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

EXAMINATION REPORT NO. 85-04 (OL)

FACILITY DOCKET NO. 50-333

FACILITY LICENSE NO. DPR-59

LICENSEE: Power Authority of the State of New York
P. O. Box 41
Lycoming, New York 13093

FACILITY: James A. FitzPatrick Nuclear Power Plant

EXAMINATION DATES: February 11 - 15, 1985

REVIEWED BY: David J. Lange for J. A. Berry 4/24/85
J. A. Berry, Lead Reactor Engineer (Examiner) Date

REVIEWED BY: for D M Johnson 4/24/85
R. M. Keller, Chief, Project Section 1C Date

APPROVED BY: H B Kister 4/24/85
H. B. Kister, Chief, Project Branch No. 1 Date

SUMMARY: Operator licensing examinations were conducted at FitzPatrick during the period of February 12-14, 1985. Three Reactor Operator candidates, four Senior Operator candidates, and three Instructor Certification candidates were administered written and oral examinations. One Reactor Operator candidate and one Instructor Certification candidate failed both the written and oral examinations. One Reactor Operator candidate and one Senior Operator candidate failed the written examination only. During the oral examinations, all candidates were noted to be very knowledgeable in the use of Technical Specifications and the Emergency Plan. Weaknesses were noted in the candidate's abilities to use piping and instrument drawings as well as basic logic diagrams. During grading of the written examinations, the Rector Operator candidates were noted to be weak in the area of Plant Design/Instrument Controls. No generic weaknesses were noted during the grading of the Senior Reactor Operator written examinations.

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REPORT DETAILS

TYPE OF EXAMS: Replacement X

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail	Inst. Cert Pass/Fail
Written Exam	1/2	3/1	2/1
Oral Exam	2/1	3/1	2/1
Overall Results	1/2	3/1	2/1

1. CHIEF EXAMINER AT SITE: D. Lange, U.S. NRC-Region I

2. OTHER EXAMINERS: W. Cliff, PNL-Battelle

3. FITZPATRICK ENTRANCE MEETING:

   NRC Attendees

   D. Lange, U.S. NRC Region I

   Facility Attendees

   F. Catella, FitzPatrick Training Department
   D. Simpson, FitzPatrick Training Coordinator
   M. Curling, FitzPatrick Training Superindentant

   An entrance meeting was conducted immediately following the start of the RO/SRO written exam.

   The tentative schedule for the oral exam assignments was discussed. The two hour exam review was scheduled on site from 3:00 PM - 5:00 PM. A tentative exit meeting was set up for Friday morning.

1. Summary of strengths and deficiencies noted on oral exams:

   A strength was noted by both examiners in the area of Technical Specifications and use of the Emergency Plan.

   An overall weakness was noted by both examiners in the candidates ability to use piping and instrument drawings and basic logic diagrams.

2. Summary of deficiencies noted from grading of written exams:

   The RO candidates had an overall weakness in the area of Plant Design and Instrument and Control.

   The SRO candidates had no generic weaknesses noted. The candidates failing the exam showed consistently lower grades in the area of procedures and administrative controls.

3. Comments on availability of, and candidate familiarization with plant reference material in the control room:

   Candidates did very well using the control room plant reference procedures and technical specifications. Overall weaknesses were noted in the candidates ability to use control room P&ID's and logic diagrams.

Exit Interview Details

Personnel Present at Exit Interview:

NRC Personnel

Dave Lange - Chief Examiner
Larry Doerflein - Senior Resident Inspector

NRC Contractor Personnel

William Cliff, PNL-Battelle

Facility Personnel

Donald Simpson - Training Coordinator
Douglas Lindsey - Assistant Operator Superintendent
W. Fernandes - Operator Superintendent
F. Catella - Nuclear Training Specialist
M. Curling - Training Superintendent
R. Converse - Superintendent of Power

Summary of Comments made at exit interview:

The Chief Examiner advised the facility of the preliminary results of the oral examinations.

The Chief Examiner noted the generic strengths and weaknesses observed during the oral exams.

The Chief Examiner commented on well written learning objectives that had been used during the candidates training program. The facility is in the process of developing lesson plans and exam bank questions based on these objectives. Emphasis should be placed on putting them in place.

Attachments:

1. Written Examination(s) and Answer Key(s) (SRO/RO)
2. Facility Comments on Written Examinations made during Exam Review

ATTACHMENT 2

During the SRO exam review, the following comments were raised by the utility. Resolution of these comm nts are incorporated in the Master Answer Key. The following are the accepted comments/additions/deletions from the test answers.

Question 5.5b Answer should be "because the high differential notch rod worth of edge rods can cause short periods OP65 pg.6."

Question 6.1a Answer should include: greater than 2.7 psig and 59.5 inches.

Question 6.1c May get the answer that if one diesel does not start, the second RHR pump will not start. Delete answer "core spray pump at speed".

Question 6.2a Manual not necessary. If not included, 0.25 points for each correct answer. If manual is included, 0.2 for each correct answer.

Question 6.2c Candidate may also say "reduced speed will reduce cooling water to barametric condenser and lube oil. (No credit taken off for this answer.)

Question 6.3a 0.5 for answer in Exam plus for "all rods in per EOP-3 (0.5).

Question 6.3b RWCU system isolates. (acceptable additional answer)

Question 6.5c Steam flow indication 9-5 panel. (acceptable additional answer)

Question 6.6 Steam flow feedflow mismatch and level decrease. Plant efficiency decrease. (Other acceptable answers.)

Question 6.7 Discharge Valve Not Open - The reason for the runback is to prevent excess axial thrust on pump.

The reason for 44% limit is to get you in the range of one feedpump to prevent low level scram.

Question 6.8 High temp in pump room should be T ambient + 40oF.

Question 7.1 Add "only if instrument nitrogen cannot be restored."

Question 7.2 Add "throttle service water flow and if still can't reduce temperature.."

Question 8.1 Add "unknown conditions, contaminated level 500,000 DPM/cm^2, 10R/hr Beta-gamma, maintenance in rad area."

ATTACHMENT 2

During the RO exam review, the following comments were raised by the utility. All comments were resolved at the review with documentation provided by the facility training department.

Question No.	Changes Necessary Change	NRC RESOLUTION REASON
1.09	Should be 10 not 40	Typo error-accepted.
2.03b	Answer Key should say "all of it".	Accepted with documentation.
2.03d	Answer Key should include (mini-purge) A&B are reset alike.	Accepted with documentation.
2.05b	Consider level control to torus in answer.	Considered during grading.
2.07b	Answer will be "yes" if candidate says from a trip from high level.	Modification accepted with documentation.
3.05b-#2	Answer should include → to give the operator enough time to respond to the steam flow to avoid a scram.	Accepted with documentation provided.
4.02b	Consider expanding on answer to include RPV Control guidelines.	Documentation provided. Considered during grading.

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: FITZPATRICK

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 05/02/12

EXAMINER: LANGE, D.

APPLICANT: MASTER

INSTRUCTIONS TO APPLICANT:

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

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

FINAL GRADE _________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 1.01 (2.25)

a. Using the attached Power to Flow operating map, name items 1 thru 14 . (explaniation not required). (1.40)
b. What is the significance of item # 10. (0.25)
c. Item # 11 (the entire line) is slightly concave, from where it intersects (starts) at item # 1 to where it ends at item # 7. Explain the reason for this. (0.60)

QUESTION 1.02 (2.00)

During a cooldown of the reactor vessel from outside the control room, reactor pressure decreased from 885 psig. to 595 psig. in one half hour.Has your reactor cooldown limit been exceeded ? { show all work } (2.00)

QUESTION 1.03 (2.00)

Concerning control rod worth during a reactor startup with 100% peak Xenon versus a startup with Xenon free conditions, WHICH STATEMENT IS CORRECT? JUSTIFY YOUR CHOICE.

a. PERIPHERAL control rod worth will be LOWER during the 100% peak Xenon startup than during the Xenon free startup.

b. CENTRAL control rod worth will be HIGHER during the 100% peak Xenon startup than during the Xenon free startup.

c. PERIPHERAL control rod worth will be HIGHER during the 100% peak Xenon startup than during the Xenon free startup.

d. BOTH CENTRAL and PERIPHERAL control rod worths WILL BE THE SAME regardless of core Xenon concentration.

QUESTION 1.04 (2.00)

Indicate whether the following will INCREASE or DECREASE reactivity during operation AND briefly EXPLAIN why?

a. Moderator temperature increases while below staturation temperature. (0.5)
b. Fuel temperature increases. (0.5)
c. Loss of a feedwater heater. (0.5)
d. A sudden reduction in reactor primary system steam pressure. (0.5)

QUESTION 1.05 (2.50)

NOTE: Answer the following question from a theoretical standpoint, not from a Fitzpatrick system design standpoint.

With the plant operating at 90% power, extraction steam to the highest pressure feedwater heater is removed. An engineer, observing that turbine load increased by 15 MWe after the extraction steam removal, concludes that this action has improved the plant's thermodynamic efficiency (NOT heat rate). Do you agree with this conclusion? Explain your answer fully. (2.0)

QUESTION 1.06 (2.50)

The concept of Subcritical Multiplication is used to describe the behavior of the reactor during refueling operations or startup.

a. In a subcritical reactor, if the source level doubles, what will happen to the neutron level ? (0.50)
b. What three variables affect the subcritical neutron level ? (0.75)
c. In a subcritical reactor, if a reactivity of 0.003 dk/k is added to the reactor,will it take longer to reach equilibrium if the initial k- effective is 0.92 or if k- effective is 0.992 ? Explain the reason for your answer. (1.25)

QUESTION 1.07 (3.00)

When the reactor is at full power and a feedwater controller malfunction results in a loss of feedwater flow, a reactor scram will occur (due to low reactor water level) within a short period of time. During the time period JUST PRIOR TO THE SCRAM, is reactor power expected to INCREASE, DECREASE or REMAIN CONSTANT? Give TWO REASONS for your answer. (3.0)

QUESTION 1.08 (3.00)

Assume the reactor is operating at 100% power and one recirculation pump trips. Indicate how each listed indicated parameter would first change (Increase or Decrease) and briefly explain why the change occurs.

a. reactor power (one reason) (1.0)

b. reactor water level (two reasons) (1.0)

c. feedwater flow (two reasons) (1.0)

QUESTION 1.09 (2.25)

For each of the pairs of conditions listed below, state which condition would have a GREATER DIFFERENTIAL ROD WORTH and briefly EXPLAIN WHY .

a. Reactor moderator temperature of 150 deg.F or 500 deg.F ? (0.75)

b. For an inserted rod next to a fully withdrawn control rod or next to a fully inserted control rod. (assume average core flux is constant) (0.75)

c. For a rod at position 10 or position 40 of a core operating at 100 % power ? (0.75)

QUESTION 1.10 (2.00)

Will the Recirculation pumps have more NPSH at 4 % or 100 % power ? (EXPLAIN YOUR ANSWER FULLY) (2.00)

QUESTION 1.11 (1.50)

For each of the events listed below, state which reactivity coefficient will respond first and if it adds positive or negative reactivity. (1.50)

a. Relief Valve opening at 100 % power . (0.50)

b. Rod drop at 100 % power. (0.50)

c. Isolation of a feedwater heater string at 75 % power. (0.50)

QUESTION 2.01 (2.50)

With the mode switch in the REFUEL position, what conditions (OTHER than those initiated by the Neutron Monitoring and Recirc.Flow Control Sys) will initiate a CONTROL ROD BLOCK ? Include setpoints as appropriate.(2.50)

QUESTION 2.02 (2.00)

Concerning the Scram Discharge Volume (SDV.)

a. Is it permissable to close the SDV vent and drain valves during normal operation ? If not, why. If so, under what conditions/limitations. (0.75)

b. Following a Scram, from full power, what will be the internal pressure of the SDV. ? (0.25)

c. On your shift, during rated power conditions, you discovered that the SDV.HI-HI water level BYPASS switch has been in the BYPASS position since startup. Could this have prevented a VALID high instrument volume scram from occuring ? (briefly explain). (0.75)

d. With the MODE switch in SHUTDOWN the HIGH -(~~18~~ 26 gal.) SDV. ROD BLOCK trip is BYPASSED ? (True or False) (0.25)

QUESTION 2.03 (3.00)

a. What are the normal values for CRD HYDRAULIC SYSTEM FLOW, DRIVE WATER DIFF. PRESS. and CHARGING WATER HEADER PRESS., indicated in the Control Room. (1.00)

b. Approximately what percentage of the flow in 'a' above is supplied to the cooling water header? (0.50)

c. Explain HOW/WHY requesting single rod insertion causes cooling header flow to vary (include by how much the flow varies). (1.00)

d. The system flow in 'a' above is less than the normal flow output of one pump. List two (2) taps off the CRDH system upstream of the flow sensing element. (0.50)

QUESTION 2.04 (3.00)

Concerning the Recirculation System ;

a. In addition to a Recirc. Pump runback, what operational conditions will cause a Recirc. Pump Trip ? (setpoints are required) (1.00)

b. Prior to starting an Idle Recirc. Pump what coolant limitations have to be adhered to ? (be specific). (0.75)

c. When starting up the Recirc. Pump and opening the discharge valve, what specific RX.parameter must be monitored and why ? (0.75)

d. For Operation of the Scoop Tube Positioner with the Handcrank :

1. How is the Electric Break released on the positioner motor for both the A-MG set and B-MG set. (0.50)

QUESTION 2.05 (3.00)

Answer the following with regard to the RHR system and its various modes of operation:

a. Match the following actions, events, or interlocks in Column A with the item in Column B that initiates that item. (0.75)

Column A	Column B
1. Shutdown cooling isolates	50 psig.(inc)
2. LPCI auto initiation (in conjuction with hi DW pressure)	75 psig (inc)
	125 psig (inc.)
3. Input to the Auto Blowdown Sys.	420 psig (inc.)
	450 psig.(dec.)

b. Explain the purpose of the RHR.- CONDENSING MODE of operation . (1.00)

c. Torus Cooling may be initiated at any time, reguardless of whether or not a LPCI initiation signal is present . (TRUE or FALSE) ?? (0.25)

d. With a LPCI initiation signal present list TWO (2) separate sets of conditions that would allow you to initiate CONTAINMENT SPRAY. (1.00)

QUESTION 2.06 (3.00)

a. What are three (3) signals that will cause a diesel generator to automatically Emergency Start (exclude manual, setpoints ARE required)? (1.00)

b. When the Emerg. D/G is in the MAINTENANCE mode of operation it can ONLY be manually started,LOCALLY ? (TRUE or False) (0.50)

c. List six (6) Emerg. D/G engine faults that would prevent automatic initiation and cause a shutdown of the Diesel, indicating whether or not the trip would occur if a VALID LOCA signal were present . (1.50)

QUESTION 2.07 (2.50)

Concerning the Reactor Core Isolation Cooling Sys. (RCIC) ;

a. List all the conditions that will cause an automatic isolation of the steam line isolation valves, 13-MOV-15 & 16 . (1.00)

b. If the RCIC sys. Turbine had been shutdown by an automatic trip signal, which inadvertantly had come in and cleared, will the Turbine re-start on a VALID initiating signal with no operator action. ? Explain. (0.75)

c. What is the reason for the 'CAUTION','Do not operate the RCIC turbine at a speed below 2200 rpm. for an extended period of time ' ? (0.75)

QUESTION 2.08 (2.75)

Concerning the Standby Liquid Control Sys.;

a. Once the SBLC. sys. has initiated, what six (6) CONTROL ROOM indications could you use to verify that the system is operating properly AND injecting into the reactor vessel ? (1.50)

b. After initiation of the SBLC. sys.,is it permissible to shut the system down ? (if not, WHY., if so, under what conditions ?) (1.25)

QUESTION 2.09 (1.75)

Concerning the OFF-GAS system ;

a. List three(3) conditions that will cause an automatic shutdown of the off-gas recombiner. (1.00)

b. What undesirable condition could exist with the loss of the recombiner and subsequent failure of the lead dilution fan to start ? (0.50)

c. Who (by title) should be notified in the event the off-gas sys. is operating with the recombiner isolated ? (0.25)

QUESTION 2.10 (1.50)

Concerning the Traversing In Core Probe Sys. ;

a. List two (2) valuable operational parameters that are developed from signals generated by the TIP. sys. (1.00)
(include in your ans. how the signals are being used and what information is being obtained)

QUESTION 3.01 (2.50)

No.1- Indicate at what RX. water level, ABOVE THE TOP OF THE ACTIVE FUEL, each of the following actions is directly initiated. If more than one level applies, indicate all of the applicable levels. (1.50)

a. Direct reactor scram

b. Standby Gas Treatment System starts

c. RCIC starts

d. Reactor Low Water level alarm annunciates

e. Recirculation pumps trip

f. RFPs trip

No.2

What is the high pressure trip setpoint for the ATWS Recirc.Pump Trip and how does this compare to pressure set point of the Relief Valves lifting? (1.00)

QUESTION 3.02 (3.00)

For EACH of the following conditions, state whether a scram, half-scram, rod block, or no action is directly generated. For conditions that produce more than one action, state the more severe action (i.e. half-scram is more severe than a rod block). (3.00)

a. Loss of one RPS MG set
b. Turbine trip at 25% power
c. Two main steam lines isolated, Mode switch in RUN
d. APRM B downscale, Mode switch in RUN
e. Scram discharge volume level is at 40 gallons, Mode switch in STARTUP
f. Recirc. Flow Comparator Inop. (mode switch in shutdown)

QUESTION 3.03 (3.00)

Match the radiation detectors in COLUMN A with the appropriate application(s) and characteristic(s) from COLUMN B. All items in COLUMN B apply. (3.0)

COLUMN A	COLUMN B
a. Ion chamber	1. Used primarily as a device for precise measurements or where high sensitivity is required.
b. Scintillation	2. The size of the electron avalanche is proportional to the original incoming radiation energy.
c. Geiger-Mueller	3. Uses a photocathode to convert light into free electrons.
	4. Normally used to set dose rates.
	5. The same size pulse is generated regardless of the type and specific ionization characteristics of the radiation.
	6. Normally reads out in counts per minute.
	7. Main Steam Line Radiation Monitors
	8. Main Stack flow Radiation Monitor .

QUESTION 3.04 (3.00)

What effect will a complete loss of Reactor Protection Sys.(RPS) power have on the following systems or parameters ? (fully explain)

a. Recirculation system. (0.75)

b. EHC system. (0.75)

c. Reactor vessel level. (0.75)

d. Reactor level instruments. (control room and local inst. racks) (0.75)

QUESTION 3.05 (2.50)

Concerning the Recirculation System;

a. Under normal operating conditions, what signals will cause a MG Set Scoop Tube Lock WITHOUT tripping the MG Set ? (1.00)

b. List two (2) conditions that will cause a Recirc. Run-Back, the percent the pumps run back to, and the reason for the runback.(1.00)

c. You are in the process of starting up (RX. Power is approx.30 %).What two (2) control room indications could you use to verify a Recirc.Run-Back. (0.50)

QUESTION 3.06 (2.25)

During the 4:00 pm. to 12:00 mid. shift , at rated power, you receive two alarms on panel 09-6 ;

1.Off GAS line high pressure. (20 psig.)

2.Off GAS line high temperature (300 deg.F)

You notice that the condenser isolation valves 38-AOV-113 A and 113 B are shut and condenser vacuum is decreasing .

a. Based on the above indications/conditions, WHAT HAS OCCURED ?,and what additional automatic actions can be expected ? (1.75)

b. Based on the above situation what would be your first IMMEDIATE action ? (0.50)

QUESTION 3.07 (3.00)

Concerning Refuel Operations and Fuel Servicing Equipment ;

a. There are no interlocks to prevent the refueling platform from traversing in the forward (from the RX.vessel to the spent fuel pool) direction. (TRUE or FALSE) (0.50)

b. What safety precaution is taken prior to using the auxiliary hoist to handle contaminated equipment that must be kept below a specific water level ? (0.50)

c. What purpose does the Refueling Bellows accomplish during refuel operations, and RX.vessel heatup and cooldown ? (0.50)

d. What two (2) conditions will prevent the refuel platform from traveling toward the core, when the mode switch is in refuel ? (0.50)

QUESTION 3.08 (2.00)

How is the integrity of ECCS piping inside the reactor vessel verified during normal operation (include sensing points, specific system(s) who's piping is verified, why its verified, and response of the instrumentation to a loss of integrity in your answer)? (2.00)

QUESTION 3.09 (2.00)

During your shift , a Relief Valve fails open . Following the Reactor Scram and full P/C Isolation the vessel rapidly depressurizes to below 500 psig.

a. What control room level instrumentation is accurate and what level instrumentation would not be reliable during the above transient ? (2.00) (BRIEFLY EXPLAIN)

QUESTION 3.10 (1.75)

Concerning the Main Steam System ;

a.List three (3) functions of the Main Steam Line Flow Restrictors.(1.00)

b.How many solenoid operated pilot valves are associated with each MSIV and what is the purpose of each. (0.75)

QUESTION 4.01 (2.00)

a. According to Fitzpatrick's Radiation Protection procedure, for selection of survey instruments, list four of the five considerations that should be taken into account prior to their use. (1.00)

b. Match the following HP. survey instruments in column # 1 with the type of radiation / contamination detection being performed in column # 2.

Column # 1	Column # 2
1. Portable GM survey meter. (Eberline- E-120)	a. Commonly used for measuring beta-gamma exposure rates. Has a range of 1 to 25,000 mrad/hr.
2. Portable Ion Chamber. (Victoreen, 740-F)	b. Very sensitive instrument used for ' sniffing ' type surveys to detect and pinpoint the presence of beta & gamma.
3. Portable ion chamber. (Teletector)	c. Not recommended for determining beta contact exposures in high radiation areas, but would be useful for checking an off-gas leak in a large area.
4. Portable G.M. detector. (digi/master- 305, and auto-digi/master-305-B)	d. Commonly used for measuring beta/gamma exposure rates in High-Rad areas. Has a range of 0.1 mr/hr to 1000 r/hr.

QUESTION 4.02 (1.75)

Concerning F- EOP-33 , Small Break Accident.

a. Under what conditions can the automatic controls of an ECCS sys. be placed in manual ? (1.00)

b. List three of the four basic objectives you are expected to achieve, in the event of a pipe break, with respect to the REACTOR CORE and its CONTAINMENT . (0.75)

QUESTION 4.03 (2.00)

Given the set of indications listed below, which exist following a valid LOCA, state whether or not adequate core cooling can be assured. Justify your answer. (2.0)

- -HPCI has ISOLATED due to low steam supply pressure.
- -All reactor water level instruments are off-scale LOW, with the exception of the fuel zone instrument which is off-scale HIGH.
- -Both core spray pumps have started, subsequently tripped on overload, and CANNOT be restarted.
- -RHR pump 'A' is RUNNING with an injection path to the RPV (minimum flow valve closed in loop 'A'. All other RHR pumps have failed to start.

QUESTION 4.04 (2.25)

During your 4:00p to12:00 mid. shift there is an unexplained (slow) decrease in Primary Containment pressure. Other than a suspected loss of Primary Containment, what additional events could have caused this pressure decrease. Explain Why . (0.25 for the event, 0.50 for the reason)

QUESTION 4.05 (1.50)

a. In accordance with the Fitzpatrick Generating Station Emergency Plan Implementation Procedures, what are the EMERGENCY EXPOSURE GUIDELINES for the following situations:(WHOLE BODY ONLY)
   1. Life Saving and Reduction of Injury (0.5)
   2. Operation of Equipment to Mitigate an Emergency (0.5)
   3. Either of the above two conditions if adaquate planning and protection permits. (0.5)

QUESTION 4.06 (2.00)

According to F-AOP-1,(Reactor Scram),list the immediate Operator actions you are to perform in the event of a reactor scram . (1.50)

QUESTION 4.07 (2.50)

While controlling reactor pressure following a reactor scram and isolation, (MSIV closure), reactor pressure approaches the safety/relief valve set-point following an earlier auto- blowdown; According to F-OP-1 (Main Steam Sys.) you are directed to reduce pressure to approximately 900 psig.

a.How could you have possibly avoided this pressure increase ? (0.75)

b.Now, having to select a safety/relief valve for pressure reduction,what is the next valve to be selected and WHY ? (0.75)

c.As soon as conditions permit , ie: Scram/Isolation reset, list two more conventional means of controlling reactor pressure.(be specific) (1.00)

QUESTION 4.08 (2.50)

When operating the RHR System in the Shutdown Cooling Mode, Procedure F-OP-1 states that anytime the reactor vessel is in a no or low flow condition, to increase vessel level to 234.5 ".

a. What level instrument should be used to verify 234.5" ? (0.50)

b. Why is this level incresse important ? (0.75)

c. List three problems that could occur if level was not increased. (1.00)

QUESTION 4.09 (2.75)

Concerning F-AOP-36, (Stuck Open Relief Valve) :

a. List five (5) control room instrument indications, including back panels that you could use to verify a Relief Valve is stuck open.(1.25)

b. Having unsuccessvily attempted to shut the stuck open R.V., by cycling the valve control switch on panel 09-4, what further action can you take to get the valve shut ? (0.50)

c. It is determined that the problem exists at the Remote Relief Valve Panel. Concerning this,answer the following;

1. Where is this panel located ? (be specific) (0.50)
2. With the panel energized, what further action can be taken ? (0.50)

QUESTION 4.10 (3.00)

Concerning Procedure F-EOP-28,"PLANT SHUTDOWN FROM OUTSIDE THE CONTROL ROOM ":

a. What are the immediate operator actions if the main control room becomes uninhabitable ?(3 required) (1.5)

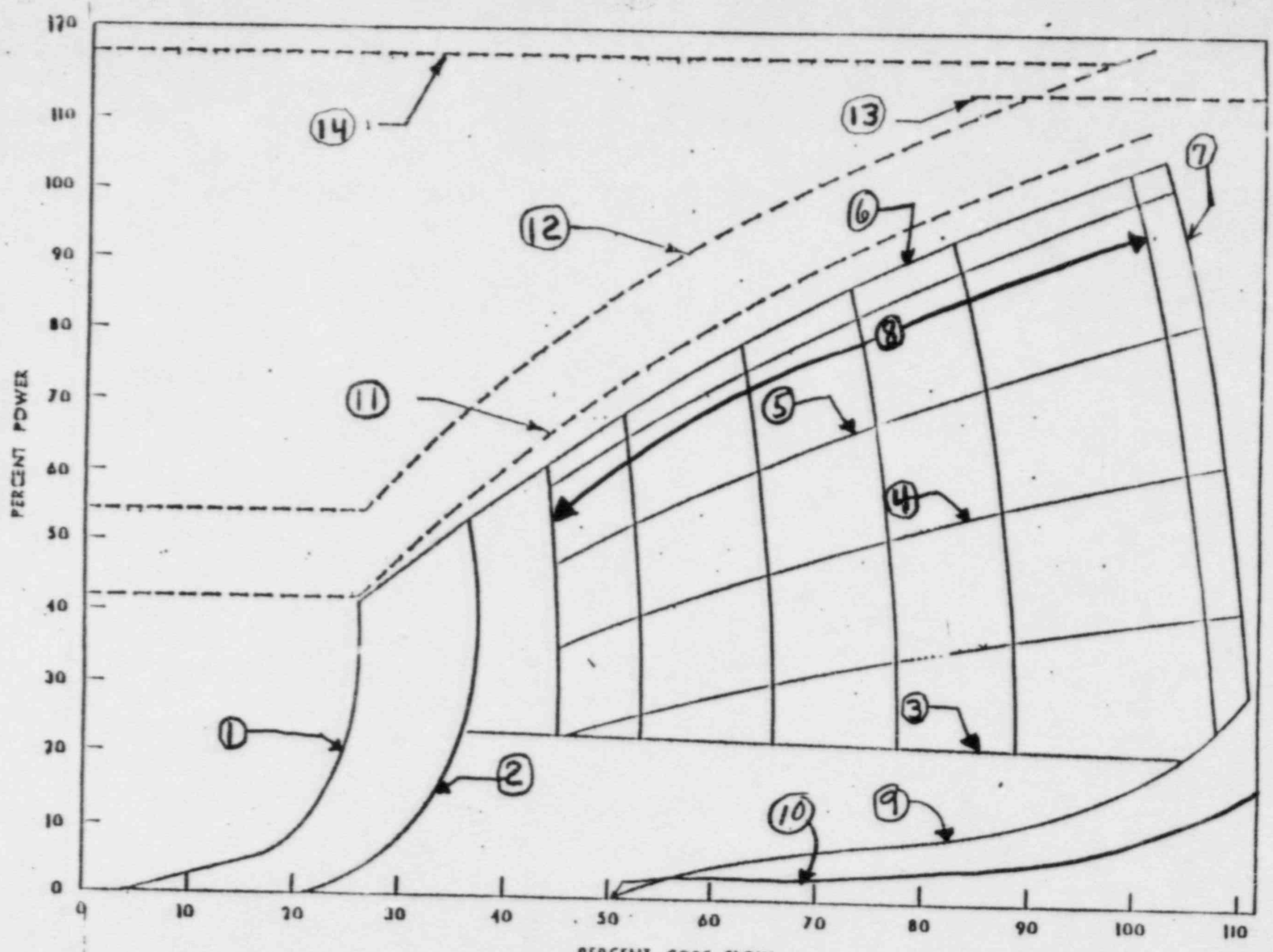
b. Where is RX. level and press. monitored outside the Control Room ? (0.5)

c. If the NCO is unable to shut down the plant prior to leaving the Control Room, list Four (4) ways to Scram the RX. (in order of prefrence).(1.00)

QUESTION 4.11 (2.75)

According to F- EOP-2 (RPV control) and F-EOP-4 (Primary Containment Control);

a. List the entry conditions for RPV Control, (include setpoints).(1.50)
b. List the entry conditions for Primary Containment Control. (1.25)
(include setpoints)



120
110
100
90
80
70
60
50
40
30
20
10
0
PERCENT POWER
0
10
20
30
40
50
60
70
80
90
100
110
PERCENT CORE FLOW
1
2
3
4
5
6
7
8
9
10
11
12
13
14

Figure 07D-5

TABLE II-3-1

PROPERTIES OF SATURATED STEAM AND SATURATED WATER (TEMPERATURE)

Temp F	Press. psia	Volume, ft³/lb Water v_f	Evap v_{fg}	Steam v_g	Enthalpy, Btu/lb Water h_f	Evap h_{fg}	Steam h_g	Entropy, Btu/lb x F Water s_f	Evap s_{fg}	Steam s_g	Temp F
32	0.08859	0.01602	3305	3305	−0.02	1075.5	1075.5	0.0000	2.1873	2.1873	32
35	0.09991	0.01602	2948	2948	3.00	1073.8	1076.8	0.0061	2.1706	2.1767	35
40	0.12163	0.01602	2446	2446	8.03	1071.0	1079.0	0.0162	2.1432	2.1594	40
45	0.14744	0.01602	2037.7	2037.8	13.04	1068.1	1081.2	0.0262	2.1164	2.1426	45
50	0.17796	0.01602	1704.8	1704.8	18.05	1065.3	1083.4	0.0361	2.0901	2.1262	50
60	0.2561	0.01603	1207.6	1207.6	28.06	1059.7	1087.7	0.0555	2.0391	2.0946	60
70	0.3629	0.01605	868.3	868.4	38.05	1054.0	1092.1	0.0745	1.9900	2.0645	70
80	0.5068	0.01607	633.3	633.3	48.04	1048.4	1096.4	0.0932	1.9426	2.0359	80
90	0.6981	0.01610	468.1	468.1	58.02	1042.7	1100.8	0.1115	1.8970	2.0086	90
100	0.9492	0.01613	350.4	350.4	68.00	1037.1	1105.1	0.1295	1.8530	1.9825	100
110	1.2750	0.01617	265.4	265.4	77.98	1031.4	1109.3	0.1472	1.8105	1.9577	110
120	1.6927	0.01620	203.25	203.26	87.97	1025.6	1113.6	0.1646	1.7693	1.9339	120
130	2.2230	0.01625	157.32	157.33	97.96	1019.8	1117.8	0.1817	1.7295	1.9112	130
140	2.8892	0.01629	122.98	123.00	107.95	1014.0	1122.0	0.1985	1.6910	1.8895	140
150	3.718	0.01634	97.05	97.07	117.95	1008.2	1126.1	0.2150	1.6536	1.8686	150
160	4.741	0.01640	77.27	77.29	127.96	1002.2	1130.2	0.2313	1.6174	1.8487	160
170	5.993	0.01645	62.04	62.06	137.97	996.2	1134.2	0.2473	1.5822	1.8295	170
180	7.511	0.01651	50.21	50.22	148.00	990.2	1138.2	0.2631	1.5480	1.8111	180
190	9.340	0.01657	40.94	40.96	158.04	984.1	1142.1	0.2787	1.5148	1.7934	190
200	11.526	0.01664	33.62	33.64	168.09	977.9	1146.0	0.2940	1.4824	1.7764	200
210	14.123	0.01671	27.80	27.82	178.15	971.6	1149.7	0.3091	1.4509	1.7600	210
212	14.696	0.01672	26.78	26.80	180.17	970.3	1150.5	0.3121	1.4447	1.7568	212
220	17.186	0.01678	23.13	23.15	188.23	965.2	1153.4	0.3241	1.4201	1.7442	220
230	20.779	0.01685	19.364	19.381	198.33	958.7	1157.1	0.3388	1.3902	1.7290	230
240	24.968	0.01693	16.304	16.321	208.45	952.1	1160.6	0.3533	1.3609	1.7142	240
250	29.825	0.01701	13.802	13.819	218.59	945.4	1164.0	0.3677	1.3323	1.7000	250
260	35.427	0.01709	11.745	11.762	228.76	938.6	1167.4	0.3819	1.3043	1.6862	260
270	41.856	0.01718	10.042	10.060	238.95	931.7	1170.6	0.3960	1.2769	1.6729	270
280	49.200	0.01726	8.627	8.644	249.17	924.6	1173.8	0.4098	1.2501	1.6599	280
290	57.550	0.01736	7.443	7.460	259.4	917.4	1176.8	0.4236	1.2238	1.6473	290
300	67.005	0.01745	6.448	6.466	269.7	910.0	1179.7	0.4372	1.1979	1.6351	300
310	77.67	0.01755	5.609	5.626	280.0	902.5	1182.5	0.4506	1.1726	1.6232	310
320	89.64	0.01766	4.896	4.914	290.4	894.8	1185.2	0.4640	1.1477	1.6116	320
340	117.99	0.01787	3.770	3.788	311.3	878.8	1190.1	0.4902	1.0990	1.5892	340
360	153.01	0.01811	2.939	2.957	332.3	862.1	1194.4	0.5161	1.0517	1.5678	360
380	195.73	0.01836	2.317	2.335	353.6	844.5	1198.0	0.5416	1.0057	1.5473	380
400	247.26	0.01864	1.8444	1.8630	375.1	825.9	1201.0	0.5667	0.9607	1.5274	400
420	308.78	0.01894	1.4808	1.4997	396.9	806.2	1203.1	0.5915	0.9165	1.5080	420
440	381.54	0.01926	1.1976	1.2169	419.0	785.4	1204.4	0.6161	0.8729	1.4890	440
460	466.9	0.0196	0.9746	0.9942	441.5	763.2	1204.8	0.6405	0.8299	1.4704	460
480	566.2	0.0200	0.7972	0.8172	464.5	739.6	1204.1	0.6648	0.7871	1.4518	480
500	680.9	0.0204	0.6545	0.6749	487.9	714.3	1202.2	0.6890	0.7443	1.4333	500
520	812.5	0.0209	0.5386	0.5596	512.0	687.0	1199.0	0.7133	0.7013	1.4146	520
540	962.8	0.0215	0.4437	0.4651	536.8	657.5	1194.3	0.7378	0.6577	1.3954	540
560	1133.4	0.0221	0.3651	0.3871	562.4	625.3	1187.7	0.7625	0.6132	1.3757	560
580	1326.2	0.0228	0.2994	0.3222	589.1	589.9	1179.0	0.7876	0.5673	1.3550	580
600	1543.2	0.0236	0.2438	0.2675	617.1	550.6	1167.7	0.8134	0.5196	1.3330	600
620	1786.9	0.0247	0.1962	0.2208	646.9	506.3	1153.2	0.8403	0.4689	1.3092	620
640	2059.9	0.0260	0.1543	0.1802	679.1	454.6	1133.7	0.8686	0.4134	1.2821	640
660	2365.7	0.0277	0.1166	0.1443	714.9	392.1	1107.0	0.8995	0.3502	1.2498	660
680	2708.6	0.0304	0.0808	0.1112	758.5	310.1	1068.5	0.9365	0.2720	1.2086	680
700	3094.3	0.0366	0.0386	0.0752	822.4	172.7	995.2	0.9901	0.1490	1.1390	700
705.5	3208.2	0.0508	0	0.0508	906.0	0	906.0	1.0612	0	1.0612	705.5

MASTER

ANSWERS -- FITZPATRICK -85/02/12-LANGE, D.

ANSWER 1.01 (2.25)

a. 1. natural circulation line .
2. min. pump speed line, 28 %.
3. min. power line.
4. 50 % load line.
5. 75 % load line.
6. 100 % load line.
7. Pump constant speed line.
8. flow control range.
9. Recirc. pump NPSH limit line.
10. Jet pump NPSH limit line.
11. APRM rod block line.
12. APRM thermal scram.
13. Thermal scram trip-Clamp
14. APRM- fixed scram.

(0.10 for each correct ans.)

b. Recirc. pump NPSH limit , to protect against cavitation. (0.60)

c. This line is slightly concave because; as core flow increases core inlet subcooling decreases, therefor core thermal power is slightly less at full flow than it would be if the core inlet temperature did not change. (0.50)

REFERENCE
JAF. LP. book #1 Tab-C, Sec. E. Recirc. and Recirc. Flow Control Sys. Performance Objective- Describe the operation of the Recirc. Flow Control System, Power to flow operating map FIG #7.

ANSWER 1.02 (2.00)

First, convert psig. to psia. by adding 14.7 psi. Then,refering to the steam tables; 900 psia. = 532 deg.F
610 psia. = 488 deg.F
532 deg.F - 488 deg.F =44 deg.F / half hour, or 88 deg./hr (1.50)

NO. The cooldown limit of 100 deg.F/hr has not been exceeded. (0.50)

REFERENCE
J.A.F. Thermodynamics; MET.-222, pg.1-15. Enabling objectives,222.7.1 thru 222.7.5 . Tech. Spec. Thermal Limitations pg. 136 .

ANSWER 1.03 (2.00)

C is the correct answer [0.5]. The highest Xenon concentration will be in the center of the core [0.5], the high flux region from the previous operating period [0.5]. This will increase the flux levels in the area of the peripheral control rods [0.5], thus increasing their worth.

REFERENCE
J.A.F. LP. # 237.5 pg. # 10 Significant effects of Xenon.- Effect on CRW .

5.10 During startup (power = 1 watt) a rod is pulled and a 60 second period is observed.

a) With no further pulling of rods, can the operator maintain this Reactor period for 30 minutes? Explain your answer. (1.5)

b) By pulling rods can the operator maintain the 60 second period for 30 minutes from the 1 watt power level? Explain your answer. (1.5)

Answer:

a) $P = P_o e^{t/T} = 1 \text{ watt } e^{1800/60} \text{ sec} = 1.06 \times 10^{13} \text{ watt}$

$= 10^7 \text{mw}$

No (0.5)
Moderator coefficient would turn it. (1.0)

b) No (0.5)
Power at end of 30 minutes would even exceed max. MWT for Fitzpatrick. (1.0)

Ref. Equation sheet.

5.11 Fitzpatrick is operating at full power when the Feedwater Controller malfunctions causing a complete loss of feedwater. A Reactor Scram will occur on low level. During the time period <u>just prior</u> to the scram, is the Reactor Power expected to increase or decrease? Give two reasons for your answer. (2.5)

Answer:

Decrease (0.5)

1) Loss of feedwater causes less subcooling, more negative $\Delta K/K$ due to more voiding. (1.0)

2) Recirc. Runback on <20% feedflow decrease core flow. (1.0)

Ref. NET 237.3 pg. 8, F-OP-27 pg. 3.

End of Section 5

ANSWER 1.04 (2.00)

a. Adds negative reactivity [0.25] due to the increase in neutron leakage - Moderator temperature coefficient. [0.25] (0.5)
b. Adds negative reactivity [0.25] due to the increase in neutron capture in the fuel - Doppler coefficient. [0.25] (0.5)
c. Adds positive reactivity [0.25] due to the decrease in neutron leakage - Moderator temperature coefficient. [0.25] (0.5)
d. Adds negative reactivity [0.25] due to the increase in neutron leakage - Void coefficient. [0.25] (0.5)

REFERENCE
J.A.F. Reactor Theory LP.# 237.4 Objective # 237.3.1.5,and1.6 .pg.2 ; and pg. 16 Effects of Core parameters.

ANSWER 1.05 (2.50)

NO.[0.5]
Thermodynamic efficiency is a comparison of energy in versus energy out. [0.5] The increase in generator output resulted from decreasing the amount of steam diverted to the HP FW heater. [0.5] This condition requires additional energy output from the reactor to raise FW temp to the same saturation temp as before [0.5] Thus, thermodynamic efficiency of the plant has gone down. [0.5] More delta T across the heater would have caused more extraction steam to have been removed from the turbine.(0.5).

REFERENCE
JAF. Heat Transfer # 228.1.1.10 and 11 .
JAF. MET. 222.9 and 222.10 Thermodynamic Cycle Analysis.
JAF. MET. 222.10.0.3 a thru d, enableing objectives. Explain the Gross Thermal Efficiency, Net Thermal Eff.,and Gross and Net plant Elec.output.

ANSWERS -- FITZPATRICK -85/02/12-LANGE, D.

ANSWER 1.06 (2.50)

a. The subcritical neutron level is directly proportional to the neutron source strength. If the source strength doubles, the neutron level doubles. (0.50)
b. The three variables are: Source Strength, K-eff., and Time. (0.75)
c. The case when K-eff.= 0.992 will take longer to reach equilibrium. The reason for this is that one must wait for the last term in the series expansion to become insignificant. That is, until K becomes insignificant. This takes more termes as K - 1. (1.25)

REFERENCE
JAF. NET. 237.7 Subcritical Mult. and 237.7.1.4 Subcritical Power Level.
JAF. Enableing Objectives 237.7.1.1&2.

ANSWER 1.07 (3.00)

DECREASE.[0.5]
REASONS: (2 of 3 required at 1.25 each)

1. Immediately the loss of feedwater flow causes a decrease in moderator subcooling which introduces negative dk/k into the core.
2. When feedwater flow drops below 20%, the recirc. pumps will auto runback to 20%. The decrease in core flow causes an increase in voiding which also adds negative dk/k into core.
3. Decreasing level in the downcomer will reduce the available head for core circulation and will result in decreased core flow, and thus reactor power will decrease.

REFERENCE
JAF.NET- 237.4 Reactor Theory, moderator coefficient .
JAF. Recirc. and Feedwater sys. 26% flow control runback < 20% FW. flow,
F-OP-27 Recirc.Flow Control.

ANSWER 1.08 (3.00)

a. Decrease (0.5) due to increased void content in the core as flow decreases (0.5).

b. Increase (0.34) due to increased voiding in the core (0.33) and recirc pump no longer taking a suction on the annulus (0.33).

c. Decrease (0.34) due to steam flow decrease (0.33) and level increase (0.33).

REFERENCE
JAF. SDLP. 02-H, 021 Recirc sys. , F-OP-27, Recirc. Sys.
SDLP. 02B-F RX. Vessel Inst. and SDLP 02-A RX. Vessel Internals.

ANSWER 1.09 (2.25)

a. At 500 deg.F (0.25),As moderator temperature increases, neutron leakage out of the fuel bundles is increased . The control rod is exposed to a higher neutron flux, thus rod worth increases. (0.50).
b. The withdrawn rod. (0.25), Neutron flux is higher in this area, thus rod worth is greater. (0.50).
c. At position ~~24~~ 10. (0.25), Voids at the top allow more fast leakage and less thermal neutron flux, therfore greater rod worth at the bottom. (0.50)

REFERENCE
JAF. Reactor Theory. NET. 237.4 , Enabling objectives, 237.4.5.3, 1 thru 5.

ANSWER 1.10 (2.00)

At 100 % power (0.75), At 4 % power you are at operating pressure but low feedwater flow rate. NPSH is low due to T- inlet being high. As power increases , pump inlet temperature is reduced due to mixing in the downcommer T- inlet is lower so P- sat. at inlet is lower, therefor NPSH is higher. (1.25)

REFERENCE
JAF. MET, 214.9.12, pg. 12.

ANSWER 1.11 (1.50)

a. VOID COEFFICIENT (0.25), adds negative reactivity (0.25)
b. FUEL TEMP. COEFFICIENT (0.25), adds negative reactivity (0.25)
c. MODERATOR TEMPERATURE COEFFICIENT (0.25), adds positive reactivity (0.25

REFERENCE
JAF. NE1. 237.4 Reactivity Coefficients.

ANSWER 2.01 (2.50)

1. Scram discharge volume high level scram bypassed
2. Refuel Platform over the core and fuel grapple not full up
3. Grapple fuel loaded (400 lbs)
4. Frame hoist fuel loaded (400 lbs)
5. Trolly hoist fuel loaded (400 lbs)
6. Service platform loaded (400 lbs)
7. SELECTION of a second ROD (with one rod not full in)

(Seven correct answers at .357 each) = 2.50 for full credit.

REFERENCE
JAF. LP. book # 2 Tab L- Reactor Man. Control & RPIS systems. Refueling Interlocks / Mode Switch Position.

ANSWER 2.02 (2.00)

a. Yes. These valves may be closed for periodic testing and maintenance under strict Administrative Controls. (0.25)
With either valve closed, if the not drained alarm should come in, the valves should be opened. If the valves cannot be opened, commence a controlled shutdown. (0.50)
b. Reactor Pressure . (0.25)
c. NO. (0.25) This switch is only active in the Refuel or Shutdown MODE switch positions. (0.50)
d. FALSE (0.25) This Rod Block trip is never bypassed .

REFERENCE
JAF.LP. Book # 1 Tab E - CRD.HYD., Tech .Specs. sec.3.3 CRD. sys. Oper. and LP. SDLP-071-RMC &RPIS sys. Objective # 4 pg.# 5.

ANSWER 2.03 (3.00)

a. 59 gpm, accept 55 - 60 gpm (0.33)for flow; 260 psid., accept 250-270psid (0.33) for flow.; 1390-1480 (0.33) for charging water press.
b. Approx. ~~50-75~~ % or ~~26 to 47~~ gpm. (0.50) - all of it.
c. When a rod is inserted, one set of stabilizing valves, 1 valves, close (0.5) to direct 4 gpm (0.5) to the CRD and away from the cooling water header.
d.- minimum flow line (0.25)
- recirculation pump seal purge (0.25)

REFERENCE
JAF- Procd. F-OP-25 and LP.#SDLP-03C CRD. HYD. SYS.

ANSWER 2.04 (3.00)

a. Reactor Low Level (-38 ") and/or Reactor High Pressure (1120 psig.) (1.00)

126.5

b. Verify the coolant temp. in the idle loop is within 50 deg. F of the coolant temp. of the RX. vessel. (0.50)
Verify that the coolant temp. between the upper and lower regions of vessel are within 145 deg.F (0.50)

c. Closely monitor the APRM chart recorders. (0.25)
The increased flow will greatly affect RX. Power. (0.25)

d. On " A ",take the break release lever to the release position. (0.25)
On " B ", tighten both red knobs on top of the break. (0.25)

REFERENCE
JAF. F-OP-27 Recirc. sys. Pg. 12 -17.

ANSWER 2.05 (3.00)

a. 1. 75 psig
2. 450 psig
3. 85 psig
(0.25 each)

b. During Reactor Isolation(.25), the RHR sys. in the Condensing Mode, is operated in conjunction with the RCIC sys.(.25), in the case of loss of main feedwater flow(.25), to reduce or maintain RX.press.(.25).

Torus Level

c. TRUE . (0.25)

d. Condition # 1- Containment Spray control switch in Manual (.166) and containment pressure of 2.00 psig.(.166) and containment spray override keylock switch in Override (.166). (0.50 total pts.)
Condition # 2- Containment spray control switch in manual (.125) AND containment pressure of 2.00 psig. (.125) AND vessel level > 0 " (.125) AND LPCI initiation signal SEALED-IN (.125). (0.50 total pts.)

REFERENCE
JAF. LP . Book # 2- Tab-Q , SDLP # 10 RHR. Sys. and F-OP-13, RHR Sys.

ANSWER 2.06 (3.00)

a.1 - high drywell pressure > 2.~~00~~ 1.7 psig.
2 - low reactor water level (+18 ") Above T.A.F.
3 - 4 kv emerg.bus low voltage.
(0.33 each)

b.TRUE (0.50)

c. 1. Engine Overspeed
2. Loss of generator field.
3. Generator protective relays
4. High circulating current (reverse power)
5. Low lube oil press. BLOCKED IF LOCA SIGNAL PRESENT
6. High jacket water temperature. BLOCKED IF LOCA SIGNAL PRESENT
(0.25 for each correct answer)

REFERENCE
JAF. Procd. # F-OP-22, D/G Emerg. Power. and LP. # SDLP-71 A & B, 93 .

ANSWER 2.07 (2.50)

a. 1.High area temp.
2.High steam line flow. (4 required at 0.25 each)
3.Low steam supply pressure.
4.High turbine exhaust diaphragm pressure.

b. NO.(0.25) Following a turbine trip from any reason, the trip throttle valve must be manually reset at the turbine. (0.50) - except Hi Level.

c. To minimize the possibility of a water hammer or flow reversal in the turbine exhaust line. (0.75)

REFERENCE
JAF. F-OP-19, pg.# 4 & 8.

ANSWER 2.08 (2.75)

a. 1. Continuity lights go out.
2. Alarm indication.
3. Milliamp meters on back of 09-03 indicate current flow to firing ckt.
4. Decrease in power.
5. Selected pump has red light indicating pump is running.
6. SBLC pressure > reactor pressure.
7. SBLC tank level decreasing. (any six at 0.25 each)

b. YES. (0.25) If the SBLC. tank level approaches zero (0.50) or the SBLC pump begins to loose discharge pressure. (0.50)

REFERENCE
JAF. F-OP-17 pg. 4 thru 7.

ANSWER 2.09 (1.75)

a. 1. R-4A or 4B high dilution flow (.33)
   2. R-4A or 4B low dilution flow (.33)
   3. R- 4A or 4B low outlet temp. (.33)

b. A detonable mixture of hydrogen and oxygen may accumulate under these conditions. (0.50)

c. Radiation Protection Department. (0.25)

REFERENCE
JAF. F-OP-24 A Off-Gas Sys. pg. 1 - 23

ANSWER 2.10 (1.50)

a. 1. Used to calibrate individual LPRM detectors. (0.25) to map the core axial flux profile. (0.25)
   2. Used by the prosess computer (0.25) to determine MCPR and local heat flux conditions. (0.25)

REFERENCE
JAF. SDLP.-07F TIP sys. pg. 6 .

ANSWERS -- FITZPATRICK -85/02/12-LANGE, D.

ANSWER 3.01 (2.50)

a. Rx. Scram = 177"
b. SGTS starts = 177"
c. RCIC Init. = 126.5" (0.25 for each correct ans.)
d. Low Level alarm = 196.5"
e. Recirc. Pump Trip = 126.5"
f. Rx. Feed Pump Trip =222.5"

No. 2- The ATWS Recirc. Pump Trip is set at 1120 psig.(0.25). The first two relief valves lift at 1090 psig., (0.25), the second two at 1105 psig. (0.25) and the last seven at 1140 psig (0.25).

REFERENCE
JAF. SDLP. 02 B, C, D, E, F,- Reactor Vessel Instrumentation

ANSWER 3.02 (3.00)

a. half-scram
b. no action
c. half scram
d. rod block
e. scram
f. rod block
(0.50 for each correct ans.) (3.00)

REFERENCE
JAF. SDLP.-071 RMCS.
SDLP.-02H,021. Recirc. sys.
SDLP.-07A-E Neutron Monitoring Sys.

ANSWER 3.03 (3.00)

a. 2, 4, 7, ~~8~~
b. 1, 3, 8
c. 5, 6
(0 ~~.28~~ .375 each)

REFERENCE
JAF. SDLP.- 17,18 (Process and Area Rad.Monitors)
SDLP.- 38 & 01A (Condenser Air Removal and Off Gas)

ANSWER 3.04 (3.00)

a. A complete loss of RPS will cause a scram . The Recirc. pumps will run back to 26 % when less than 20 % feed flow is sensed, or vessel level is less than 196.5 without both feedwater pumps running. (0.75)

b. The EHC pressure regulator will sense a decreasing pressure at the avg. press. manifold and close the TCV,s . The turbine will trip on reverse power. The BPV,s will open if the avg. manifold press. rises above 920 psig. (0.75)

c. Level will initially shrink due to void collapse and then return to near normal with the smaller void reformation following the scram . (0.75)

d. RPS does not supply power to RX. level instrumentation.No effect.(0.75)

REFERENCE
JAF. SDLP. 71 a & b , & 93. RX. Level & Inst.
SDLP- 94-C, EHC sys.
F-OP-27, Recirc. Sys.

ANSWER 3.05 (2.50)

a. Loss of control signal.
MG Set control power transfer (3 correct ans @ .33 each)
Loss of power to the scoop tube positioner

b. 1. FW flow < 20 % runback to 26 % speed to protect against cavitation. (0.50)
2. RX. level < 196.5" without (ONE) both feedwater pumps running, runback to 44 % to protect against cavitation. (0.50)

c. Red runback light on panel 09-4 (0.25)
Recirc. Flow Limit Annunciator. (0.25)

REFERENCE
JAF. Procd. # F-OP-27 Recirc . Sys. pg. 11-13 .

ANSWER 3.06 (2.25)

a. EXPLOSION in the Air Ejector discharge piping. (0.75)
Automatic Actions:
1.Air Ejection steam supply valve, 29-PCV-107 shuts. (0.25)
2.Possible isolation of Turbine Bldg. on high airborn activity. (0.25)
3.Main Turbine and RX. feed pump trip. (22.5"hg) (0.25)
4.MSIV and Bypass valve closure on low vacuum (8"hg) (0.25)

b. Manually Scram the Reactor. (0.50)

REFERENCE
JAF. Special Procd.F-AOP-4,'Explosion in the Air Ejection Off Gas Line.'
(Symptoms / Automatic Actions / Operator Actions)

ANSWER 3.07 (3.00)

a. TRUE (0.50)
b. A stainless steel jaming button must be installed on the hoist cable.(.5
c. During refueling it provides a seal between the vessel flange and the drywell. (0.25)
During heatup and cooldown it accomodates the differential expansion that occurs. (0.25)
d. 1. More than one control rod withdrawn. (0.25)
2. Refueling platform position switch open. (0.25)

REFERENCE
JAF. F-OP-66,Refueling Equipment. pg. 51
JAF. SDLP-19, Fig. 19-2 Refueling Bulkhead and associated Bellows.
JAF. SDLP-97/A, Fuel Handeling Equipment.

ANSWER 3.08 (2.00)

A differential pressure sensor is used to confirm the integrity of the CORE SPRAY piping within the reactor vessel (between the inside of the vessel and the core shroud).
To continuously monitor the integrity of the core spray piping, a Delta P switch measures the pressure difference between the two loops, which is effectively the inside of each Core Spray sparager pipe, just outside of the Rx vessel. If the core spray sparager is intact, this pressure difference will be zero. If integrity is lost, this pressure differential will include the pressure drop across the steam seperator. Alarms at .5 psid in the control room (2.00)

REFERENCE
JAF. SDLP.-14 , Core Spray Sys.
JAF. Performance Objective # 4014 .3-02, Lic. Qual Std.
JAF. Annunciator response procd. Vol.-1, alarm 9.3.3-1

ANSWER 3.09 (2.00)

a. GEMAC. and FUEL ZONE level inst. are accurate. (0.50 each)
The WIDE RANGE YARWAY level inst. are not reliable. (0.50)
reason; Flashing in the refrence leg. (0.50)

REFERENCE
JAF. F-EOP-1 pg. # 15 .

ANSWER 3.10 (1.75)

a. 1. Restricts discharge and protects vessel internals from large D/P. (0.33)
   2. Provides signal for MSIV closure. (0.33)
   3. Provides steam flow signal to FWCS. (0.33)

b. Three solenoid operated pilot valves. (0.25)
   1. One AC and one DC for normal operation. (0.25)
   2. One test pilot solenoid for slow closure testing. (0.25)

REFERENCE

JAF. F-OP-1 Main Steam, pg. 3-6.

ANSWER 4.01 (2.00)
6- Battery Check. O.K.

a. 1. Sensitivity of the instrument to the types of radiation and/or contamination present.
2. Ranges of the instrument with respect to expected levels of radiation.
3. Limitations of the instrument with respect to humidity, temperature, etc.
4. Current calibration of the instrument.
5. Proper response to check sources. (any four at .25 each)

b. #1=b, #2=a, #3=d, #4=c (0.25 for each correct ans.)

REFERENCE
JAF. Radiation Protection Procedure 2.4.3

ANSWER 4.02 (1.75)

a. 1. Misoperation in automatic is confirmed by at least two independent process parameter indications. (0.50)
2. Core cooling is assured. (0.50)

b. 1. Maintain Core Cooling.
2. Limit the release of off-gas radiation.
3. Place the Reactor in a safe stable condition.
4. Keep the Torus bulk temp. below 120 deg. F. (any 3 at .25 ea.)

REFERENCE
JAF. F- EOP-33, Small Break Accident.

ANSWER 4.03 (2.00)

Adequate core cooling is not assured. (1.00)
With both Core Spray loops out of service and only one RHR pump available for injection, the definition of adaquate core cooling cannot be met.

REFERENCE
Mitigation of Core Damage , JAF. RX. Core Cooling Bases. Sec. 3.5, pg. 126 of Tech. Specs.

ANSWER 4.04 (2.25)

Three events;

1. Decrease in Supp. pool level; As level decreases the nitrogen must occupy a larger volume, resulting in decreased Drywell pressure.
2. Increase in barometric pressure; The Drywell pressure instruments are all refrenced to atmosphere therefore an increase in atmospheric press. causes a decrease in D/W press.
3. A decrease in D/W or chilled water temp; An increase in cooling capacity in the D/W from the D/W coolers and chillers will decrease D/W pressure. (Other reasonable ans.accepted if substantiated)
(0.25 for each event; 0.50 for each reason)

REFERENCE
JAF. Procedure,F-AOP-9 Loss of Primary Containment Integrity.

ANSWER 4.05 (1.50)

a. 1) 75 R
2) 25 R
3) 12 R

REFERENCE
JAF. Radiation Protection Procedures 2.7.4 , Emergency Exposure Guidelines.

ANSWER 4.06 (2.00)

Immediate Operator Actions;

1. Place the Mode Switch in Refuel. (0.33)
2. Insert IRM and SRM detectors into the Core. (0.33)
3. Verify the Rx. is shutdown by observing Power decrease. (0.33)
4. Confirm all control rods are inserted to or beyond position 00 by ;
   a. bypass the SDV. high level trip. (0.165)
   b. Attempt to reset the Scram signal. (0.165)
5. Once the RX. is shutdown Trip the Main Turbine. (0.33)
6. Reset N-2 supply to the drywell if it had isolated from a group -2 isolation. (0.33)

REFERENCE
JAF. F- APO-1, Reactor Scram.

ANSWERS -- FITZPATRICK -85/02/12-LANGE, D.

ANSWER 4.07 (2.50)

a. With HPCI or RCIC, if available . (0.75)
b. The SRV that discharges to the torus as far away as possible from the first SRV that actuated. (0.25) To minimize local heating of the torus water. (0.50)
c. #1- Open MSIV,s and control pressure with the turbine by-pass valves. OR, (0.50)
#2- Operate the RCIC. sys. in the Steam Condensing mode.(0.50)

REFERENCE
JAFNPP F-OP-1 sec 5 pg 15 & 16

ANSWER 4.08 (2.50)

a. Refueling GEMAC level indication. (0.50)
b. To assure adaquate coolant mixing thru natural circulation. (0.75)
c. Vessel stratification .
Loss of valid temperature indication. (2.5)
Extreme case of vessel boiling and pressurization.
(any two (2) at .75 each)

REFERENCE
F-OP-13 RHR System pg 13

ANSWER 4.09 (2.75)

a. 1. Acoustic monitor (alarm).
2. Tailpipe temp. (alarm).
3. Generator (electrical output). (any five (5) @ 0.25 each)
4. Torus temp.
5. RPV. pressure and water oscillations.
6. SORV solenoid indicating light (showing energized state).
b. Direct an operator to panel 09-45 (relay room) (0.25) and have him pull the control power fuses for the SORV. (0.50)
c. 1. Reactor building - 300 ft. elev. North wall. (0.50)
2. Try cycling the C/S again, and if no action,open PS. ckt.brk inside the Remote Relief Valve Panel. (0.50)

REFERENCE
JAF. Procd. F-AOP-36,Stuck Open Relief Valve.

ANSWERS -- FITZPATRICK -85/02/12-LANGE, D.

ANSWER 4.10 (3.00)

a. 1-Scram the reactor (0.5)
   2-Trip the main turbine (0.5)
   3-Leave the Mode switch in Run (this insures MSIV closure) (0.50)
b. RX. bldg. 300 ft. elev. NE. inst. racks. (0.50)
c. Manually trip the turbine at the front standard. (0.25)
   De-energize RPS from Dist. panels in the relay room. (0.25)
   Open RPS MG set supply breakers. (0.25)
   Isolate and vent inst. air to scram valves. (0.25)

REFERENCE
JAF. F- EOP-28 Plant Shutdown From Outside The Control Room.

ANSWER 4.11 (2.75)

a. 1. RPV water level below 177"
   2. RPV press.above 1045 psig.
   3. D/W press.above 2.7 psig.
   4. MSIV isol. required.
   5. RX. scram required, with RX. power > 2.5 % or cannot be determined.
   6. When directed by another EOP.
   (0.25 for each correct ans.)
b. 1. Supp. pool avg. water temp.> 75 deg.F
   2. D/W avg.temp. > 135 deg. F
   3. D/W press. > 2.7 psig.
   4. Supp. pool water level > 0.0 in.
   5. Supp. pool water level < 1.5 in.
   (0.25 for each correct ans.)

REFERENCE
JAF. EOP-2 and EOP-4.

TEST CROSS REFERENCE

QUESTION	VALUE	REFERENCE
01.01	2.25	DJL0000090
01.02	2.00	DJL0000091
01.03	2.00	DJL0000092
01.04	2.00	DJL0000093
01.05	2.50	DJL0000094
01.06	2.50	DJL0000095
01.07	3.00	DJL0000096
01.08	3.00	DJL0000116
01.09	2.25	DJL0000129
01.10	2.00	DJL0000130
01.11	1.50	DJL0000131
	25.00	
02.01	2.50	DJL0000097
02.02	2.00	DJL0000098
02.03	3.00	DJL0000099
02.04	3.00	DJL0000100
02.05	3.00	DJL0000101
02.06	3.00	DJL0000102
02.07	2.50	DJL0000121
02.08	2.75	DJL0000122
02.09	1.75	DJL0000133
02.10	1.50	DJL0000134
	25.00	
03.01	2.50	DJL0000103
03.02	3.00	DJL0000104
03.03	3.00	DJL0000105
03.04	3.00	DJL0000106
03.05	2.50	DJL0000124
03.06	2.25	DJL0000125
03.07	3.00	DJL0000126
03.08	2.00	DJL0000127
03.09	2.00	DJL0000128
03.10	1.75	DJL0000136
	25.00	
04.01	2.00	DJL0000108
04.02	1.75	DJL0000110
04.03	2.00	DJL0000111
04.04	2.25	DJL0000112
04.05	1.50	DJL0000113
04.06	2.00	DJL0000117
04.07	2.50	DJL0000118
04.08	2.50	DJL0000119
04.09	2.75	DJL0000123
04.10	3.00	DJL0000132

TEST CROSS REFERENCE

QUESTION	VALUE	REFERENCE
04.11	2.75	DJL0000135
	25.00	
	100.00	

Question and Answers to James A. Fitzpatrick SRO Exam - 2/12/85

5.0 Theory of Nuclear Power Plant Operation, Fluids and Thermodynamics (25.0)

5.1 Assume Fitzpatrick is at 100% power and flow. Condenser vacuum decreases thus causing the hotwell temperature to increase to the saturation temperature of the new vacuum. This in turn causes the temperature of the feed water to increase.

a) Would Reactor power increase or decrease? Explain. (1.0)

b) Would Core Flow increase or decrease? Explain. (1.0)

Answer:

a) Decrease (0.25) Increased temperature of feedwater would increase negative reactivity insertion of moderator coefficient (0.75).

b) Increase (0.25) Core Flow would increase due to reduced two phase pressure drop across core. (0.75).

Ref. NET 237.4 pg. 13 H228.8 pg. 31 H228.8 fig. 3-4.

5.2 Will the recirculation pumps have more NPSH at 4% or 100% power? Explain your answer fully. (2.0)

Answer:

More NPSH at 100% power. (0.5)

At 4% you are at operating pressure but low feedwater flow the temperature at the recirculation pump inlet is nearer to saturation than at 100% power where there is a higher percentage of feedwater in the downcomer causing the temperature at the inlet of the recirculation pumps to be lower. (1.5)

Ref. MET 214.9 pg. 16.

5.3 State whether the following increase, decrease or have no effect on Control Rod worth.

a) Rod Density decrease. (2.5)

b) Gadolinium burning out.

c) Void fraction decrease.

d) Moderator temperature decrease.

e) Fuel temperature increase.

Answer:

a) Increase

b) Increase

c) Increase

d) Decrease

e) No effect

(0.5 for each correct answer)

Ref: NET 237.4 pg. 19, 18.

5.4 While Fitzpatrick is operating at 90% power, extraction steam to the highest pressure feedwater heater is removed. An engineer observed that the turbine load increased by 20 MW electric and concluded that this action has improved (increased) the plant's thermodynamic efficiency (not heat rate).

Is this conclusion correct? Explain your answer fully. (Include what caused electrical output to increase). (2.5)

Answer:

No (0.5) thermo efficiency is a comparison of Energy In to Energy Out (0.5). The increase in output results from no steam being diverted to the high pressure feedwater heater (0.5). Because the feedwater is now cooler, more energy from the reactor is required to bring the water up to saturation temperature (0.5) thus thermo efficiency is down (0.5).

Ref. H-228.8 pgs. 22-23.

5.5 Fitzpatrick has been operating at 100% power for 3 weeks, when the Reactor scrams. You begin startup within 3 hours.

a) Which control rods have the highest worth? Explain. (1.0)

b) Briefly explain why you must approach criticality very carefully under these conditions. (0.5)

Answer:

a) Edge Rods (0.5) Xenon poisoning is highest where the highest flux was, in the center of the core - thus edge rods have the highest worth.(0.5) (1.0)

b) Because the high differential notch rod worth of edge rods can cause short periods. (0.5)

Ref. NET 237.5 pgs. 2 and 4, OP65 pg. 6.

5.6 What are the names of the two major neutron reactions which occur with U^{238} fuel during core life? For each reaction state whether the reaction produces a positive or negative reactivity effect and which (if any) of the reactions occur as a result of the Doppler effect? (2.0)

Answer:

Fast Fission (0.25); Resonance Capture.(0.25) (0.5)

Fast Fission → Positive reactivity (because it contributes more neutrons to overall neutron population). (0.5) Resonance Capture → negative reactivity (because it absorbs neutron - thus reducing overall neutron population). (0.5)

Resonance Capture. (0.5)

Ref. NET 227.1 pg. 4 NET 227.2 pg. 4.

5.7 During a backshift, Standby Liquid Control (SLC) was inadvertently initiated and almost immediately stopped while the plant was at 90% power. You determine that the total SLC pump run time was [15 seconds]. While making your reports to plant management, you are asked to determine if SLC ran long enough to inject any boron into the Reactor vessel. From other sources available in the control room, you have the following information:

1) Length of pipe from SQUIB VALVE to REACTOR VESSEL - 100 FEET

2) SLC piping - 2 INCH INSIDE DIAMETER PIPING

3) SLC pump capacity - 40 GPM

a) Did boron reach the vessel? (1.0)

b) How long did it, or would it take for boron to reach the vessel? (Show all work, consider instantaneous pumn capacity). (1.5)

Answer:

a) No (1.0)

b) Volume - $\pi r^2 L = (3.14) \frac{1}{12}^2 (100) = 2.18 \text{ ft}^3$

$(2.18 \text{ ft}^3) \times 7.48 \frac{\text{gal}}{\text{ft}^3} = 16.3$ gallons. (1.0)

(16.3 gallons)/(40gpm) = .408 min = 24.5 seconds. (0.5)

Ref. NRC Exam Bank 0005421.

MET - 214.3 pg. 5.

5.8 Fitzpatrick has three (3) thermal safety limits. List these limits and briefly describe the purpose of each. (3.0)

Answer:

1) MCPR (1.07) - to insure 99.9% of the fuel rods do not experience transition boiling during a transient.

2) LHGR - to 13.4kw/ft EOC - to insure 1% plastic strain on cladding is not exceeded.

3) MAPLHGR - to insure that clad temperature remains below 2200°F during loca.

(0.25 for each limit, 0.75 for proper description).

Ref. Heat Transfer H-228.9 pg. 39.

5.9 Concerning the figure given below (for thermodynamic cycle analysis):

a) Identify the points corresponding to the high pressure turbine inlet, <u>and</u> low pressure exhaust to the hotwell. (0.5)

b) <u>Explain</u> why there is no temperature increase between points 4 and 1. (0.5)

c) **What does the horizontal portion of the line between** point 2 and point 3 correspond to. (Be specific). (0.5)

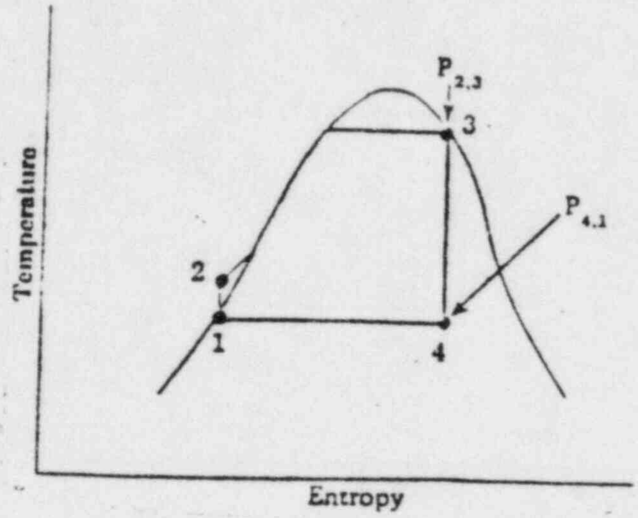


Figure 1

Answer:

a) 3, 4 (0.5)

b) Vapor is condensing to liquid in a saturated environment which is at the saturation temperature in the condenser. (0.5)

c) The vapor generation portion of the core where liquid and vapor are saturated. (0.5)

Ref. Thermo Lesson Plan MET 222.10 pg. 5.

5.10 During startup (power = 1 watt) a rod is pulled and a 60 second period is observed.

a) With no further pulling of rods, can the operator maintain this Reactor period for 30 minutes? Explain your answer. (1.5)

b) By pulling rods can the operator maintain the 60 second period for 30 minutes from the 1 watt power level? Explain your answer. (1.5)

Answer:

a) $P = P_o e^{t/T} = 1 \text{ watt } e^{1800/60 \text{ sec}} = 1.06 \times 10^{13}$ watt

$= 10^7$ mw

No (0.5)
Moderator coefficient would turn it. (1.0)

b) No (0.5)
Power at end of 30 minutes would even exceed max. MWT for Fitzpatrick. (1.0)

Ref. Equation sheet.

5.11 Fitzpatrick is operating at full power when the Feedwater Controller malfunctions causing a complete loss of feedwater. A Reactor Scram will occur on low level. During the time period just prior to the scram, is the Reactor Power expected to increase or decrease? Give two reasons for your answer. (2.5)

Answer:

Decrease (0.5)

1) Loss of feedwater causes less subcooling, more negative $\Delta K/K$ due to more voiding. (1.0)

2) Recirc. Runback on <20% feedflow decrease core flow. (1.0)

Ref. NET 237.3 pg. 8, F-OP-27 pg. 3.

End of Section 5

6.0 Plant System Design, Control and Instrumentation (25.0)

6.1 Concerning the diesel generator system.

a) With the engine control switch in standby what conditions or actions will cause Emergency Diesel Generator No. 2 to Start? (Include set points). (1.5)

b) If the field flash circuitry of the diesel generator failed to operate, will the diesel generator still start and produce power? Explain. (1.0)

c) Explain the ECCS loading sequence on the Diesel generator from the time of initiation to full loading. (Give time, in seconds. Include times that diesel and loads start and are at speed). (1.0)

Answer:

a) High Drywell Pressure (0.25); greater than 2.7 psig (0.25)
low Reactor level (0.25); 59.5 inches (0.25)
4KU emergency bus low voltage. (0.25)
Control room switch to start. (0.25)

b) Yes (0.5)
The generators are still capable of energizing due to residual magnetism. (0.5)

c)

0 second	Diesel initiation.
10 seconds	Diesel up to speed.
11 seconds	First RHR pump starts.
16 seconds	Start second RHR pump. (First RHR pump up to speed).
21 seconds	Start core spray pump. (Second RHR pump at speed).

(0.2 for each correct answer, values acceptable ±10%).

(may also get that if one diesel does not start, the second RHR pump will not start.)

Ref. F-OP-22 pg. 6, 7, and 39.

6.2 With regard to the Reactor Core Isolation Cooling System (RCIC).

a) What five (5) signals cause automatic isolation of the RCIC Steam line isolation valves 13-MOV-15 and 13-MOV-16? (1.0)

b) If the RCIC turbine received and inadvertent trip signal, which immediately cleared, would the turbine restart on a valid auto initiation with no operator action? (Briefly explain). (1.0)

c) There is a caution which states "Do not operate the RCIC turbine at a speed below 2,200 RPM for an extended period of time". What is the reason for this caution? (1.0)

Answer:

a) Manual (only if auto initiated)
High Area temperature.
High Steam line flow.
Steam Supply Lower Pressure.
Turbine Exhaust Diaphragm High Pressure.

(0.2 for each correct answer)

(Manual not necessary. If not included 0.25 for each correct answer)

b) No, (0.25)

Following a Turbine trip from any cause, the trip throttle valve must be manually reset at the turbine. (0.75)

c) To minimize the possibility of water Hammer (0.5) or flow reversal (0.5) in the turbine exhaust

(may also say reduced speed will reduce cooling water to barametric condenser and lube oil - no credit taken off for this answer)

Ref. F-OP-19 pg. 4 and pg. 8.

6.3 As regards the Standby Liquid Control System (SBLC) Operating Procedure (F-OP-17).

a) After initiation of the SBLC system, is it permissable to shut the system down? (If not why, if so under what conditions?) (1.5)

b) After a valid initiation, list six (6) control room indications that you could use to verify that the SBLC system is operating properly and injecting into the reactor vessel. (2.0)

Answer:

Yes, (0.5)

a) Control tank level approaches zero, pump begins to loose discharge pressure, or all rods in per EOP-3.
(0.5 for each of two correct answers)

b) Continuity lights go out
Alarm indication
Milliamp meters mounted on back of 09-03 panel indicate
Low current flow to firing circuits
Power should decrease
Selected pump has red light indicating running
SBLC pressure >RV pressure
SBLC tank level decreasing
RWCU system isotates.

(0.33 per correct answer, any 6).

Ref. F-OP-17 pg. 4, 5, 6 and 7.

6.4 With regard to HPCI:

a) Where are the HPCI rupture diaphrams located <u>and</u> what are their purpose? (1.0)

b) Describe the valving sequence of the HPCI suction valves from the CST and torus during an auto transfer from the CST to the torus. Briefly state the reason for this sequence. (1.0)

Answer:

a) On the exhaust line in the HPCI room (0.5) to protect the exhaust line from over pressure. (0.5)

b) The CST suction valve stays open till the torus suction valves are open. (0.5) This is to insure a continuous suction to HPCI during the change over. (0.5)

Ref. F-OP-15 pg. 6, SDLP 23 pg. 13.

6.5 Concerning the Main Steam System:

a) Following a failure of the air supply to the accumulators how many times can an operator actuate (manually cycle open and close) a relief valve from the air supplied by the valve's accumulator? (0.5)

b) What are the two (2) purposes of the valve interlock which permits two of the four steam lines to be closed without scramming the Reactor. (1.0)

c) List Three (3) functions of the Main Steamline Flow restrictors. (1.5)

Answer:

a) Five (5). (0.5)

b) To provide flexibility for testing (0.5) and to provide for possible malfunction of steam line isolation valves during plant operation. (0.5).

c) 1) Restricts discharge

2) Protects vessel internals from large Δp.

3) Provides signal for MSIV closure.

4) Provide steam flow measurement to FWCS.

5) Steam flow indication on 9-5 panel

(0.5 for each of three correct answers)

Ref. F-OP-1 Pg. 4.

6.6 You suspect that a relief valve is stuck open. List five (5) instrument indications in the control room, including back panels, that you would use to confirm that a relief valve was stuck open. (2.5)

Answer:

1) Acoustic monitor (alarms).

2) Tailpiece temperature (alarm).

3) Generator (electrical output).

4) Torus temperature.

5) RPV pressure and water oscillations.

6) Relief valve solenoid indicating light (showing energized state).

(0.5 per correct answer, any 5).

Ref. FSP-06 pg. 2 of 3.

7) Steam flow feedflow mismatch and level decrease.

8) Plant efficiency decrease.

9) (Other acceptable answers.)

6.7 Concerning the Recirculation System (Operating Procedure F-OP-27).

List the two (2) conditions that will cause a recirc runback, the percent of pump speed the runback is set for and explain the reason for the runback. Also which of the runbacks seal in? (2.0)

Answer:

Feedflow less than 20% (0.25) or discharge valve not full open (0.25) causes a recirc runback, to 26% recirc (0.5) to protect against cavitation, prevent axial thrust on pump with discharge valve not full open (0.5).

Reactor Vessel level less than 196.5" without both feedwater pumps running (0.5), limit is 44% (0.5) to get you into the range of one feedpump to prevent low level scram.

The 44% runback.

Ref. F-OP-27 pg. 4, 15, and 13

6.8 The reactor water cleanup system (F-OP-28) is operating in the blowdown mode and discharging to the condenser. Indications are as follows:

Cleanup inlet flow = 120 GPM
Cleanup flow to condenser = 50 GPM
Cleanup flow to reactor is = 0 GPM
(leak in pump room)

Non Regen outlet = 225°F

Using these indications list three (3) auto actions that should have occurred in the reactor water clean up system (include the parameter(s) which should have caused each action. Setpoints are required. (3.0)

Answer:

a) System should have isolated, inboard and outboared; isolation valves should have shut (0.5) due to high nonregenerative Hx outlet (0.5) (140°F) (0.5). Leak should have caused high temperature isloation in the pump room (0.5) at T ambient +40°F (0.5), RWCU pumps should have tripped due to closing of any isolation valve (0.5).

Ref. F-OP-28 pgs. 3 and 8.

6.9 Concerning neutron monitoring instrumentation:

a) What is required regarding LPRM inputs to an APRM, for the APRM to be considered operable? (1.0)

b) If the requirements of part "a" were not met, would you be required to manually trip the circuitry? Explain why. (1.5)

Answer:

a) To consider an APRM operable there must be at least two (2) LPRM's per level (0.5) in addition to a total of at least 11 LPRM inputs must be operable. (0.5)

b) Yes, (0.5). If the total number of LPRM's falls below 11, an APRM inop. signal would automatically be generated by the circuitry but if the 2 LPRMS per level is not met, the circuitry will not auto generate an inop. This must be manually initiated. (1.0).

Ref. F-OP-16 pg. 9

End of Section 6

7.0 <u>Procedures - Normal, Abnormal, Emergency and Radiological Control</u> (25.0)

7.1 In accordance with F-AOP-1 "Reactor Scram":

a) Following a reactor scram from full power, <u>if</u> reactor vessel is <u>not</u> isolated, <u>how</u> is reactor water level and pressure to be maintained? With the Reactor vessel isolated, <u>how</u> is the reactor water level and pressure to be maintained? (1.5)

b) If instrument nitrogen is lost to the drywell instrumentation what must be done? (0.75)

Answer:

a) Level-Feedwater pumps if possible, RCIC and/or HPCI. (0.75)
Pressure - Bypass valves

Level - RCIC and/or HPCI. (0.75)
Pressure - steam condensing mode of RHR, HPCI, RCIC and/or safety relief valves.

b) Line drywell instrumentation up to instrument air, only if instrument nitrogen can not be restored. (0.75)

Ref. NRC Exam bank RRP0000203 F-AOP-1 pg. 3 and 2.

7.2 With regard to the Reactor Building Closed Loop Cooling System (RBCLC):

a) Your are cautioned against large temperature swings in the RBCLC. What are the two (2) purposes of this caution (i.e., what adverse effect could a rapid temperature increase have and what adverse affect could a large temperature decrease have)? (2.0)

b) During the winter, if the temperature of RBCLC can not be held at or above 75°F, with TCV-101 fully open and two coolers in operation. What action should you take? (1.0)

Answer:

a) A rapid increase in temperature could raise drywell pressure significantly. (1.0)
decrease could cause drywell to torus differential to go out of specification low ($\geq$ 1.7 psig). (1.0)

b) Throttle service water flow and if still can't reduce temperature, remove one of the operating Hx from operation. (1.0)

Ref. F-OP-40 pg. 5

7.3 A severe fire causes the NCO to leave the control room before he can shut the plant down. List the four (4) methods, in order of preference, that the NCO is supposed to do to scram the Reactor. (2.5)

Answer:

1) Manually tripping the turbine at the front standard.

2) De-energizing the RPS from the distribution panels in the relay room (panel 5-6A, RPS A; Panel 5-6B, RPSB).

3) Opening the RPS MG set supply or output breakers in the electric bay.

4) Isolating and venting instrument air to the scram valves.

Underlined items not required; 0.5 points per right answer, 0.5 points for proper sequence).

Ref. F-AOP-43 pg. 2 of 4.

7.4 According to F-AOP-12 (Loss of Instrument Air):

a) Identify four (4) automatic actions that should have occurred as instrument air header pressure decreased to 80 psig. (2.0)

b) Does a loss of instrument air require a manual scram? If not - why and if so under what conditions? (1.0)

Answer:

a) Loading of operating air compressors at 110 psig decreasing.
Start of standby air compressors at 100 psig decreasing.
Closure of 39-FCV-110, service air isolation, at 95 psig decreasing.
Closing of 39-FCV-111, breathing air isolation at 85 psig decreasing.

(0.5 per correct answer, set points not required).

b) Yes - If pressure continues to decrease and corrective actions cannot be accomplished immediately, or if an automatic scram occurs. (1.0)

Ref. USNRC Exam bank RRP0000201 F-AOP-12 pg. 2.

7.5 During rapid RPV depressurization to below 500 psig:

a) What level instrumentation will provide an accurate measure of vessel level? (1.0)

b) What level instrumentation is not reliable under this condition and why? (1.0)

Answer:

a) Gemac and Fuel Zone.

(0.5 for each correct answer)

b) Wide Range Yarway; Flashing in the reference leg.

(0.5 for each correct answer).

Ref. F-EOP-1 pg. 15.

7.6 According to EOP cautions (F-EOP-1) under what condition(s) can an ECCS system be secured or placed in manual mode? (1.5)

Answer:

If by at least two independent in, cations (0.5)

1) Misoperation in automat., mode is confirmed (0.5) or
2) Adequate core cooling is assured (0.5)

Ref. F-EOP-1 pg. 21.

7.7 According to EOP cautions (F-EOP-1) under what three (3) conditions would you as an SRO, be permitted to exceed the cooldown rate permitted by technical specifications? (1.5)

Answer:

1) Conserve RPV water inventory.

2) Protect primary containment.

3) Limit radioactivity release to the environment.

(0.5 per answer).

Ref. F-EOP-1 pg. 27.

7.8 According to F-EOP-2 (RPV control) and F-EOP-4 (Primary Containment Control):

a) List the RPV Control (F-EOP-2) entry conditions (including setpoints). (1.5)

b) List the entry conditions for Primary Containment Control (F-EOP-4) including setpoints. (1.25)

Answer:

a) 1) RPV water level below 177 in.

2) RPV pressure above 1045 psig.

3) Drywell pressure above 2.7 psig.

4) When MSIV isolation is required.

5) Whenever a reactor scram is required and reactor power is above 2.5% or cannot be determined.

6) When directed by another emergency operating procedure.

(0.25 points per answer).

Ref. F-EOP-2.

Answer:

b) 1) Suppression pool water average temperature >95°F.

2) Drywell average temperature >135°F.

3) Drywell pressure >2.7 psig.

4) Suppression pool water level >0.0 inches.

5) Suppression pool water level <-1.5 inches.

(0.25 for each correct answer).

Ref. F-EOP-4 pg. 3.

7.9 How is Boron injected upon the failure of SBLC valves to actuate? (0.5)

Answer:

CRD (0.5)

Ref. F-EOP-3 pg. 35 or F-AOP-37.

7.10 According to F-AOP-24 (Stuck Control Rod):

a) List the actions and any specific precautions/limitations this procedure directs you to do to move or free a stuck rod. (1.0)

b) If you are unsucessful in freeing the control rod what **two (2) actions must be taken.** (1.0)

Answer:

a) Increase drive water pressure.(0.25) If R_x is <650 psig limit drive water ΔP to $\leq$600 psid.(0.25) Attempt to purge air from drive (0.25) (can only be done above 20% power with RSCS out of service).
Vent drive piping.(0.25)

b) Electrically disarm directional control valves (0.5) rearrange Control Rod Pattern or shut reactor down.(0.5)

Ref. F-AOP-24

7.11 According to F-AOP-31 (Loss of Condenser Vacuum) what actions can be taken to minimize the rate at which vacuum is decreasing? (2.0)

Answer:

Trip the recombiner.
Reduce reactor power.
Place spare air ejectors inservice.
Shut the vacuum drag valve.

(0.5 for each correct answer).

Ref. F-AOP-31 pg. 3.

7.12 In the event that a fuel assembly is dropped during fuel handling, list the actions you would take to mitigate the consequences of the dropped assembly. (2.0)

Answer:

Cease operation of the refueling equipment; refuel floor shall be immediately evacuated.
The Control Room shall be notified by the licensed operator.
The Control Room Operator shall sound the evacuation alarm and evacuate the Drywell and Reactor Building.
(plus any other reasonable actions)

Ref. F-EOP-31 pg. 2
or F-AOP-44

End of Section 7

8.0 Administration Procedure, Conditions, and Limitations (25.0)

8.1 According to Radiation Protection Procedures:

a) When are extended RWP's used and what is the maximum allowable time an extended RWP can be issued for? (1.5)

b) Under what two (2) conditions would you not use an extended RWP? (0.5)

Answer:

a) For certain routine repetitive functions throughout the plant, (1.0) 1 year (0.5)

b) Where neutron radiation (0.5) is present and where airborn radiation (0.5) signs are posted. (1.0)

(also acceptable: unknown conditions, contaminated level >500,000 DPM/cm^2, 10R/hr Beta-Gamma, maintenance in radiation area)

Ref. Radiation Protection Procedures pg. 14 and 15, 29.

8.2 Define or explain the following term.

Secondary Containment Integrity (2.0

Answer: Reactor Building is in tact (0.5).

At least one door in each access opening is closed (0.5).

The Standby Gas Treatment System is operable (0.5).

All automatic ventilation system isolation valves are operable or secured in the isolated position (0.5).

Ref. Tech Specs pg. 5.

8.3 According to Shift Relief and Log Keeping (ODSO-4):

a) ODSO-4 states that each normally accessible pump will be checked for abnormal conditions. List four (4) of these abnormal pump conditions that are to be checked. (1.0)

b) The off going shift supervisor or assistant shift supervisor shall prepare Section "C" of the shift turnover checklist. What is included in Section "C" of the shift turnover checklist? (Include four (4) items that must be entered). (2.0)

Answer:

a) Excessive packing leak off.
Excessive vibration.
Proper oil levels in bearings.
Proper suction and discharge pressure and flow.
Excessive bearing temperatures.
Lubrication.

(0.25 for each correct answer up to four (4) correct answers).

b) Technical Specification items (1.0)
Brief description, action statement, technical specification no., date and time of expiration, actual time, date.

(0.25 for each correct up to four (4) correct)

Ref. ODSO-4 pgs. 5, 6, and 15.

8.4 Which of the following events must be reported to the NRC <u>within</u> 1 hour after the event has been discovered? (2.5)

a) Receipt of Special Nuclear Material.

b) Theft of Source Material.

c) Changes in Security Plan made due to attempted theft of Special Nuclear Material.

d) Attempted theft of Special Nuclear Materials.

e) Unaccounted for SNM (special nuclear materials) exceeding applicable limits.

f) Report of radioactive contamination on package of radioactive materials.

g) Severe accident involving licensed materials.

h) Loss of licensed materials.

i) Licensed operator becomes disabled due to a auto injury.

j) Personnel exposure of a terminated employee.

Answer:

b, d, f, g, and h.

(0.5 for each correct answer).

Ref. Rules of Practice PSO #6 pg. 21.

8.5 Fitzpatrick is operating at 90% power. You are to be the Shift Supervisor From 12:00 midnight - 8:00 a.m. shift. During shift change you are informed that during your shift the pressure suppression - Reactor building suppression chamber vacuum breaker instrumentation surveillence is to be performed. You are further told it has been five (5) months since this surveillence test was last performed. The instrument technicians performing the surveillance tests inform you that one of the vacuum breaker's pressure switch is in-operable and the vacuum breaker pressure switch has a as-found setpoint of 1.0 psid. They also inform you that the in op pressure switch must be replaced and it will take two (2) weeks to get a new switch.

XXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXX
X NOTE: USE THE ATTACHED SECTION OF THE TECHNICAL SPECIFICATIONS X
X TO ANSWER THIS QUESTION. FULLY REFERENCE ALL APPLICABLE SECTIONS X
X OF THE T.S. YOU USE TO DEVELOP YOUR ANSWER. X
XXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXX

a) What continued plant operation situation exists due to the above conditions? (1.5)

b) What action must be taken as a result of this surveillence test? (1.5)

Answer:

a) You are in violation of Technical Specifications on the maximum surveillance interval. This is a reportable occurance. Section 4.7.4, pg. 177. (1.5)

b) You are above the setpoint permitted by Technical Specifications on the 1.0 psig. Orderly shutdown shall be initiated and the Reactor shall be in cold shutdown within 24 hrs. Section 3.7.8.a, pg. 180a. (1.5)

Ref. Technical Specifications pgs. 177 and 180a.

8.6 Define or explain the following term.

Primary Containment Integrity (2.5)

Answer:

a) Primary containment integrity means that the drywell and pressure suppression chamber are intact (0.5) and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the Reactor Coolant System or containment which are not required to be open during plant accident conditions are closed. These valves may be opended to perform necessary operation activities. (0.5)

2. At least one door in each airlock is closed and sealed. (0.5)

3. All automatic containment isolation valves are operable or de-activated in the isolated position. (0.5)

4. All blind flanges and manways are closed. (0.5)

Ref. Technical Specifications 4, 5

8.7 Concerning plant staffing requirements.

a) What is the minimum operations shift crew composition during full power operation, and during refuel or cold shutdown? (1.0)

b) Does a shift technical advisor have to be in the control room at all times? (Explain). (1.5)

Answer:

a)

Full Power	Cold Shutdown/refuel
2 SRO (licensed)	1 SRO (licensed)
2 SRO (licensed)	1 RO (licensed)
2 O (unlicensed)	1 O (unlicensed)
STA	

(0.5 for full power correct, 0.5 for cold shutdown correct)

b) No. (0.5) He is required to be on site and readily available to the control room except during the cold shutdown or refuel mode. (1.0)

Ref. Technical Specifications 247, 247a, ODSO-1.

8.8 In accordance with Technical Specifications the reactor was scrammed due to suppression temperature >110°F. The reactor is now in hot shutdown suppression pool cooling is on and the suppression pool water temperature = 98°F. (2.25)

Can you place the reactor mode switch into startup? Explain your answer fully.

Answer:

No, (0.5) you must be below 95°F in suppression pool (0.75). Technical Specifications prohibits you from entering into an operational condition while relying on an action statement. (1.0)

Ref. Technical Specifications 3.0.D pg. 30a; 3.7A pg. 166.

8.9 Fitzpatrick is in the process of starting up, (mode switch in startup), with all systems and components normal except that "A" IRM is inoperable and bypassed. The "C" IRM now loses power and declared inoperable. Can Fitzpatrick continue startup in this condition? (Explain). (1.25)

Answer:

a) Yes (0.5). Place the RPS A channel in the tripped position within one (1) hour. (0.75)

Ref. Technical Specifications Table 3.1-1 pg. 42, 3.0.C pg. 30.

8.10 The technical specification limit on reactor coolant chloride concentration is different at low steaming rates (less than 100,000 lb/hr) than at high steaming rates (greater than 100,000 lb/hr).

a) Which condition (low steaming or high steaming) permits the highest reactor coolant chloride concentration? (0.5)

b) Explain the basis for your answer in part a. (1.0)

Answer:

a) High steaming rate (0.5)

b) At low steaming rates a more restrictive limit is established to assure chloride-oxygen contamination are kept within limits at steaming rates above 100,000 lb/hr boiling causes deoeration of the reactor water thus maintaining oxygen concentration at low levels. (1.0)

Ref. Technical Specification 3.6C pg. 140, Bases pg. 149.
Figure 4.6.1 pg. 164

8.11 <u>Define</u> <u>or</u> <u>explain</u> the following terms:

a) LCO (0.5)

b) Core alteration (0.5)

c) Startup/Hot Standby (0.5)

d) operable (0.5)

Answer:

a) The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. (0.5)

b) The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. (0.5)

c) The reactor mode switch is in the Startup or Hot Standby mode position, the reactor coolant is greater than 212 degrees F and the reactor pressure is less than 1005 psig (0.5)

d) System operable when it is capable of performing its intended function. (0.5)

Ref. Technical Specifications Pg. 3, 1, 2, 4.

End of Section 8

End of Test