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

EXAMINATION REPORT NO. 50-309/85-13 (OL)

FACILITY DOCKET NO. 50-309

FACILITY LICENSE NO. DPR-36

LICENSEE: Maine Yankee Atomic Power Company
83 Edison Drive
Augusta, Maine 04336

FACILITY: Maine Yankee

EXAMINATION DATES: May 14 to 16, 1985

CHIEF EXAMINER: N F Dudley — N. F Dudley, Lead Reactor Examiner — 6-21-85 Date

REVIEWED BY: RM Keller — R. M. Keller, Chief, Projects Section 1C — 6/27/85 Date

APPROVED BY: HB Kister — H. B. Kister, Chief, Projects Branch No. 1 — 6/26/85 Date

SUMMARY: Four candidates were examined and four SRO licenses were issued. The requalification program was inspected and a facility administered simulator examination was observed. Some licensed operators had not been attending all requalification lectures. Other program deficiencies have been identified by the training department and are being addressed in a revised training procedure being developed for submittal to INPO.

REPORT DETAILS

TYPE OF EXAMS: Replacement

EXAM RESULTS:

	SRO Pass/Fail
Written Exam	4/0
Oral Exam	4/0
Simulator Exam	4/0
Overall	4/0

CHIEF EXAMINER AT SITE: N. Dudley

OTHER EXAMINERS: G. Streier, EG&G

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

   All candidates showed weaknesses in evaluating plant conditions after a reactor trip and taking the corresponding remedial actions described in EOP-70-0 Emergency Shutdown From Power.

2. Personnel Present at Exit Interview:

   NRC Personnel

   N. Dudley, Lead Reactor Engineer (Examiner)
   C. Holden, Senior Resident Inspector

   Facility Personnel

   J. Garrity, Plant Manager
   R. Nelson, Nuclear Safety Section Head
   S. Nichols, Licensing
   A. Shean, Manager, Training
   R. Bickford, Operations Training Section Head
   M. Evringham, Operations Training Supervisor

3. Summary of NRC Comments made at exit interview:

   All candidates were clear passes on the operational and simulator portion of the examination. No generic weaknesses were noted and individual weaknesses have been discussed with the training department.

   The requalification program is well organized and documented. Some licensed operators are not actively participating in the requalification program as indicated by attendance at lectures and timely completion of quizzes and required reading. The annual examination adequately identifies individual and generic weaknesses. The use of the operations department to evaluate the crew and individual performance on the annual simulator requalification examination is noted as a good practice. Other weaknesses in the requalification program have been identified by the training department and are being corrected.

   The NRC stated that completion of all weekly quizzes and the annual examination in a two week period was not an acceptable means of participation in the requalification program.

   The NRC assured the facility that the resolution of comments made during the examination review would be incorporated into future examination reports, however, the format of the information might change.

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

The facility requested clarification on what was considered timely completion of block quizzes. The facility requested continued feedback on resolution of comments made during the examination review. The facility noted that it would be beneficial to have the author of the examination present during the examination review.

5. Changes made to written exam during examination review:

Facility comments were taken into consideration during the grading of the examination. However, not all facility comments resulted in changes to the examination answer key.

Answer No.	Change	Reason
5.3A	Add "immediate prompt drop to $P \sim \frac{\beta}{\beta - P} Po$ or about 5% power".	Recognizes theoretical prompt drop in addition to concept of power level attributable to delayed neutrons.
5.8	Delete "Tave remained constant for a given power (Q)."	Statement not required to answer question.
6.5C	Delete.	The correct answer was not provided as a choice.
6.7A	Add "(computer alarm)".	Recognizes thermocouples can produce a computer alarm.
6.7B	Add "Other pressure gauges used to measure primary pressure are acceptable".	Primary pressure can be measured by more than just the pressurizer pressure gauges.
6.7C	Add "Incore flux".	During steady state operations incore flux can be used to measure core power.
6.8C	Add "Steam dumps will open due to SG pressure".	Steam dumps do not depend on temperature signal from RRS.
6.10 and 8.10	Define partial credit.	More definite explanation of how question was graded.

7.1	Change to "Drive group 5A and 5B to maintain S/O below the upper control line until rods reach PDIL (>50%) or 90 steps (<50%). This drives flux toward bottom of core and reduces S/O."	Corresponds to discussion section of the abnormal procedure.
7.4B	Add "HP condenser receiver; Atm. steam dumps; Air ejectors (monitored release)".	Provides other possible release paths.
7.10A	Change "indirect (or pull tag and direct)" to "direct through BAM 36 or 37".	Boration will be done using direct path when two CEA's fail to insert.
7.13C	Add "Verify SG is being fed by main feed".	Verification that SGs are being fed by mainfeed should be completed before securing auxiliary feed pumps.
8.7B	Change to "No tag".	Conforms to present procedure.
8.14	Change "D" to "B".	Provides correct definition of criticality in accordance with Technical Specifications.

6. Inspection of Requalification Program

a. Scope

A review of the requalification training program for licensed operators was conducted. The review involved verification of the program's conformance to the requirements of 10 CFR 55 and included an audit of personnel training records, formal lesson plans, training schedules, requalification block quizzes, and an annual requalification examination (June 1984). A facility administered requalification simulator examination was observed.

b Findings

Some licensed personnel have not attended all requalification lectures or completed all the quizzes which are given at the end of each requalification block. One licensed individual has not attended any lectures, completed any quizzes, or signed for completing the required reading assignments since August of 1984. The periodic memos sent by the training department to Department Heads detailing licensed personnel who were behind in the requalification program have not been effective in bringing all licensed personnel up-to-date.

In a telephone conversation on June 19, 1985, the facility stated that all licensed operators were up-to-date on requalification lectures and required reading assignments.

There were three sets of annual requalification examinations given in June 1984. The first two examinations were separate examinations with less than 20% similar questions. The majority of questions were short answer essay type with less than 20% true-false multiple choice type questions. The questions were well written and required a detailed knowledge of the Maine Yankee facility to answer. The third examination was administered to a single operator and was a combination of the other two examinations. The examinations were adequate to meet the requirements of 10 CFR 55, Appendix A, 4.a.

Six operators failed at least a portion of the annual requalification examination. Appropriate actions were taken in accordance with the training programs Procedure 18-20-1. One of the licensed personnel who failed the examination sat on the Training and Qualification Review Board which reviewed actions to be taken concerning examination failures. This selection of board membership was inappropriate even though the person in question was eventually removed from all licensed duties due to medical problems. The enrollment of all licensed personnel who failed the annual examination and their subsequent reevaluation was adequate to meet the requirements of 10 CFR 55, Appendix A, 4.e.

Weekly quizzes given at the end of each requalification block are adequate to evaluate licensed operator's knowledge of subjects covered during the week. The same quiz is given to each of the six crews for any given requalification block and no attempt is made to maintain security of the quiz over the six week period in which the block is taught.

The annual simulator examination which is administered for 2 hours to a crew of four licensed operators was adequate to meet the requirement of 10 CFR 55, Appendix A, 4.c. The use of senior operations personnel to conduct the evaluation provided operational input into the evaluation process.

In the past, the evaluation of the competency of licensed operators was performed by contracted simulator instructors during non-plant specific simulator training conducted to meet the requirements of 10 CFR 55, Appendix A, 3.a. This inappropriate practice was discontinued when the plant specific simulator was installed.

Nine licensed persons are assigned to the training department. This provides an adequate number of qualified instructors to conduct the requalification training program.

The Licensed Operator and Operations Instructor Training Programs procedure 18-20-1 provides flexibility in the areas of evaluation of proficiency in subject areas covered during requalification block lectures, the removal of licensed operators from licensed duties due to annual examination failures, and exemption of instructors from portions of the requalification program. The flexibilities incorporated in the procedure have not been abused.

Other deficiencies in the requalification program have been identified by the training department and are being addressed in a submittal for INPO accreditation. These deficiencies include lesson plans which are not formally reviewed or approved, instructors signing certification for themselves for completion of required simulator manipulations, and evaluation of instructor classroom techniques.

7. Summary

The administration of the requalification program conforms to the requirements in 10 CFR 55, Appendix A. Some licensee personnel had not maintained active involvement in the requalification program. The annual examination provides an indication of generic and individual weaknesses. Some inappropriate evaluation practices have been corrected by integrating the newly acquired plant specific simulator into the requalification program. Other inappropriate practices have been identified by the facility and are being corrected.

Attachments:
1. Written Examination and Answer Key (SRO)
2. Facility Comments on Written Examinations Made After Exam Review

MASTER

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

Facility:	Maine Yankee
Reactor Type:	PWR
Date Administered:	May 14, 1985
Examiner:	K. Ferlic, N. Dudley
Candidate:	

INSTRUCTIONS TO CANDIDATE:

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

Category Value	% of Total	Candidate's Score	% of Category Value	Category
25	25	M. Swartz		5. Theory of Nuclear Power Plant Operation, Fluids and Thermodynamics
25	25	D. Seule		6. Plant Systems Design, Control, and Instrumentation
25	25	J. Kirsch		7. Procedures - Normal, Abnormal, Emergency, and Radiological Control
25	25	N. Everhart		8. Administrative Procedures, Conditions, and Limitations
100				Totals

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

Candidate's Signature

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

1. The reactor is at full power early in core life. All power range safety channel meters read exactly 100 % power. The upper detector of one channel fails. The switch inside the power range safety channel drawer on the failed channel is adjusted to block the signal of the upper detector resulting in the doubling of the lower detector to provide total channel power.

   A. If no further adjustments are made will the total power reading on this channel be greater than, less than or equal to 100 % power. Explain the basis of your answer. (1.0 pt)

   B. If the above were to occur late in core life would there be any difference in the power reading on the failed channel compared to the reading on the channel at beginning of core life? Explain. (1.0 pt)

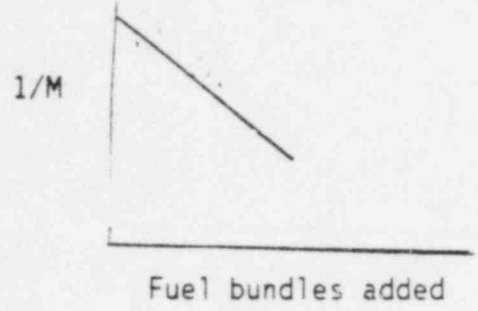
ANSWER

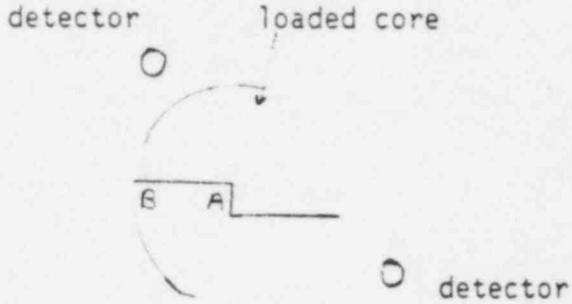
A. Greater. At BOL the core flux is shifted towards the bottom of the core because of the colder water at the core bottom. Therefore power at the bottom of the core is higher than at the top. The lower detector reads greater than 50% power. By doubling the lower detector the final total power would be greater than 100%.

B. Over core life the fuel is slightly more depleted in the bottom of core and the flux shifts up splitting evenly between top and bottom. Consequently if this event occurred late in core life the bottom detector would be reading approximately 50% power. Doubling the the bottom detector would result a total channel reading of about 100% power.

REFERENCE: Excore Instrumentation p-13

2. During a fuel load the following 1/M plot and core load has been obtained.





Two identical bundles are to be placed in the core; one at position A and one at position B. Which position, A or B, will result in a data point farthest below the 1/M plot (closest to zero)? Explain the basis of your answer. (2.0 pts)

ANSWER

1/M So/S, As the count rate S increases 1/M decreases. (0.7 pt) To get the 1/M data point significantly below the 1/M plot one needs a large increase in the count rate (or a large multiplication). (0.6 pt) Position A will result in the higher multiplication since less neutrons will be lost to leakage. (0.7 pt)

REFERENCE: Fuel Load, Section 12.3 Reactor Operations Book

3. A reactor scram occurs from full power. Immediately after the scram reactor power is observed on the wide range log channels and core delta T ($Q = m\ c\ \Delta T$).

   A. How far, in terms of reactor power, will the wide range log channel indication initially drop after the reactor scram? Explain the basis for your answer. (1.0 pts)

   B. Based on core delta T measurements what will reactor power be immediately after the scram? Explain the basis for your answer. (1.0 pts)

   C. One hour after the scram, which indication is representative of the real reactor power and why? (1.0 pts)

ANSWER

IMMEDIATE PROMPT DROP TO $P \approx \frac{\beta}{\beta-\rho} P_0 \Rightarrow 5\%$ pow [0

A. Excore Instrumentation detects neutrons. ✓ Following a scram, the delayed neutron fraction is approximately 0.007 (0.7%) of the previous neutron population. Power would therefore drop to about 0.7% of full power. [0.5]

B. Decay heat from fission fragments results in an energy addition equivalent to approximately 6-7% of full power. Core delta T will reflect this energy addition and removal.

C. Neutron population decays to the source range. Fission product decay continues and will be about 1% after one hour. Core delta T will be indicative of the correct reactor power.

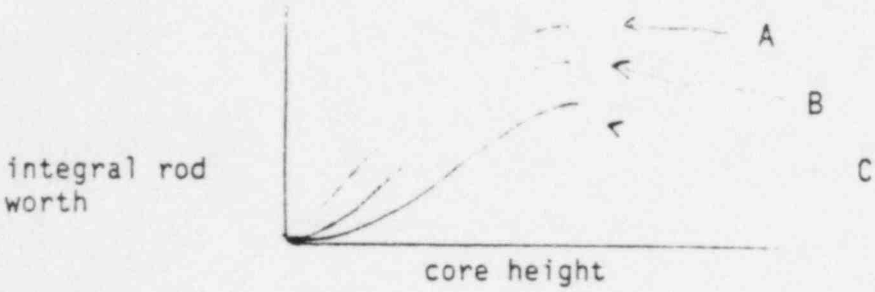
REFERENCE

A. Reactor Operations, Chapter 5, Effective Delayed Neutrons

B. Reactor Operations, Chapter 14, Shutdown Cooling

C. Reactor Operations, Chapters 5 and 14

4. The following is a plot of three integral rod worth curves for control rods in different core locations.



A. Which curve A, B, or C, would be representative of the integral rod worth for a control rod at the center of the core and which curve is representative of a control rod near the edge of the core. Explain the basis of your answer (1.0 pt).

B. Assuming the above figure is drawn for BOL conditions, will the integral rod worth curve for the rod in the center of the core shift up or down as the core ages? Explain. (1.0 pt)

ANSWER

A. Curve A: center of core (0.3 pt)
Curve C: edge of core (0.3 pt)
More neutrons are in the core center, therefore the center rod has the higher intergral rod worth (0.4 pt)

B. As the core ages, neutron population shifts outward. (0.5 pt) The rod in the center of the core will be worth less than at BOL. Therefore the curve will shift down. (0.5 pt)

REFERENCE

A. Reactor Operations p 7.5-4
B. Reactor Operations p 11.4-3

5. A reactor trip occurs from 100% power after an extended full power operation. The reactor is brought critical within one half hour. What rod motion is necessary to maintain the reactor power at 10E-4% for the next 16 hours? (1.5 pts)

ANSWER

Xenon dominates rod motion. Rods are withdrawn as xenon builds in until maximum (approximately 9.5 hrs). (0.75 pt) After xenon peaks the rods will then be driven in as xenon decays away. (0.75 pt)

REFERENCE: Reactor Operation p 15-19

6. A. Explain which phenomenon, over moderation or under moderation, is responsible for a positive moderator temperature coefficient. Explain the basis for your answer. (1.0 pt)

B. Under what plant condition can/does an over moderation condition exist. (1.0 pt)

ANSWER

A. Overmoderation: High density with high concentration of soluble poisons causes neutrons to rapidly slow and travel shorter distances before being absorbed. As temperature increases, neutron travel farther and more become available to cause fission (less poison capture) consequently, positive reactivity is added.

B. Beginning of life, high boron concentration.

REFERENCE

A. Reactor Operations, Chapter 9.6
B. Reactor Operations, p 9.6-3

7. According to procedure 1-2, Reactor Startup, the "point of adding heat" for the Maine Yankee reactor is 10E-2% power.

   A. What does the "point of adding heat" mean? (0.75 pt)

   B. If the reactor is critical, compare how the reactor power would respond to a rod withdrawal for initial power levels above and below the point of adding heat. (1.75 pts)

ANSWER

A. Power level where the power generated equals the heat loses to ambient.

OR

Point at which the reactor is producing enough heat to cause a temperature increase in the coolant. (0.75 pt)

B. Below the point of adding heat power will increase at a constant rate until the point of adding heat is reached. (0.75 pt)

Above the point of adding heat, as power increases the temperature increases. (0.5 pt) Negative reactivity is inserted because of the higher fuel and moderator temperature. The power will continue to rise until the increase in temperature is sufficient to allow the fuel and moderator coefficients to insert enough negative reactivity to balance the initial rod withdrawal. (0.5 pt)

REFERENCE

A. Reactor Operations p 13.5-2, 13.5-4
B. Reactor Operations p 13.5-2, 13.5-3, 13.5-4

8. Following a LOCA hot leg recirculation is required after 20 to 24 hours to prevent increases boron concentrations from fouling the heat transfer surfaces of the fuel. Using the appropriate heat transfer equations, explain why boron fouling of the heat tranfer surfaces is undesired. (1.5 pt)

ANSWER

Q=UA (Tfuel - Tave) U decreases since the heat transfer ability is reduced due to an additional layer of material. (0.75 pt) ~~Tave remains constant for a given power (Q)~~ Since A is constant, Tfuel - Tave must increase to balance the equation. Since Tave is constant, Tfuel must rise. If Tfuel increases too much, fuel damage may occur. (0.75 pt)

REFERENCE: Plant Performance, APP B; Heat Transfer

9. When starting a centrifugal radial pump is more current drawn when the discharge valve is open or shut? Explain the basis for your answer. (1.5 pt)

ANSWER

More energy is required in starting a centrifugal pump with the discharge valve open. (0.5 pt)

Valve closed: no work is done in pumping the water, energy is only required to turn the impellor. (0.5 pt)

Valve open: work is done in both pumping the fluid and bringing the pump up to speed. (0.5 pt)

REFERENCE: Plant Performance, p 6.4-5

10. Doubling a centrifugal pump's speed will: (0.5 pt)

A. Increase the flow by 2 and head by 8
B. Increase the flow by 8 and head by 2
C. Increase the head by 2 and power by 4
D. Increase the head by 4 and power by 8
E. Increase the flow by 2 and power by 4

ANSWER: D

$S \propto m$, $h \propto m^2$, $p \propto s^3$

REFERENCE: Pump laws

11. Explain why a high pressurizer pressure trip may result if the reactor is generating more power than the secondary system is removing. (1.5 pts)

ANSWER

When reactor power is greater than secondary power, heat is added to the primary and the coolant expands into the pressurizer compressing the steam bubble. (0.75 pt) As the steam bubble is compressed the steam pressure rises. If the pressure rise is great enough a reactor trip will occur. (0.75 pt)

REFERENCE: Reactor Protection System, p 4

12. A. List two causes of waterhammer. (0.8 pt)

B. Give two examples of how waterhammer can be minimized by the operator. (1.2 pts)

ANSWER

A. valve operation; opening or closing
pump starting
pump stopping
auto oscillation of valves

B. slowly opening of valves between voided and full systems
proper venting of components
ensure adequate level on tanks in systems where the tanks provide supply or surge function
proper use of steam traps and vents
proper sequencing of valves in pressurized system high or low, vice low to high
follow the operating procedure for the system in question (any 2 @ 0.6 pt each)

REFERENCE: Plant Performance p 2.1-4

13. At what pressure would the formation of a bubble in the vessel be expected if a rapid decrease in primary pressure occurred while operating at 100% power. Assume no automatic safeguards actions. State all assumptions and basis for your answer. (1.0 pt)

ANSWER

Pressurizer; T ~ 650°F, P = 2235 psia

Vessel; Tave ~ 574°F, T hot 600°F, P at T hot = 1543 psia

Pressure would have to drop to approximately 1543 psia

REFERENCE: Steam tables and plant parameters

14. A. Why does nucleate boiling heat transfer remove more heat than non-boiling heat transfer? (1.0 pt)

B. Why does film boiling remove less heat than nucleate boiling (1.0 pt)

ANSWER

A. Nucleate boiling creates turbulent flow which promotes more mixing. Coolant picks up latent heat of vaporization and carries it to cooler parts of the channel.

B. In film boiling a film of steam coats the clad surface and form an insulating layer which drastically reduces the heat transfer coefficient.

REFERENCE: Plant Performance, Chapter 3, p 3.3-2, 3.3-3

6. PLANT SYSTEMS DESIGN, CONTROL, INSTRUMENTATION

1. Arrange the following events/actions in order of increasing pressurizer pressure. (2.0 pt)

   A. Backup heaters on
   B. TM/LP trip
   C. Technical Specification Safety Limit
   D. Third safety valve lifts
   E. First safety valve lifts
   F. SIAS initiation
   G. Spray valve opens
   H. PORV's opens

ANSWER

F SAIS initiation
B TM/LP trip
A Back up heaters on
G Spray vavle opens
H PORV's opens
E First Safety opens
D Third Safety opens
C Tech Spec Safety Limit

REFERENCE

Tech Data Book Fig 1.2.1
Tech Spec p 2.3-1

2. Concerning the allowable symmetric offset control band:

   A. By maintaining the limits on the symmetric offset control band what core design parameter will be maintained? (0.75 pt)

   B. What ultimate potential consequences could result if the symmetric offset limits are exceeded for an extended period of time? (0.75 pt)

ANSWER

A. Maintain fuel design limit on linear heat generation rate (0.75 pt)

B. Maintain the integrity of the fuel cladding (0.75 pt)

REFERENCE: TS 2.1 and 2.2 pages 2.1-2 and 2.2-1

3. Concerning the Main Control Board (MCB) charging pump four position control switch, in which one of the following cases may the charging pump be started? (0.4 pt)

   A. The MCB switch is turned to "start" and the switchgear key lock is in "emergency".

   B. The MCB switch is in "pull to lock", local control is assumed with the switchgear key lock switch in "normal".

   C. The MCB switch is in "auto" and a manual containment spray ac uation occurs (with no safety injection actuation signal).

   D. The MCB switch is in the "pull to lock" and local control .s assumed with the switchgear key lock in "emergency".

ANSWER: D (0.4 pt)

REFERENCE: ECCS P 51,52,53

4. The following detectors are used in radiation measurements. Match the appropriate detectors to the statement. A detector may be used for more than one answer (2.1 pt).

Detector

1) Geiger Muller
2) BF3 Proportional
3) Compensated Ion Chamber
4) Uncompensated Ion Chamber
5) Scintillation Detector
6) Gas Avalanche Ion Chamber
7) Fission Chamber
8) Rhodium Detector
9) Beta emission detector

A. ______ Used in Maine Yankee source range instrumentation

B. ______ A higher voltage applied to an ion chamber will produce this detector

C. ______ This detector produces light pulses in proportion to the ionizing event

D. ______ Used in the dual linear power range instrumentation

E. ______ In this detector, thermal neutrons are readily absorbed by the nuclei of the detector, subsequent beta emissions cause a current flow

F. ______ Provides start up rate indication over the entire range of reactor power

G. ______ Fixed incore instrumentation

ANSWER

A. Fission chamber
B. Geiger Muller
C. Scintillation detector
D. Uncompensated ion chamber
E. Rhodium detector
F. Fission chamber
G. Rhodium detector - (0.3 pt each)

REFERENCE

Excore Instrumentation p 2,3
Incore Instrumentation p 1

5. Concerning the Reactor Protection System.

A. Explain the main difference in operation between a shunt trip coil and an undervoltage coil in opening a reactor trip breaker. Which type is fail-safe and why? (1.5 pt)

B. Using a one line diagram explain how the Maine Yankee reactor trip breaker arrangement prevents a partial reactor trip on the loss of a motor-generator set. (1.5 pt)

~~C. Which accident listed below will cause the non trippable control rods to drop into the core? (0.5 pt)~~

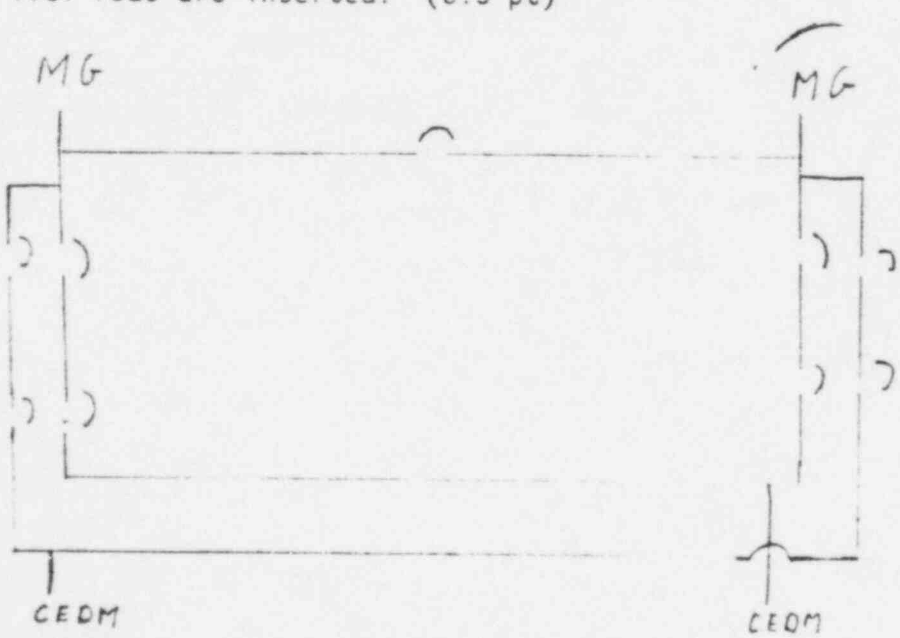
~~1) loss of coolant~~
~~2) steam generator tupe rupture~~
~~3) loss of all offsite power~~
~~4) steamline break~~
~~5) main generator hydrogen explosion~~

ANSWER

A. Undervoltage trip coils mechanically open the breaker when deenergized, (0.5 pt) shunt trip coils mechanically open the breaker when enegized. (0.5 pt)

Fail-safe: the UV open on loss of power resulting in a safe condition i.e. rods are inserted. (0.5 pt)

B.



If the circled train is lost, the CEDM are still supplied by the other MG set. Both MG sets are full capacity MG sets.

~~C. 3 (0.5 pt)~~

REFERENCE: RPS p 30, figure NS-12-13 and CEA

6. The following postulated accidents require Emergency Core Cooling System response

- intermediate break LOCA
- large break LOCA
- steam generator tube failure
- CEA ejection incident
- excess load incident
- steam line rupture

A. Which of the above accidents may result in a containment isolation signal in addition to a safety injection actuation signal (SIAS)? Explain. (1.0 pt)

B. If a SIAS occurs on a high containment pressure will a containment spray actuation signal also occur? Explain. (1.0 pt)

C. Is containment spray needed for the ECCS to fulfill its design purpose? Explain. (1.0 pt)

ANSWER

A. All containment isolation valves close on SIAS except those essential for operation (they close on CIS only). CIS occurs only if manually initiated or high containment pressure (> 5 psig) (0.6 pt)

CIS will probably occur on:

- large break LOCA (0.1 pt)
- intermediate break LOCA (0.1 pt)
- CEA ejection (0.1 pt)
- steam line rupture if in containment (0.1 pt)

B. No; (0.3 pt) SIAS high containment pressure > 5 psig
CSAS high containment pressue > 20 psig with SIAS

therefore CSAS only if > 20 psig (0.7 psig)

C. Yes; (0.3 pt) containment spray pump provides both the net positive suction head for the HPSI pump and cooled water for recirculation. (0.7 pt)

(Design purposes are: - inject borated water to cool core on LOCA,
- inject borated water to provide negative reactivity on steam line break,
- provide post accident core cooling)

REFERENCES

A. ECCS p 10 , simulator text

B. ECCS p 12

C. ECCS p 1,7

7. Most of the indications and controls which are used to monitor and control the operations of the reactor core and warn the operators of potential problems are not directly located within the reactor core.

   A. Identify the two types of detectors located in the core, whether each provides control functions and/or indications, and what each measures (1.2 pts).

   B. Identify two independent methods for measuring core pressure. (A second example of measuring the same physical parameter is unacceptable.) (0.8 pt)

   C. Identify two independent methods of measuring core power (0.8 pt).

   D. What method is available for identifying fuel element damage? (0.7 pt)

ANSWER

A. Thermocouples (0.2 pt), indication only (0.2 pt), core temperature (0.2 pt) (computer alarm)
Flux detectors (0.2 pt), indication (0.2 pt), incore flux (0.2 pt)

B. Pressurizer pressure (0.4 pt)
Core temperature/pressurizer temperature using steam tables (0.4 pt)
OTHER PRESSURE GAUGES USED TO MEASURE [illegible] PRESSURE ARE ACCEPTABLE

C. Excore flux (0.4 pt)
Primary heat balance/calorimetric (0.4 pt)
INCORE FLUX

D. Step increase in fission product activity in coolant (0.7 pt)

REFERENCES

Reactor Core, p 23
RPS, p 4

8. The Reactor Tave Control Program is generated by the Reactor Regulating System (RRS).

A. How does the generation of the Tref signal differ from the T_{ave} signal? (1.0 pt)

B. How large a deviation will occur before a T_{ave} - T_{ref} Hi/Lo alarm is received? (0.5 pt)

C. If the RRS fails and a turbine trip occurs would any adverse plant conditions exist? Explain. (1.5 pt)

ANSWER

A. T_{ref} is determined from the first stage turbine pressure T_{ave} is from the Reactor Coolant hot leg and cold leg temperatures

B. 5 degrees (0.5 pt)

C. Fail High: steam dumps open all the way and remain open with no operator action. On reactor trip steam dumps are planned to open [.5]

Fail Low: The loss of the turbine steam dump control from the RRS could prevent the opening of the dump valves resulting in the lifting of the steam generator code safeties. Although the normal heat sink (condenser) is not available, this in itself in not adverse. If a code safety were to remain open, an uncontrolled cooldown could occur. STEAM DUMPS WILL OPEN DUE TO SG PRESSURE. [1.0]

REFERENCES

CEA control system
CEA p 10
CEA/AOP 2-20 , simulator reference RX09

9. Explain how the use of a Main Steam manifold assists in maintaining a uniform neutron flux distribution within the reactor core. (2.0 pts)

ANSWER

- Main Steam manifold allows even sharing of the steam loads between the three steam generators
- Even steam loads allow uniform loop temperatures since the same energy is removed from each loop
- uniform loop temperatures allow for uniform reactor vessel mixing and therefore uniform core flux.

REFERENCE

Standard design

10. Emergency bus 5 feeds Primary Component Cooling pumps P-9A and P-9B. Emergency bus 6 feeds Secondary Component Cooling pumps P-10A and P-10B. All other redundant pumps on the emergency buses (charging, containment spray, LPSI, and steam generator aux charging) are split between buses 5 and 6 (bus 5: P14A, P61A, P12A, P25C/bus 6: P14B, P61B, P12B and P25A).

    A. Explain why the design and operation allows the above electrical arrangement. (1.0 pt)

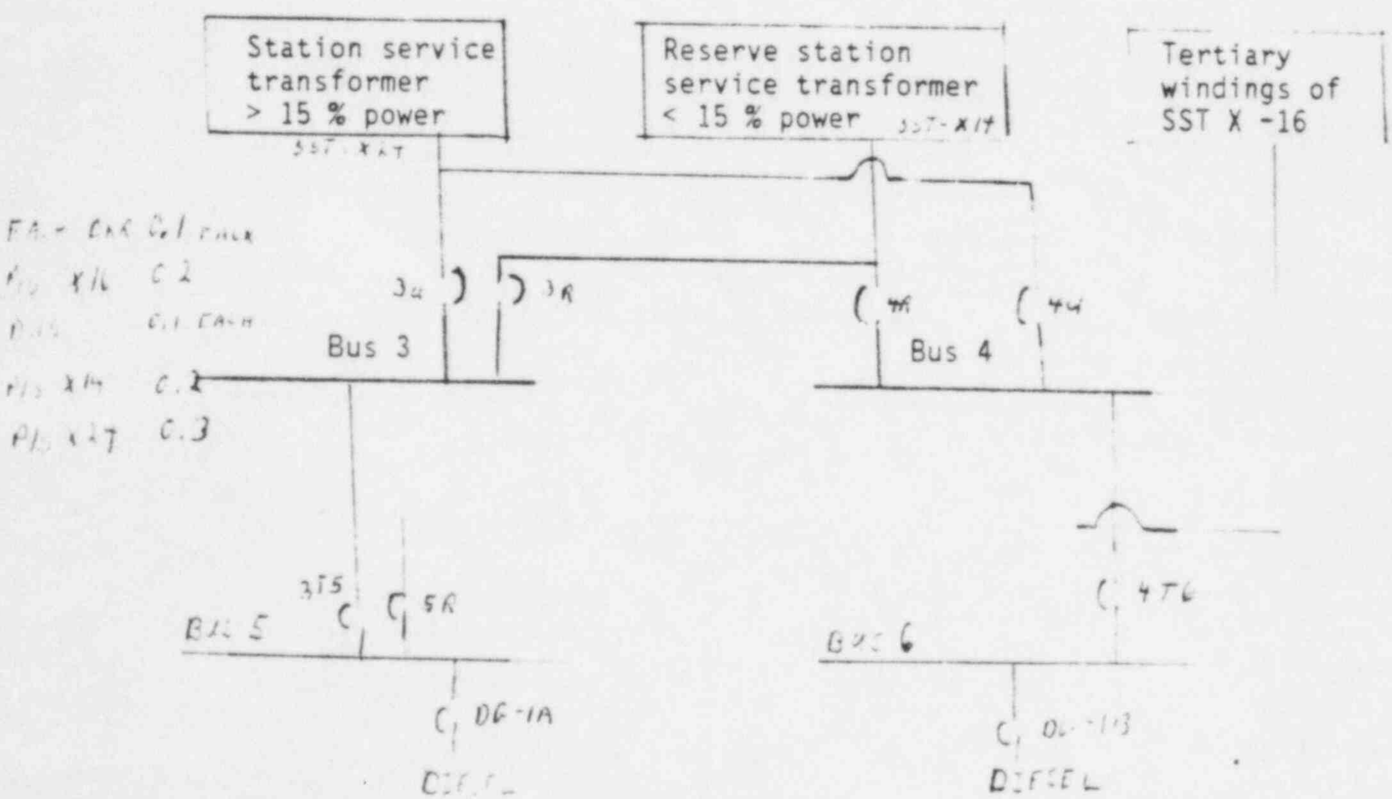
    B. Using a one line diagram, identify the following sources of power for buses 5 and 6. Show location of breakers (breaker numbers not necessary). (2.0 pt)

        1) Normal supply (> 15 % power)
        2) Normal supply (< 15 % power)
        3) Emergency Power

ANSWER

A. Under emergency conditions the PCC and SCC are redundant trains of Component Cooling. (0.5 pt) Each is a separate system supplying all redundant components required for core cooling following a loss of coolant accident. (0.5 pt)

B.



~~Bus 5~~ ~~Bus 6~~

~~Diesel~~ Diesel

grading: 9 breakers, 5 flow paths, 2 diesels, 3 transformers, 4 buses

REFERENCES

Component Cooling System p 3
High Voltage Electrical 4.16 KV bus p 39, figure PGS 17-1

7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL

1. According to Procedure 1-8-1, Power Distribution Control, all control rods are withdrawn unless a significant xenon oscillation occurs. If a significant oscillation develops, when should the control rods be inserted? Explain the basis of your answer. (2.0 pt)

ANSWER

~~Insert when offset is at positive peak (lower core power less than upper core power, U-L/U+L). (1.0 pt) On a positive peak more power is produced in the upper half of the core producing xenon and high clad temperatures. By driving the rods, the flux is thrown towards the bottom of the core causing upper cladding temperatures to decrease (less power) and reducing the xenon production. In addition, since flux is thrown to the bottom of the core, xenon is burnt out. Overall result is termination of the xenon oscillation. (1.0 pt)~~

REFERENCE

Procedure 1-8-1, Reactor Protection Systems p 24

DRIVE GROUPS 5A + 5B ~~WHEN S/C~~ TO MAINTAIN S/C BELOW THE UPPER CONTROL LINE UNTIL RODS REACH PDIL (> [illegible]) OR 90 STEPS (< 50%). THIS DRIVES FLUX TOWARD BOTTOM OF CORE AND REDUCES S/C. (2.0 pt)

2. If a charging pump started automatically and there is no SIAS, which one of the following sets of conditions will trip the pump? (0.5 pt)

   A. Pressurizer level drops 11% below the RRS setpoint.

   B. Pressurizer suction pressure decreases to 15 psia for half a second.

   C. Header discharge pressure increases to 2400 psig.

   D. Charging flow increases to 200 gpm.

ANSWER

C

REFERENCE

F[illegible] p 53

3. According to AOP 2-25, High Radiation Levels, several radiation monitors have automatic functions.

   A. For the following alarms, identify the automatic functions associated with each (1.0 pt):

      1) Primary Component Coolant monitor
      2) Waste Gas Vent monitor
      3) Spent Fuel Pool monitor

   B. A high alarm is received on the letdown high detector and the letdown low detector is pegged low. Is this indicative of a failure/problem with the letdown monitoring system or indicative of a plant operational problem? Explain. (1.0 pt)

ANSWER

A. 1) closes PCC surge tank vent
   2) closes waste gas release valve
   3) stops upward movement of the new fuel elevator

B. Indicative of a plant operational problem, potentially indicative of fuel cladding failure. The letdown low scintillation detector meter fails low when saturated.

REFERENCE

A. AOP 2-25
B. AOP 2-25

4. Concerning radioactive gas release:

   A. Identify three of the four sources of potentially radioactive gases that can be directed to the vent stack. (1.0 pt)

   B. In the event of a steam generator tube rupture, are any uncontrolled release paths possible if plant procedures are followed? Explain. (neglect small steam or feedwater leaks) (1.5 pt)

ANSWER

A. Waste Gas System
Containment Purge
Air Ejectors
Blowdown vents (any 3 @ 0.33 pt each)

B. Plant procedures require that affected steam generator to be isolated when identified. Until the faulted steam generator is identified and the Non Return Valve and Excess Flow Check Valve are closed (RCS pressure < 985 psig) the faulted steam generator will supply steam. Potential release paths are: (0.7 pt)

- Turbine driven aux feed pump (unless MS 59, 77 and 99 supply lines are closed) (0.4 pt)

- Aux steam system via auxiliary boiler condensate reciever (0.4 pt)

If the above are isolated, no releases should be present unless the safety valves lift.

CAF above answer.

H.P. CONDENSER RELEASE [0.4]
ATM. STEAM DUMPS [0.4]
AIR EJECTORS (MONITORED RELEASE) [0.2]

REFERENCE

Procedure 3.7.1.2
EOP 2-70-3, Steam Generator Tube Rupture; and FSAR

5. Identify three of the five plant conditions that adequately vent the Reactor Coolant System and allow the removal of the LTOP protection. (1.5 pt)

ANSWER

- Reactor head off
- Pressurizer manway off
- One pressurizer safety valve removed
- one PORV removed and its isolation valve disabled in the open position
- Both PORV control switches in "open" and their isolation valved open

REFERENCE

OP 1-1, Plant Heatup

6. Which of the following two items must be calculated daily (once per shift) on loss of the computer? (0.50 pt)

   A. Power based on 0.85(max kw/ft)/(latest kw/ft) and calorimetric
   B. symmetric offset and azimuthal power tilt
   C. calorimetric and symmetric offset
   D. calorimetric and azimuthal power tilt
   E. power based on 0.85(max kw/ft)/(latest kw/ft) and azimuthal power tilt
   F. power based on 0.85(max kw/ft)/(latest kw/ft) and symmetric offset

ANSWER

Calorimetric and azimuthal power tilt

REFERENCE

Procedure 1-4-1, Plant Operations with the Plant Computer

7. Concerning fire pump testing in accordance with Surveillance Procedure 3.1.9:

A. Why is it necessary to run the diesel driven fire pump for one hour at full load where as a 15 minute run is satisfactory for the motor driven pump? (0.7 pt)

B. During routine observation of the diesel driven fire pump a yellowish residue is evident on some of the diesel components. Is this observation of any significance? Explain. (0.7 pt)

C. What adverse condition is possible if the recirculation valve to the fire pond (F-19) is not closed prior to securing the only running fire pump? (0.7 pt)

D. Which of the following items are Technical Specification items? (0.4 pt)

1) operability testing of the diesel fire pump
2) operability testing of the motor dirven fire pump
3) fuel oil storage tank

ANSWER

A. Diesel driven pump needs 1 hour to reach full operating temperatures (0.7 pt)

B. Yes, chromates are used as a corrosion inhibitor. Residue may indicative of a coolant leak. (0.7 pt)

C. Prevent depressurizing the fire system by drawing back through F-19 with the pump shutdown. (0.7 pt)

D. All

REFERENCE

Procedure 3.1.9 Fire Pump Test

8. Explain how a Reactor Coolant System leakage evaluation is performed during steady state operations (T_{ave} is constant). Include the parameters recorded and any required calculations. (2.0 pts)

ANSWER

Record T_{ave}
Record start time, VCT level, pressurizer level (0.4 pt)
Record finish time, Vct level, pressurizer level (0.4 pt)
Convert change in VCT level and change in pressurizer level to gallons and compare difference (0.8 pt)
Difference divided by the elapsed time is the leakage (0.4 pt)

REFERENCE

Procedure 3.1.19, Reactor Coolant System Leakage Evaluation

9. Explain why it is necessary to have eight Emergency Operating Procedures for fires at Maine Yankee. (2.0 pts)

ANSWER

Each procedure, except for 2-90-0, Plant Fire Assessment, identifies the operator action for a fire in a given plant area. (0.8 pt) The different procedures are needed because each fire has the potential to remove from service certain equipment. Plant shutdown or operation may be impacted by the loss of this equipment. Guidance is provided for:

- Identifying the equipment which may be lost or produce erroneous readings (if applicable)
- Action to prevent potentially spurious actuation due to the fire (if applicable)
- Post fire ventilation requirements

REFERENCE

EOP for fires: 2-90-0 through 2-90-7

10. If emergency boration is required due to the failure of two or more control element assemblies to insert:

   A. What is the preferred boron injection path? Include source, major components, and injection point. (1.2 pts)

   B. Why is using the RWST and an auxiliary charging pump the least preferred method? (0.8 pt)

ANSWER

DIRECT THROUGH BAM [illegible] 37 [illegible]

A. BAST to BA transfer pumps ~~(0.6 pt) indirect (or pull tag and direct)~~ to charging pump to normal charging line ~~(0.6 pt)~~.

B. Auxiliary charging pump will not provide a boration rate sufficient to satisfy Technical specifications (0.8 pt)

REFERENCE

Emergency Boration 2-70-5

11. According to EOP 2-70-7, Total Loss of Forced Reactor Coolant Flow/Natural Circulation:

    A. What are three of the four conditions which, if established, will allow restarting of a Reactor Coolant Pump? (1.5 pts)

    B. Identify the THREE indications which are used to establish the effectiveness of natural circulation and identify how they should be trending. (1.5 pts)

ANSWER

A. LOCA does not exist or is isolated
subcooling is greater than 50 F
RCP seal water return flow
PCC flow available to RCP (any 3 @ 0.5 pt each)

B. Tc constant or decreasing
Th constant or decreasing
Incore thermocouples trending with Th RTDs (0.5 pt each)

REFERENCE

EOP 2-70-7

12. A. Under what circumstances is it possible to damage the reactor vessel by allowing safety injection to continue during a loss of coolant accident? (0.7 pt)

B. What four criteria must be evaluated prior to terminating HPSI? Numerical values are not required? (0.8 pt)

ANSWER

A. Following a rapid depressurization and cooldown if pressure is rapidly increased pressurized thermal shock may occur. (0.7 pt)

B. RCS Subcooling (> 50°F)
PZR Level ($\gtrless$ 50%)
SG Level (one > 150" WR)
HPSI not required to maintain PZR pressure or level (0.2 pt each)

REFERENCE

EOP-70-2 p 1

13. What actions, if any, should be taken for each of the following situations if the reactor has tripped after an extended run at 100% power? Consider each situation separately.

A. CEA 55 in regulating Group 1 indicates 70% withdrawn. All other CEA's indicate inserted. (1.0 pt)

B. Pressurizer level is at 40% and slowly increasing. RCS pressure is 1850 psig and T_{ave} is 570°F. (1.0 pt)

C. The SG pressures are 880 psig, 905 psig and 900 psig respectively. The turbine stop, governor, intercept, and reheat stop valves are shut. Main condenser ~~condenser~~ pressures is 15" Hg. SG level is increasing with only the standby feed pump running. The main feed reg valve is closed and the bypass is 30% open. (1.0 pt)

ANSWER

A. Open MG set output breakers (0.5 pt)
Continue with procedure (0.5 pt)

B. Reduce charging (0.3 pt)
Increase letdown (0.3 pt)
Prevent unnecessary heatup (0.3 pt)
Utilize alternate letdown (0.1 pt)

C. Open atmospheric steam dump valve and isolate MOV. (0.4 pt)
Return MOV isolation switch to neutral. (0.2 pt)
Shutdown auxiliary feed pumps. (0.4 pt)
AFTER VERIFYING ~~[illegible]~~ SG BEING FED BY MAIN FEED

REFERENCE

EOP 2-70-0 p 1, 2, 5

8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS

1. Identify four of the five conditions that must be met for a steam generator to be considered operable for decay heat removal, in accordance with Technical Specifications. (2.0 pts)

ANSWER

- The reactor coolant system must be closed and pressurized to 100 psi above saturation pressure
- The steam generator must have both the cold and hot leg stop valves fully open
- The steam generator water level must be above the top of the tube bundle
- An inventory of over 100,000 gallons of primary grade feedwater must be available
- A feed pump must be operating or available for operation (any 4 @ 0.5 pt each)

REFERENCE

TS 3.8.B, p 3.8.2

2. According to Technical Specification 3.6.A.2 one operable Emergency Core Cooling system train consists of five operable subsystems. Identify the five subsystems that are needed to make one ECCS train operable. (2.0 pts)

ANSWER

- Service Water pump subsystem
- Component Cooling Water pump sybsystem
- Low Pressure Safety Injection pump subsystem
- High Pressure Safety Injection pump subsystem
- Containment Spray pump and RHR Heat Exchanger subsystem (0.4 pt each)

REFERENCE

TS 3.6.A.2, p 3.6-1

3. Concerning control of plant drawings:

   A. How does an individual using plant drawings know what drawings are to be maintained to reflect the latest as-built information? (0.7 pt)

   B. Are copies of uncontrolled drawings allowed to be used for performance of an activity requiring plant drawings? Explain. (0.7 pt)

   C. A drawing is obtained for use in system work and it is stamped "Document under revision, refer to EDCR 126". May this drawing be utalized for the system work? Explain. (0.7 pt)

   D. Who is authorized to request a drawing change? (0.4 pt)

ANSWER

A. Stamped "Controlled Drawing". (0.7 pt)

B. Permitted only if the drawing is verified against a controlled set of drawings. (0.7 pt)

C. The referenced Engineering Design Change Request must be consulted for the actual status of the print. A change to the print may or may not be necessary before using. (0.7 pt)

D. Any individual who notices a needed change to a drawing. (0.4 pt)

REFERENCE

0-01-2 Drawing Control

4. For the following three events classify them as an Unusual Event, Alert, Site Emergency, General Emergency or not an emergency plan event. (1.5 pt)

   A. Effluent monitors detect greater than 1 R/hr at site boundary.
   B. Reactor Coolant System leakage is 1 gpm.
   C. Initiation of SIAS due to low pressurizer pressure. Pressurized level restored in less than 20 minutes.

   ANSWER

   A. General Emergency
   B. Not Emergency Plan
   C. Unusual Event

   REFERENCE

   Emergency plan and TS

5. What is the minimum number of personnel required on shift for a cold shutdown plant according to the Technical Specifications. Include the number and position of the personnel. (1.5 pt)

ANSWER

1 PSS
1 RO
1 Auxiliary Operator
1 HP
5 Emergency Brigade Members (0.3 each)

REFERENCE

Technical Specification

6. During a plant tour an unescorted operator enters a locked high radiation area with a survey meter. While in the area he finds the radiation levels less than those required for posting a high radiation area. He removes the posting and leaves the entrance open.

   A. According to plant procedures, was the operator permitted to enter the locked high radiation area? Explain. (0.75 pt)

   B. Was the operator authorized to remove the posting? Explain. (0.75 pt)

ANSWER

A. Yes, operations personnel are the only exemption from the requirement that HP coverage is necessary in high radiation areas and the requirement that only HP personnel may open HRAs.

B. No, only HP personnel are permitted to take down HRAs barricades and signs.

REFERENCE

Ops memo 9-J-3 Rev 3
High Radiation Area Controls

7. For each of the following situations, should a <u>white tag</u>, <u>yellow tag</u>, or <u>no tag</u> be used?

   A. Reactor Protection System high pressurizer pressure channel A is bypassed.

   B. The S charging pump is being used as a A pump in Standby.

   C. Following maintenance on a service water heat exchanger a relief valve is blocked in accordance with a hydrostatic test procedure.

   D. A valve on the service water system is to be repacked.

ANSWER

A. No tag
B. ~~White tag (on replaced pump)~~ No TAG
C. Yellow tag
D. White tag (0.5 pt each)

REFERENCE

Procedure No. 16-1

8. What three requirements must be met for a tag to be temporarily lifted for the purpose of testing of flushing? (1.0 pt)

ANSWER

- PSS/SOS and man for whom the apparatus is tagged and anyone having a hold on the tagging order must agree on the testing or flushing that is to be accomplished.
- Testing/flushing must commence immediately after lifting the tag.
- After testing or flushing is completed, tags must be immediately replaced or the associated tagging order cleared. (0.33 pt each)

REFERENCE

9. During review of a completed plant procedure, the shift operating supervisor (responsible management reviewer) observes several procedural steps marked with a handwritten "N/A" and no other notation.

   A. What is the acceptable form/format for identifying procedural steps which may not be required? Explain. (1.0 pt)

   B. What action must the responsible reviewing manager take in the above case? (0.5 pt)

ANSWER

A. If in the performance of a procedure, various step(s) or sections are not required, the steps or sections will be crossed out or otherwise neatly identified, initialed by the individual performing the procedure, and written justification shall be noted in the procedure margin or otherwise appropriate location. (1.0 pt)

B. Supervisory or management personnel reviewing such procedures shall insure that the written justification is completed and accurate. (0.5 pt)

REFERENCE

0-06-2, Procedures review, approval, distribution and adherence

10. Under what three conditions may a safeguards annunciator be disabled? (1.5 pt)

ANSWER

- Continuous intermittent alarming [0.4] and plant maintenance personnel are working on the problem under a valid DR/RO [0.1]

- Failure of a piece of safeguards equipment resulting in continuous annunciator alarming [0.5]

- Approval of a senior license operator [0.3] on shift to disable the annunciator and a log entry should be made as to why it it has been disabled and a DR/RO issued to correct the problem. [0.2]

REFERENCE

Ops memo 9-K-2, Rev 1

11. What THREE steps must be taken when a Reactor Protection System bypass key is used for removing a spurious alarm while at full power? (1.5 pts)

ANSWER

- Make a log entry as to the condition that exists and the channel that the key bypasses.
- Immediately notify I & C to investigate and repair the affected channel.
- The keys are not to be left inserted in the bypass mode for more than thirty minutes. If this condition occurs, then the key is to be removed from the RPS and the channel left in the alarm while awaiting repairs.

REFERENCE

Ops memo 9-K-1

12. While operating a full power a diesel generator is removed from service. Is it necessary to declare the associated high pressure and low pressure safety injection trains inoperable due to the loss of their emergency power supply? Explain. (1.5 pts)

ANSWER

No. Technical Specifications allow the exception that if a system, subsystem, train, component or device is determined to be inoperable soley because its normal or emergency power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition of Operation, provided:

- its corresponding normal or emergency power source is operable and all of its redundant systems, trains, components, and devices, are operable, or likewise satisfy the requirement of this specification.

Canidate is to demonstrate some knowledge of the existence of the above exception and what it involves.

REFERENCE

TS p 3.0-2

13. Concerning reportable events and the Emergency Plan:

An event occurs in which the reportability is questionable. The Plant Shift Supervisor disagrees with the Shift Engineer and the Senior Reactor Operator on shift.

A. Who is responsible for the reportability or non reportability of this event? (0.5 pt)

B. If the event is reported, who is responsible for notifying the Maine Yankee Duty Call Officer? (0.5 pt)

C. Who is responsible for notifying the Plant Manager? (0.5 pt)

ANSWER

A. PSS
B. PSS
C. Duty Call Officer

REFERENCE

Op 1-26-1

14. According to Techincal Specifications the reactor is critical when: Choose the most correct answer. (0.5 pt)

    A. When ever group 5 achieves 140 steps.

    B. Wide range log channels read 10E-4% power.

    C. When Keff = 1.0000.

    D. When the reactor operator declares the reactor critical.

ANSWER

~~D~~ B

REFERENCE

TS Definitions p-2, procedure 1-2

15. For each of the following maintenance requests, indicate whether the request SHOULD or SHOULD NOT be approved. Justify your answer. Assume reactor power is 100% and consider each request separately.

    A. A request to replace the gasket on the outer personnel air lock door.

    B. A request to install a SG level instrumentation modification which requires bypassing SG level instruments E-1-1 A, E-1-2 A and E-1-3 A. SG level instrument E-1-1 B is bypassed.

    C. A request to deenergize the entire containment spray high pressure indication subsystem to perform maintenance. (0.75 pt)

    D. A request to replace the condenser air ejector discharge radiation monitor. (0.7 pt)

ANSWER

A. SHOULD (0.3 pt), only one hatch of the personnel air lock is required to be shut. (0.45 pt)

B. SHOULD (0.3 pt), must trip Channel E-1-1 B then allowed to bypass single safety channel on each SG. (0.45 pt)

C. SHOULD (0.3 pt) allowed to remove one engineered safeguards subsystem from service for up to 24 hr. (0.45 pt)

D. SHOULD NOT (0.3 pt) should not intentionally enter an action statement. (Must take grab samples if channel inoperable). (0.45 pt)

REFERENCE

T.S. 3.11-1, 3.9-1, 3.9-3, 3.17-3

8 [illegible] ?

13 ? use [illegible]

---

(1) A BOL = - SOTC [illegible]

A. [illegible] should depend upon SOT

B. also depends on SOT

could discuss Xe transients

(2) ans A don't require CR ↑ causes [illegible] to ↓ (0.7) out of 2.0 or 35%

(3) A Prompt drop equation shows ≃ 6-7% past trip

B. ✓

C ✓

(4) A, B } OK except we have changed our fuel loading pattern putting hi power assemblys inboard somewhat could cause confusion on diff [illegible]

$$\frac{\phi_R N_R}{\phi_f N_f} = \psi$$

(5) OK

(6) A. less absorption in moderator relative to more resonance abs in fuel

7 A ✓
B ✓

8 T_{AV} remaining constant = not a concern (as a function of power)
reason that situation

9 ✓

10 ✓

11 ✓

12 A. add some water in steam line hitting elbow
B. ✓

13 ✓

14 ✓

NRC SRO EXAM REVIEW

SECTION 6.

1. OK
2. a. OK
   b. OK
3. OK
4. a. OK
   b. OK
   c. OK
   d. OK
   e. OK
   f. OK
   g. OK
5. a. OK
   b. OK
   c. NONE OF THE ACCIDENTS LISTED WILL CAUSE THE NON-TRIPPABLE CEAS TO DROP INTO THE CORE. (NON-TRIP GRIPPERS ENGAGED W/O ELECT. PWR)
6. a. QUESTION DOESN'T ASK FOR WHAT IS LISTED IN FIRST PART OF ANSWER (.6). ONLY ASKS FOR WHICH ACCIDENTS WOULD RESULT IN CIS AND WHY (IE HI CONT. PRESSURE)
   b. OK
   c. OK
7. a. OK
   b. CORE/PZR TEMP COULD BE SUPERHEATED OR READING METAL TEMPERATURE IF NO LEL
   c. OK INCORE FLUX SHOULD BE ACCEPTABLE IN PLACE OF EXCORE FLUX.
   d. OK

8. a. OK

b. OK

c. FAIL HIGH - REMAIN OPEN UNTIL STEAM HEADER DROPS TO 775 PSIG. COOLDOWN TOO FAR.

FAIL LOW - NORMAL & QUICK OPEN SIGNALS TO STM. DUMP VALVES ARE LOST - VALVES STAY SHUT.

- TURBINE BYPASS VALVES WILL STILL OPEN ON NORMAL SIGNAL (STEAM HEADER PRESSURE) - MIGHT STILL OPEN SAFETIES DEPENDING ON INITIAL POWER LEVEL.

9. OK

10. a. OK

b. OK

7.1 Answer should reflect instructions given in 1-8-1 concerning keeping Symmetric Offset between the "control lines".

4. A more paths available
PAB/SPRAY/Fuel Building ventillation

6. Daily or once per shift?

Azimothal Power Rqd only >50%

10 A - Refer to Procedure

13 B - Actions in Answer not necessarily elicited by the symptoms given - Indications Given and Procedure reference are (CRITICAL Safety Functions) Confusing

C - Not enough indications given to shut off Aux Feed Pumps

ADMIN Procedures, Conditions, Section 8

QUESTION

7B - answer should be - no tag required

10. Candidate should only need to demonstrate some knowledge of the existence of the above Op memos.

11 same as 10

14 answer key says D - it should be B.

15 these answers could be answered either way if the justification is correct - these type of decisions are the individual's choice - there are no hard set rules to follow.