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

EXAMINATION REPORT NO. 84-15

FACILITY DOCKET NO. 50-271

FACILITY LICENSE NO. DPR-28

LICENSEE: Vermont Yankee Nuclear Power Corp.
RFD #5, Box 169 Ferry Road
Brattleboro, Vermont 05301

FACILITY: Vermont Yankee

DATES: June 4-7, 1984

CHIEF EXAMINER: Original Signed By: John Berry — 7/11/84 Date

APPROVED BY: Original Signed By: Noel J. Dudley for — Chief, Project Section 1C — 7/11/84 Date

SUMMARY: Written and oral examinations were administered to two Reactor Operator and two Senior Reactor Operator candidates. All four individuals passed the exams.

REPORT DETAILS

TYPE OF EXAMS: Initial ____ Replacement X Requalification ____

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail	Inst. Cert Pass/Fail	Fuel Handler Pass/Fail
Written Exam	2/0	2/0	/	/
Oral Exam	2/0	2/0	/	/
Simulator Exam	/	/	/	/
Overall	2/0	2/0	/	/

1. CHIEF EXAMINER AT SITE: K. M. Henry, ORNL

2. PERSONS EXAMINED

RO

Bronson, Kevin H.
Swanson, Roger B.

SRO

Keith, Ronald M.
Pichette, Brian R.

1. Summary of generic strengths or deficiencies noted on oral exams:

   The examiner noted weaknesses in all four candidates in the areas of Nuclear Theory and Thermodynamics. Deficiencies were also noted regarding the availability and use of the second offsite power source.

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

   None

3. Comments on availability and candidate familiarization with plant reference material:

   Availability and use were generally satisfactory. Some minor deficiencies noted in the use of P&IDs.

4. Comments on availability and candidate familiarization with plant design, procedure, T. S. changes and LERs:

   Availability and use were satisfactory.

5. Comments on interface effectiveness with plant training staff and plant operations staff during exam period.

   No comments.

6. Improvements noted in training programs as a result of prior operator licensing examinations/suggestions, etc:

   N/A

7. Personnel Present at Exit Meeting:

   NRC Personnel

   None

   NRC Contractor Personnel

   K. M. Henry, ORNL

Facility Personnel

D. A. Reid
L. W. Anson
E. Lindamood

8. Summary of NRC Comments made at exit interview:

   The examiner indicated that all candidates passed the oral examinations. Deficiencies in the areas of Nuclear Theory, Thermodynamics and offsite power sources were noted.

9. Summary of facility comments and commitments made at exit interview:

   The facility expressed their concern over the examination review procedure.

10. CHANGES MADE TO WRITTEN EXAM

    The facilities comments on the written exam are attached. Our resolution of those comments are attached.

Attachment:

Written Examination(s) and Answer Key(s) (SRO/RO)
Facility Comments
Resolution of Facility Comments

U. S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

Facility: Vermont-Yankee

Reactor Type: BWR

Date Administered: 6/ /84

Examiner: K. M. Henry

Applicant: ____________________
PRINT

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 & THERMODYNAMICS (25.0)

5-1 Fuel storage and fuel handling procedures are designed to prevent criticality accidents. What are three (3) parameters (in terms of reactor physics considerations) which affect whether a group of fuel rods can become a critical assembly (i.e., go critical)? Include for each parameter the change in that parameter necessary to reduce the risk of criticality. (3.0)

5-2 Figure 5.2 shows a schematic diagram of the recirc. pump seal system. For each of the following symptoms (a-e), select the most likely cause from the choices given below (1-5). (2.5)

Symptoms

a. No. 2 pressure would go toward zero and flow thru FS "A" would approach zero and alarm low at 0.25 gpm.

b. No. 2 seal pressure would drop dependent upon magnitude of failure. Leakage thru FS "β" would exceed 0.25 gpm, and alarm HI.

c. No. 2 seal pressure would approach No. 1 seal pressure. Leakage thru No. 2 orifice will go to appx. 1.1 gpm and FS "A" will alarm HI at $\geq$0.9 gpm.

d. Total leakage out of the seal assembly would approach 60 gpm as limited by the breakdown bushing. Both FS "A" and FS "B" would alarm high. Pressure in both seals would drop depending upon magnitude of failure. (No. 1 pressure might not drop significantly unless failure was large).

e. No. 2 seal pressure would approach No. 1 seal pressure. Controlled leakage would approach zero and alarm low at 0.25 gpm.

Causes

1. Failure of No. 1 seal only.
2. Failure of No. 2 seal only.
3. Failure of both seals.
4. Plugging of No.1 internal "RO".
5. Plugging of No. 2 internal "RO".

5-3 Step 4 of the immediate actions of OP 3124, Loss of Reactor Coolant Outside Primary Containment, states "Continuously monitor vessel water level using all available instrumentation." Explain why the operator is cautioned about rapid depressurization effects using reference and variable legs and the characteristics of water. (2.0)

(continued on next page)

5-4 a. Draw a curve (on Figure 5.4 a.) showing the approximate axial neutron flux profile you would expect in the reactor during normal full power operation. (Numerical values are not required). (0.5)

b. Draw a curve (on Fig. 5.4 b.) showing the approximate shape of the differential control rod worth over the axial length of the core (i.e., reactivity worth versus axial length). (Numerical values are not required). (0.5)

c. Draw a curve (on Fig. 5.4 c.) showing the approximate axial Xenon concentration profile following extended operation at full power (equilibrium Xenon conditions). (Numerical valves are not required.) (0.5)

d. The reactor scrams from the conditions stated in part c. Show on the same plot (Figure 5.4 c.) the expected axial xenon concentration profile at approximately eight (8) hours following shutdown. (Numerical values are not required). (0.5)

e. If the reactor is restarted from the conditions stated in part d, (eight hours after a scram from full power equilibrium xenon conditions), show on Figure 5-4 b. the affect this would have on the curve you have drawn for differential control rod worth and explain why. (0.5)

5-5 Use the supplied steam tables to determine the maximum temperature that could occur in the tailpipe monitoring system if a relief valve is leaking to atmospheric pressure. Describe HOW you arrived at your answer. (2.0)

5-6 Indicate whether each of the following conditions will result in an increase, decrease, or have no effect, on the fuel centerline temperature, assuming a constant power level (heat flux) and bulk coolant temperature: (2.0)

a. Increased coolant flow rate.
b. Helium gap distance (fuel pellet to clad) is reduced.
c. Boundary layer "dries out" forming a steam blanket.
d. Increased coolant inlet temperature.

5-7 A sudden closure of all MSIV's occurs from 100% power. Describe the response of the following parameters and reasons for parameter changes during the transient: (Assume no operator action & continue the discussion until appx. 20 seconds into the transient. Assume all other controls and safety system function properly). (3.0)

a. reactor power
b. reactor level (sensed)
c. reactor pressure
d. feedwater flow

(continued on next page)

5-8 State the immediate response (increase, decrease, or no change) of each of the following parameters to an increase in recirc. pump speed (from 65% power) and explain why for each:

a. critical power. (1.0)
b. indicated reactor level. (1.0)
c. natural circulation contribution to core flow. (1.0)

5-9 How does the void coefficient of reactivity change with increasing core age? Explain why. (2.0)

5-10 How does the immediate percentage of delayed neutrons being produced change (compared to total neutron production) following a step insertion of positive reactivity? Explain. (1.5)

5-11 Pertaining to the reactivity worth of control rods:

a. Which statement is correct. A rod will experience its greatest total worth when: 1) it is inserted individually while all others rods are withdrawn or 2) it is fully withdrawn and all other rods are inserted. (0.75)

b. Generally, the first rod (or first few rods) in a RWM group have more reactivity worth than the remaining rods in that group. Answer TRUE/FALSE and if FALSE state why. (0.75)

6. PLANT SYSTEMS: DESIGN, CONTROL & INSTRUMENTATION (25.0)

6-1 a. Describe the normal and alternate power supplies to the RPS system. (1.0)

b. What automatic actions will occur on switching from the normal to alternate power supply? List three (3). (1.5)

6-2 Describe three (3) conditions which will result in a Reactor Water Cleanup pump trip. (1.5)

6-3 a. The drywell equipment drains are equipped with a leakage rate alarm system. Describe two (2) methods by which this leakage rate alarm system can determine that excessive leakage is occurring (include applicable instrumentation in your description). (2.0)

b. What provides cooling for the drywell equipment drain sump? (0.5)

6-4 Certain indications (in the control room) should be received with an automatic scram (to verify that control rods are inserted). List four (4) of these indications and for each, describe the mechanism by which that indication is produced (i.e., limit switch on...,etc.) (3.0)

6-5 What interlocks/conditions must be met in order for the feedpump in standby to auto start following trip of one of the running feedpumps? (2.0)

6-6 On a return from loss of normal power, both RBCCW pumps will automatically start and Isolation Valve RCW-118 will close to isolate "non-essential" items serviced by the system. For each of the following, state whether they will be isolated (I) or not-isolated (N) on closure of RCW-118: (2.0)

a. Containment Air Compressor
b. Drywell Equipment Drain Sump Cooler
c. Non-regenerative Heat Exchanger
d. Fuel Pool Heat Exchangers
e. Reactor Building Equipment Drain Sump Coolers
f. RRU 1
g. CRD Pump Coolers
h. Cleanup Pump Coolers

6-7 An APRM Flow Transducer malfunctions. What indication(s) should an operator expect to observe for the following conditions:

a. Flow transducer produces a reading low by 3% (at 75% power)? (1.0)

b. Flow transducer fails downscale low from 75% power? (1.0)

6-8 What will be the response and/or indication(s) following the loss of AC power to a MSIV (i.e., loss of AC power to one MSIV, not a plant wide loss of AC power)? (1.0)

(continued on next page)

6-9 What are the specific actions (e.g.,interlocks, functions initiated, trip bypasses, etc.) which result from placing the mode switch to the following positions:

a. Shutdown (during operation)? List three (3). (1.5)
b. Startup/Hot Standby? List three (3). (1.5)

6-10 a. What are six (6) automatic valve and/or component actions which should occur on receipt of an initiation signal for the HPCI system? (3.0)

b. A HPCI turbine trip occurs due to high exhaust pressure. If the HPCI initiating signal is still present will the HPCI turbine restart automatically? Explain. (0.5)

6-11 Describe which rods an operator may select without receiving a select error under the following conditions: (2.0)

a. With all permissives (i.e.,no rod blocks imposed). List three (3).

b. If a withdraw block exists. List two (2).

7. PROCEDURES-NORMAL, ABNORMAL, EMERGENCY & RADIOLOGICAL CONTROL (25.0)

7-1 As a result of a loss of Instrument Air Header Pressure, the Feedwater flow control valve fails as is (subsequent to the reactor scram). Describe a method for controlling reactor water level under these conditions. (2.0)

7-2 An MPR failure to maximum output occurs during startup.

a. What effect will this have on reactor pressure control if no operator action is taken? (Include a description of any automatic plant responses to control pressure.) (0.50)

b. What are the three (3) immediate operator action steps required by O.P. 3104 on failure of MPR to maximum output? (1.5)

7-3 A caution in O.P. 3116 (Loss of Reactor Coolant) states that "automatic controls should not be placed in Manual mode unless: ......". Give two (2) conditions under which it is permissible to place automatic controls in the Manual mode. (2.0)

7-4 What are four (4) indications and/or component actions which should be observed following initiation of the Standby Liquid Control System (i.e, turning SLC keylock switch to System 1 position)? (2.0)

7-5 a. What are five (5) breakers which should be open following a generator trip (according to O.P. 3103, Loss of Normal AC Power)? (1.5)

b. Other than implementing scram procedures, what are four (4) immediate operator actions which should be taken following a total loss of normal station AC power? (2.0)

7-6 What are two (2) possible symptoms of a tube leak in a feedwater heater? (1.0)

(continued next page)

7-7 a. Define high radiation area. (1.0)

b. Requirements in Tech. Specs. regarding entry into high radiation areas state "any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following....". Give two (2) items which fulfill the requirements (other than requiring a RWP). (1.5)

7-8 Shutdown Operations Procedure 0112 lists four methods for venting the reactor vessel. List three methods (specific valve numbers are not required) and include required authorizations if applicable for each method. (3.0)

7-9 a. What are three (3) observations which may be made to verify the validity of an ATWS event (according to immediate actions of O.P. 3109)? (1.5)

b. What are three (3) other immediate actions which should be taken once the validity of the ATWS event has been verified? (1.5)

7-10 Describe the required operator action for the following conditions:

a. Two (2) IRM channels in one trip system become inoperable (Startup Mode, APRM's downscale). (0.5)

b. Two (2) APRM channels in each trip system become inoperable (run mode). Give two (2) possible courses of action. (1.0)

7-11 One of the immediate operator actions for the Reactor Scram Emergency Procedure states ...."If more than 2 adjacent control rods or a total of 22 or more control rods fail to insert below position 06, determine the cause of failure by observing....". List five (5) indications or observations which could be used to determine the cause of failure (to insert). (2.5)

8. ADMINISTRATIVE PROCEDURES, CONDITIONS & LIMITATIONS (25.0)

8-1 What three (3) conditions are necessary for Secondary Containment Integrity to exist? (2.0)

8-2 a. Who is authorized to initiate the Standby Liquid Control System injection into the reactor? (0.5)

b. Under what conditions is the initiation of Standby Liquid Control authorized. List three (3). (2.0)

8-3 What are four (4) conditions which would warrant evacuation of all personnel from the refueling floor? (2.0)

8-4 a. The fire brigade is composed of what personnel? (1.0)

b. For each of the personnel listed in part a., state where they should report upon hearing the announcement of a fire and their initial duties. (2.0)

8-5 What are four (4) Limiting Conditions for Operation for the temperature of the suppression pool water? For each include 1) applicable setpoints, 2) conditions under which the limits apply, and 3) actions required on exceeding the temperature setpoints. (3.0)

8-6 List four (4) actions which must be taken if a safety limit is exceeded. (3.0)

8-7 a. What communication requirements exist during refueling operations? (2.0)

b. Under what conditions is a SRO required to be on the refuel floor? (1.5)

8-8 a. Who may implement the temporary installation of a jumper or lifting of a lead? (0.5)

b. Whose authorization's are required prior to installing a jumper or lifting a lead (i.e., outline the necessary administrative requirements)? (2.0)

8-9 Explain the following:

a. MCPR (1.0)
b. OPERABLE (Tech. Specs. definition) (1.0)

(continued next page)

8-10 a. For implementation of the Vermont Yankee Emergency Plan, who has the responsibility and authority for classifying the level of emergency? (0.75)

b. In the absence of the person listed in part a, who assumes this responsibility? (0.75)

END

EQUATIONS/DATA SHEET

$A(x) = A_o e^{-\mu x}$

$1Ci = 3.7 \times 10^{10} d/s$

$\alpha_D \approx -1 \times 10^{-5} \frac{\Delta K}{K}/°F$

$\alpha_v \approx -1 \times 10^{-3} \frac{\Delta K}{K}/\%$ voids

$\alpha_M \approx -1.0 \times 10^{-4} \frac{\Delta K}{K}/\%°F$

$\alpha p \approx -4.5 \times 10^{-4} \frac{\Delta K}{K}/\%$ power

$I(t) = Io\ e^{-\lambda t}$

$T1/2 = \ln(2)/\lambda$

$Cp = (CP_{base})\ (Ks)\ (K_A)$

$\frac{V_2}{V^1} = \left(\frac{\Delta P_2}{\Delta p_1}\right)^{1/2}$

$\dot{Q} = MCp\ \Delta t$

$\Delta\rho = f \frac{L}{D} \frac{\rho V^2}{2g_c}$

$f = 64/Re$

$\rho = \frac{k(eff) - 1}{K(eff)}$

$\frac{1}{M} = \frac{CR1}{CR2} = \frac{1-K(eff)2}{1-K(eff)1}$

$\dot{Q} = M\Delta h$

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

$\lambda \approx 0.1$

$h_L = k\dot{m}V^2$

$P = P_o\ 10^{SUR(t)}$

$SUR = \frac{26.06}{\tau}$

$P = P_o\ e^{t/\tau}$

$A = \lambda N$

$M = 1/(1-k) = \frac{CR_1}{CR_0}$

$N(t) = No\ e^{-\lambda T}$

$\alpha_r \propto (L_f + L_s) \frac{(\phi rod)^2}{(\phi avg)}$

$n = \nu/(1 + d)$

$P = \Sigma\ \phi\ \nu/(3.7 \times 10^{10})$

$\tau = (\beta-\rho)/\lambda\rho$

$\tau = \bar{\ell}/\rho + (\beta-\rho)/\lambda\rho$

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

$v = v_f + xv_{fg}$

$H = xh_g + (1-x)\ h_f$

$S = xS_g + (1-x)\ S_f$

1 in = 2.54 cm

1 gal. = 3.785 liters

1 kg = 2.205 lb

$N = \rho A_O/A$

17.58 watts = 1 BTU/min
1psi = 6.895 Pa
1psi = 2.036 " H_g (@ 0C)
1psi = 27.68 " H^2O (@ 4C)

$\bar{\beta} \approx .0071$

$\bar{\ell} \approx 2 \times 10^{-5}$ sec

$\ell d \approx 12.5$

$RR = \Sigma_f \phi_{th}$

$SCR = \frac{S}{1-Keff}$

$\rho = \frac{\beta}{\lambda\tau + 1}$

Figure 5-4

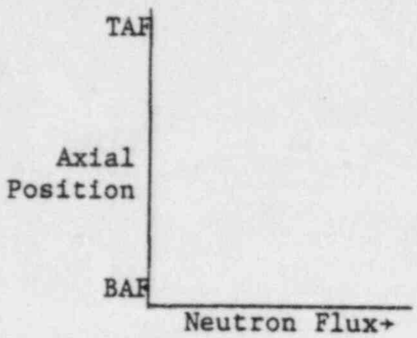

Fig. 5-4a.

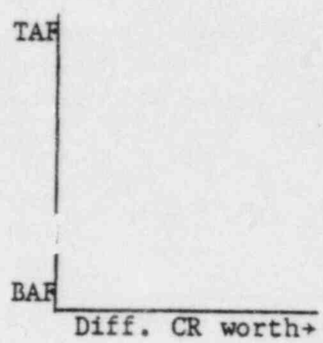

Fig. 5-4b

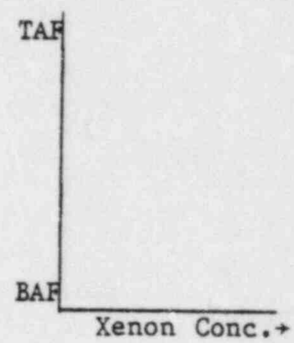

Fig. 5-4c

TAF - Top of Active Fuel
BAF - Bottom of Active Fuel

6/84

GREEN = PREREVIEW CORRECTIONS

5-1 (0.5 pts each for parameter, 0.5 pts each for change for 3 of below answers) 3 for (3.0)

1. Geometry-greater spacing
2. Fuel enrichment-decrease
3. Absorbers-Poisons-increase concentration
4. Moderation-Moderators present-reduce concentration or remove

REF.: General Theory

5-2 .5@ (2.5)

a. 4
b. 2
c. 1
d. 3
e. 5

REF.: VY Lesson Plans, Recirc. System p. 5

5-3 Do not rely on Yarway or analog indicators if erratic behavior, indicative of reference leg flashing, is observed. (2.0)

REF.: OP 3124 page 4

Flashing is caused when the fluid pressure becomes less than the saturation pressure for a specific temperature. (0.5)

The variable leg is kept full by connection to ~~the~~ AFTER the vessel. The reference leg is filled by condensing steam, ~~if~~ flashing (oscillating response), THE LEG may be slow refilling. (A level error.) (0.5)

REF.: VY Lesson Plan Rx Vessel Process Instru.

5-4 a.

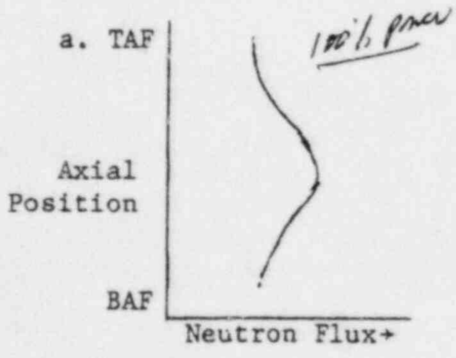

Fig. 5-4a.

100% Power

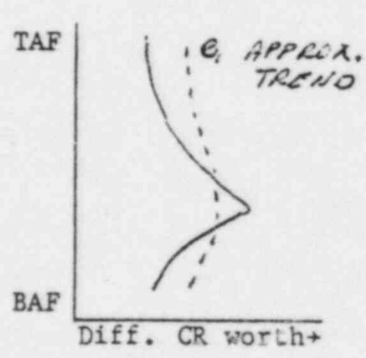

Fig. 5-4b

b.

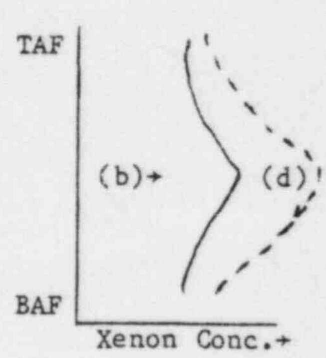

Fig. 5-4c

100% Full power
d - 8 hr after shutdown

(continued on next page)

e Shapes may vary due to core age

b. see Fig. 5-4b.

c. follows neutron flux profile, see Fig. 5-4c.

d. see Fig. 5-4c.

e. Increased rod worths where Xenon concentrations are lowest - at top and bottom, decreased rod worths where Xenon concentrations peak -- Xenon changes flux profile on which control rod worth is highly dependent.

REF.: General Theory

5-5 From the Mollier diagram take peak of saturation enthalpy to horizontally to 14.696 psia. Read temperature or superheat. (2..)

Credit will be given for any other logical method.

REF.: Steam Tables

5-6
a. decrease
b. decrease
c. increase
d. increase

REF.: VY Lesson Plans, Heat Transfer, General Theory

5-7
a. Initial spike upward due to sudden pressure increase followed by decrease to decay levels due to scram on MSIV closure. (MSIV CLOSURE SCRAM MAY KEEP SMALL)

b. Level drops initially due to void collaspe caused by scram and pressure increase, then begins to recover when SRV's open to control pressure and feedwater flow increases due to low sensed level.

c. Reactor pressure increases rapidly and lifts SRV's which then control pressure near SRV setpoints.

d. Feedwater flow will increase due to low sensed level (initially receives signal due to decreasing steam flow, prior to level decrease).

REF.: VY Reactor Theory p. 16-10, Figure 16-4
MSIV

(continued on next page)

5-8 a. Increase, increasing pump speed increases core flow. Increased core flow increases ability to remove h at so power at which onset of transition boiling (critical power) occurs will increase. (1.0)

b. Decreases initially as increased water flow is pumped out of downcomer. Feedwater flow increases eventually as a result of level drop, but lags recirc. speed increases. (1.0)

I QUESTION THIS ANSWER → ASK FOR REVIEW

c. Increase, power increases results in increased natural circulation contribution (increase in quality in core results in increased driving force for natural convection). (1.0)

REF.: General Theory

5-9 [redacted] (0.5 pt.) Becomes less negative ---
[redacted] pts.) 1.5 → as control rods are withdrawn there will be less loss of neutrons to control rods. Even if leakage from the bundle gets very high, little effect on total neutron population will be experienced since the neutrons will leak out of one bundle and into the adjacent bundle (i.e, neutrons which aren't moderated are not as likely to be absorbed by control rods so moderating ability loses some importance).

REF.: VY Rx Physics p. 12-~~6~~ 8 - says same as α tm (pg 5)

5-10 Total neutron population has increased, but the additional delayed neutron power level has not had adequate time to decay, so the immediate precentage of delayed neutrons decreases. (1.5)

REF: VY Reactor Physics p. 11-14

5-11 a. 2 ~~x~~ VY SAY EITHER 1 OR 2 BUT DOES NOT REFERENCE MATERIAL (0.75)

b. True (0.75)

→ No why! NOT REQUIRED

REF: Reactor Theory

6-1 a. Normal - RPS MG sets (1.0)
Alternate - 480V MCC 8B

(3 of below for full credit)

b. 1) half scram 3 for (1.5)
2) start SGTS
3) Trip reactor building ventilation
4) May isolate Off-Gas System valves OG-516A&B

REF.: O.P. 2134, p.1,2

6-2 (3 of below for full credit) (1.5)

1) full closure of CU-68
2) CU-15 and CU-18 not full open
3) bearing outlet cooler (RBCCW) temperature is $\geq$ 120°F
4) system flow rate at discharge of filter demineralizer drops below 50 gpm and filter demin. bypass valve (CU-74) is shut.

REF.: O.P. 2112, p.1

6-3 a. (2 numbered items below for full credit) (2.0)

There are sump level switches used for a leakage rate alarm system. 1) Leakage rate measurement is accomplished by measuring the time interval between two different level switches in the sump as the sump fills with water. Whenever the time interval decreases to a prescribed setpoint (indicating an increase in leakage rate), an alarm annunciates in the Control Room. 2) The alarm will also sound if a sump pump runs longer than a preset time interval.

b. RBCCW cooling coils

REF.: O.P. 2152 p.1

6-4 (4 of below for full credit, 0.5 pts. for indication, ~~0.5~~ 0.25 pts. for mechanism for each) (3.0)

1) Controls rods at full-in position — reedswitch
2) Green background lights-~~[illegible]~~ reed switch
3) White scram lights--limit switches on scram inlet & outlet valves
4) Amber accumulator lights--either water leak (level) or nitrogen pressure switch — local verification
5) Rod Drift lights on--timer initiated
KM added — 6) 8 RPS lights out -

REF.: O.P. 3100, p.2

(continued next page)

6-5 (5 of below for full credit) (2.0)

1) pump switch in AUTO
2) any condensate pump running
3) pump suction valve open
4) suction pressure greater than 200 psig.
5) running pump in tripped condition with its switch in AUTO
→ 6) Lube oil > 40#

REF.: O.P. 2172, p.1

6-6 (2.0)

a. N
b. N
c. I
d. I
e. N
f. N
g. N
h. I

REF.: O.P. 2182, p.1

6-7 a. No affect other than lowering the reference flow signal which lowers Upscale Trip (0.66w + 54). (1.0)

b. Rod block due to flow comparator picking up >10% difference. (1.0)

REF.: VY Lesson Plans, Nuc. Inst. p. 55

6-8 Only indication of a loss of AC power is the inability to test close a MSIV on the twice per week tests using the test pushbutton (MSIV's need both AC & DC power deenergized to close a valve in the normal mode of operation). (1.0)

REF.: O.P. 2113, p.4

(continued on next page)

6-9 a. (3 of below for full credit) (1.5)

1) Scram is initiated
2) Power to control rod drives is removed
3) Reactor protection system trip systems are de-energized

b. (3 of below for full credit) (1.5)

1) low turbine condenser vacuum trip bypassed ENABLED. when condenser vacuum <12 in. Hg and both turbine stop valves & bypass valves are closed.

2) low pressure MSIV closure trip bypassed

3) 10% closure MSIV closure trip bypassed

4) reactor protection system energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM neutron monitoring system operable.

REF.: VY Tech. Specs. p. 3,4

6-10 a. (6 of below for full credit) (3.0)

1. steam supply valve HPCI-14 opens
2. auto start to Auxiliary Oil Pump which supplies oil pressure to turbine stop & control valve, allowing turbine to start.
3. starts GSC exhaust blower
4. closes test bypass valve to SCT HPCI-21 and HPCI-24
5. opens pump suction from CST HPCI-17
6. opens pump discharge HPCI-19
7. closed steam line drain valves
8. closes condensate pump discharge to clean rad-waste valves HPCI-39

b) Yes, since shutdown of the turbine eliminates the condition which caused the shutdown (0.5)

REF.: VY Lesson Plans, HPCI-p. 14,16

(continued on next page)

6-11 (3 of below at 0.4 pts. each) (1.2)

a.) 1) Any rod in the sequence and group displayed

2) Any insert error

3) Any rod in next higher group if all rods in current latched groups are at their withdraw or alternate withdraw limit (excluding 2 insert errors).

4) Any rod in next lower group if all rods in current latch group are at their insert or alternate insert limit.

b. (2 of below @ 0.4 pts. each) (0.8)

If a withdraw block exists:

1) Only the withdraw error or,

2) Any rod with an insert error (any other rod will also result in an insert block when selected).

REF.: VY Lesson Plans, RWM p. 12

7-1 By throttling open HP Feedwater Heater Bypass Valve (FDW-5) and closing HP Feedwater Heater Inlet Valve (FDW-6A and FDW-6B) (2.0)
ALSO USE OP 2172. — USE 10% VALVE.

REF.: O.P. 2190 p. 8

7-2 a. This will cause the MPR to attempt to cause reactor pressure to decrease and bypass valves to open. (0.5)

b. (3 of below for full credit) (1.5)

1. Attempt to regain control of the MPR by placing the MPR control switch to raise.

2. If this is unsuccessful, manually shut the MSIV's to prevent a rapid cooldown and commence a reactor shutdown.

3. Carryout appropriate steps of O.P. 3100, Reactor Scram.

REF.: VY O.P. 3104, page 2

7-3 1) Misoperation in AUTOMATIC is confirmed by at least two (2) independent process indications or, (2.0)

2) core cooling is assured <u>and</u> subsequent procedural action states specifically to do otherwise.

REF.: O.P. 3116, p.2

7-4 (4 of below for full credit) (2.0)

1. One SLC pump will start. Red indicating light "ON".
2. Verify that cleanup system isolates automatically.
3. One squib valve will fire, amber monitor light goes out. "Sqb. Vlv. Con. Loss" annunciates.
4. SLC pump discharge pressure increases.
5. Red flow indicator light "ON" at $\geq$30 gpm.
6. SLC tank level decreases

REF.: O.P. 2114, p. 5

(continued next page)

7-5 a. (5 of below @ (0.3) pts. each) (1.5)

ACB's 12, 22, 3T1, 4T2, 1T, 81-1T, and Exciter field breaker

b. (4 of below for full credit) (2.0)

1. Verify both diesels start and supply power to Busses 3 & 4 at normal voltage.

2. Verify at least two SW pumps start automatically. Close SW-20 and verify transfer of station air compressor cooling water supply to alternate cooling.

3. Restart Station Air Compressors A&B from CRP 9-6 and secure all SJAE Valves.

4. Verify Turbine Emergency Bearing Oil Pump, Emergency Seal Oil Pump and Recirc. MG DC L.O. Pumps have auto started and that the vital MG Set has shifted to D.C. drive.

REF.: O.P. 3103

7-6 (2 of below for full credit) (1.0)

1) heater shell level
2) terminal temperature difference # (temp. of steam in minus temp. of feedwater out)
3) drain cooler approach # (temp of condensate out (condensed steam) minus temperature of feedwater in).

REF.: R.P. 2170, p.8

(continued on next page)

7-7 a. Area where person may receive >100 mrem in one hour (1.0)

b. (2 of below for full credit) (1.5)

1) A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2) A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. (Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.)

3) A H.P. qualified individual with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and who will perform periodic radiation surveillances.

REF.: VY Tech. Specs. p. 200,201

7-8 Methods for Venting the Reactor Vessel: 3 for (3.0)

a. Through the main steam lines, both second stage air ejectors, AOG dilution steam and both AOG trains with vacuum pumps used to assure direction of flow.
(NOTE: With iodine in the main condenser this flow path should be used to vent the main condenser by opening a 516 valve A or B.)

b. Through standby gas charcoal filters; (via RV-FCV-17 and RV-FCV-18) into the drywell equipment drain sump; then through the ventilation system to standby gas train A or B.

c. With authorization from Chemistry and Health Physics Supervisor, vent to the main condenser via the main steam lines or main steam line drains.

d. With authorization from Chemistry and Health Physics Supervisor by removal of the reactor vessel head.

REF.: VY OP. 0112

(continued next page)

7-9 a. 3 for (1.5)

1) Auto scram A and Auto Scram B annunciators, and
2) Control rod position, and
3) Reactor flux and pressure indicators, or
4) Any reactor protection signal which should cause a scram, but does not.

b. (3 of below @ 0.5 pts. each) 3 for (1.5)

1) Verify recirc. MG field breakers have tripped or immediately trip them by depressing the RPT/ARI trip pushbutton, BX and DX and/or AX and CX.

2) Attempt to manually scram the reactor.

3) Initiate SLC System injection into the reactor vessel if the control rods cannot be scrammed.

4) If an MSIV isolation has occurred, attempt to reset the isolation as quickly as possible and open sufficient bypass valves to control reactor pressure.

REF.: VY OP. 3109

7-10 a) Trip the system (causing a half-scram) (0.5)

b) 1) Commence insertion of control rods and complete the insertion within four (4) hours, <u>or</u> (1.0)

2) Reduce power to IRM range and place mode switch in STARTUP within eight (8) hours.

REF.: VY O.P. 0101 p.3, precautions

7-11 (5 of below for full credit) (2.5)

1. Scram Valve Indication
2. Scram Instrument Volume Water Level Indicators
3. Scram Instrument Volume Level Annunciators
4. Scram pilot air header indicator and annunciator
5. Auto Scram Annunciators (Both A and B)
6. Manual Scram Annunciators (Both A and B)
7. Reactor Protection Bus group Solenoid lights (CRP 9-15 and 9-17).

REF.: O.P. 3100 p. 3

8-1 (3 of below at 0.66 pts. each) (2.0)

1) At least one door in each access opening is closed
2) SBGT system is operable
3) All reactor building automatic ventilation system isolation valves are operable or are secured in the isolated position.

ALSO BLDG INTACT

REF.: VY Tech. Specs. p. 3

8-2 a. Shift Supervisor OR DESIGNATED TO ASSUMING (ACTING) PERSON (0.5)

b. (3 of below for full credit) (2.0)

1. If, at any time, it becomes apparent there has been a failure of more than two (2) control rods such that they cannot be scrammed or manually inserted into the core.

2. Normal rod exercise indicates rods stuck and not able to scram such that it is apparent that Shutdown Margin Requirements of Technical Specifications cannot be met.

3. The Reactor Engineer determines the Shutdown Margin Requirements of Technical Specifications cannot be met.

4. Any time reactor water level cannot be maintained above the top of the core or torus water temperature cannot be maintained below the torus scram limit (110°F), and the control rod drive system is unable to maintain the reactor subcritical.

5. ATWS

REF.: O.P. 2114, p. 1

8-3 (4 of below for full credit) (2.0)

1) High rad. alarms on the refueling floor area rad. monitors
2) Any incident that results in a ruptured fuel assembly
3) High activity alarms from airborne activity devices in the Reactor Building
4) Sounding of the plant evacuation alarm
5) Decreasing reactor cavity water level
6) Upon instruction of the Control Room for any reason
7) Upon instruction of the HP representative on the refuel floor

REF.: A.P. 1000, p. 2

(continued next page)

(1.0)

8-4 a. 1-STA
2-AO's
1-HP Tech
1-Security Guard
AP - 0036

(2.0)

b. STA-scene of fire, assume duties of Brigade Commander and size up situation
AO's-Brigade Room, don protective equipment, pick up breathing apparatus and report to scene. Also bring equipment for Brigade Commander.
HP-Scene of fire to assist brigade commander
Security Guard-Scene of fire to assist brigade commander
REF: VY OP 3020

8-5 (4 of below for full credit) (3.0)

a. Maximum Water Temperature during normal operation 90°F.

b. Maximum Water Temperature during any test operation which adds heat to the suppression pool 100°F and shall not be above 90°F for more than 24 hours.

c. If Torus Water Temperature exceeds 110°F, initiate an immediate scram of the reactor. Power operation shall not be resumed until the pool temperature is reduced below 90°F.

d. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the torus water temp. exceeds 120°F.

e. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant pressure >200 psig, an external visual examination of the suppression chamber shall be conducted prior to resuming power operation.

REF.: Tech. Specs. p. 126

(continued on next page)

8-6 (4 of below for full credit) (3.0)

1. reactor shall be shutdown immediately
2. immediate report to Manager of Operations (Vice President of oper.)
3. Analysis of circumstances leading up to and resulting from the situation together with recommendations by the PORC shall be prepared and submitted to Manager of Operations and Chairman of the Nuclear Safety Audit and Review Committee
4. reactor operation shall not be resumed until authorized by USNRC
5. Notification of NRC within 1 hour

REF.: Tech Specs. P. 199 & A.P. 0010

8-7 a. (3 numbered items below for full credit) (2.0)

1) The Control Room will be notified of all component movement by the refueling crew and 2) the refueling crew will be notified of any intended rod motion by the operators in the Control Room. 3) Whenever only two communication channels exists between the refueling floor and the Control Room, and one fails, the other channel shall be used to notify all personnel to halt refueling operations. The Shift Supervisor may reinitiate the refueling operation after two communication channels are reestablished and checked.

USED only this to grade →

b. 1) During movement of irradiated fuel and,
2) during any core alteration which involves reactivity manipulation (1.5)

REF.: VY A.P. 1000, p.2,5

8-8 a) Any person who has determined the need (0.5)
Consider possible responses due to implement

b) (4 of numbered items below at 0.5 pts. each) (2.0)

1) ESS review and approval (if safety evaluation determined necessary by originating department supervisor.)

2) Originating department supervisor approval.

3) Operations Supervisors review and approval.

4) Shift supervisor approval to implement.

REF.: VY A.P. 0020 p.4,5

8-9 a. The ratio of that power in a fuel assembly which is calculated to cause some point in that assembly to experience boiling transition (as calculated by application of the GEXL correlation to the actual assembly operating power).

b. Operable: A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing it specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable or performing their related support function(s).

REF.: Tech. Spec. p. 2

8-10 a. duty Shift Supervisor
b. duty Supervisory Control Room Operator
REF: VY A.P. 3125 p.2