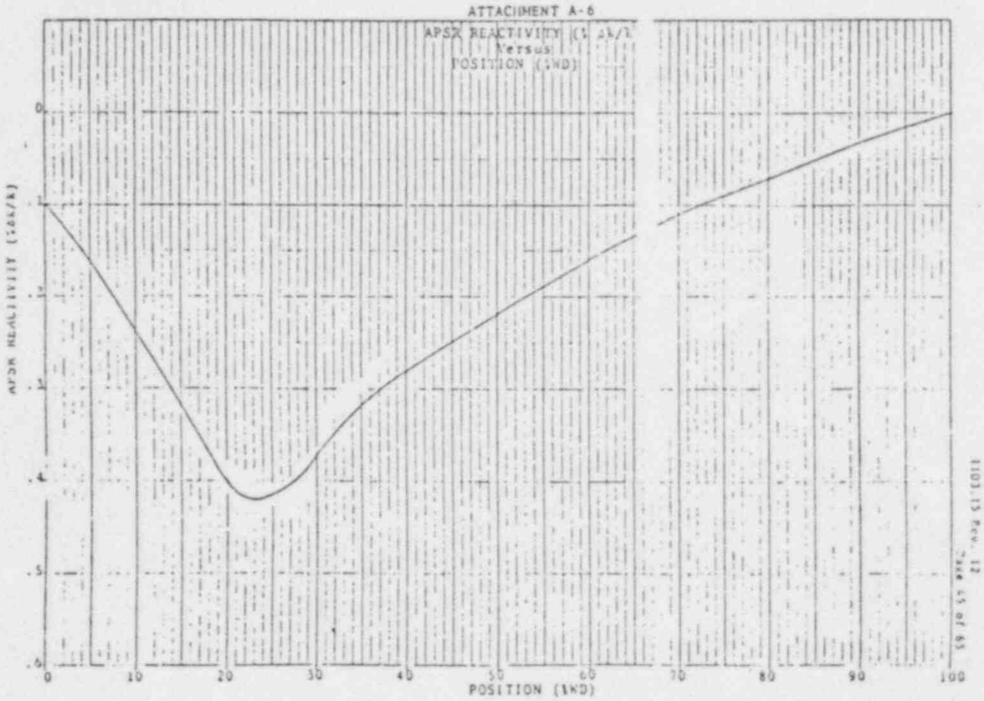
Answer: Imbalance

Reference: Tech Specs. - Pg 8

Prepared By:	Date:	Project:

Title/Subject:

1.18 Is the plot of APSR reactivity shown below a differential plot or an integral plot of reactivity? Explain your answer by drawing a sketch of what the opposite kind of plot would look like. (2.5)



ARKANSAS NUCLEAR ONE

PLANT MANUAL SECTION: REACTOR COOLANT SYSTEM OPERATING

PROCEDURE/WORK PLAN TITLE: REACTIVITY BALANCE CALCULATION

NO. 1103.15

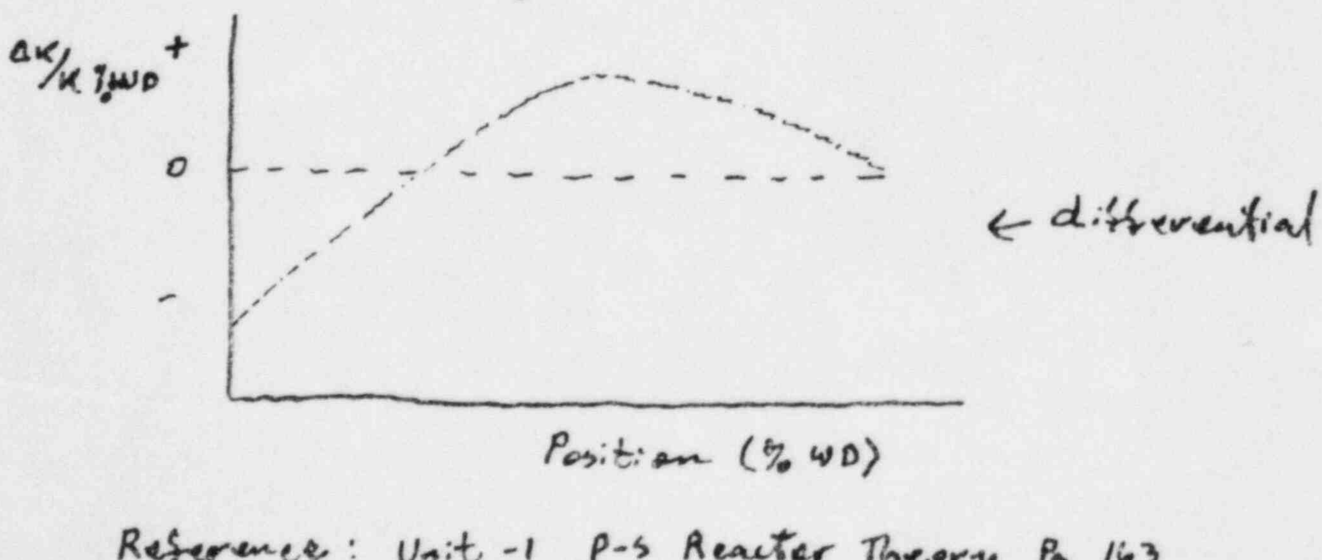
PAGE 45 of 65

REVISION 12 DATE 07/13/83

CHANGE DATE

- End of Section 1 -

~~Reference~~ Answer: Integral



Reference: Unit -1 P-5 Reactor Theory Pg 163
OP 1103.15

2.0 PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS (25 Points)

2.1 The Auxiliary Feedwater Pump is capable of supplying about _______ of full load feedwater capacity. (Select one.) (0.5)

a) 0.3%
b) 3%
c) 30%

Answer: b Reference: STM-1-19 Feedwater Pg 2

2.2 List five (5) of the ten (10) conditions that will cause a MFW pump to trip. (1.5)

Answer:

a) Thrust bearing wear forward (5 mils) or reverse (35 mils).
b) Rotor vibration (6.5 mils).
c) Turbine overspeed (6215 rpm).
d) Bearing oil pressure low (10 psig).
e) FW pump discharge pressure high (1150 psig).
f) Control Room trip (HS-6709 on C02).
g) Pushbutton trip (local).
h) FW pump suction pressure low (230 psig).
i) ~~FW pump low flow (1600 gpm) provided low-flow protection bypass switch (HS-6719 on C12, HS-6720 on C12) not in "Bypass" position.~~ IAW facility comments
j) Low exhaust vacuum (15 in. Hg).
k) Preferred pump will trip when the main turbine trips.

#'s not required.

Reference: STM-1-19 Pg 5

Prepared By: | Date: | Project:

Title/Subject:

2.3 a) At what position (i.e., % open) of the Startup Feedwater control valve will the associated low load block valve begin to open? (0.5)

b) At what power level (increasing) will the the Main Feedwater block valves open, shifting the Main Feedwater Pumps to speed control? (0.5)

c) At what power level (decreasing) will the Main Feedwater block valves go shut, shifting the Main Feedwater pumps to Delta P control? (0.5)

Answer:

a) 80%

b) 50% FP

c) 45% FP

Reference: STM-1-19 Pg 15

2.4 TRUE or FALSE. A level of 100% on the OTSG Operate Range is less than one-half of the distance to the upper tube sheet. (0.5)

Answer:

False

Reference: STM-1-19

2.5 The size of the condensate water storage tank is based upon the water inventory that will supply _____ hours of decay heat removal operation with the EFW system. Select the correct answer. (1.0)

a) 8

b) 24

c) 80

d) 800

Answer: a

Reference: STM 1-20 Pg 13

2.6 List the conditions that will result in automatic starting of the EFW pump. (2.0)

Answer: Loss of all four RCPs
Loss of both MFWP's when reactor power is >5%.
Either OTSG startup level is less than 18".
SLBIC actuation.

Reference: STM-1-19 pg 21

2.7 TRUE or FALSE. The Unit-one instrument air system may be interconnected with the Unit-2 instrument air system. (0.5)

Answer:

True.

Reference: Instrument Air System Pg 16

2.8 List those Makeup and Purification System (CV) valves that automatically close and those that automatically open upon receiving an ES channel 1&2 (HPI) actuation signal. Common names are acceptable. (3.0)

Answer:

(see next page)

Prepared By:	Date:	Project:
Title/Subject:		

Answer: (2.8 continued)

OPEN

1) 2 1/2-inch motor-operated valves (CV-1219, CV-1220, CV-1227 & CV-1228) open on actuation signal from ESFAS on low RCS pressure or high reactor building pressure

d. Throttle valves in each HPI line (MU-1231, MU-1232, MU-1233, & MU-1234) throttled to prevent makeup pump runout due to abnormal increase in flow.
8
Outlet header valves (CV-1407, CV-1408) from BWST automatically open on actuation signal

1) 14-inch valves open on signal from ESFAS Channel 1 and Channel 2.

Reference: HPI Pg 2+3

System Valves CLOSE

a. On system actuation of HPI, the isolation valves in the purification letdown line, the RCP controlled bleed-off line, and the normal makeup line close.

1) Purification letdown valves are:
1 { CV-1214 (Letdown Cooler A)
CV-1216 (Letdown Cooler B)
2 CV-1221 (Letdown Coolers Iso.)
2) RCP controlled bleed-off valves are:
3 { CV-1270 (RCP 32D return)
CV-1271 (RCP 32C return)
CV-1272 (RCP 32B return)
CV-1273 (RCP 32A return)
4 CV-1274 (RCP's seal return)
3) Normal makeup valves are:
5 CV-1234 - (MU Block Valve)
6 { CV-1301 - (MU pumps recirc iso. valve)
CV-1300 -

b. All valves motor-operated.

c. Inlet valves in each HPI line open on actuation.

2.9 Sketch the HPI system piping between the reactor coolant system and the motor operated loop A and B HPI isolation valves (CV-1219, 1220, 1227, and 1228) showing the cross connection piping. 2.0 (~~1.5~~)

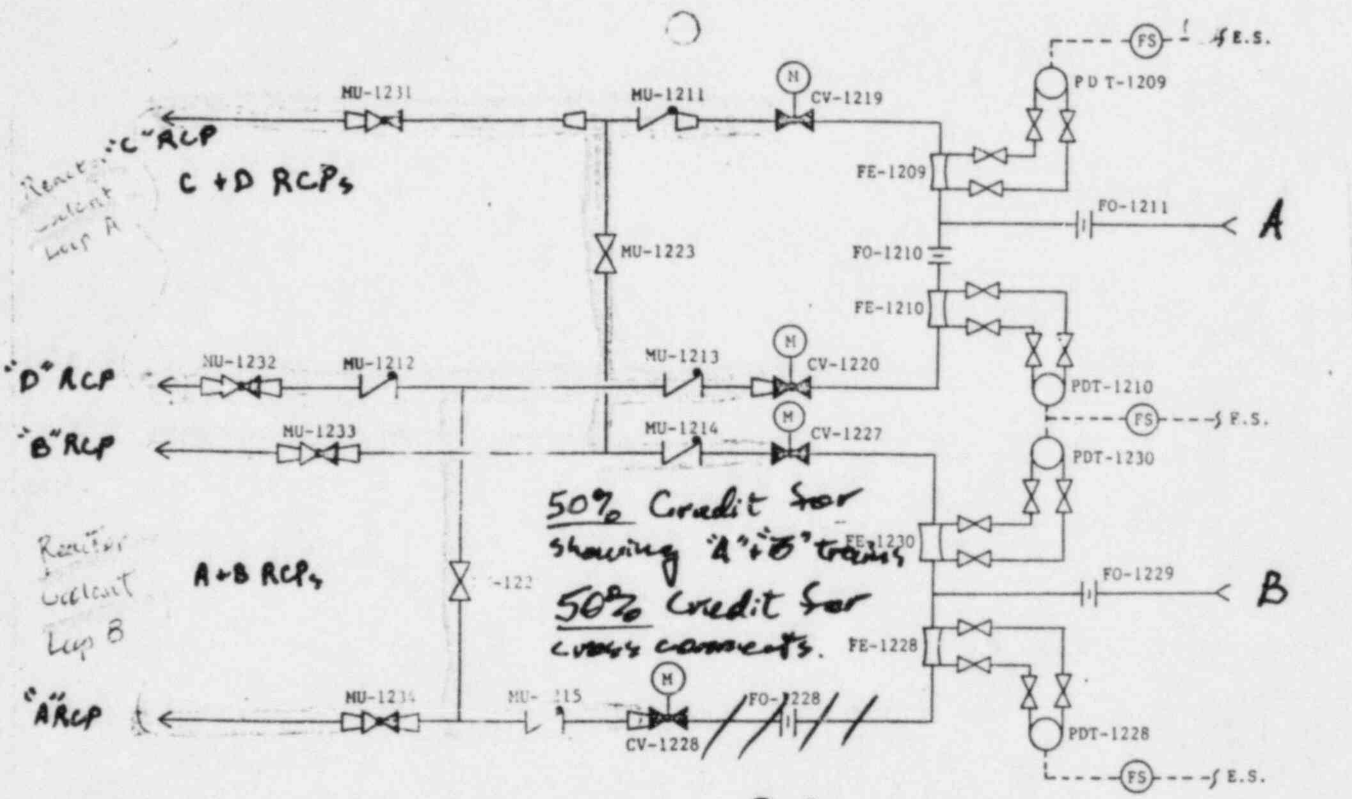
Answer:

(See next page)

Prepared By:	Date:	Project:

Title/Subject:

Answer: (2.9 continued)



Reference:

Exact placement of check-valves is not necessary.

FIGURE 4.3

HPI PIPING ONE LINE			
LP TITLE HIGH PRESSURE INJECTION			
LP No	FIGURE	DATE	REV
AA-51002-004	4.3	3-82	0

2.10 The letdown flow rate is based upon: (Select one) (1.0)

a) purifying one RCS volume/day.

b) purifying ~~three~~ twenty four RCS volumes/day.

c) purifying ~~nine~~ one RCS volumes/~~day~~ week.

d) purifying ~~24~~ one RCS volume~~s~~/~~day~~ month

Answer: a Reference: MUP Pg 3 + Pg 6

2.11 TRUE or FALSE. Because normal makeup demands are so low, the makeup pumps are designed to operate for prolonged periods at 40 gpm. (0.5)

Answer:

False

Reference: MUP Pg 13

2.12 TRUE or FALSE. One gallon per minute of makeup flow is maintained at all times to preclude thermal shock to the injection nozzle. (0.5)

Answer:

True.

Reference: MUP Pg 15

2.13 Why does allowable makeup tank pressure decrease with decreasing makeup tank level? (2.0)

Answer:

The pressure in the tank is limited to ensure that gas in the tank is not injected into the suction of a running HPI pump in the event of an actuation of HPI. A higher pressure is allowed at higher levels because the pressure will fall as level falls.

Reference: OP 1104.02 pg. 3 + Attachment C

Prepared By: | Date: | Project:

Title/Subject:

2.14 Show, using a basic sketch, how the low pressure injection headers are cross connected prior to entering the core flood nozzles. (2.0)

Answer :

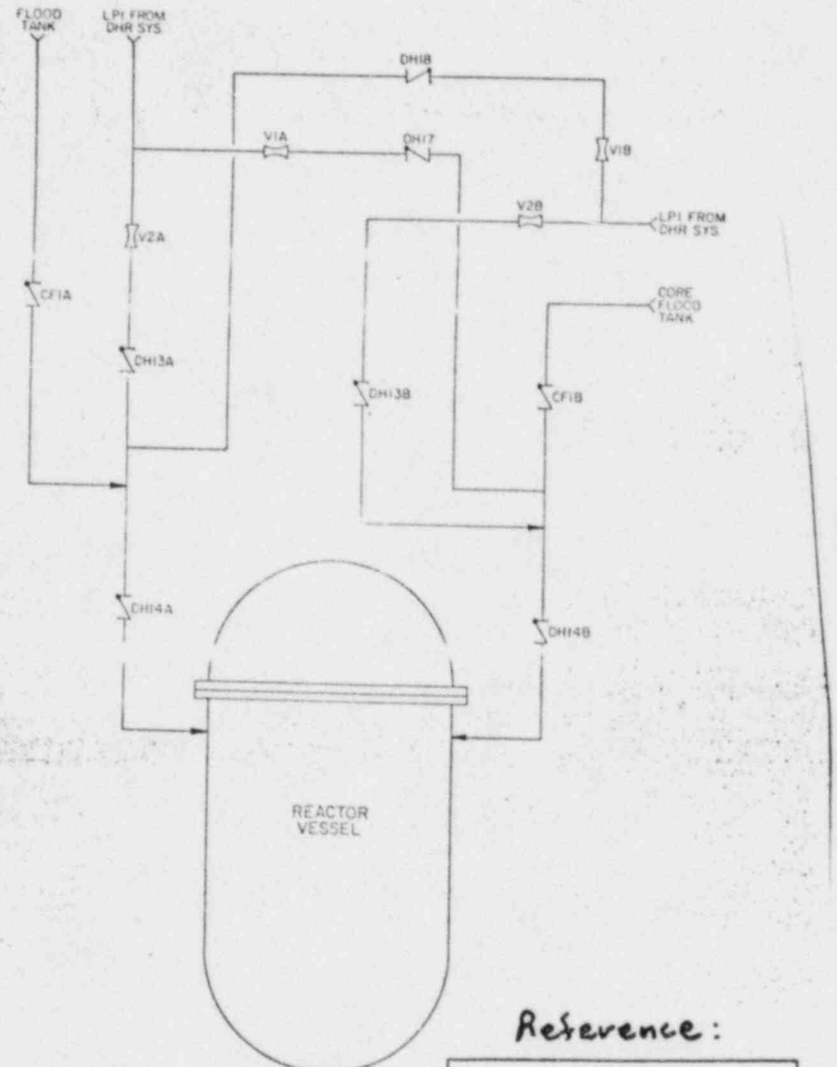


FIGURE 20.7

Reference:

LOW PRESSURE INJECTION CHECK VALVE CONFIGURATION			
LP TITLE DECAY HEAT REMOVAL SYSTEM UNIT 1			
LP No.	FIGURE	DATE	REV
AA-51002-020	20.7	9-83	0

2.15 The shut-off head for the low pressure injection pumps is approximately: (Select one) (1.0)

a) 100 psig

b) 150 psig

c) ~~260~~ 290 psig

d) ~~380~~ 600 psig.

Answer: b Reference: OP 1104.04, Pg 18

2.16 Which of the following components are a direct interface between RCS water and ICW water? (More than one answer maybe correct) (1.0)

a) Letdown coolers
b) RCP seal cooling heat exchangers
c) RCP seal return coolers
d) Control rod drive mechanisms
e) RCP motors
f) Reactor building coolers
g) Pressurizer sample cooler

Answer: a, b, c, g Reference: ICW

2.17 The level in the intermediate cooling water surge tank (T37B) has been continuously increasing for several hours. Samples indicate no detectable activity or boron concentration. RCS leak rate determinations indicate no unaccountable leakage. What is a probable source of non-nuclear in-leakage into the nuclear ICW loop? (1.5)

Answer:

Service water leaking from a tube leak in the ICW cooler servicing the nuclear ICW loop.

Reference: ICW Pg 4

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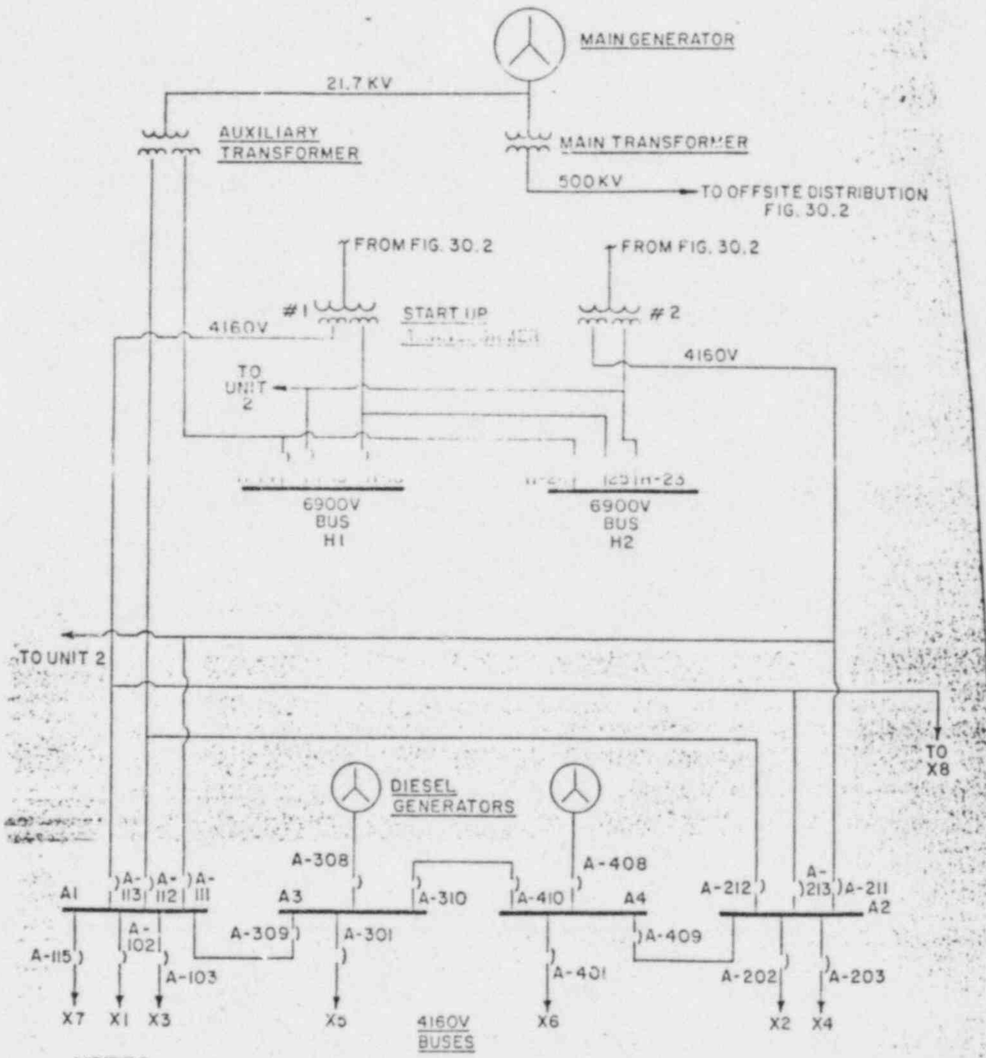
2.18 Sketch the 6900/4160 V distribution system from the main generator and startup transformers to the 480 V load center transformers. Show which of the seven 480 volt load centers come off of each 4160 V bus. No other loads need be shown. Label each bus. Individual breaker designations are not required. (3.0)

Answer:



NOTES:
1 REFER TO TABLE 30.1 FOR BUS LOADS
2. X1 THRU X7 ARE 4160V/480V XFMRS TO MCC'S B1 THRU B7 SEE FIGURE 30.3
3. A3 AND A4 ARE ESSENTIAL 4160V BUSES

Reference: →

6900/4160V DISTRIBUTION SYSTEM			
LP TITLE ELECTRICAL DISTRIBUTION			
LP No.	FIGURE	DATE	REV
AA-51002-007	7.8	10-83	0

FIGURE 7.8

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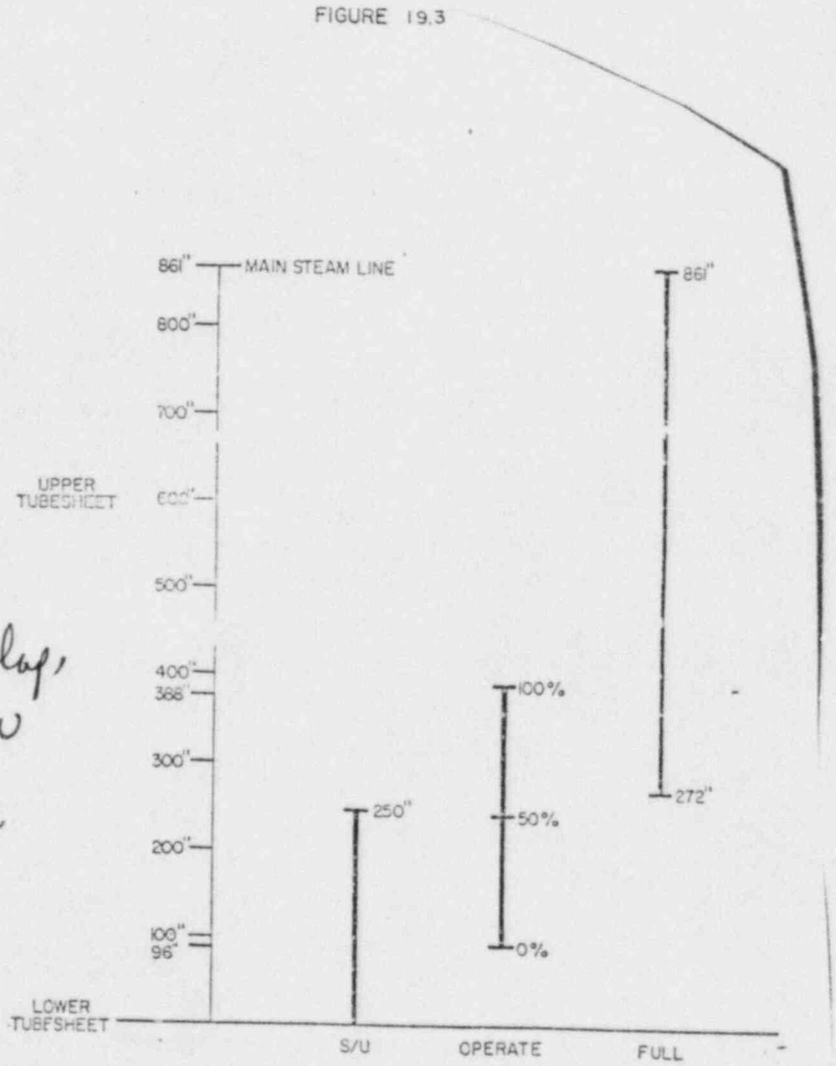
3.0 INSTRUMENTS AND CONTROLS (25 Points)

3.1 Using the figure given on page ~~18~~ 19 of this exam, sketch in the ranges for the three OTSG level indicators showing the overlap between the ranges. (3.0)

FIGURE 19.3

Answer:



80% credit for general overlap, showing that SU and full range don't overlap.

Reference: →

OTSG RANGE OVERLAP
LP TITLE FEEDWATER SYSTEM

Prepared By: | Date: | Project:

Title/Subject:

3.2 Which of the following statements is true regarding the function of the "cross limit" circuity of the ICS? (Select one.) (1.0)

a) When feedwater flow is greater than or less that feedwater demand by more than 5%, the reactor is feedwater limited.

b) When neutron power is greater than or less than feedwater demand by more than 5%, the feedwater is reactor limited.

c) When feedwater flow is less than feedwater demand by more than 5%, the reactor is feedwater limited.

d) When neutron power is less than feedwater demand by more than 5%, the feedwater is reactor limited.

Answer

c

Reference : STM-1-19 Pg 15

3.3 Describe the interlock associated with the decay heat isolation valve (CV-1050) and the decay heat pump suction valve (CV-1410). (1.5)

Answer:

DH isolation valve CV-1050 and DH pump suction valve CV-1410 interlocked with RCS pressure.

a) CV-1050 cannot be opened if RCS pressure > 290 psig and will automatically shut if RCS press. increases above 320 psig.

b) CV-1410 cannot be opened if RCS pressure > 290 psig and will automatically shut if RCS press. increases above 385 psig.

Reference: DHR Pg 5

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3.4 List five (5) of the nine (9) interlocks that must be satisfied prior to starting a reactor coolant pump. (1.5)

Answer:

PLANT MANUAL SECTION: REACTOR COOLANT SYSTEM OPERATING	PROCEDURE/WORK PLAN TITLE: REACTOR COOLANT PUMP OPERATION	NO: 1103.06
ARKANSAS NUCLEAR ONE	PAGE 14 of 15	
	REVISION 2 DATE 03/14/83	
	CHANGE DATE	

ATTACHMENT B

Reserence

Reactor Coolant Pump Interlocks

Rx. power <22.

RCP seal injection flow >5 gpm.

RCP Motor ICW cooling flow >250 gpm.

RCP pump seal cooling ICW flow >30 gpm.

RCP motor upper brg. oil reservoir level >9 inches.

RCP HP oil lift pressure >1750 psig.

RCP reverse rotation >12.7 gpm return oil flow.

RCP motor lower brg. oil reservoir level >6.5 inches.

RCS temp >500°F to start fourth RCP.

3.5 a) What is the SLBIC actuation set-point? (0.5)

b) What actions result upon actuation of SLBIC? (1.5)

Answer:

a) 600 ± 25 psi

b) Both MSIVs shut.
EFW pump supply valve from affected OTSG opens.
Feedwater isolation valve for affected OTSG shuts.

Reference: Steam Systems Pg 10
~~SLIB~~ SLBIC Inst and Control Pg 6.

3.6 At what vacuum are condenser dump valves interlocked shut? (0.5)

Answer: 20" Vacuum Reference: Steam System Pg 7

3.7 TRUE or FALSE. The unit controls for ESAS actuated equipment allow taking manual control of a component even with an ES actuation signal present. (0.5)

Answer: True Reference: ESAS Pg 4

3.8 TRUE or FALSE. Loss of power to a digital cabinet does not cause ES actuation. (0.5)

Answer: True Reference: ESAS Pg 10

3.9 Which of the following is consistent with the Engineered Safeguards Actuation System logic? (Select one.) (1.0)

a) Analog subsystem #1 experiences both an RCS low pressure trip (1500 psig) and an RB high pressure trip (4 psig) resulting in the actuation of ESAS Channel 1, 3, and 5.

b) Analog subsystem #1 experiences an RCS low pressure trip (1500 psig) and Analog subsystem #3 experiences an RB high pressure trip (4 psig). This results in the actuation of ESAS Channels 1, 2, 3, and 4.

c) All three analog subsystems experience an RCS low pressure trip (1500 psig) resulting in the actuation of ESAS Channels 1 through 8.

d) All three analog subsystems experience an RCS low pressure trip (1500 psig) resulting in the actuation of ESAS Channels 1 through 6 only.

Answer: b Reference: ESAS Figs 12.1, 12.2

3.10 When less than 15% power, the "measured variable" position on the main feedwater pump hand-auto station: (Select one) (1.0)

a) has the same interpretation as it does when greater than 50% power.

b) indicates demanded position of the main feedwater pump turbine governor value.

c) indicates 0-100 psid across the startup and low load control valves.

d) indicates 0-100% demanded main feedwater pump speed.

Answer: c Reference: ICS Pg 5

3.11 When latching the main turbine and bringing it up to speed, a "NOTE" in the plant startup procedure (OP 1102.02) requires manual control of the steam dump valves. Why is this necessary? (1.5)

Answer:
When the turbine is latched, turbine header pressure input is switched from the instrument on that side to the selected turbine header pressure instrument. Both header pressures will not necessarily equalize until the turbine is at speed; therefore, to maintain proper header pressures the dump valves must be in hand.

Reference: ICS Pg 7

3.12 What happens to the low load and startup control valves as the main block valve starts coming open at 50% feedwater loop demand? (1.0)

Answer:

The low load control valve and startup control valve "freeze" in position.

Reference: ICS · Pg 12

3.13 Which of the following conditions most correctly defines a situation resulting in the 50 psi bias being applied to the steam dump valve circuitry? (SELECT ONE) (1.0)

a) The turbine is synchronized to the grid and the steam dump valves are partially open with unit load demand less than or equal to 15%

b) The turbine is synchronized to the grid and the steam dump valves are partially open with unit load demand greater than 15%.

c) All turbine by-pass valves are closed and unit load demand is at 13% with header pressure exceeding header pressure setpoint by 15 psi.

d) The turbine is in ICS auto with the steam dump valve hand - auto controllers in auto.

Answer:

b

Reference: ICS Pg 8 and Fig 15.30

3.14 As the main feed block valve opens, what do you expect to see happen to the main feed pump speed? (Assume loop demand remains constant.) (Select one.) (0.5)

a) It remains constant
b) It slows down
c) It increases slightly.

Answer:

b

Reference: MFW Page #

3.15 The high level limit alarm is received. Which of the following is most correct? (Select one). (1.0)

a) OTSG level may or may not be at the high level limit.

b) Feedwater loop demand is frozen as is.

c) Main feed pump speed is frozen as is and the reactor is feedwater limited.

d) Main feed pump speed is frozen as is, but the reactor is not feedwater limited.

Answer:

a

Reference: ICS Pg 18.

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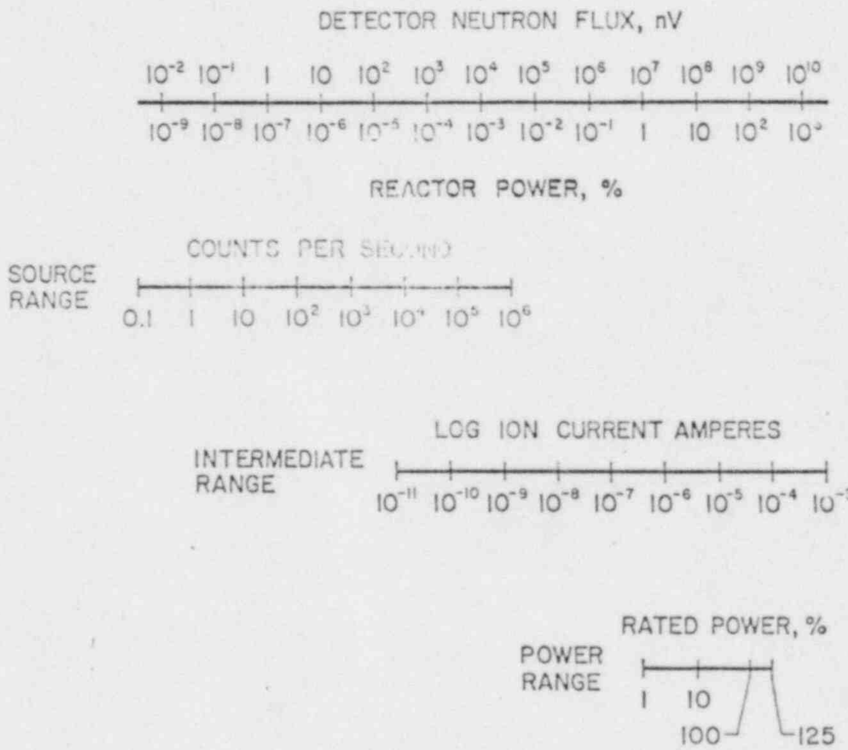
3.16 Using the figure given on page ~~15~~ 20 of this exam, sketch in the ranges for the three nuclear instrumentation indicators showing the overlap between the ranges. (3.0)

Answer:



Reference

FIGURE 14.2

NUCLEAR INSTRUMENTATION FLUX RANGES			
LP TITLE: NUCLEAR INSTRUMENTATION			
LP No	FIGURE	DATE	REV
AA-51002-014	14.2	9-83	0

3.17 The amount of time needed for the incore instruments to indicate a step change in actual incore power within 10% accuracy: (select one) (1.0)

a) is approximately 30 seconds
b) is approximately 3 minutes
c) is approximately 30 minutes
d) is very heavily dependent upon whether the power change was positive or negative.

Answer: b Reference: In-core Inst Sys Fig 19.6

3.18 TRUE or FALSE: The range of the source range SUR indication on the control board is sufficient to verify the stable negative SUR expected following a reactor shutdown. (0.5)

Answer: True. Reference: NI Pg 3

3.19 TRUE or FALSE: The reason that power range nuclear instrumentation is not compensated for gamma radiation is because the detector is shielded by 4 inches of lead. (0.5)

Answer: False. Reference: NI Pg 6 + 12

3.20 What value of source range and intermediate range instrumentation startup rate (SUR) will result in actuation of a SUR rod withdrawal interlock. (1.0)

Answer:

Source range SUR > 2 DPM

Intermediate Rg SUR > 3 DPM

Reference: NI

3.21 TRUE or FALSE: The radiation detector on the line from the Reactor Building Coolers will cause the isolation of its respective coolers even if an ES actuation signal is present. (0.5)

Answer: True Reference: SW and ACW Pg 20
Process Rad Mon. Table 19.6

3.22 Which of the following is true concerning the uncertainty associated with the RCS saturation margin monitoring instrumentation during an increase in RCS temperature? (Select one.) (1.0)

a) Uncertainty increases for both the temperature margin to saturation and pressure margin to saturation.

b) Uncertainty decreases for the temperature margin to saturation, but increases for the pressure margin to saturation.

c) Uncertainty decreases for the pressure margin to saturation but increases for the temperature margin to saturation.

d) Uncertainty decreases for both the temperature margin to saturation and the pressure margin to saturation.

Answer: b Reference: OP 1105.12 Attachment 1

ENGINEERING WORKSHEET

Prepared By: | Date: | Project:

Title/Subject:

4.0 PROCEDURES: NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL (25 Points)

4.1 A manual trip of the reactor is required following any automatic reactor trip or if the reactor protection system fails to function upon reaching any of its parameter setpoints. List 5 other specific conditions requiring a manual reactor trip as itemized in the reactor trip procedure (EP1202.01). (2.5)

Answer:

Pressurizer level > 290"

Pressurizer level < 100" with no indication of level control being restored.

Actuation of SLBIC or any MSIV closure at power.

Steam generator level less than 15" or greater than 375" (95% on the Operate Range).

Reactor Building Pressure > 3 psig.

Reference: EP 1202.01 Pg 2

4.2 If RCPs are inadvertently left on for >2 minutes following a loss of subcooling margin: (Select one) (1.0)

a) Secure all but one of the RCPs.

b) Secure both RCPs in the loop with the highest subcooling margin.

c) Secure both RCPs in the loop with the lowest subcooling margin.

d) Leave only one pump running in each loop.

Answer: d Reference: EP 1202.01 Pg 4.

4.3 The auxiliary lube oil pump for a makeup pump: (Select one) (1.0)

a) should be run for at least three minutes prior to starting a makeup pump and should remain in operation until the makeup pump is stopped.

b) should be run for at least one minute and then secured prior to starting a makeup pump.

c) should be run until oil pressure is at least 10 psig prior to starting a makeup pump.

d) should be run for at least one minute prior to starting a makeup pump and then stopped after the makeup pump is running.

Answer: d Reference: EP 1202.01 Pg 8
OP 1104.02 Pg 4
MUP Pg 24

4.4 List the four (4) verifications for the presence of natural circulation decay heat removal. (2.0)

Answer:

RC temperature (including incore thermocouples) decreasing

RC hot and cold leg $\Delta T < 50°$ and decreasing with decreasing decay load.

Continued demand for EFW to maintain steam generator levels.

Continued demand for operation of turbine bypass, atmospheric dump or main steam safety valves to limit secondary pressure.

Reference: EP 1202.01 Pg 10

reactor vessel

4.5 Should excessive thermal stresses be experienced, a 3 hour soak period is required with one exception. What is that exception? (1.0)

Answer:

Steam generator tube rupture.

Reference: EP 1202.01 Pg 21

4.6 During an OTSG tube rupture, what is the maximum allowable (emergency) cooldown rate, and what two (2) conditions must exist before it is allowed? (2.0)

Answer:

Limit - 240°F/hr

RCS temp > 500°F and the tube leak is greater than HPI capacity.

Reference: EP 1202.01 Pg 67 + 72

4.7 Which of the following is true regarding the expected behavior of the RCS temperature following a reactor trip? (Select one.) (1.0)

a) RCS temperature should be at 532°F within five to ten minutes.

b) RCS temperature should be at 545°F within two to three minutes.

c) RCS temperature should be at 532°F within two to three minutes.

d) RCS temperature should be at 545°F within five to ten minutes.

Answer: b Reference: EP 1202.01 Pg 11

4.8 What constitutes a loss of subcooling margin? (1.5)

Answer:

A loss of the subcooling margin is defined as being less than 50° subcooled because of possible instrument errors. However, during normal power operation, the subcooling margin is less than 50°, therefore, for the immediate action section only, a subcooling margin less than 30°F will be used as a loss of subcooling margin.

Reference: EP 1202.01 Pg 16

4.9 What conditions determine whether the reactor vessel has been subjected to excessive thermal stresses? (2.0)

Answer: Ref- 1202.01 pg 20

RCS < 500°F and cooldown rate > 100°F/hr or HPI is on and RCPs are off.

4.10 Assume that reactor coolant pumps are stopped because of a loss of subcooling margin. What must be done to prevent a resultant overcooling? (1.5)

Answer: EFW will auto-start and begin feeding the OTSG's to 50% on the Operate Range. Manual control of EFW must be taken allowing OTSG's to fill to 355" (95%) slowly.

Reference: EP 1202.01

4.11 The maximum allowable heatup rate during a normal heatup is interpreted as being: (Select one) (1.0)

a) 100°F in any one hour period
b) 50°F in any thirty minute period
c) 25°F in any fifteen minute period
d) No more than 1.67°F per minute.

Answer: d Reference: OP 1102.02 Pg 9

4.12 TRUE or FALSE: During a heatup, deboration may continue while withdrawing one group of safety rods provided that shutdown margin is not lowered below 1.5% Delta K/K. (0.5)

Answer:

False

Reference: OP 1102.08 Pg 1

4.13 TRUE or FALSE: Because of the uncertainty involved in the ECP calculation (+0.5% Delta K/K) it is possible to go critical in the restricted region of the rod withdrawal curve even though the ECP is ~~calculated properly.~~ Performed according to procedure. (0.5)

Answer: False Reference: OP 1102.08 Pg 2

4.14 TRUE or FALSE: When a CAUTION card's instructions conflict with requirements specified in procedures, the CAUTION card takes priority. (0.5)

Answer: False Reference: AP 1000.27 Pg 2

4.15 TRUE or FALSE: Unless stated otherwise, valves on which a HOLD card is installed will be shut and breakers on which a HOLD card is installed will be open. (0.5)

Answer: True Reference: Ap 1000.27 Pg 8

4.16 Which of the following is the ~~The~~ proper action to take upon discovering a fire ~~is too~~? (Select one) (1.0)

a) Immediately attempt to control the fire.

b) Summon the fire brigade by using the plant paging system.

c) Evaluate the situation, and immediately report the fire to the control room.

d) Immediately evacuate the area then notify the control room.

Answer: C Reference: AP 1000.30 Pg 4

4.17 a. What is the maximum background tolerable for using a "frisker"? (0.5)

b. While frisking, a countrate of greater than ________ indicates the possible presence of contamination. (0.5)

Answer:

a) 300 cpm

b) 100 cpm above background

Reference: RP 1612.02 Pg 3.

4.18 Match the four substances given below with their respective tenth thicknesses for gamma radiation: (1.0)

a) Lead	1) 24"
b) Steel	2) 2"
c) Water	3) 14"
d) Concrete	4) 4"

Answer: a-2 b-4 c-1 d-3

Reference: RP-1612.06 Pg 7

4.19 In the event of a remote shutdown AP-1203.29, four operators have responsibilities assigned by procedure. Which operators are these, and where do they station themselves? (Assume auxiliary feed is available.) (2.0)

Answer:

Waste Control Operator - standby at MCC-61, 62

Assistant Plant Operator - lower south electrical penetration room.

Auxiliary Operator - lower south electrical penetration room.

Plant Operator - Dasey Panel

Reference: AP 1203.29 Attachment 1-4.

4.20 What three (3) conditions in combination require the establishment of Reactor Building Integrity? (1.5)

Answer: Reactor Coolant Pressure $\geq$ 300 psig
Reactor coolant temperature $\geq$ 200°F
Nuclear fuel in the core.

Reference: Tech Specs Pg 54.