U. S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT

EXAMINATION REPORT NO. 50-271/85-01 (OL)

FACILITY DOCKET NO. 50-271

FACILITY LICENSE NO. DPR-28

LICENSEE: Vermont Yankee Nuclear Power Corporation

FACILITY: Vermont Yankee

EXAMINATION DATES: January 21-24, 1985

CHIEF EXAMINER: J. A. Berry [signature] 4-29-85
J. A. Berry, Lead Reactor Engineer (Examiner) Date

REVIEWED BY: J. Berry for 4-30-85
R. M. Keller, Chief, Reactor Projects Section 1C No. 1C Date

APPROVED BY: H B Kister 5/10/85
H. B. Kister, Chief, Projects Branch No. 1 Date

SUMMARY: Operator licensing examinations were conducted at Vermont Yankee January 21-24, 1985. Two Senior Reactor Operator candidates and two Instructor Certification candidates were administered written and oral examinations. The two Senior Reactor Operator candidates and one Instructor Certification candidate failed the written examination. Low scores were noted on sections five and seven for all three failures. Specific areas of weakness included RPV level instrumentation, refueling procedures, and fire control procedures. The Instructor Certification candidate who failed the written also failed the oral portion of the examination. Weaknesses in knowledge of radiation protection procedures and personnel exposure limits were noted during all oral examinations.

REPORT DETAILS

TYPE OF EXAMS: Replacement

EXAM RESULTS:

	SRO Pass/Fail	Inst. Cert Pass/Fail
Written Exam	0/2	1/1
Oral Exam	2/0	1/1
Overall	0/2	1/1

1. Chief Examiner At Site: B. Hajek, NRC Consultant Examiner

2. Summary of deficiencies noted on oral exams:

   Weaknesses were noted in the areas of radiation protection procedures and personnel exposure limits.

3. Summary of generic strengths or deficiencies noted from grading of written exams:

   All failures received low scores on sections five and seven. Specific weaknesses included RPV level instrumentation, refueling procedures and fire control procedures.

4. Exit Interview Details:

   NRC Contractor Personnel Present

   B. Hajek

   Facility Personnel Present

   E. V. Lindamood, Operations Training Supervisor
   L. W. Anson, Training Department Supervisor
   R. W. Spiney, Training Manager
   D. A. Reid, Operations Supervisor

Summary of comments made at exit interview:

a. The Chief Examiner indicated that three of the four candidates would be recommended as clear passes on the oral examinations.

b. The Chief Examiner noted the deficiencies observed during the oral examinations (detailed in this report).

5. Examination Review

All facility comments were resolved during the examination review. Comments which required changes to the examination have been incorporated into the Master Exam and Answer Key attached.

Attachment: Written Examination and Answer Key (SRO)

U. S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

Facility: Vermont-Yankee

Reactor Type: BWR

Date Administered: January , 1985

Examiner: B. K. Hajek

Applicant: MASTER

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple questions 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 Cat. Value	Category
25.0	25			5. Theory of Nuclear Power Plant Operation, Fluids & Thermodynamics
25.0	25			6. Plant Systems: Design, Control & Instrumentation
25.0	25			7. Procedures-Normal, Abnormal, Emergency & Radiological Control
25.0	25			8. Administrative Procedures, Conditions and Limitations
100.0	100			TOTALS

Final Grade ______ %

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

____
Applicant's Signature

5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS (25)

5.1 The Rod Worth Minimizer is required to be operational below 20 percent power, but not above 20 percent. Fully explain why this is the case. (2.0)

5.2 The Reactor Recirculation System consists of two main centrifugal pumps and 20 jet pumps. If one jet pump nozzle assembly hold down mechanism should fail,

a. Would the indicated flow rate in the affected loop increase or decrease? Briefly explain the reason for this change in loop flow rate. (1.0)

b. How would the actual and indicated total core flow rates be affected? Briefly explain the reasons for these changes in flow rate. (1.0)

pt wasn't indicated.
I told them to use the
100% rod line @ 90% power.
Doug had already used a 90/90 point.
I said this was ok.

c. If the reactor were operating at 90 percent power at the position indicated on the Power to Flow map in Figure 5.1 prior to the failure of the jet pump, where would the reactor be operating after the failure occurred? Assume the Percent Rated Core Flow data is derived from the measured loop flow rates. Briefly explain why operation would move in the direction you indicate. (1.0)

5.3 The reactor startup and heatup procedures warn against operations that might result in cold water stratification in the bottom of the vessel. What three situations, occurring together, would increase the potential for this coldwater stratification if they occur simultaneously? (3.0)

5.4 O.P. 0100, Reactor Startup to Criticality, warns of the possibility of unexpected high notch worths during withdrawal sequences. Briefly explain under what conditions these high notch worths can be expected, and in what part of the core you would expect relatively higher notch worths to be located? (3.0)

CATEGORY CONTINUED ON NEXT PAGE

5.5 Reactor water chemistry limits are specified for radioiodine concentrations, conductivity, and chloride levels in the water.

a. Since chlorides are not measured directly, what parameter is monitored to detect the presence of chlorides? Why can this method be used? (1.0)

b. Why are chloride levels permitted to be higher at steaming rates in excess of 100,000 lbs/hr, and conductivity levels permitted to be higher during startup periods? (1.0)

5.6 Core orificing is used to assure uniform flow through all fuel elements, somewhat independent of the power variations in individual fuel bundles.

a. Explain how the resistance to flow changes as the power increases in a fuel bundle. (1.0)

b. How does core orificing assure relatively uniform flow independent of plant operating conditions and location in the core? (2.0)

5.7 The HCU accumulators are charged with nitrogen to assure rapid insertion of the control rods.

a. When the accumulators are precharged, instructions in O.P. 2111, Control Rod Drive System, recommend that the gas be permitted to enter at a slow rate. What is the purpose of this instruction, and what would result if the gas were permitted to enter at a fast rate? (2.0)

b. What is the approximate accumulator nitrogen pressure during normal reactor operation? Why is it so different from the precharge pressure? (1.0)

5.8 With the reactor operating at full power, and a feedwater controller malfunction resulting in a loss of all feedwater, the reactor will scram on low water level in about eight seconds.

a. Prior to the reactor scram, would you expect the reactor power to increase, decrease, or remain the same? (0.5)

b. Give two reasons for your expectations for the behavior of the reactor power. (2.0)

CATEGORY CONTINUED ON NEXT PAGE

5.9 Indicated reactor water level at 100 percent power differs from the actual water level directly above the core.

a. Which level (actual or indicated) is higher, and by how many inches? (1.0)

b. Explain why the above difference occurs. (1.5)

5.10 Indicate whether each of the following will increase or decrease (add positive or negative) reactivity during reactor operation.

a. Moderator temperature increases while below saturation temperature. (0.5)

b. Fuel temperature increases. (0.5)

END OF CATEGORY

6. PLANT SYSTEMS: DESIGN, CONTROL, AND INSTRUMENTATION (25)

6.1 For each of the following failures of the controlling pressure regulator, state what will happen to reactor pressure and power. Also indicate whether a reactor scram might occur, and what signal would cause the scram.

a. MPR output decreases toward zero during startup. (0.75)
b. EPR output decreases toward zero during steady state operation at 90 percent power. (0.75)
c. EPR output increases to maximum during steady state operation at 90 percent power. (0.75)

6.2 The RBCCW System provides cooling to several essential and non-essential systems in the Reactor Building.

a. How is the temperature of the water in the RBCCW System regulated? (1.0)

b. How is the temperature of individual components cooled by the RBCCW System regulated? (1.0)

c. On a loss and subsequent return of power to the RBCCW pump busses, will one or both RBCCW System pumps restart? Explain. (1.0)

6.3 The Containment Instrument Air System provides a source of compressed gas to air-operated drywell equipment. If the CAC should shutdown because of a high discharge air temperature, what two other sources of gas supply might be available? Be sure to indicate under what conditions each of these two alternate supplies would be available. (1.5)

Normally CAC is S/D. ∴ attack the question from point of view that if CAC is supplying, what are the two alternatives.

CATEGORY CONTINUED ON NEXT PAGE

6.4 The Nuclear Instrumentation is designed to provide for channel overlap during startup and shutdown operations. Various interlocks are provided.

a. For the SRMs, two interlocks are part of the system. For the Retract Permissive Interlock, (1) what three signals are monitored, and (2) what four conditions must be violated simultaneously for a rod block to occur? (3) Will the interlock stop detector movement? (2.0)

b. When the Mode switch is transferred to RUN, what IRM scram functions are bypassed? (0.5)

c. What three APRM trips will occur if the transition to RUN is not made within the necessary power range? (0.75)

6.5 Following the occurence of a small leak into the Drywell, HPCI auto initiates and operates as designed. Explain how each of the following parameters is affected (increase, decrease, or stay the same) as reactor pressure decreases from 1000 to 400 psig.

a. HPCI flow to the reactor (1.0)
b. HPCI pump discharge head (assume constant NPSH) (1.0)
c. HPCI turbine RPM. (1.0)

6.6 The Diesel Generators are designed to automatically start on receipt of undervoltage or certain plant emergency signals. Because they are emergency backup systems, they are also designed to be operated manually for periodic testing.

a. If a DG fails to start (doesn't reach minimum speed), the engine is stopped and "Locked Out". What is required to enable the DG to be restarted? (0.5)

b. For parallel operation of the DGs during testing, the droop setting must be adjusted to be different from what it is for normal, or isochronous, operation. Why is there a difference in these settings? (1.5)

CATEGORY CONTINUED ON NEXT PAGE

6.7 While the reactor is operating at 100 percent power in three element level control (See Figure 6.1.), one of the steam flow inputs is lost. Discuss the effect this will have on the reactor level. Be sure to include in your discussion how the various inputs are derived, how error signals are developed and used, and what the final approximate indicated and actual parameter values (reactor power, steam flow, FW flow, and level) will be. (3.0)

6.8 While you are assisting a new operator with a full flow test of the A Core Spray loop, a major leak develops in one loop of the Recirculation System, the reactor pressure quickly decreases to about 500 psig, and then very slowly continues to decrease. Core Spray receives an immediate auto-initiation signal.

a. Explain what will happen to the (1) Test Bypass Valve (CS-26A), (2) the Minimum Flow Valve, and (3) the Discharge Valves (CS-11A and 12A). (1.5)

b. Will Core Spray inject? If not, why? If so, under what conditions? (1.5)

6.9 The Reactor Water Cleanup System is operating in the letdown mode with discharge to the condenser. System indications are:

Dump Flow:	95 gpm
NRHX Outlet Temperature	142 °F

3 pts within RWCU only?
.5 /action or cause

Using these indications, list the auto actions that should have occurred in the RWCU System. Include the parameters/parameter values that will have caused each action, including setpoints. (3.0)

6.10 What provides the motive force for opening and closing the Main Steam Isolation Valves? (1.0)

END OF CATEGORY

7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL (25)

7.1 During plant heatup after reaching critical, according to O.P. 0101,

a. How and when is warmup of HPCI and RCIC completed? (1.0)

b. Satisfactory MPR operation is to be demonstrated after pressure has reached about 900 psig. However, steam flow cannot be read accurately at these rates. How is it estimated? (0.75)

c. After transferring to RUN, the IRM detectors are to be withdrawn. How can you be sure that proper withdrawal is occurring? (0.75)

7.2 With fuel in the vessel and the reactor pressure vessel head not installed, according to the Admin. Limits in O.P. 1101, Fuel Assembly Movement,

a. What personnel location requirements must be met for performing (1) subcritical checks of a fuel cell, and (2) shutdown margin tests of the reactor core? (1.5)

Relative to part a.

b. What verification is required prior to control rod motion? (1.0)

7.3 If a high temperature and an ionization signal are received from the MG Set area, an alarm is sounded in the control room and a timer is started. Quick action is required to be taken according to O.P. 2186, Fire Suppression Systems. After arriving at the MG set area and determining that the alarm is false or the fire is small, what is the operator required to do? Why? (2.0)

CATEGORY CONTINUED ON NEXT PAGE

7.4 According to O.P. 3105, Relief Valve Stuck Open Emergency Procedure,

a. What are four parameter changes you would be able to check to verify that a relief valve was open? Do not list alarms or lights that initiate or change state. (1.0)

b. What immediate actions are you required to take according to this procedure? Be sure to include any time considerations. Do not include any actions that may be required after a reactor scram may occur. (2.0)

7.5 As a Senior Reactor Operator, you may be in a position where you will be supervising individuals working in areas where they may receive exposure to radiation. According to A.P. 0501, Radiation Protection Standards, what are your responsibilities as a supervisor to assure exposures are kept ALARA? Answer this question by completeing the following sentences.

a. It is the responsibility of each supervisor to be cognizant of . . . (0.75)

b. Individual work assignments should be made so as to limit . . . (0.75)

c. Factors which may affect the selection of a specific individual are . . . (three items) (1.0)

7.6 According to O.P. 3126, Shutdown Using Alternate Shutdown Methods,

a. What level of emergency should be declared if the Control Room must be evacuated and the reactor shutdown from alternate locations? (0.5)

b. What four methods of communications, in order of decreasing reliability, should be considered for maintaining contact among the Shift Supervisor and the alternate operating stations? (1.0)

c. Why should the diesels be started as soon as possible? (1.0)

CATEGORY CONTINUED ON NEXT PAGE

7.7 Prior to operating the RHR System in the Shutdown Cooling Mode, O.P. 2124, Residual Heat Removal System, advises to flush the system to Rad Waste with about 5500 gallons of water from the reactor.

a. What are two reasons for requiring this flush? (1.0)

b. During the flush, water bypasses the RHR heat exchanger through RHR-65. But the procedure warns that the flush can still affect the reactor cooldown rate, and cautions to control the cooldown by intermittently shutting and opening the drain valve to Rad Waste, RHR-66. Briefly explain how flushing can affect the vessel cooldown rate. (1.0)

7.8 The Recirculation Pumps are provided with two full pressure mechanical seals, each of which operates at one-half reactor system pressure. According to O.P. 2110, Reactor Recirc System,

a. For each of the following seal failures, state the effect on seal temperatures, pressures, and controlled leakage flow rates (i.e., increase, decrease, or remain the same).

(1) Failure of the Number 1 seal. (1.0)

(2) Plugging of the Number 1 internal orifice. (1.0)

for single or double failure?

b. When and what operator action is required if recirculation pump seals fail? (1.0)

CATEGORY CONTINUED ON NEXT PAGE

7.9 O.P. 3111, the Loss of Condenser Vacuum Emergency Procedure, states that the first indication of a loss of vacuum will be the alarm at 5" HgA backpressure.

a. What three additional automatic actions will occur as a result of continuing decreasing condenser vacuum? (1.5)

b. The immediate action steps require that reactor power be reduced concurrently with other efforts to recover vacuum.

(1) Under what conditions is the turbine to be tripped? (1.0)

(2) When may the mechanical vacuum pump be started to help maintain vacuum? (0.5)

7.10 According to O.P. 2134, Reactor Protection System, four automatic actions may occur should it be necessary to shift the RPS from its normal power source to its alternate source. What are these four automatic actions? (2.0)

END OF CATEGORY

8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS (25)

8.1 During a refueling outage, the HPCI turbine was torn down for seal repairs. According to the Technical Specifications, what action is required prior to full power operation?
NOTE: Use the provided sections of the Technical Specifications to answer this question. Be sure to reference each section you use. (2.0)

8.2 a. According to A.P. 0036, Shift Staffing, under what conditions is the licensed reactor operator at the controls permitted to be outside the "line of site"? (1.0)

b. According to A.P. 0150, Responsibilities and Authorities of Operations Department Personnel, what condition must be met for the on-duty Supervisory Control Room Operator to perform a plant tour? (1.0)

8.3 According to the Administrative Limits in the Startup and Heatup Procedures (O.P. 0100 and O.P. 0101), while the reactor is below 20 percent power,

a. Who by title must provide authorization for bypassing the RWM, and (1.0)

b. What three conditions must be met for this authorization to be granted? (1.5)

8.4 At about 8 pm with the reactor operating at 80 percent, an instrument technician, who has been working on the quarterly functional checks and calibrations since early in the day, informs you that he has just determined the RBM Channel A count circuit to be malfunctioning. He has checked the spare parts availability at the plant, and has determined that parts will have to be obtained from off site. He does not know whether they can be obtained the next day. According to the Technical Specifications, what action must be taken immediately, and what must be done if the spare parts cannot be obtained for three days?
NOTE: Use the provided sections of the Technical Specifications to answer this question. Be sure to reference each section you use. (3.0)

CATEGORY CONTINUED ON NEXT PAGE

8.5 During a shift turnover, with the plant operating at 45 percent power, you are informed that the MSIV Full Closure Time Surveillance Test has exceeded the maximum allowable extension interval and will be performed on your shift. Halfway through the test, the "C" Outboard MSIV fails to meet the specified closing time. In accordance with the Technical Specifications (NOTE: Use the provided sections of the Technical Specifications to answer this question. Be sure to reference each section you use.),

Not sure of what is meant by a situation.

a. What situation exists due to continued plant operation in the above condition? (1.0)

b. What actions must be taken due to the fact that the MSIV has failed its closing time test? (2.0)

8.6 According to the Technical Specifications, what constitutes Secondary Containment Integrity? (2.0)

8.7 A frequency control failure results in a trip of the UPS A inverter while the reactor is operating at reduced power (50 percent) on a weekend. Preliminary investigation by an operator seems to indicate that instrument technicians will need to be called in, and that repair may take several days. (On Monday, it is found that a replacement part _may_ not be received for one full week.) However, the operator is able to transfer MCC-89A Power from UPS to the Maintenance Tie.
NOTE: Use the provided sections of the Technical Specifications to answer this question. Be sure to reference each section you use.

a. Is the UPS A module considered to be inoperable for purposes of satisfying Technical Specification requirements? (0.5)

b. Independent of your answer to Part (a), if the UPS A module _is inoperable_, what actions are required to be taken by the Technical Specifications. (2.0)

8.8 According to A.P. 0150, Responsibilities and Authorities of Operations Department Personnel, when is the Shift Supervisor authorized to order and the Supervisory Control Room Operator authorized to shutdown the reactor? (1.0)

CATEGORY CONTINUED ON NEXT PAGE

8.9 According to A.P. 0156, it is the responsibility of the duty Shift Supervisor to insure that notification of significant events is performed in a timely manner.

a. When does the one-hour or four-hour clock start? (0.5)

b. List three events that would need to be reported on the one-hour clock. (1.5)

8.10 One reactor recirculation pump has been operating with an elevated leakage rate (but well within acceptable limits) for most of the previous shift. About one hour after your shift starts, while the reactor is operating at about 25 percent power, you are informed that the measured leakage rate has increased to about 75 gpm. You immediately order the tripping and isolation of the suspected recirculation pump. The measured leakage rate decreases immediately to less than 15 gpm. According to A.P. 3125, would you classify this event as an Alert, as an Unusual Event, or as no event at all. Explain your decision. (2.0)

8.11 According to the Admin. Limits in O.P. 2111, Control Rod Drive System,

a. What action is required if reactor pressure is less than 800 psig with more than one HCU in alarm status and both CRD pumps out of service? (0.5)

b. With the reactor operating at full power and both CRD pumps out of service for more than two minutes, the recirc pump seal purge must be secured prior to restarting the CRD pumps. Why? (1.0)

8.12 During RCIC testing, the suppression pool temperature was inadvertently permitted to increase to above 110 degrees F. The reactor was scrammed in accordance with Tech Specs, and is now in Hot Shutdown. Suppression Pool cooling is on, and the Suppression Chamber water temperature has been reduced to 93 degrees F. Can you start up the reactor and enter the RUN mode?
NOTE: Use the provided sections of the Technical Specifications to answer this question. Be sure to reference each section you use. (1.5)

END OF EXAMINATION

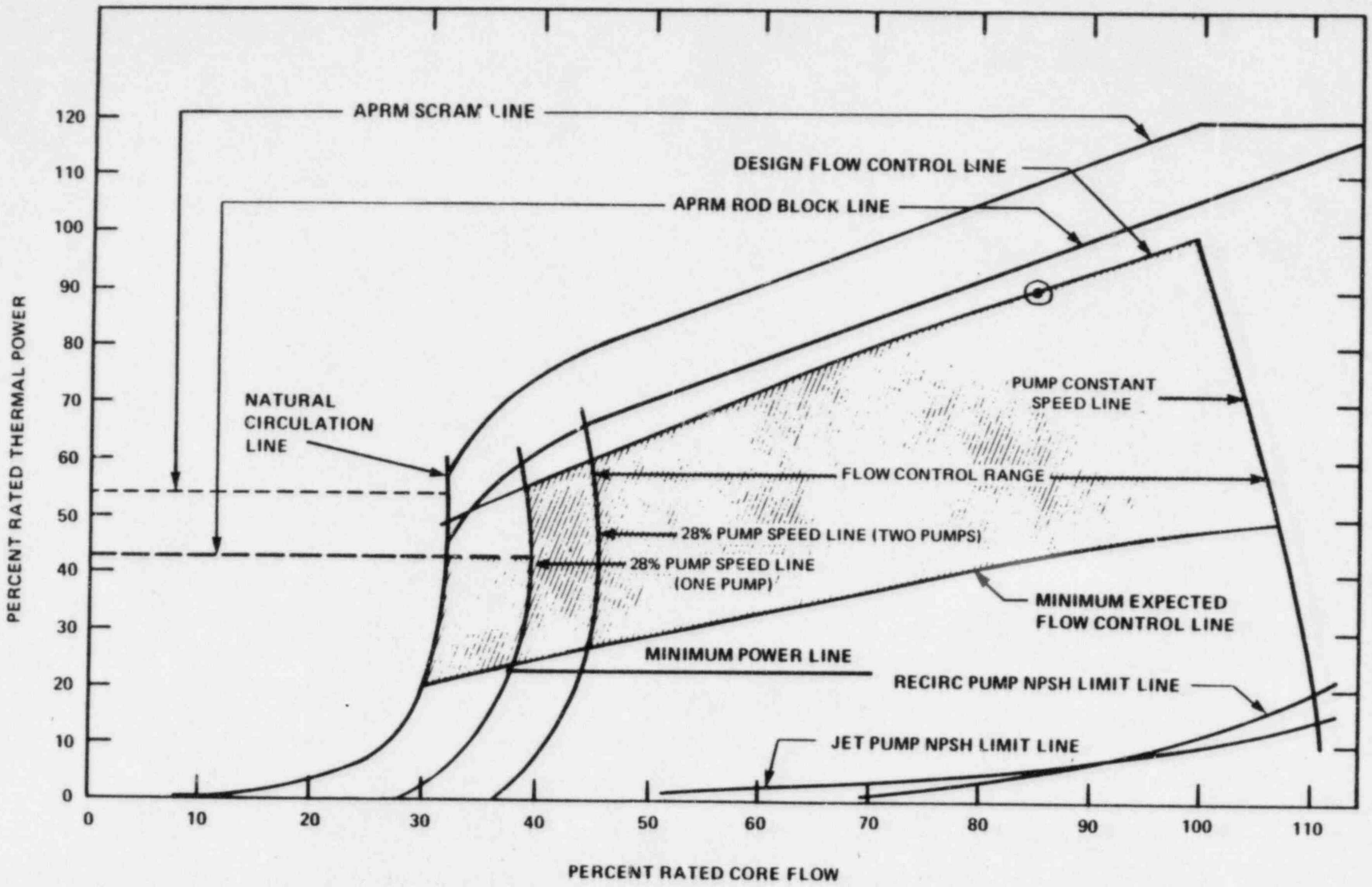

Figure 5.1

0280

Figure I
Feedwater Control System Simplified Diagram

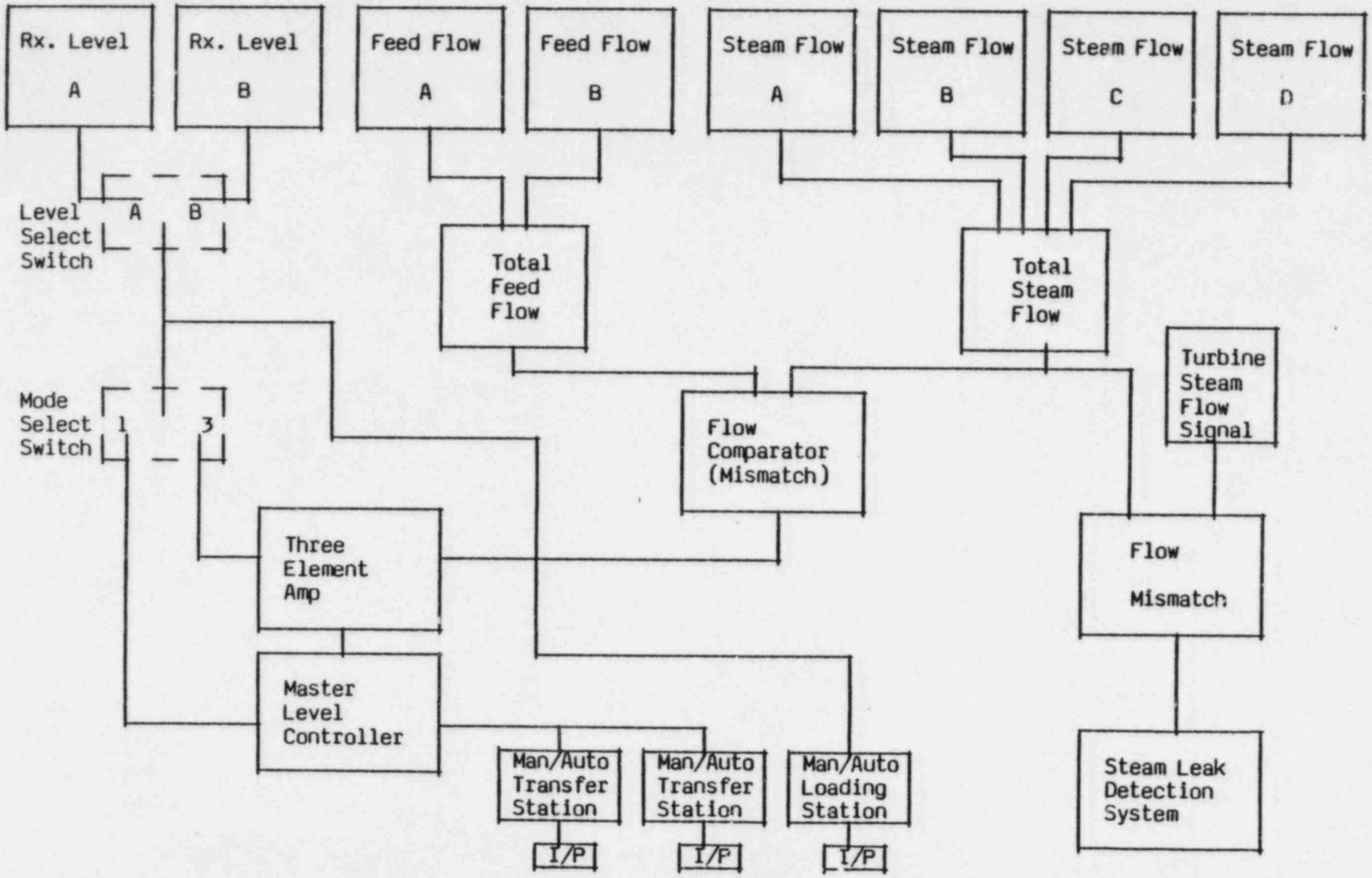

Figure 6.1

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Figure I
Feedwater Control System Simplified Diagram

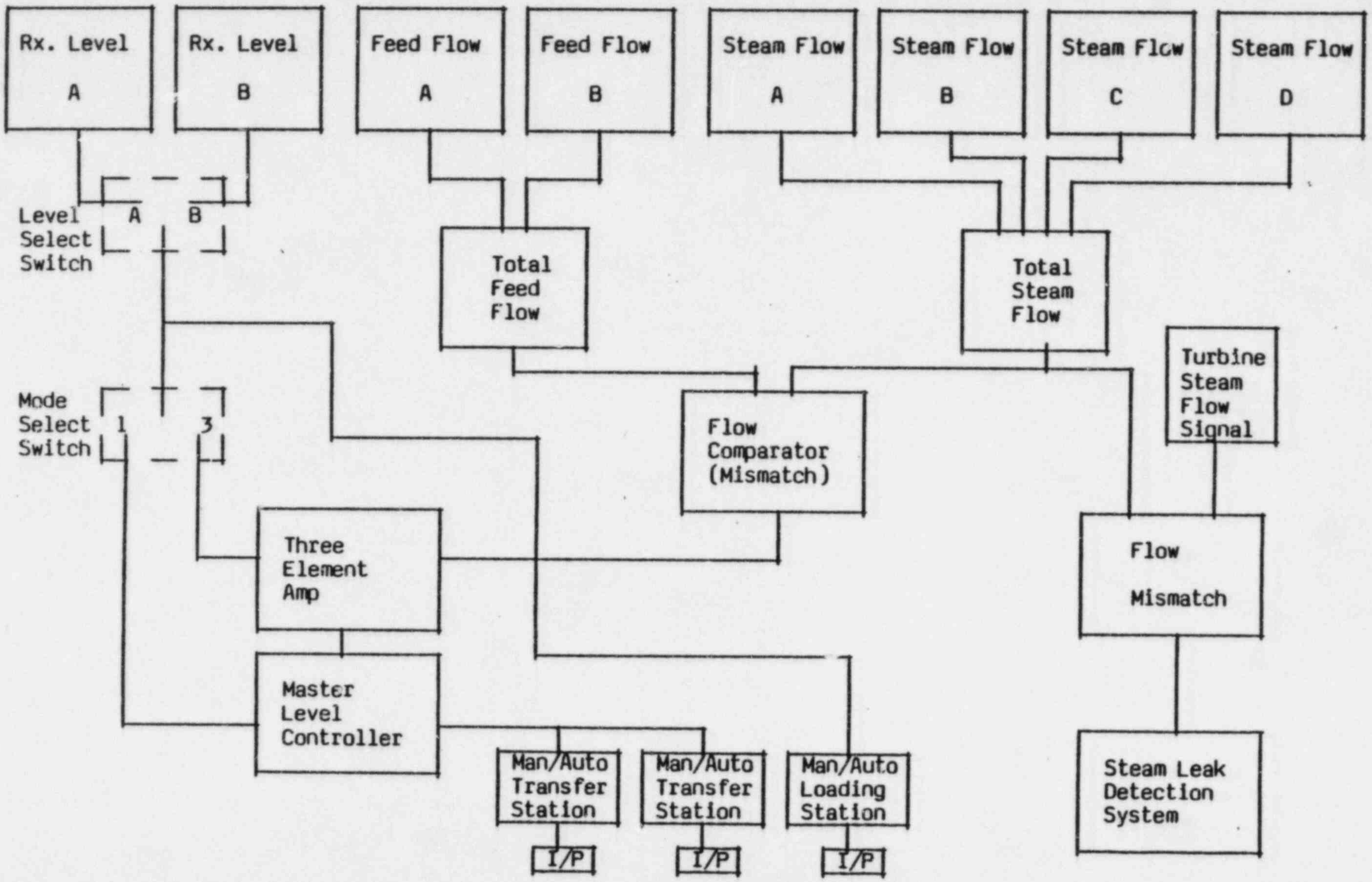


Figure 6.1

EQUATION SHEET

$f = ma$ $v = s/t$ Cycle efficiency = (Network out)/(Energy in)

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

$E = mc^2$

$KE = 1/2\ mv^2$ $a = (V_f - V_o)/t$ $A = \lambda N$ $A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$ $w = \theta/t$ $\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

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

$\Delta E = 931\ \Delta m$

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

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

$\dot{Q} = UA\Delta T$ $I = I_o e^{-\mu x}$

$Pwr = W_f \Delta n$ $I = I_o\ 10^{-x/TVL}$

$TVL = 1.3/\mu$

$P = P_o 10^{sur(t)}$ $HVL = -0.693/\mu$

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

$SUR = 26.06/T$ $SCR = S/(1 - K_{eff})$

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

$SUR = 26\rho/\ell^* + (\beta - \rho)T$ $CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$T = (\ell^*/\rho) + [(\beta - \rho)/\bar{\lambda}\rho]$ $M = 1/(1 - K_{eff}) = CR_1/CR_o$

$T = \ell/(\rho - \beta)$ $M = (1 - K_{effo})/(1 - K_{eff1})$

$T = (\beta - \rho)/(\lambda\rho)$ $SDM = (1 - K_{eff})/K_{eff}$

$\rho = (K_{eff}-1)/K_{eff} = \Delta K_{eff}/K_{eff}$ $\ell^* = 10^{-4}$ seconds

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

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

$I_1 d_1 = I_2 d_2$

$P = (\Sigma\phi V)/(3 \times 10^{10})$ $I_1 d_1^2 = I_2 d_2^2$

$\Sigma = \sigma N$ $R/hr = (0.5\ CE)/d^2(meters)$

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.

Miscellaneous Conversions

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

Answer parts in [ ] are not required.

5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS (25) - ANSWERS

5.1 (1) Increased power causes increased voiding in the top of the core. This causes a flattening of the peak-to-average flux profile surrounding a control rod. Since the rod worth is proportional to the square of this ratio, rod worth decreases. 1.0

(2) At low power levels when water is dense or no voiding is present, the neutron migration length (or diffusion length) is shorter, and the resultant transient is confined to a local area. 0.5

Add 280cal/gm discussion (0.5)

REFERENCE: 1. Lesson Plan, RWM, page 3
2. Tech Specs, page 76

5.2 a. The ~~actual~~ indicated flow rate in the affected loop would increase. (0.25)
for recirc loop flow
This is because the break in the jet pump reduces the resistance to flow in the affected loop, and the recirc pump will operate at a higher flow condition for the same controller setting. (0.75)

Must watch for an answer that would be based on jet pump sum rather than recirc loop

b. The actual total core flow rate would decrease and the indicated flow rate would decrease slightly. (0.25)
This is because the failed jet pump would provide a bypass flow path around the core. The indicated core flow will be higher than actual core flow because the reverse flow through the jet pump will be indicated as a positive flow by the instrumentation. (0.75)

c. The measured flow would ~~increase~~ (decrease) and the power would decrease. Therefore, the new operating position would be to the lower right on the map. (0.25)
The loop flow would increase for the reason given in the Part (a) answer, but the actual core flow would decrease as stated in Part (b) answer. The decrease in actual core flow would result in increased voiding in the core, and therefore an insertion of negative reactivity, resulting in a power decrease. (0.75)

They will interpret the loop flow will go down because jet pump sum is actually used, so the flow will decrease & the line will move to the lower left.

REFERENCE: Tech Specs Bases Section 3.6.F
Reactor Theory, Chapter 28

5.3 (1) Core flow less than approximately 30 percent rated
(2) Normal CRD cooling water flow in service
(3) No cleanup flow out of the bottom vessel drain

REFERENCE: O.P. 0101, pg. 12, Section II.A.1.

5.4 (1) Relatively higher notch worths would be found in core regions that had relatively low neutron flux levels during the previous reactor operation (or a trip after a high power run). This would be in the peripheral areas of the core, and particularly near the top of the core where boiling had been occuring. (Other reasonable exlanations also acceptable.) (1.5)
(2) Xenon production is a function of local power or flux level. After shutdown, it will peak highest in these previously high flux regions. This will tend to depress the flux in these same regions during the subsequent startup, and cause the flux to be higher in previously low flux regions to maintain an equivalent power level. Since control rod worth is a function of the square of the flux, these regions will have relatively higher worth notches. (1.5)

REFERENCE: O.P. 0100, pg. 3
Reactor Theory, Chapters 31 and 33.

5.5 a. Conductivity is monitored.
They are ions.

REFERENCE: Chemistry Lesson Plans, pg. 3-2.

b. At higher steaming rates, the high oxygen levels necessary for stress corrosion cracking are not present, and higher chloride levels are permissible. (0.5)
At startup, conductivity may be higher because of increased initial evolution of gases, and because at lower temperatures, corrosion rates are reduced. (0.5)

REFERENCE: Tech Spec Bases, Section 3.6.B.
Chemistry Lesson Plans, pg. 1-10.

5.6 a. As power increases, the amount of boiling in a flow channel increases, and causes an increase in the resistance to flow - that is, core dP increases.

b. (1) Two regions are used for core orificing. The central region consists of all but the outermost ring of perimeter fuel with the smaller or fewer orifices in the outer fuel positions. taught as zoning

(2) The orifices installed in the fuel supports add a relatively large dP to the total dP across the core. The dP added by boiling is then insignificant compared to the dP of the orifices, and so the dP is nearly the same (as is flow) to all fuel cells under any condition of power.

REFERENCE: System Lesson Plan, Reactor Vessel and Internals, pgg. 13 - 14.

5.7 a. (1) As the high pressure gas in the supply bottle is released into the accumulator, it expands rapidly and thus cools. By filling the accumulator slowly, the gas will tend to reach equilibrium temperature and establish the proper pressure more rapidly. (1.5)

(2) If it were permitted to fill rapidly to the pressure given in the procedure [620 - 650 psig] , as it warmed to ambient conditions the gas pressure would increase to a value higher than the procedure calls for. (0.5)

b. Normal pressure at operating conditions is about 1100 psig. - normally runs ~ 1050 psig @ VY (0.25)
The charging water decreases the accumulator gas volume by about half. Note that the gas pressure is not the same as the charging water pressure. (0.75)

REFERENCE: O.P. 2111, ppg. 6-7.

5.8 a. The power would decrease.

b. (1) Loss of FW would result in a decrease of inlet subcooling and insertion of negative reactivity. (1.0)
(2) When FW flow falls to 20 percent [in about 4-5 sec], recirc runs back, causing increased voiding and a further influence on a power decrease. (1.0)

REFERENCE: O.P. 3112 and Reactor Theory, Chapters 26, 27, and 28.

5.9 a. (1) Indicated level is higher. (0.5)

(2) Approximately 7 - 10 inches. (0.5)

b. The indicated water level is sensed outside the dryer skirt. (0.75)
Steam flow at 100 percent power (0.25)
causes a backpressure on the dryers. (0.50)

REFERENCE: System Lesson Plan, Reactor Vessel Process Instrumentation

5.10 a. Decrease or negative reactivity

b. Decrease of negative reactivity

REFERENCE: Reactor Theory, Chapter 26, 27, and 29.

End of Category

6. PLANT SYSTEMS: DESIGN, CONTROL, AND INSTRUMENTATION (25) - ANSWERS

6.1 a. Will cause a pressure increase [by closing the bypass valves]
Power will increase.
A scram may occur from the power spike.

b. Turbine inlet pressure will increase, and therefore reactor pressure. Control will switch to the MPR which is usually set about five percent higher.
Reactor power will increase slightly.
A scram should not occur.

In actuality, MPR at VY does not catch the pressure transient and they get a scram on flux. Those with operating experience will probably say this

c. This will cause a pressure reduction.
Power will decrease.
A scram will occur if pressure decreases to the level of the [850 psig] MSIV isolation signal.
changed to 800 about 1/15/85

REFERENCE: O.P. 3104, PR Malfunction EP

6.2 a. RBCCW System temperature is adjusted and regulated by throttling service water flow through the heat exchanger.

b. Temperature of individual components being serviced is regulated by using the return globe valve from that component to adjust the flow.

c. Normally the system is operated with both pump control switches in the Auto position. If power should be returned to both pump busses at the same time, both pumps will start. If power is returned to one bus prior to the other (normal), one pump will start. This will satisfy the auto start pressure switch of the other pump so that it will not start when power is returned to its bus.

REFERENCE: O.P. 2182
System Lesson Plan - RBCCW.

6.3 If the containment is inerted, the Containment Makeup System is available for cross-connect. (0.75)

When the containment is not inerted, the Instrument Air System should be cross-connected. (0.75)

REFERENCE: System Lesson Plan - Air Systems, pg. 13-30
O.P. 2191, ppg. 7-8.

6.4 a. (1) SRM Log Count Rate (0.25)
SRM Detector Position (0.25)
IRM Range Switch Position (0.25)

Ref due

(2) Any one of the following four conditions will preclude a rod block. That is, all four must be violated simultaneously for the rod block to occur.

SRM detector full in (0.25)
SRM log count rate above 100 cps (0.25)
All IRM range switches on or above Range 3 (0.25)
Mode switch in RUN with APRMs on scale (0.25)

(3) Interlock does not stop detector movement. (0.25)

b. (1) Hi Hi [120/125 of scale]
(2) INOP [H. V. low, Function Sw., Module unplugged]

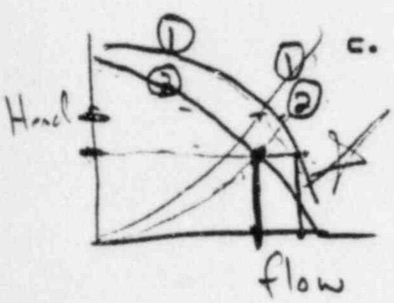
c. (1) < 5 % in RUN gives Rod Block
(2) > 12 % not in RUN gives Rod Block could be 13% per OP-0100
(3) > 15 % not in RUN gives scram
Ref due (4) < 5% in RUN w/ corresponding IRM upscale or inop will give a scram

REFERENCE: System Lesson Plan - NIs, ppg. 24-25, 36, 98-99.

6.5c

6.5 a. Remains constant. (0.25)
Flow is controlled by the flow controller which will maintain constant flow by controlling steam flow to the turbine. (0.75)

b. Decreases. (0.25)
The flow controller operates to maintain a constant flow as measured at the flow element. As the reactor pressure head decreases, the pump discharge head decreases to maintain constant flow. (0.75)

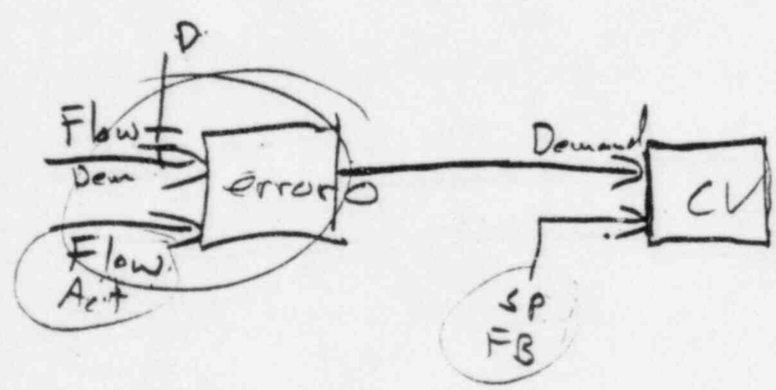


c. Remains constant. (0.25)
The turbine speed is a function of the pump flow at the flow element. The speed changes as a function of the error signal between the control room set point and the actual pump flow. (0.75)

REFERENCE: System Lesson Plan - HPCI, ppg. 4 and 11.

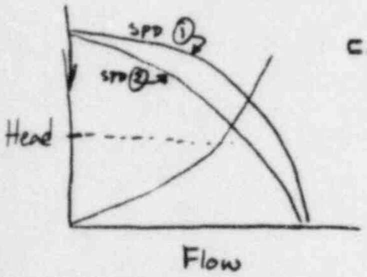
6.6 There is a difference because at zero droop the frequency will be controlled be the DG. As droop is increased, a second generator on the line, in this case, the main generator, is able to control the DG unit speed, and thus system frequency.

REFERENCE: O.P. 2126, ppg. 4 - 5.
System Lesson Plan - EDGs, ppg. 31, 32, 53.



6.5 a. Remains constant. (0.25)
Flow is controlled by the flow controller which will maintain constant flow by controlling steam flow to the turbine. (0.75)

b. Decreases. (0.25)
The flow controller operates to maintain a constant flow as measured at the flow element. As the reactor pressure head decreases, the pump discharge head decreases to maintain constant flow. (0.75)



c. ~~Remains constant.~~ Decrease (0.25)
The turbine speed is a function of the pump flow at the flow element. The speed changes as a function of the error signal between the control room set point and the actual pump flow. (0.75)

REFERENCE: System Lesson Plan - HPCI, ppg. 4 and 11.

6.6 b There is a difference because at zero droop the frequency will be controlled by the DG. As droop is increased, a second generator on the line, in this case, the main generator, is able to control the DG unit speed, and thus system frequency. Also, as droop is increased, load sharing (division) is enabled and assures load sharing stability. At zero droop, a DG will will be able to change load without changing speed.

⇒ load sharing

REFERENCE: O.P. 2126, ppg. 4 - 5.
System Lesson Plan - EDGs, ppg. 31, 32, 53.

6.6. a. You have to reset the Lockout at the local panel.

6.7 (1) (a) The steam flow signals from the four main steam lines are summed, and (b) the FW flows from two parallel FW lines are summed. (c) With loss of one of the four steam signals, a 25 percent mismatch will exist. (0.6)

(2) (a) This mismatch will produce an error signal which (b) will reduce feed flow to correct the error signal. (0.4)

(3) (a) Since the actual steam flow has not been changed, (b) the level will decrease, and (c) a second error signal will be generated. (d) Since the system is level dominant, (e) a signal will be sent to open the FCVs and increase feed flow to maintain level constant. (1.0)

At VY, the 100% loss of stm signal is worth a 7" level change. ∴ a 1/4 loss will result in ≈2" lower.

(4) The final indications will be: (1.0)

Reactor power: 100 %
Steam flow: 100 % actual, 75 % indicated
FW flow: 100 %
Level: Constant, but lower than prior to the fault

REFERENCE: System Lesson Plan - FW Control

6.8 The Core Spray system is already operating, in the full flow test mode. Therefore the test bypass valve is open and the min-flow valve is closed.

a. (1) The test bypass valve [CS-26A] will close with no delay. (0.5)
(2) The min-flow valve will open when low flow [less than 300 gpm] is sensed in the flow element. When [300 gpm] flow is sensed by the flow element, the min-flow valve closes. (0.5)
(3) When reactor pressure decreases to 350 psig, the discharge valves CS-12A and CS-11A will be signaled OPEN. (0.5)

one of these is already open (the one that is the throttle valve).

b. No. ← Need to consider context. (0.5)
When reactor pressure decreases to below the Core Spray System pressure, injection to the vessel begins. (1.0)

REFERENCE: O.P. 2123, Core Spray, pg. 3
System Lesson Plan - Core Spray pg. 6.

6.9 (1) The CU isolation valves, CU-15, 18, and 68, will shut when the demin inlet temperature is 140 degrees F. (1.0)

(2) The RWCU pump will trip whenever any one of the isolation valves is not fully open (15 + 18 not full open and 68 full shut) (1.0)

REFERENCE: O.P. 2112, ppg. 1,2, 14.
System Lesson Plan - RWCU

Ref due for

6.10 (1) Gas [nitrogen inboard, air outboard] to open (0.5)

(2) Spring and/or gas to close (0.5)

REFERENCE: System Lesson Plan - Main Steam, ppg. 14 - 15.

END OF CATEGORY

the dump valve will trip on low pressure [5 psig] after the the RWCU isolates to keep the sys from draining down. pg 2 center

6.9 (3) The hold pump will start when system flow falls to 25 gpm [and the hold pump disch vlv (CU-37A(B) will open] (0.5)

4. CU-55, the drain flow regulating valve, may close if upstream pressure decreases to below 5 psig to keep from draining the system. (0.5)

7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL (25) - ANSWERS

7.1 a. Prior to the reactor pressure reaching 100 psig, keylock switches [on CRP 9-39 and 9-30] are positioned to HPCI and RCIC WARMUP, respectively, to open the steam inlet valves [HPCI/RCIC 15 and 16]. These low pressure bypass switches are to be restored to NORMAL prior to the reactor pressure reaching 150 psig.

b. It is estimated by observing Bypass Valve No. 1 to be positioned between 7 and 15 percent open.

c. By selecting each recorder to IRM and observing that the flux level is decreasing.

REFERENCE: O.P. 0101, Section I.A. Normal Operation
a. Steps 5 and 11.
b. Step 19. a.
c. Section I.B.1.d.

7.2 a. (1) All personnel should be out of the line of site of the open vessel.
(2) All personnel should be clear of the refueling floor.

b. These facts must be verified by the senior licensed operator on the refueling floor and transmitted to the Shift Supervisor prior to control rod motion.

REFERENCE: O.P. 1101, References B.8., pg. 2.

7.3 In either case, the operator must rapidly proceed to the manual control panel, break the glass, shift the lever to ABORT (0.5) in order to preclude the automatic initiation of the deluge system (0.5), and call the control room (0.5).
For a small fire, he should then fight it with a locally available fire extinguisher (0.5).

REFERENCE: O.P. 2186, pg. 8.

7.4 a. (1) 12 - 15 % drop in generator output
(2) 12 - 15 % drop in steam flow
(3) Suppression pool temperature increasing
(4) Suppression pool level increasing
(5) Feedflow adjusts to maintain level
(6) Distinct audible level change in Reactor Building
(7) Change in D/W-Torus dP (goes low)
(8) tail pipe temperature
(9) tail pipe pressure
(0.25) for each of any four

b. (1) Determine which relief valve and cycle the control switch from AUTO to OPEN to AUTO. If successful, no further action required. (0.5)

(2) If not successful,
(a) transfer station loads to S/U Transformer (0.5)
(b) and manually scram the reactor (0.5)
(c) within 2 min of the relief valve lifting. (0.5)

REFERENCE: O.P. 3105, ppg. 1 - 2.

7.5 a. It is the responsibility of each supervisor to be cognizant of . . .the current radiation exposure of each person under his direct supervision.

b. Individual work assignments should be made so as to limit . . . the total absorbed dose and the individual's accumulated exposure.

c. Factors which may affect the selection of a specific individual are . . . the individual's physical condition, skill level, and year-to-date exposure.

HP training tells them also consider age. I said this should be ALARA related.

REFERENCE: A.P. 0501, pg. 2.

7.6 a. An Alert [per O.P. 3501]

b. (1) Runners
(2) Sound powered phones
(3) Radios from the guard force
(4) Gai-tronics and telephones

c. The diesels should be started as soon as possible to provide LPCI capability in case of a stuck open relief valve.

REFERENCE: O.P. 3126, ppg. 3 - 4.

7.7 a. (1) To prevent the injection of suppression pool water into the vessel.

(2) To preheat the Shutdown Cooling System.

b. When flushing the Shutdown Cooling System, hot reactor water is drawn from the vessel and must be made up with relatively cool condensate water. Too high a flush rate may cause too high a cooldown rate. [Cooldown to be limited to 90 degrees F/hr.]

REFERENCE: O.P. 2124, ppg. 3 and 16.

See Ref 17 Attached

7.8 a. (1)

Number 1 seal pressure	No change
Number 2 seal pressure	Increases
Number 2 seal leakage (staging flow)	Increases (not the flow to DWEDS)
Both seals temperature	~~Increase~~ decrease

(2)

Number 1 seal pressure	No change
Number 2 seal pressure	Decreases
Number 2 seal leakage	Decreases
Both seals temperature	Increase — Logic vs. procedure #2 should increase #1 should stay the same

~~b. Operator action is not required until both seals fail (0.5)~~

b. Operator action is not required until both seals fail (0.5)
and then the recirc pump should be shutdown and the reactor shutdown and depressurized. (0.5)

REFERENCE: O.P. 2110, pg 16. See also sh 5 of G-19 1159

7.9 a. (1) Turbine stop valve closure at 10" HgA
(2) MSL isolation at 12" HgA backpressure
(3) Bypass valve closure at 23" HgA

b. (1) If 30 percent power is reached and vacuum cannot be maintained below 5" HgA.
(2) After the mode switch is placed in startup. or out of RUN

REFERENCE: O.P. 3111, ppg. 1 - 2.

7.10 (1) Reactor Building HVAC will trip.
(2) Standby Gas Treatment will start.
(3) AEOG System may isolate with consequent closure of Off Gas System valves [OG 516A(B)].
(4) A half scram will occur.

REFERENCE: O.P. 2134, pg. 3.

8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS (25) - ANSWERS

8.1 If HPCI was inop while at cold shutdown, it must be demonstrated operable immediately upon achieving a steaming rate sufficient to sustain operation.

REFERENCE: Tech Specs Section 3.5.E
Tech Spec Interpretation # 9

8.2 a. In the event of an emergency affecting the safety of operations, the Operator at the controls may momentarily enter the back panel area to initiate corrective action provided he remains within the control room.

REFERENCE: A.P. 0036, B. 1. Note.

b. The SCRO must be relieved of his Control Room duties by the Shift Supervisor.

REFERENCE: A.P. 0150, pg. 5.

8.3 a. 1. Operations Supervisor
2. Operations Superintendent
3. Plant Manager

b. Only after

1. Withdrawal of at least 12 rods, and
2. Provided a second licensed operator monitors and documents further rod movements, and
3. The rod select template is in place.

REFERENCE: O.P. 0101, pg. 2.

8.4 (1) T.S. 3.2.E requires the instrumentation that initiates control rod blocks to be operational in accordance with Table 3.2.5. (0.5)

(2) Table 3.2.5 requires one channel per trip system when in RUN mode with three notes. One note permits bypassing at less than 30 percent power (which doesn't apply here) and states the channel is to be considered inop if there are less than half the total number of normal inputs. (0.5)

(3) Note 10 requires an immediate action: Verify that the reactor is not operating in a limiting control rod pattern. If it is, you must satisfy T. S. 3.3.B.6. This requires that both RBMs be operational, or the block must be taken, or the operating power level be limited s.t. MCPR will remain above limit assuming a single error results in the complete withdrawal of any operable control rod. (1.0)

(4) Otherwise Note 10.b and Note 6 permit bypassing the channel for 24 hours if it hasn't been bypassed for 24 hours during the past 30 days. If this cannot be met, it must be tripped within the next hour. (1.0)

REFERENCE: T.S. 3.2.E, Table 3.2.5 with notes, and T.S. 3.3.B.6.

8.5 a. At least once per quarter, the MSIVs must be shown to meet the specs of Table 4.7.2. Operation in violation of the specs of this table is a Tech Spec violation (in an LCO) [and a reportable occurrance].

b. T.S. 3.7.D.2 requires that if any isolation valve specified in Table 4.7.2 becomes inoperable, reactor power operation may continue provided at least one valve in the line with the inop valve is in the isolated condition. Then T.S. 4.7.D.2 requires the position of at least one other valve in each line having an inop valve to be logged daily. This implies that even if the MSIV that fails the test can be closed, the other MSIV in the line must also be closed.

Need Plant Management Interpretation

OK - received verbally at exit interview, 1/24/85

REFERENCE: T.S. 3/4.7.D and Table 4.7.2.

8.6 (1) Reactor Building is intact
(2) At least one door per access opening is closed
(3) Standby Gas Treatment is operable
(4) All RB auto isolation valves are operable or are secured in the isolated position

REFERENCE: Tech Specs, pg 3.

8.7 a. It is inoperable.

REFERENCE: O.P. 2143 Precautions (pg. 3) and Note (pg. 8).

b. (1) When it is determined that one UPS is inop, the requirements of T.S.3/4.5.A shall be satisfied. Effectively says that LPCI is inop. (0.5)

REFERENCE: T.S.3/4.10.B.4, PG. 177.

(2) Reactor operation may continue for seven days provided the 1 other LPCI is operable, and the 2 Containment Cooling subsystem, 3 the Core Spray subsystems, and the 4 DGs are operable. (0.5)

REFERENCE: T.S.3.5.A.4, PG. 87.

(3) The systems listed in (2) must be shown to be operable immediately and daily thereafter. (0.5)

REFERENCE: T.S.4.5.A.4, PG. 87.

(4) If the requirements of 3.5.A cannot be met, an orderly shutdown shall be initiated, and the reactor shall be in cold shutdown within 24 hours. (0.5)

REFERENCE: T.S.3.5.A.6, PG. 87.

8.8 (1) When he determines that the safety of the reactor is in jeopardy, or
(2) when operating parameters exceed reactor protection setpoints and automatic shutdown does not occur.

REFERENCE: A.P. 0150, ppg. 4 - 5.

8.9 a. The clock starts the moment the Shift Supervisor recognizes, or is notified of, a condition that requires notification.

b. Any three items of the 17 items listed on the Appendix A list attached.

REFERENCE: A.P. 0156, pg. 2 and Appendix A.

8.10 (1) Unusual Event. (0.5)

~~(2) The initial event if it continued would be classified as an Alert be~~ ~~(0.5)~~

(2) The initial event if it continued would be classified as an Alert because the leakage rate is above 50 gpm inside the containment. (0.5)

However, for events that have occurred and are no longer present, the procedure states to classify them at the lowest level consistent with the General Criteria. (0.5)

And if an event has occurred and is no longer present, the minimum classification is Unusual Event. (0.5)

REFERENCE: A.P. 3125, pg. 2 and Table I, Section 3 and General Criteria.

8.11 a. The reactor should be manually scrammed [per I. E. Info Notice].

b. Secure the recirc pump seal purge to limit thermal stress to the seals.

REFERENCE: O.P. 2111, pg. 3.

8.12 Power operation is not permitted until the suppression pool water temperature is reduced below 90 degrees F.

However a reactor startup may be performed without going to RUN.

REFERENCE: T.S. 3.7.A.1.c.

END OF EXAMINATION

APPENDIX A

EMERGENCY EVENTS REQUIRING NOTIFICATION WITHIN ONE HOUR

1. Declaration of any emergency class of the Vermont Yankee Emergency Plan [50.72(a)(i)].

2. Initation of a plant shutdown required by the Technical Specifications [50.72(b)(1)(i)(A)].

3. Deviation from Technical Specifications as allowed by 10CFR50.54(x) [50.72(b)(1)(i)(B)].

4. An event or condition during plant operation that results in the plant (including the principal safety barriers) being seriously degraded, or results in the plant being:

   a. In an unanalyzed condition that significantly compromises plant safety, or

   b. In a condition outside the design basis of the plant, or

   c. In a condition not covered by the operating and emergency procedures [50.72(b)(1)(ii)].

5. Any natural phenomenon or other external condition that poses an actual threat to the safety of the plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant [50.72(b)(1)(iii)].

6. Any event that results in or should have resulted in ECCS discharge into the Reactor Coolant System as a result of a valid signal [50.72(b)(1)(iv)].

7. Any event that results in a major loss of emergency assessment capability, off-site response capability, or communications ability [50.72(b)(1)(v)].

8. Any event that poses an actual threat to the safety of the plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant (including fires, toxic gas, radioactive release) [50.72(b)(1)(vi)].

9. Any event that causes or threatens to cause:

   a. Exposure of the whole body of any individual to ≥25 rems (≥5 rems requires notification within 24 hours).

   b. Exposure of the skin of the whole body of any individual to ≥150 rems (≥30 rems requires notification within 24 hours).

   c. Exposure of the feet, ankles, hands, or forearms to ≥375 rems (≥75 rems requires notification within 24 hours).

APPENDIX A

d. The release of radioactive material in concentrations which, if averaged over a 24-hour period, would exceed 5,000 times the limits specified for such materials in 10CFR20, Appendix B, Table II (>500 times the limits requires notification within 24 hours).

e. A loss of one working week or more of the operation of any facilities affected (loss of 1 day or more requires notification within 24 hours).

f. Damage to property in excess of $200,000 (>$2,000 requires notification within 24 hours).

10. A package containing radioactive material received on-site which has been found to have removable radioactive contamination on the external surfaces of the package in excess of 0.01 microcuries (22,000 disintegrations per minute) [10CFR20.205(b)(2)].