ENCLOSURE 1

EXAMINATION REPORT - 50-259/OL-88-01

Facility Licensee: Tennessee Valley Authority
6N 38N Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Facility Name: Browns Ferry Nuclear Plant

Facility Docket No.: 50-259, 50-260, and 50-296

Written examinations and operating tests were administered at the Browns Ferry Nuclear Plant near Decatur, Alabama.

Chief Examiner: Jesse A. Arildsen — 8/23/88 Date Signed

Chief Examiner:* for John F. Munro — 8/23/88 Date Signed

Approved by: Kenneth E. Brockman, Chief, Operator Licensing Section 2 — 8/22/88 Date Signed

*Operating Test Only

Summary:

Examinations on March 23 and May 2-12, 1988.

Operating tests were administered to 22 candidates, 19 of whom passed. 22 candidates were administered written examinations, 18 of whom passed.

Based on the results described above, 9 of 14 RO's passed and 7 of 8 SRO's passed.

10 of 29 (34%) of the changes made to the written examination as a result of facility comments were due to inadequate or incorrect facility training material supplied to the examiners for examination preparation. Reference material submitted to the NRC should accurately reflect the current plant configuration so that post-examination modifications are minimized. Your attention is also invited to additional concerns, problems, and two Inspector Followup Items discussed in the following report details.

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REPORT DETAILS

1. Facility Employees Contacted:

   *R. J. Johnson, TVA, Director, Nuclear Training
   *R. G. Jones, BFNP, Supervisor, Operations Training
   *N. C. McFall, BFNP, Licensing Staff
   *C. T. Dexter, BFNP, Training Staff
   *T. Albright, BFNP, Training Staff
   *T. E. Mayfield, BFNP, Training Staff
   *E. Howard, BFNP, Training Staff
   *J. Marshall, BFNP, Training Staff
   M. A. Morrow, BFNP, Training Staff
   C. Leach, BFNP, Training Staff

   *Attended Exit Meeting

2. Examiners:

   *J. A. Arildsen, NRC, RII
   M. A. Sullivan, Sonalysts, Inc.
   G. T. Hopper, NRC, RII
   K. E. Brockman, NRC, RII
   M. W. Parrish, EG&G, Idaho
   J. F. Munro, NRC, RII
   W. Cliff, Battelle
   M. Daniels, Sonalysts, Inc.

   *Chief Examiner

3. Examination Review Meeting

   At the conclusion of the written examinations, the examiners provided your training staff with a copy of the written examination and answer key for review. The NRC Resolutions to facility comments are listed below.

   Question 1.02/5.14

   Comment accepted. Partial credit was awarded for the acronyms recommended by the facility, since these relate to the actual thermal limits and are calculated by the Process Computer to easily identify proximity to thermal limit violation. In addition, "Pin power at any node" was not a required part of the answer for LHGR.

   Question 1.04/5.02

   Comment not accepted. The question specifically stated that heating power was at 40 on range 7. Therefore, the assumption that POAH was 40/125 is incorrect. In addition, the minimum permissible stable positive period allowed is 60 second per GOI-100-1. A 30 second period is not the "minimum permissible stable positive period".

Question 1.16/5.16

Comment accepted. Full credit was awarded for the statement "EHC follows increasing pressure by opening CVs" for part (c).

Question 2.02/6.01

Comment accepted. The answer to part (c) was changed to false.

Question 2.05/6.02

Comment partially accepted. Part (a) of the question was deleted. The point value of the question was adjusted to .50 points.

Question 2.07

Comment accepted. The answer key was changed to reflect acceptable answers based upon assumptions made by each individual candidate on total CRD hydraulic system flow.

Question 2.08/6.04

Comment accepted. The answer to part (b) was changed to allow full credit for "reduction of core spray performance".

Question 3.1/6.07

Comment not accepted. The question explicitly stated "for each trip function in column A" and did not imply that all column A conditions existed simultaneously.

Question 3.05

Comment accepted. The answer to part (b) was changed to accept "all bypass valves closed" as an additional answer.

Question 3.08

Comment accepted. The answer to part (a) was changed to reflect the facility's recommendation.

Question 3.13/6.14

Comment partially accepted. Answers to parts (a) were checked for validity and given credit as appropriate. Part (b) specifically asked for "power supplies" and full credit was only given for those answers which included all normal, alternate, and backup power supplies and associated components. Part (c) was deleted and the point value of the question was adjusted to 1.75 points total.

Question 4.04/7.03

Comment partially accepted. Partial credit was given for the facility's recommended answer for part (b). The question specifically required setpoint values; therefore, partial credit point values were readjusted such that setpoints on the original answer key were worth 0.2 points.

Question 4.13/7.12

Comment accepted. Full credit was awarded for the facility's recommended answer. It is recommended that BF 14.9 learning objective A be changed to read "List three systems..." since the "Nitrogen Isolation valves" are subdivided into two systems; the CAD and Nitrogen Purge and Makeup systems.

Question 4.14/7.16

Comment accepted. Point values were adjusted for part (b) to emphasize the significance of the reduction to 100% loop flow. It is strongly recommended that the facility change the procedure in OI-68 to reflect the current actual procedural practices in use by the operators and taught by the Training Department.

Question 8.03

Comment noted. This question was taken directly from facility learning objective (g) of RCI-9, and satisfies requirements set forth in 10 CFR 55.43.b (4). Efforts will continue to be made in the future to ensure questions are more job related, however, the NRC maintains that numbers pertaining to levels of contamination are required knowledge.

Question 8.09

Comment partially accepted. Full credit was given for either the original set of answers or the facility's recommended set of answers.

Question 8.10

Comment accepted. "Rod Group" was deleted from the answer key and point values were readjusted to .30 each. The facility is advised to ensure that Training Lesson Plans are accurate and updated to reflect changes in procedures.

Question 8.12

Comment accepted. Full credit was allowed for "yellow or red" for the color on the area to be circled.

NOTE:

Additional changes/deletions were made to the examination during the grading which were not the result of the facility's comments. These alterations were made to improve the test's accuracy and clarity and are included in Enclosure 2.

4. Exit Meeting

At the conclusion of the site visit the examiners met with representatives of TVA and the plant staff to discuss the results of the examination.

There were several generic weaknesses noted during the oral examination. The areas of below normal performance were:

(1). Candidates expressed an inability to deliniate which Browns Ferry Administrative Procedures held various items.

(2). Candidates were lacking in the use of ARP's.

(3). Candidates had difficulty determining the correct unit prints in the control room. Two of two candidates directed into the prints in the Unit 3 control room incorrectly referenced a Unit 1 print for several minutes, and the specific item being checked differed in the two prints.

(4). Candidates displayed noteworthy weakness in proper control board manipulations.

There were additional operational and procedural items of concern noted during the examination. Those items included:

Candidates stated that they would classify any ATWS as a General Emergency. This was based upon their interpretation of EP-1 which states that a General Emergency should be declared when Core Thermal Power indicates some significant value while the reactor is required to be shutdown. Clarification and more specificity with this procedure appears to be necessary.

Candidates were observed to use pieces of mylar of the approximate dimensions of a credit card as switch holders in the simulator. This action appeared to be a standard practice, and the NRC is concerned that it may be considered by the operators to be permisible in the control room. Allowing simulator operation in this manner reinforces this inappropriate action to be taken in the control room.

Some candidates stated that there exists an alternate means of injecting Boron into the core when the normal SLC system path is unavailable. Other candidates and training staff personnel stated that this capability does not exist. There appears to be no procedural guidance in this area. The NRC is concerned that if an alternate means of injection does exist, proper procedural guidance on its implementation is necessary; and that all operators are properly trained with respect to this capability.

Several candidates demonstrated a lack of familiarity with many of the procedural changes implemented in the 30 to 60 days prior to the administration of the examination. Recognizing the need that the Browns Ferry Training Department states it requires for cut-off times in training and testable material, the examiners noted the items to be numerous and significant. The NRC is concerned that a program must be in place to ensure the operators are fully updated on these items prior to their assumption of licensed operator duties.

Two procedures currently exist to delineate responsibilities for fire fighting within the plant. While no direct conflict exists between these procedures, their guidance has led to confusion amongst the operating staff as to who has direct fire fighting responsibility, at the scene. This confusion could delay the implementation of effective fire fighting practices, in an emergency.

The Evacuation Alarm Panel on the unit operator's desk is not provided with explicit instructions on its operation in the control room (Unit 3).

There appears to be no succinct listing of the secondary containment isolation valves for the use of operators.

The following two items are listed in this report as Inspector Followup Items:

1. Browns Ferry Technical Specification Table 3.2.A note 12 states that:

   "A channel contains four sensors, all of which must be operable for the channel to be operable.

   Power operations permitted for up to 30 days with 15 of the 16 temperature switches operable.

   In the event that normal ventilation is unavailable in the main steam line tunnel, the high temperature channels may be bypassed for a period of not to exceed four hours. During periods when normal ventilation is not available, such as during the performance of secondary containment leak rate tests, the control room indicators of the affected space temperatures shall be monitored for indications of small steam leaks. In the event of rapid increases in temperature (indicative of steam line break), the operator shall promptly close the main steam line isolation valves."

Browns Ferry AOI 99-1- 4.2, "Loss of Power to one RPS Bus," states a CAUTION that:

"The main steam line tunnel PCIS high temperature trip setpoint is 194°F. In the event that normal ventilation is unavailable in the main steam line tunnel (Reactor Building ventilation is isolated) the high temperature channels may be bypassed (jumpered) in accordance with Technical Specification Section 3.2.A, Note 12. Main steam line tunnel temperatures should not be allowed to exceed 200°F to prevent operational problems with the MSIV solenoid pilot valves."

Examiner discussions with licensed operator candidates and the Browns Ferry Training staff have led to the NRC concern that Browns Ferry personnel appear to have been trained to use the Abnormal or Emergency procedures to bypass the main steam line tunnel high temperature channels, when inappropriate. The NRC is concerned that the basis of bypassing the high temperature channels is not clearly stated, nor is it uniformly understood by the operators. This NRC concern is listed by this report as Inspector Followup Item 259/OL-81-01 and remains an open item pending facility clarification and completed training on the basis and appropriate use of the allowance for bypassing the main steam line tunnel high temperature channels.

2. All examiners noted that the licensed operator candidates encountered significant difficulty in tracking through the EOIs during the simulator examination. This deficiency had been previously recognized in the last two emergency exercises conducted at the facility. The physical construction of the EOIs and the training given licensed operator candidates on the use of EOIs (not annotating the steps that were completed) led the candidates to leaf back through the EOIs, appearing uncertain of the steps previously covered. Additionally, EOI 1, RC/Q-4, "Power Control," uses the condition of "shutdown" to determine necessary procedural actions. "Shutdown" does not appear to be clearly defined in the text, and the licensed operator candidates varied in their definitions. This NRC concern is listed by this report as Inspector Followup Item 259/OL- 81-02 and remains an open item pending facility clarification of exact parameters specifying a "shutdown" condition, and facility evaluation of the useability of Browns Ferry EOIs.

The cooperation given to the examiners and the effort to ensure an atmosphere in the control room conducive to oral examinations was also noted and appreciated.

The licensee did not identify as proprietary any of the material provided to or reviewed by the examiners.

SRO

MASTER

Nuclear Regulatory Commission

Operator Licensing

Examination

This document is removed from
Official Use Only category on
date of examination.

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

FACILITY:	BROWNS FERRY 1, 2&3
REACTOR TYPE:	BWR-GE4
DATE ADMINISTERED:	88/03/23
EXAMINER:	HOPPER, G
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
25.75 ~~26.25~~	24.76			5.	THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
26.0 ~~26.50~~	25.00			6.	PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
26.75	25.24			7.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
[illegible] ~~26.50~~	25.00			8.	ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
[illegible] ~~106.00~~		Final Grade	%		Totals

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

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

3. Use black ink or dark pencil <u>only</u> to facilitate legible reproductions.

4. Print your name in the blank provided on the cover sheet of the examination.

5. Fill in the date on the cover sheet of the examination (if necessary).

6. Use only the paper provided for answers.

7. Print your name in the upper right-hand corner of the first page of <u>each</u> section of the answer sheet.

8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a <u>new</u> page, write <u>only</u> <u>on</u> <u>one</u> <u>side</u> of the paper, and write "Last Page" on the last answer sheet.

9. Number each answer as to category and number, for example, 1.4, 6.3.

10. Skip at least <u>three</u> lines between each answer.

11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.

12. Use abbreviations only if they are commonly used in facility <u>literature</u>.

13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.

15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

16. If parts of the examination are not clear as to intent, ask questions of the <u>examiner</u> only.

17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

(2) Exam aids - figures, tables, etc.

(3) Answer pages including figures which are part of the answer.

b. Turn in your copy of the examination and all pages used to answer the examination questions.

c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 5.01 (1.50)

a. DOES the delayed neutron fraction INCREASE or DECREASE from the beginning of cycle (BOC) to the end of cycle (EOC)? (0.5)

b. LIST the two (2) major causes for the change in delayed neutron fraction from BOC to EOC. (1.0)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.02 (1.00)

Reactor power is 60 on IRM range 2 with the MINIMUM permissible stable positive period allowed by procedure GOI-100-1. Heating power is determined to be 40 on IRM range 7. CALCULATE how long it will take for power to reach the point of adding heat if the period remains constant.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.03 (2.00)

SELECT the appropriate response for EACH of the following statements concerning Control Rod Worth:

a. (MORE/LESS) control rods would need to be pulled to make the reactor critical at 545 degrees F, as opposed to 140 degrees F.

b. An INCREASE in the Void Fraction will result in an (INCREASE/DECREASE) in individual control rod worth.

c. Control Rod Worth at End of Cycle would be (LESS/GREATER) than at the Beginning of Cycle.

d. Control Rod Worth will (INCREASE/DECREASE) with an INCREASE in moderator temperature.

e. Control Rod Worth will (INCREASE/DECREASE) as the adjacent control rods are withdrawn.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.04 (1.00)

Briefly EXPLAIN HOW a control rod withdrawal of one or two notches can result in a decrease in bundle power.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.05 (~~2.00~~) 1.5

For each of the following events, STATE which COEFFICIENT of reactivity (MODERATOR, VOID, DOPPLER) would act FIRST to change reactivity.

a. control rod drop at 25 percent power

b. SRV opening at 50 percent power

Deleted ~~c.~~ loss of shutdown cooling when removing decay heat

d. one recirc pump trips while at 50 percent power

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.06 (2.00)

Following a reactor scram from power, several control rods fail to insert to the full-in position. One hour after the scram, the Reactor is determined to be subcritical with a shutdown margin (SDM) of 0.22 % delta K/K.

a. If reactor coolant temperature and control rod positions remain constant during the next hour (from 1 - 2 hours after the scram), would SDM INCREASE, DECREASE, or REMAIN the SAME? Briefly EXPLAIN your answer.

b. During the next hour (2 - 3 hours after the scram) you notice reactor pressure is decreasing. WHAT effect would ONLY the pressure decrease have on SDM. Briefly EXPLAIN your answer.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.07 (1.00)

The reactor has been operating at 95 percent power for several days. An operator RAPIDLY reduces reactor power to 60 percent by reducing the speed of the recirculation pumps. During the next 2-3 MINUTES the operator notices that the reactor power slowly increases to 63 percent (with no operator action). EXPLAIN the cause of the power increase.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.08 (1.25)

As a reactor operator coming on shift, you are told that the previous shift performed a reactor shutdown and commenced a cooldown from 630 psig at 0630. It is now 0730 and you note that wide range reactor pressure is 200 psig. Your shift is to place the reactor in shutdown cooling.

a. HAS the previous shift exceeded the Technical Specification maximum allowable cooldown rate (YES or NO)? INCLUDE in your answer the Technical Specification Cooldown Limit and the assumptions and calculations used.) (0.75)

b. HOW many more degrees of cooldown are necessary before RHR can be initiated for shutdown cooling? (INCLUDE your assumptions and calculations.) (0.5)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.2[illegible] (1.00)

Concerning the bypass flow in the reactor core, STATE the two (2) most significant consequences that would occur if bypass flow were significantly reduced at full power.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.10 (1.00)

A centrifugal pump is operating at rated speed with a discharge head of 240 psig and power consumption of 0.5 MW. The speed of the pump is then decreased until the power consumption is 1/64 of its original value. CALCULATE the new discharge head. SHOW ALL WORK.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.11 (1.00)

A leak develops in the low pressure side of a flow transmitter. HOW will this effect the flow indication for the instrument? (CHOOSE one.)

a. The indicated delta pressure would decrease causing the indicated flow to decrease.

b. The indicated delta pressure would decrease causing the indicated flow to increase.

c. The indicated delta pressure would increase causing the indicated flow to decrease.

d. The indicated delta pressure would increase causing the indicated flow to increase.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.12 (2.00)

Answer EACH ONE of the following statements TRUE or FALSE regarding reactivity coefficients.

a. An increase in flow through the reactor core will add negative reactivity by decreasing the void fraction and thus increasing reactor power.

b. As the burnable poison within a fuel bundle burns out, the VOID coefficient becomes more negative.

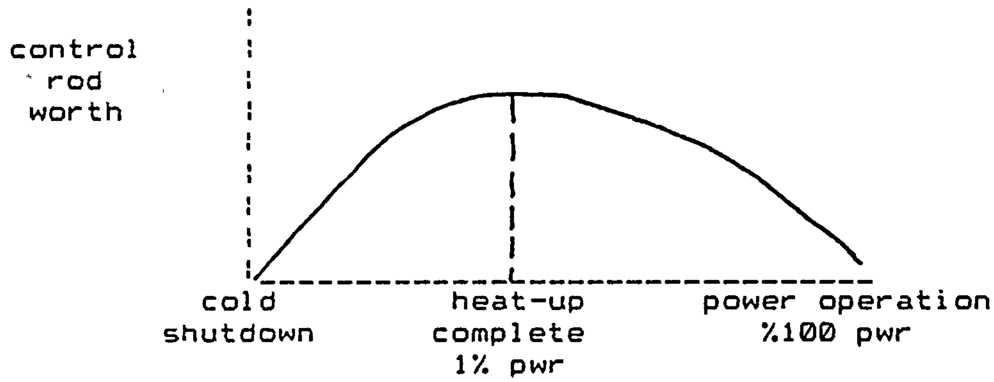
c. LATE in core life, the large reduction in fuel molecules and the decrease in moderator density during a plant HEAT-UP can lead to a positive reactivity addition.

d. As core age progresses, the DOPPLER coefficient becomes more negative due to plutonium-240 buildup.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.13 (2.00)

Answer EACH ONE of the following questions TRUE or FALSE concerning the graph of Contol Rod Worth During a Startup.



a. Control rod worth increases during heatup due to density decreases of the moderator which causes longer slowing down and thermal diffusion lengths, resulting in more thermal flux around a control blade.

b. Control rod worth decreases as power exceeds 1% due to the effects of rod shadowing. Withdrawal of rods increases the thermal diffusion length thereby increasing the flux around a control blade.

c. While heating-up, rod worth increase is due mainly to the effects of Bundle Coupling. Rod withdrawal couples fuel cells together making their effective size larger, resulting in increased leakage and a reduction in thermal flux.

d. Since control rods are worth more when the moderator is hot, fewer control rods must be withdrawn to go critical when the reactor is hot than when cold.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.14 (3.00)

LIST the three (3) "thermal limits" observed during reactor operation and STATE the limiting condition for each.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.15 (1.50)

During startup you have established a stable 160-second period. By definition IS the reactor CRITICAL, SUBCRITICAL or SUPERCRITICAL? EXPLAIN the difference between subcritical and supercritical. ASSUME no startup sources are present. (1.5)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.16 (2.00)

The attached FIGURE (GTH-747) represents parameter changes for a plant transient on UNIT TWO. Use this figure and the following information to answer EACH of the questions below:

(1) Initial Power Level = 100 %
(2) Bypass Valves go to Full Open position
(3) No operator action is taken

a. The DECREASE in turbine steam flow. (POINT 4)

b. The INCREASE in power. (POINT 7)

c. The INCREASE in turbine steam flow. (POINT 5 and AREA 6)

d. The DECREASE in pressure. (POINT 2)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.17 (1.00)

CALCULATE the equilibrium neutron count rate in a subcritical reactor after FOUR (4) generations given the following initial conditions:

Source = 100cps
Keff = .2
ASSUME: generation 0 consists of only source neutrons and equilibrium is achieved after four generations

(***** END OF CATEGORY 05 *****)

QUESTION 6.01 (2.00)

STATE whether the following statements concerning the Primary Containment Isolation System are TRUE or FALSE:

a. Most of the PCIS motor operated valves fail closed on loss of power to the valve.

b. The containment isolation reset switches on panel 9-5 must be operated to manually reset a RCIC turbine steam supply isolation.

c. Loss of RPS Bus A will NOT cause any PCIS isolation valves to close.

d. The TIP guage tube ball valve will isolate on a high radiation signal.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.02 ~~(1.00)~~ .50

During your shift the Drywell Air System (DWAS) isolates. You verify a Group VI isolation has not occurred.

Deleted ~~a. NAME one other signal that could have caused the DWAS isolation.~~

b. WHAT air system valves close when the DWAS isolates?

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.03 (3.00)

For EACH of the following statements regarding the High Pressure Coolant Injection System, INDICATE whether the statement is TRUE or FALSE, and EXPLAIN your answer.

a. In the event low HPCI booster pump suction pressure is sensed during HPCI system operation, the turbine will trip, and the signal must be manually reset before the turbine will restart, even if initiation signals are still present.

b. Upon a HPCI system isolation due to low steam pressure, the system cannot restart until the pressure rises above the isolation setpoint and the isolation signal is reset.

c. If the HPCI turbine trips due to an overspeed condition, it will restart when the speed coasts down to less than 5000 rpm.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.04 (1.50)

A Core Spray line breaks inside the shroud.

a. WILL the break cause an alarm in the control room (YES or NO)? (0.5)

b. HOW will the break affect core spray performance for that loop? (1.0)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.05 (2.00)

The RCIC system is in operation on your shift to demonstrate operability for Technical Specifications.

a. DESCRIBE what occurs to the RCIC system (components) if reactor water level exceeds 54 inches.

b. A low-low reactor water level condition occurs after the high level condition described in part a. above. DESCRIBE the operator actions required to permit the RCIC system to respond to this low-low level condition.

QUESTION 6.06 (2.00)

For the Rod Block Monitor (RBM), answer EACH of the following:

a. WHAT adverse condition is the system designed to prevent?

b. When the Meter Function Switch on the Back Panel 9-14 Meter Section is in the "count" position, WHAT are the "units" of the indication on the meter and WHAT can be calculated by utilizing the indicated value?

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.07 (2.50)

The plant is operating at 100% power and 100% core flow when the "A" flow converter output fails to zero. MATCH from column B the action that will exist for each trip function in Column A given the above conditions.

NOTE: REPSONSES MAY BE USED MORE THAN ONCE

COLUMN A	COLUMN B
a. "A" APRM Hi-Hi thermal	1. Rod Block
b. "B" APRM Hi-Hi thermal	2. Half Scram
c. "C" APRM Hi	3. Full Scram
d. "D" APRM Hi	4. None
e. "E" APRM Hi-Hi neutron	

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.03 (2.00)

LIST four (4) conditions that will initiate the annunciator "RPS ATU TROUBLE" on Panel 9-5.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.09 (2.50)

A reactor startup of Unit 2 is in progress. The plant is operating at 7% in the process of power ascension to 100% Rated Thermal Power. The Mode Switch has just been placed in "RUN" and the following equipment is out of service:

Condensate Pump A
CRD Pump B
APRM B Bypassed (FAILED DOWNSCALE)
RBCCW Pump C

A RED Hi-Hi/Inop light suddenly illuminates on the apron section of Panel 9-5 for IRM H. A check of Panel 9-14 determines a WHITE INOP light is NOT illuminated for IRM H.

Answer EACH of the following with regards to the above situation:

a. LIST three (3) causes for the indications on IRM H. (1.5)

b. STATE what automatic trips should occur. JUSTIFY your response. (1.0)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.10 (1.50)

Answer EACH of the following given the below data for APRM Channel C:

LPRM Level:	A	B	C	D
No. of LPRMs assigned:	6	5	5	5
No. of LPRMs bypassed:	3	0	3	0

a. If APRM Channel C selector switch on the local (back) panel was placed to the COUNT position, STATE the expected meter reading. (0.5)

b. Based on the above information, STATE whether APRM Channel C is operable or inoperable. JUSTIFY your response. (1.0)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.11 (1.00)

Unit 2 is operating at 100% Rated Thermal Power with "A" feedwater level control selected. All other controls are in Normal/Automatic. A failure in a reactor vessel level narrow range instrument has occurred and resulted in the following related trips/indications:

REAC VESSEL WATER LEVEL LOW-LOW CHAN A ANNUNCIATOR - OFF
REAC VESSEL WATER LEVEL LOW-LOW CHAN B ANNUNCIATOR - OFF
"A" NR level indication reads at MINIMUM
"B" & "C" NR level indications read at MAXIMUM
REACTOR WTR LEVEL A ABNORMAL ANNUNCIATOR - ON
Feedwater flow is at ZERO
Two channels of Level 8 have TRIPPED
One channel of Level 3 has TRIPPED

STATE which NR level transmitter has failed and STATE in which direction it has failed (HIGH/LOW).

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.12 (1.50)

Answer EACH of the following regarding the Reactor Protection System (RPS) for Unit 1:

a. STATE the alternate source of power to the RPS bus. (BE SPECIFIC! INCLUDE VOLTAGE AND BOARD NUMBER) (0.5)

b. LIST two (2) interlocks associated with this alternate power supply. (1.0)

QUESTION 6.13 (1.50)

Answer EACH of the following with regard to the Source Range Monitors (SRMs):

a. STATE the two (2) permissives that must be satisfied before an operator may withdraw the SRMs without producing a rod block. (1.0)

b. DESCRIBE the indications available to the operator on Panels 9-5 and 9-12 that signify the SRMs may be withdrawn without causing a rod block. (0.5)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.14 (2.50)

Answer EACH of the following with regard to the 250v Unit and Plant DC power system:

a. LIST three (3) major types of loads supplied by this system. (0.75)

b. EXPLAIN how a reliable source of DC power is maintained to these loads. INCLUDE ALL NORMAL, ALTERNATE & BACKUP POWER SUPPLIES AND ASSOCIATED COMPONENTS! (1.0)

c. EXPLAIN why DC power is preferred for these types of load (other than for improved reliability). BE SPECIFIC! THREE RESPONSES REQUIRED FOR FULL CREDIT! (0.75)

(***** END OF CATEGORY 06 *****)

QUESTION 7.01 (2.00)

Entry into a HIGH RADIATION AREA is required. To complete the task, the operator will receive an estimated 70 mrem whole body dose. You have the following information on available operators. Time constraints will not permit authorization of an increase in administrative limits. NRC Form 4s are on file unless otherwise indicated.
STATE your REASONS for accepting or rejecting each operator for the job.

OPERATOR	1	2	3	4
SEX	male	male	female	male
AGE	29	30	24	20
WK EXP	200 mrem	0 mrem	5 mrem	90 mrem
QTR EXP	1190 mrem	950 mrem	435 mrem	5 mrem
ANN EXP	2170 mrem	3990 mrem	750 mrem	2500 mrem
LIFE EXP	----	55370 mrem	2735 mrem	10050 mrem
Remarks	History Unavailable		3 months Pregnant-- Signed Prenatal Document on File	

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.02 (1.00)

With Unit 2 operating at 90% power, an Off Gas Hydrogen [H-2] High alarm is received. Both Hydrogen analyzers indicate a Hydrogen level of ~ 5.3%. Which one of the following responses most accurately reflects the proper course of action you should take?

a. Change over to the alternate Recombiner.

b. Change over to the Alternate Off Gas Train.

c. Start an additional SJAE to assist in Condenser H-2 removal.

d. Manually Scram the reactor.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.03 (2.50)

The following parameter changes / annunciators are observed by the reactor operator:

RBCCW temperature Lower than normal
RBCCW Surge Tank HI Level alarm
(No other alarms present)

a. WHICH one (1) of the following malfunctions would most likely cause these indications: (1.0)

1. Raw Cooling Water leak in the RBCCW Heat Exchanger(s).
2. Reactor Coolant leak into RBCCW via NRHX.
3. Fuel Pool Cooling System leak from RBCCW.
4. RBCCW Makeup Valve (fill valve) leak
5. DWEDS Heat Exchanger leak into RBCCW.

b. LIST three (3) of the conditions/circumstances that will cause the isolation valve to non-essential equipment (MOV-48) to automatically close. (1.5)

NOTE: BE SPECIFIC AND INCLUDE SETPOINT VALUES

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.04 (2.00)

A Reactor Startup is in progress on UNIT 2 when the operating CRD Pump trips on motor overload. The Backup Pump is under repair and is expected to be operable within 45 minutes. The following initial conditions exist:

Reactor Power = 20 %
Reactor Pressure = 610 psig
Charging Water Pressure = 1490 psig (decreasing slowly)
1 ACCUM light on Full Core Display Illuminated

ANSWER EACH of the following questions TRUE or FALSE.

a. A Manual Scram is required if Reactor Pressure drops below 600 psig.

b. A Manual Scram is required if Charging Water Pressure cannot be maintained above 1410 psig.

c. A Manual Scram is required if 30 Control Rods receive High Temperature alarms with a Low CRD Water Pressure alarm.

d. A Manual Scram is required if the backup CRD Pump is started, Charging Water Pressure is 1500 psig, and a second ACCUM alarm comes in due to low pressure.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.05 (1.50)

Answer EACH of the following questions TRUE or FALSE concerning Fuel Handling:

a. A Rod Block will be inserted if the Service Platform Hoist is loaded with the Mode Switch in the Shutdown position.

b. A 1/M Plot is used to predict the number of bundles needed to produce criticality.

c. Refuel Interlocks are the single and only means of protection against an inadvertent criticality during refueling operations.

L 00k U:

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

X

QUESTION 7.06 (1.50)

Answer EACH of the following questions TRUE or FALSE concerning procedural limitations applied to squirrel cage induction and synchronous motors, 200 horsepower and larger:

a. The number of starts should be kept to a minimum since the life of a motor is affected by the number of starts.

b. A motor shall be limited to two (2) starts is succession, coasting to rest between starts, if the motor is initially at normal operating temperature.

c. Following a motor start at normal operating temperature, the motor should be allowed to cool for approximately 20 minutes while running at no load before an additional restart is attempted.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.07 (1.25)

Answer the following question regarding EPIP-2 (NUE) by FILLING IN THE BLANKS.

After an event is declared the ODS shall be notified within _____1______.

The SE/SED shall notify the NRC immediately or within ____2_____ by using the _____3_____.

Reanalysis of the current situation will be done by the ____4_____ at least every _____5_____ or more frequently if conditions warrant to determine if the NUE should be cancelled.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.06 (2.00)

LIST eight (8) different indications that should be observed in the Control Room following SLC initiation.

NOTE: DO NOT INCLUDE REACTOR POWER OR PRESSURE IN YOUR ANSWER

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.09 (1.50)

Answer EACH of the following questions concerning EOI-1:

a. LIST the conditions under which SLC injection is mandatory. (0.5)

b. Permanent disabling of ADS is required when Reactor Shutdown is contingent upon SLC (ATWS condition), because core damage could occur. STATE two (2) methods of causing core damage if ADS is allowed to actuate. (1.0)

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.10 (2.00)

LIST eight (8) initial operator actions to be taken when Control Room abandonment is required as specified in Emergency Plans Manual Six (EPM-6).

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.11 (1.50)

LIST six (6) CONDITIONS which must be met to transfer the Reactor Mode Switch to the RUN position as required by GOI 100-1 (Integrated Plant Operations).

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.12 (1.00)

LIST two (2) systems that require tagging prior to entry into the Primary Containment. INCLUDE in your answer the required status or position of the system.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.13 (1.00)

Operating Instruction No. 23 " RHRSW " requires that anytime a RHRSW heat exchanger is placed in service the chemical laboratory takes a sample every 4 hours to verify compliance to 10CFR20 release rate limits.

EXPLAIN WHY is it necessary for the chemical laboratory to conduct a routine sampling for verification of compliance to 10CFR20?

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.14 (1.00)

EOI-2, "Containment Control" , states in step PC/P-1 :

> Operate SBGT and Containment Atmosphere Dilution to vent as required only when the temperature in the space being evacuated is below 210 DEG F...

EXPLAIN WHY the Primary Containment is not vented at temperatures above 210 DEG F.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.15 (1.00)

EXPLAIN the reason for resetting a Scram as soon as possible as stated in GOI 100-11 (Reactor Scram Procedure).
NOTE: SETPOINT REQUIRED FOR FULL CREDIT

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.16 (2.00)

A single Recirculation Pump trips while operating at 100 % power in automatic control.

a. STATE the immediate action(s) that should be performed on the RUNNING PUMP.

b. EXPLAIN WHY the Running Pump speed must be reduced to < 50 % of rated speed prior to starting the idle pump.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.17 (1.00)

During Shell Warming of the High Pressure Turbine, the operator should maintain the 1st Stage Pressure between 60-100 psig in accordance with OI-47, since a Reactor Scram would occur if pressure became > 142 psig. EXPLAIN WHY this would initiate a Reactor Scram.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.10 (1.00)

Diesel Generator Fast Starts should be avoided during the time period of 15 minutes to 3 hours after Diesel shutdown, according to GI-82.

a. EXPLAIN the reason for this precaution.

b. Briefly DESCRIBE the method by which a Diesel Generator should be started during this time interval and from where it would be started.

(***** END OF CATEGORY 07 *****)

QUESTION 8.01 (1.00)

WHICH one (1) of the following scenarios would result in the assumption that the Fuel Cladding Integrity Safety Limit of the Unit 1 Technical Specifications had been exceeded ?

a. Reactor power is at 42% RTP; the main turbine trips due to an EHC malfunction; the reactor SCRAMS on HIGH PRESSURE; the BPV's control pressure thereafter.

b. Reactor power is at 70% RTP; a steam leak to the Drywell occurs and Drywell pressure rises; the reactor SCRAMS at 1.85 psig; HPCI auto-actuation does not occur, but manual start is successful; the reactor is brought to a cold shutdown condition.

c. Reactor is in Start-Up, at 12% RTP; power is increased by rod pull; the reactor SCRAMS at 12.5% power, by APRM's; level and pressure are maintained by normal systems for the plant status.

d. The reactor is at 18% RTP; 1-1/2 BPV's are open in preparation for turbine warmup; controller failure reduces pressure to 875 psig; MSIV's close; reactor SCRAMS; level and pressure are maintained by normal systems for the plant status.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.02 (1.00)

While acting as the Shift Engineer, you encounter circumstances that are not specifically addressed in Technical Specifications, which result in the inability to satisfy an LCO for the EECW System.

WHICH one (1) of the following statements is MOST CORRECT. (CHOOSE ONE STATEMENT)

a. An orderly shutdown shall be initiated and the Reactor placed in the Cold Shutdown condition within 24 hours.

b. Notify the Unit Superintendent and request a Technical Specification interpretation.

c. The Reactor may remain in operation for a period not to exceed 7 days, at which time the unit must be in the Hot Standby Condition.

d. Shutdown the Reactor within 6 hours and be in the Cold Condition within 36 hours.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.03 (2.50)

STATE whether a Radiation Work Permit (RWP) is " REQUIRED " or " NOT REQUIRED " for EACH of the situations given below:

a. An employee will need to work in an area having airborne radioactivity of 15 % MPC.

b. Work will be done in a designated " RADIATION AREA ".

c. Work is to be done in an area with 1500 DPM/100 cm2 loose surface contamination.

d. A radiological survey inside a Contamination Zone will be performed while standing outside the Zone.

e. Trash and protective clothing will be removed from a Contamination Zone while standing outside the Contamination Zone on the stepoff pad.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.04 (~~2.00~~) 1.5

Answer EACH of the following questions TRUE or FALSE.

Deleted
~~a. The~~ CO2 Fire Protection System shall be operable with a minimum of 50 % (.5 Tank) in storage units 1,2, and 3.

b. If CO2 fire protection is lost to a Cable Spreading Room, a continuous Fire Watch must be stationed until it is restored.

c. Reactor operation may continue with the High Pressure Fire Protection System inoperable, provided patrolling Fire Watches with portable fire equipment are available to patrol all areas hourly.

d. During Reactor operation, welding is permitted in the Cable Spreading Room, provided a continuous Fire Watch is stationed in the immediate vicinity where the work is in progress.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.05 (2.00)

Answer EACH of the following questions concerning BF 14.25 (Clearance Procedure):

a. Clearances on plant equipment shall be issued only to those people whose names appear in the _______1________ , which is approved by the ______2______. (FILL IN THE BLANKS) (0.5)

b. EXPLAIN the provision for performing short-term Emergency Maintenance if the workers are not authorized to hold a clearance. Specifically address to whom the clearance is issued and what "Remarks" are required. (1.0)

c. What is a " GROUND DISC " ? (0.5)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.06 (1.00)

ANSWER EACH of the following questions by FILLING IN THE BLANKS:

a. The Reactor shall be shutdown if Ph is less than __1__ or greater than __2__ for a 24 hour period.

b. At steaming rates > 100,000 lb/hr, the following limits shall apply:

i) Conductivity (umho/cm @ 25 Deg C) ___3___.

ii) Chloride (ppm) ___4___.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.07 (1.50)

An Operability Test is conducted on a safety-related system, following the installation of an approved modification. LIST three (3) separate criteria which would procedurally require that a TEST DEFICIENCY be initiated (documented).

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.06 (2.00)

Brown's Ferry Standard Practice 12.17, "Administrative Control for Plant Operation," establishes plant policy for the control of containment isolation and safety systems during an emergency.

a. STATE the evaluation which must be made prior to resetting a Primary Containment Isolation. 1.0

b. LIST the two (2) conditions which allow operators to override automatic operations of engineered safety features. (1.0)

NOTE: DO NOT CONFUSE THIS WITH THE GUIDANCE FOR MANUALLY SECURING AN ECCS SYSTEM.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.09 (1.50)

Unit 1 Technical Specifications specify for REACTIVITY CONTROL ...

"A sufficient number of control rods shall be oper-
able so that the core could be made subcritical in the
most reactive condition during the operating cycle ..."

LIST the three (3) conditions/assumptions which are used to verify this "Reactivity Margin" (Adequate Shutdown Margin).

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.10 (1.50)

STATE the six (6) ITEMS to be recorded in the daily journal when the Reactor is declared "critical" during a Reactor startup in accordance with GP 100-1.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.11 (3.00)

You have just assumed the 0000-0800 (3/23/88) shift as the UNIT 2 Shift Engineer. The plant is operating at 100 % power with 98 % core flow and the following equipment out of service:

	Date OOS
SLC Tank Remote Level Indication	02/25/88
RHR Service Water Pump B2	03/20/88
RBCCW Pump C	03/12/88
CRD Pump	02/28/88
Core Spray System I Room Coolers	03/18/88
Core Monitor	02/05/88
Turning Gear Motor (Main Turbine)	03/15/88
Condensate Pump A	02/05/88
RHR Pump 3C	03/14/88

Answer EACH of the following questions based on the above information:

a. LIST the out of service equipment which should have Limiting Conditions for Operation (LCOs) IN EFFECT and STATE the LCO and CURRENT SURVEILLANCE REQUIREMENTS that must be performed for each per Technical Specifications in order to allow continued operation in these conditions. (1.5)

b. STATE how long the Reactor may remain in operation if NO repairs are completed. (0.5)

c. While performing the RHR quarterly full flow test, RHR Pump 2C is declared inoperable. STATE the Tech Spec requirements Concerning plant operability. (BE SPECIFIC AND REFERENCE THE TECH SPEC BEING APPLIED!) (1.0)

NOTE: APPLICABLE TECHNICAL SPECIFICATIONS ATTACHED

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.12 (2.00)

DESCRIBE the four (4) standards (i.e. symbols/colors) used in marking TEMPORARY ALTERATIONS on plant drawings and WHAT they mean.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.13 (2.00)

The following data was taken during recent testing of the Standby Liquid Control System:

Pump Flow Rates	A: 45gpm	B: 40gpm
Relief Valve Setting	A: 1375psig	B: 1325 psig
SLC Tank Level	4200 gal @ 13.0 % Boron concentration	
Solution Temperature	72 Deg F	

LIST any paramaters which do not meet Technical Specification requirements. INCLUDE, in your answer, any applicable Limiting Conditions for Operation.

NOTE: APPLICABLE TECHNICAL SPECIFICATIONS ATTACHED

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.14 (1.50)

STATE the criteria that are used to determine if an MR should be checked EMERGENCY, IMMEDIATE ATTENTION, or ROUTINE MAINTENANCE. (i.e., Define each classification)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.15 (2.00)

a. STATE which two areas/systems must be intact to satisfy Primary Containment Integrity. (1.0)

b. LIST four (4) conditions which must be satisfied, according to Technical Specifications, to have Primary Containment integrity. (1.0)

(***** END OF CATEGORY 08 *****)
(************* END OF EXAMINATION ***************)

MASTER

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 5.01 (1.50)

a. decrease [+0.5]

b. The contribution to the delayed neutron population by U-235 decreases as the U-235 is burned out [+0.5] and the contribution from plutonium increases [+0.5], decreasing the delayed neutron fraction.

REFERENCE
BFNP: Reactor Theory, pp. 3-29
Chapter 3, Objective 4.6.

2.5/2.5
292003K104 ...(KA'S)

ANSWER 5.02 (1.00)

60 on range 2 is equal to 0.06 on range 7 [+0.25]

P(t) = P(o)e**-t/T [+0.25]

P(o) = 0.06, P(t) = 40, period = 60 seconds

t = 60 ln 40/0.06 [+0.25]

= 390 seconds or 6.5 minutes [+0.25]

REFERENCE
BFNP: Reactor Theory, pp. 3-17 and 3-19.
Chapter 3, Objective 3.2.
GOI-100-1, p. 13.

2.7/2.8
292003K108 ...(KA'S)

ANSWER 5.03 (2.00)

a. more
b. decrease
c. less
d. increase
e. increase

[+0.4 each]

REFERENCE
BFNP: Reactor Theory, pp. 5-9 through 5-16.
Chapter 5, Objective 2.4.

2.5/2.6
292005K109 ...(KA'S)

ANSWER 5.04 (1.00)

The steam bubbles generated by the withdrawal of a shallow rod increase the void fraction [+0.5], which adds negative reactivity, offsetting the positive reactivity effects of the rod withdrawal [+0.5].

REFERENCE
BFNP: Reactor Theory, p. 5-23.
Chapter 5, Objective 3.3.

3.1/3.2
292008K119 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 5.05 ~~(2.00)~~ 1.5

a. Doppler
b. void
~~c. moderator~~ Deleted
d. void

[+0.5 each]

REFERENCE
BFNP: Reactor Theory, pp. 7-2 through 7-23.
Chapter 7, Objective None.

3.3/3.3
292004K114 ...(KA'S)

ANSWER 5.06 (2.00)

a. increase [+0.5] due to Xe build in [+0.5]

b. The decrease in reactor pressure is directly related to a decrease in reactor coolant temperature [+0.25]. Due to the negative value of the moderator temperature coefficient, reactivity increases (adds positive reactivity with decreasing moderator temperature [+0.25].
Hence, SDM decreases [+0.5].

REFERENCE
BFNP: Reactor Theory, pp. 1-35, 4-5, and 6-9.
Chapter 1, Objective 4.1; Chapter 4,
Objective 2.2; Chapter 6, Objective 2.3.d.

3.2/3.5 2.6/2.7
292002K110 292002K114 ...(KA'S)

ANSWER 5.07 (1.00)

Less feedwater heating will occur resulting in colder feedwater entering the vessel [+0.5] which will cause reactor power to increase (about 3 percent) from the positive reactivity addition (alpha m) [+0.5].

REFERENCE
BFNP: Heat Transfer and Fluid Flow, pp. 5-48.
Reactor Theory, pp. 7-18 and 7-19.
Chapter 7, Objective 8.1.

3.3/3.4 2.9/3.0 2.7/2.8
292008K120 292008K121 293005K105 ...(KA'S)

ANSWER 5.08 (1.25)

a. YES [+0.25], (the previous shift DID EXCEED the cooldown limit of) 100 degrees F/hr [+0.25].

331.8 or 381.8

Tsat for 630 psig = 494 degrees F;
Tsat for 200 psig = 388 degrees F;
cooldown rate = (494-388) degrees F/1 hour
= 106 degrees F/hr [+0.25]

b. 47 +/- 2 degrees F (of cooldown required) [+0.25]

Tsat for 200 psig = 388 degrees F;
Tsat for 105 psig = 341 degrees F;
(388-341) = 47 degrees F [+0.25]

ECF WILL BE APPLIED TO PART b.

REFERENCE
BFNP: Heat Transfer and Fluid Flow, pp. 3-1, 3-13.
Chapter 3, Objective 1.1
OPL171.044, RHR, pp. 14 and 16.
Objectives V.D.3, V.E.3, and V.E.4.
Technical Specifications, 3.6.A.

3.7/3.8 2.8/3.1
205000K402 293003K123 ...(KA'S)

ANSWER 5.09 (1.00)

1. (excessive voiding in bypass region resulting in) unreliable LPRM readings [+0.5]

2. inadequate cooling of LPRM detectors (resulting in premature LPRM detector failures) [+0.5]

REFERENCE
BFNP: Heat Transfer and Fluid Flow, pp. 8-49 through 8-52
Chapter 8, Objectives 9.4 and 9.5.

2.5/2.6 2.4/2.6
293008K132 293008K133 ...(KA'S)

ANSWER 5.10 (1.00)

use pump laws: power proportional to (speed)**3 [+0.4]
and head proportional to (speed)**2 [+0.4]

then: power decreased to 1/64 implies speed decrease by 1/4.

hence head decreased to (1/4)**2, which is 1/16
240 psig x (1/16) = 15 psig [+.2]

REFERENCE
BFNP: Heat Transfer and Fluid Flow, p. 6-96.
Chapter 6, Objective 7.11.

2.8/2.9
291004K105 ...(KA'S)

ANSWER 5.11 (1.00)

d

REFERENCE
BFNP: Heat Transfer and Fluid Flow, p. 6-68.
Chapter 6, Objective 6.1.

2.9/3.1 3.1/3.1
291002K104 291002K105 ...(KA'S)

ANSWER 5.12 (2.00)

a. False
b. True
c. True
d. True (.5 each)

REFERENCE
G.E. Reactor Theory, Chpt. 4, LO 1.5,3.6,4.3,6.3, Chpt 7, LO 5.6
2.5/2.6 , 2.1/2.2 , 2.5/2.6 , 1.9/2.12
292004K102 292004K109 292004K111 292004K113 ...(KA'S)

ANSWER 5.13 (2.00)

a. True
b. False
c. False
d. False (.5 each)

REFERENCE
G.E. Reactor Theory, Chpt. 5, LO 2.5
2.5/2.6 , 2.6/2.9
292005K109 292005K112 ...(KA'S)

ANSWER 5.14 (3.00)

1. LHGR - Linear Heat Generation Rate [+0.5]
(designed to limit the pin power at any node in the reactor to a value that) limits the fuel clad strain to less than one percent plastic strain. [+0.5]

2. APLHGR - Average Planer Linear Heat Generation Rate [+0.5]
(designed to limit average pin power at any node to a value such that following a design basis accident the) maximum fuel clad temperature will not exceed 2200 degrees F. [+0.5]

3. MCPR - Minimum Critical Power Ratio [+0.5]
(designed to limit the power of any fuel element to below the value that will) prevent any point in the bundle from experiencing the onset of transition boiling. [+0.5]

other answers: LHGR - MFLPD or CMFLPD
APLHGR - MAPRAT or CAMPR .35 each
MCPR - CMFCP or MFLCPR

REFERENCE
BFNP: Heat Transfer and Fluid Flow, pp. 9-16 through 9-26.
Chapter 9, Objectives 2.3, 3.3, and 4.3.

2.8/3.6 2.8/3.6 2.8/3.6
293009K107 293009K111 293009K119 ...(KA'S)

ANSWER 5.15 (1.50)

supercritical [+0.5]

subcritical - reactor power is decreasing (OR neutrons per generation are decreasing) [+0.5]

supercritical - reactor power is increasing (OR neutrons per generation are increasing) [+0.5]

REFERENCE
BFNP: Reactor Theory, pp. 1-31 and p. 7-7.
Chapter 1, Objective 3.1.
Chapter 7, Objective 2.3.

3.5/3.5 3.9/3.9 4.1/4.1
292002K107 292008K107 292008K108 ...(KA'S)

ANSWER 5.16 (2.00)

a. BPV'S open causing EHC to close Turbine CV's. (0.5)

b. Power increased due to lower feedwater temperature. (0.5)
(Less steam to the Turbine)

c. (All BPV,s are open at point 5.) ~~(0.25)~~ EHC follows increasing pressure by opening CV's. (~~0.25~~) (0.5)

d. Pressure decreases due to BPV's opening. (0.5)

REFERENCE
BFNP: OPL 171.055 LO A
4.1/4.2 3.6/3.7 4.1/4.1
241000A203 241000K301 241000K610 ...(KA'S)

ANSWER 5.17 (1.00)

S/1-Keff 100/(1-.2) = 125

or

	fission	source	total
0	0	100	100
1	20	100	120
2	24	100	124
3	25	100	125
4	25	100	125

REFERENCE
G.E. Reactor Theory, Chpt 3, LO1.2,1.5
2.9/3.0 , 2.1/2.3
292003K101 292003K102 ...(KA'S)

ANSWER 6.01 (2.00)

a. False (MOV's fail as-is)
b. False (separate reset switch for RCIC)
c. FALSE ~~True (both logic channels must deenergize)~~
d. False (only high D/W pressure or low RPV level)

[+0.5 each]

REFERENCE
BFNP: OPL171.017, PCIS, pp. 6, 17 and 18.
Objectives V.D. and V.E.

3.4/3.6 2.7/2.9 3.4/3.5 3.5/3.7
223002K107 223002K113 223002K406 223002K608 ...(KA'S)

ANSWER 6.02 0.5 (1.00)

~~a. reactor zone ventilation radiation signal~~ Deleted ~~[+0.5]~~
b. D/W air compressor suction valves (63,62) [+0.5]

REFERENCE
BFNP: OPL171.054, Control and Station Air Systems, p. 11.
Objective V.C.

3.3/3.2
295019A104 ...(KA'S)

ANSWER 6.03 (3.00)

a. False [+0.5] - once the low suction pressure signal is clear, the turbine will auto restart if the initiation signals are still present [+0.5].

b. False [+0.5] - the low steam pressure isolation signal does not seal in [+0.5].

c. True [+0.5] - the oil pressure will be restored when the turbine coasts down, thereby causing the stop valve to open [+0.5].

REFERENCE
BFNP: OPL171.042, HPCI, pp. 17, 25, and 26.
Objectives V.D.1, and V.D.2.

3.8/3.9 3.9/4.0 4.2/4.1 4.0/3.9
206000K401 206000K402 206000K403 206000K404 ...(KA'S)

ANSWER 6.04 (1.50)

a. No [+0.5]. (If a core spray line breaks inside the shroud, the differential pressure indicating switch will detect reactor pressure inside the shroud as usual; therefore, no abnormal differential pressure will be indicated.)

b. The core spray loop will have degraded performance [1.0] ~~can perform a flooding function [+0.5] but its spray will not provide full core spray coverage [+0.5].~~

REFERENCE
BFNP: OPL171.045, Core Spray, pp. 15 and 16.
Objective V.K.

2.8/3.0 3.0/3.2
209001K113 209001K404 ...(KA'S)

ANSWER 6.05 (2.00)

a. trip and throttle valve closes [+0.5]
min flow valve closes [+0.5]

b. Must manually open trip and throttle valve (71-9) from panel 9-3 by running motor generator to "close" position to relatch trip valve to trip solenoid. Then run motor operator to "open" position to open valve. [+1.0]

REFERENCE
BFNP: OPL171.040, RCIC.
Objective V.B.3.

3.5/3.5 3.3/3.3 3.8/3.7
217000G007 217000K102 217000K402 ...(KA'S)

ANSWER 6.06 (2.00)

a. local fuel damage (by generating a rod withdrawal block) [+1.0]

b. units = volts [+0.5], number of operable LPRM inputs can be calculated (by using 1 volt per operable input) [+0.5]

REFERENCE
BFNP: OPL171.035, RBM, pp. 4 and 28.
Objective V.K.

2.9/2.9 3.3/3.4 3.2/3.1
215002A402 215002G004 215002K102 ...(KA'S)

ANSWER 6.07 (2.50)

a. 2 (1 ok)

b. 4

c. 1

d. 4

e. 4 (0.5 each)

REFERENCE
BFNP: LP22, L.O. D

3.3/3.4 3.6/3.6 3.2/3.3
215005K116 215005K505 215005K607 ...(KA'S)

ANSWER 6.08 (2.00)

(1) Gross failure of a trip unit

(2) Card out of card file

(3) Calibration in progress

(4) Power supply failure (0.5 each)

REFERENCE
BFNP: LP 600, L.O. B.5

3.0/3.1 2.9/3.0
216000K118 216000K318 ...(KA'S)

ANSWER 6.09 (2.50)

a. (1) High voltage low

(2) Module unplugged

(3) Function switch not in operate (0.5 each)

b. None (0.5)

With the Mode Switch in "RUN", bypassing the companion APRM also bypasses IRM H. (IRM H's companion is APRM B) (0.5)

REFERENCE
BFNP: LP 20. L.O. C,G & H
LP 22, L.O. E & G
LP 28, L.O. K & L

3.9/3.9 3.9/4.0 3.9/4.0 3.7/3.8 3.7/3.7
4.0/4.0 3.2/3.4 3.5/3.7 3.7/3.6
215003K101 215003K106 215003K301 215003K305 215003K401
...(KA'S)

ANSWER 6.10 (1.50)

a. 75% (0.5)

b. Operable (0.5)

There are greater than 13 operable inputs on APRM C (0.5)

REFERENCE
BFNP: LP 21, L.O. A & B

3.6/3.6 3.6/3.4
215005G009 215005K104 ...(KA'S)

ANSWER 6.11 (1.00)

1. Level transmitter "A" (LIT 3-53)
2. Failed LOW (0.5 each)

REFERENCE
GGNS: OP-B21-501
BFNP: LP 3, L.O. J
LP 12, L.O. E.3
ARP 9-5: XA-55-5A-8, XA-55-5A-30, XA-55-5B-4, XA-55-5B-5

3.6/3.7 3.7/3.8 3.4/3.5 3.9/4.1
216000K112 216000K312 216000K313 216000K324 ...(KA'S)

ANSWER 6.12 (1.50)

a. 480v Shutdown Board XB (via RPS regulating transformer) (0.5)

b. (1) Both RPS buses cannot be simultaneously fed from the alternate power source. (0.5)

(2) Prevent paralleling RPS MG set with alternate power source. (0.5)

REFERENCE
BFNP: LP 28, L.O. C

3.2/3.3 3.0/3.1 3.1/3.1
212000K201 212000K403 212000K404 ...(KA'S)

ANSWER 6.13 (1.50)

a. (1) SRM count rate > 100 cps

(2) IRMs are Range 3 or greater (0.5 each)

b. 9-5: Retract Permit Light - ON

9-12: Retract Permit Light - OFF (0.25 each)

REFERENCE
BFNP: LP 19, L.O. G & H

3.7/3.7 2.8/2.8 3.1/3.1 3.1/3.2 3.2/3.1
215004A106 215004A405 215004A406 215004K401 215004K503
...(KA'S)

ANSWER 6.14 (2.50)

a. (1) DC motor operated valves

(2) DC motor operated pumps

(3) Control power for ECCS

(4) Logic power for ECCS

or any examples of that
eg Control pwr: 480V SD Boards, cooling tower swgr
4KV SD Boards. Logic pwr: specific systems
(any 3 @ 0.25 each)

b. The DC bus normally is supplied by a battery charger (0.25) powered from the 480v AC shutdown board (0.25).

Alternate power to the charger is from the 480v common board 1 (manual transfer only) (0.25).

Backup power is supplied by a (120 cell lead-acid) battery on a float charge (0.25).

c. (1) Provides more constant pull on coils

(2) Absence of hysterisis effects

(3) Absence of eddy current losses (0.25 each)

or any reasonable benefit of using DC power systems.

REFERENCE
BFNP: LP 37, L.O. A,B & C

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

3.3/3.5 3.2/3.3 3.4/3.6 3.3/3.5
263000G004 263000G007 263000K101 263000K102 ...(KA'S)

ANSWER 7.01 (2.00)

Operator 1 - Rejected - Would exceed 10 CFR 20 limit 1250 mrem/qtr with no Form 4 on file (0.5)
Operator 2 - Rejected - Would exceed 4000 mrem limit per year (0.5)
Operator 3 - Rejected - Would exceed pregnancy limit of 500 mrem (0.5)
Operator 4 - Accepted - Would not exceed quarterly limit of 1.25 rem whole body. 5(N-18) is limiting only when it is desired to exceed 1.25 rem/qtr whole body. (0.5)

REFERENCE
GPC: 60AC-HPX01-0, 10 CFR 20
EIH: SR-301, LP 300.3, LO #4
BFNP: RCI-1 LO B
3.3/3.8
294001K103 ...(KA'S)

ANSWER 7.02 (1.00)

d (1.0)

REFERENCE
BFNP: OI-66 L.O. I
3.3/3.6 3.8/4.1
271000G015 271000K404 ...(KA'S)

ANSWER 7.03 (2.50)

a. 1 (1.0)

b. 1) Low Reactor Water Level (.1) <= -114.5 ~~(0.2)~~ (.1) and (0.1) DG voltage applied to SD board(s) (0.2)

2) Drywell Pressure (.1) > 2.45 psig ~~(0.2)~~ (.1) and (0.1) DG voltage applied to SD board(s) (0.2)

3) Low discharge header pressure < 57 psig. (0.5)
50 psig Header pressure

REFERENCE
BFNP: OI 70 LO A , AOI-70 , LP 171.047
3.8/4.1 3.3/3.4 2.9/3.2
295018AA10 295018AK30 295018G011 ...(KA'S)

Alternate answers for 1) & 2)
Accident signal (.2) and (.2) loss of normal AC power (.4)
Initiation of unit 1 & 2 480V load shed logic (.8)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 7.04 (2.00)

a. TRUE (AOI) False (OI)
b. FALSE
c. FALSE
d. FALSE (0.5 each)

REFERENCE
BFNP: OI 85 LO H,I , 3-AOI-85-3
3.3/3.4 3.9/4.0 3.6/3.8 3.2/3.3
201001A201 201001G015 295022AK10 295022G011 ...(KA'S)

ANSWER 7.05 (1.50)

a. ~~FALSE~~ TRUE
b. TRUE
c. FALSE (0.5 each)

REFERENCE
BFNP: OPL171.053 LO A,C,D
3.1/3.7 3.4/3.7
234000G007 234000K502 ...(KA'S)

ANSWER 7.06 (1.50)

a. TRUE
b. FALSE
c. TRUE (0.5 each)

REFERENCE
BFNP: OSIL 28 LO K
3.3/3.6
294001K106 ...(KA'S)

ANSWER 7.07 (1.25)

1. 5 min
2. 1 hr
3. ENS - (red phone)
4. SE/SED
5. 2 hrs. (0.25 each)

REFERENCE
BFNP: EPIP-2 LO B,C,D
2.9/4.7
294001A116 ...(KA'S)

ANSWER 7.08 (2.00)

Red Pump Running Light illuminated
Squib Valve Hold Ready (Amber) Lights extinguish
SLC Loss of Continuity Annunciator alarms
SLC Pump Dicharge Pressure is greater than Reactor Pressure
Flow Light (White) is illuminated
SLC Injection Flow Annunciator alarms
RWCU isolates and Pumps trip
SLC Storage Tank Level decreasing

(0.25 each)

REFERENCE
BFNP: OI 63 LO A
3.8/3.8 4.1/4.2 4.0/4.1 4.2/4.2
21100A303 21100A305 211000A306 211000A308 ...(KA'S)

ANSWER 7.09 (1.50)

a. If the Reactor cannot be shutdown prior to Suppression Pool Temperature reaching 110 DEG F. (0.5)

b. ADS initiation may result in injection of cold (unborated) water from Low Pressure Injection Systems (0.5) and Boron Dilution (0.5).

REFERENCE
BFNP: EOI-1, OPL 171.057 LO B.6
4.4/4.7 3.7/3.9 4.0/4.2 4.2/4.4
295037EK10 295037G007 295037G011 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 7.10 (2.00)

1. Manually Scram the Reactor
2. Trip the Recirc Pumps
3. Start the DG's
4. Check at least 2 EECW Pumps operating
5. If operating, Trip the Main Turbine,(continue Bypass Valve operation for as long as possible)
6. Announce Control Room Evacuation to all plant personnel.
7. Proceed to Backup Control Center (in the Shutdown Board Room.)
8. Place all MSRV Disconnects and Transfer Switches to Emergency.
9. Close all MSIV'S.

(any 8 @ .25 each)

REFERENCE
BFNP: EPMM-6 , OPL 174.711 EO 2
3.8/3.6 4.4/4.5 4.2/4.3
295016AA10 295016AK20 295016G010 ...(KA'S)

ANSWER 7.11 (1.50)

RFP's warmed and ready for service
HPCI and RCIC steam lines warmed and Low Pressure Isolations reset.
SJAE in service
APRM's reading between 5 % and 12 % .
Inboard and Outboard MSIV's open
Cond. Vacuum > 24" Hg.
Reactor Pressure > 850 psig.
All section 3 signoffs completed

(any 6 @ .25 each)

REFERENCE
BFNP: OPL 174.724 LO B.9
4.0/4.1 4.3/4.5
212000A216 212000G001 ...(KA'S)

ANSWER 7.12 (1.00)

TIP (0.25) withdrawn (0.25)
Nitrogen Isolation Valves to Primary Containment (0.25) closed (0.25)
or
CAD system isolated
or
N_2 purge valves isolated

REFERENCE
BFNP: BF 14.9 LO A
3.2/3.7 3.2/3.4

294001K105 294001K114 ...(KA'S)

ANSWER 7.13 (1.00)

Because the BHRSW on-line process monitors lack required sensitivity (.25) to detect a small leak (.5) that could be in excess of 10CFR20 release rate limits. ~~(1.0)~~ (.25)

REFERENCE

BFNP: OI 23 LO B
3.3/3.8 3.3/3.6 3.7/3.9
294001K104 294001K103 290001G001 ...(KA'S)

ANSWER 7.14 (1.00)

At higher temperatures, venting could lead to removal of all non-condensibles and result in a saturated steam environment (0.5). Subsequent condensing of the steam may reduce pressure (to less than 2 psig vac) leading to potential collapse of the containment (0.5).

REFERENCE
BFNP: EOI-2 , OPL 171.057 EO 10
3.8/4.0 3.6/3.8
295010AK30 295010G007 ...(KA'S)

ANSWER 7.15 (1.00)

To limit cold water entry into the Vessel Bottom Head (0.5) to avoid exceeding 145 deg F delta T (0.1) between Steam Dome temperature (0.2) and Bottom Head temperature (0.2) (if immediate Reactor cooldown is not anticipated).

REFERENCE
BFNP: OPL 174.725 LO B.1
4.2/4.2
295006AA10 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 7.16 (2.00)

a. Place Recirculation Subpanel in Manual (~~0.5~~) (.25) and reduce speed to establish 100 % Loop Flow (45,200 gpm) (~~0.5~~) (.75).

b. Prevents excessive Jet Pump vibration. (1.0)

REFERENCE
BFNP: OI-68, LO E
3.4/3.4 3.9/3.5 3.1/3.5 3.6/3.7
202001A203 202001K411 202002A201 202002G014 ...(KA'S)

ANSWER 7.17 (1.00)

The Scram off of the Turbine Stop Valve position is armed when power is above 30 % (142 psig) as sensed by the 1st Stage Pressure (0.5). In Shell Warming, the Stop Valves are closed (0.5) (and a scram would occur).

REFERENCE
BFNP: OPL 171.010 LO D , OI-47
3.6/3.7 3.3/3.5 3.2/3.4 4.0/4.1
212000A212 212000K110 245000G001 245000K104 ...(KA'S)

ANSWER 7.18 (1.00)

a. Minimizes the possibility of damage to the Turbocharger Thrust Bearing. (0.5)

b. Manual Slow Starts (0.25) from the Engine Control Cabinet (0.25).
Local control cabinet/panel

REFERENCE
BFNP: OI-82 LO A
3.7/3.7 3.7/4.2
264000A404 264000G001 ...(KA'S)

ANSWER 8.01 (1.00)

a

REFERENCE
EIH: U1 TS, Section 1.1.c
BFNP: U1 TS, Section 1.1.B , OPL174.728 LO 3
3.2/3.7
293009K118 ...(KA'S)

ANSWER 8.02 (1.00)

d

REFERENCE
BFNP: OPL174.728 LO 2 , TECHNICAL SPECIFICATIONS
2.7/3.5
295018G004 ...(KA'S)

ANSWER 8.03 (2.50)

a. not required
b. not required
c. required
d. not required
e. not required (0.5 each)

REFERENCE
BFNP: RCI 9 LO G
3.3/3.8
294001K103 ...(KA'S)

ANSWER 8.04 (~~2.00~~ 1.5)

~~a. TRUE~~ Deleted
b. TRUE
c. FALSE
d. FALSE (.5 each)

REFERENCE
BFNP: OPL174.728 LO 66,67,68 , TECHNICAL SPECIFICATIONS
3.4/3.6 3.8/4.0 3.2/4.1
286000G001 286000G011 286000K401 ...(KA'S)

ANSWER 8.05 (2.00)

a. 1) Applicable OSIL (0.25)
   2) Plant Manager (0.25)

b. The clearance for equipment shall be held by the SE (0.25) and he shall note the person's NAME (0.25), PURPOSE (0.25), TIME and DATE (0.25) under the " Remarks " section.

c. A numbered identification tag used for administrative control of electrical grounds (~~0.25~~) (.45) in conjunction with a clearance (~~0.25~~) (.1).

REFERENCE
BFNP: 14.25 LO B,C
3.9/4.5
294001K102 ...(KA'S)

ANSWER 8.06 (1.00)

1. 5.6
2. 8.6
3. 1.0
4. 0.2 (0.25 each)

REFERENCE
BFNP: OPL174.728 LO 42 , TECHNICAL SPECIFICATIONS
2.9/3.4
294001A114 ...(KA'S)

ANSWER 8.07 (1.50)

(1) System fails to operate.

(2) System operates in a suspected adverse manner.

(3) System operates outside of the limits of the documented acceptance criteria. (0.5 each)

REFERENCE
BFNP:(Standard Practice 10.9, L.O. "B") , PMI-17.1
3.9/4.5
294001K102 ...(KA'S)

ANSWER 8.08 (2.00)

a. Ensure inadvertent transfer of significant amount of containment fluids will not occur. ~~(0.5)~~ 1.0

b. Continued operation of the engineered safety features will result in an unsafe plant condition (with regard to either personnel or operability of safety features). (0.5)

The plant is in a stable condition (in which technical specifications clearly indicate that) operability of the engineered safety feature is no longer required. (0.5)

REFERENCE
BFNP: BF SP 12.17 LO A
3.8/4.0 3.6/3.8
223001G001 223001K102 ...(KA'S)

ANSWER 8.09 (1.50)

(1) Highest worth rod (0.25) fully withdrawn (0.25)
(2) Xenon free core (0.5)
(3) Cold core (68 deg F) (0.5)

or
1) same
2) all other RODS inserted
3) .38 % $\frac{\Delta K}{K}$ SD margin

REFERENCE
EIH: U2 TS, 1.0 "SDM"
BFNP: U1 TS, 3.3/4.3.A , OPL174.728 LO 9
3.2/3.5
292002K110 ...(KA'S)

ANSWER 8.10 (1.50)

Time
~~Rod Group~~ Delete
Rod Number
Rod Notch
Period
Recirc Loop Temperature (~~0.25~~ .30 each)

REFERENCE
BFNP: OPL 174.724 LO 6 , GP 100-I
3.8/3.9 4.3/4.3
292008K101 292008K105 ...(KA'S)

ANSWER 8.11 (3.00)

a. Core Spray System I Room Coolers => Core Spray System I
is Inoperable. (~~0.25~~) (.5)

LCO - (3.5.A.2) The Reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS Loop and the RHR system (LPCI Mode) and the Diesel Generators are OPERABLE. (~~0.25~~) (.5)

CURRENT SURVEILLANCE RQMTS - (4.5.A.2) The other Core Spray Loop (SYS II) shall be demonstrated to be operable daily. (~~0.25~~) (.5)

RHR Pump 3C Inoperable (0.25) deleted

LCO - (3.5.B.12) If one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit is INOPERABLE for any reason, the reactor may remain in operation for a period not to exceed 30 DAYS provided the remaining RHR Pump and associated diesel generator are OPERABLE. (0.25)

CURRENT SURVEILLANCE RQMTS - (4.5 B.12) Remaining RHR Pump (3A) and associated Heat Exchanger on the unit cross-connection and the associated Diesel Generator shall be demonstrated to be OPERABLE every 15 DAYS until the inoperable pump and associated heat exchanger are returned to normal service. (0.25)

b. 2 days (7 days from 3/18/88) or 3/25/88 (0.5)

c. LCO - (3.5.B.8) Since specification 3.5.B.3 cannot be met:
An orderly shutdown shall be initiated and the reactor shall be shutdown and placed in the cold condition within 24 hours. (1.0)
or 3.5.A.3

REFERENCE
BFNP: OPL 174.728 LO 71 , TECHNICAL SPECIFICATIONS
3.6/4.5 3.4/4.2
203000G011 209001G011 ...(KA'S)

ANSWER 8.12 (2.00)

1. Green - information deleted (by the TEMPALT)

2. Red - information added (by the TEMPALT)

3. Red circle - surrounds area affected (by the TEMPALT)
or yellow circle

4. TACF # - (number assigned to track the TEMPALT) and is placed beside the red circle

(0.5 each)

REFERENCE
BFNP: PMI-8.1, L.O. F
(3.0/3.7)
294001A107 ...(KA'S)

ANSWER 8.13 (2.00)

Relief Valve " B " setting of 1325 psig (<1350 psig) (0.5)
LCO - (3.4.B) Continued operation permitted provided that the component is returned to an operable condition within seven days. (0.5)

Sodium Pentaborate Solution LESS than minimum required by figure 3.4-1 (0.5)
LCO - (3.4.D) The Reactor shall be placed in a shutdown condition with all operable control rods fully inserted within 24 hours, (0.5) (unless condition can be corrected within specified time allowed.)

REFERENCE
BFNP: OPL 174.728 LO 71,20 , TECHNICAL SPECIFICATION 3.4
3.4/4.1 3.6/4.4
21000G005 21000G011 ...(KA'S)

ANSWER 8.14 (1.50)

EMERGENCY - Action is required to prevent imminent major equipment damage or to protect personnel from any imminent threat of bodily injury (0.35/0.15)

IMMEDIATE ATTENTION - Work is to be performed within the next 24 hours (, the next scheduled workday, or upon completion of necessary technical evaluations or material procurement) (0.5)

ROUTINE MAINTENANCE - Work is to be performed as manpower and circumstances permit. (0.5)

REFERENCE
BFNP: SP 7.6 L O A
3.9/4.5
294001K102 ...(KA'S)

ANSWER 8.15 (2.00)

a. Drywell
Pressure Suppression Chamber (0.5 each)

b. (1) All non-automatic containment isolation valves (on lines connected to reactor coolant systems or containment) which are not required to be open during accident conditions are closed.

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

(3) All automatic containment isolation valves are operable or deactivated in the isolated position.

(4) All blind flanges and manways are closed.
(0.25 each)

REFERENCE
BFNP: OPL174.728 LO 1.J , TECHNICAL SPECIFICATIONS 1.0
3.6/3.7 3.0/4.0
223001G004 223001G006 ...(KA'S)

QUESTION	VALUE	REFERENCE
05.01	1.50	GTH0000720
05.02	1.00	GTH0000721
05.03	2.00	GTH0000723
05.04	1.00	GTH0000724
05.05	2.00	GTH0000726
05.06	2.00	GTH0000727
05.07	1.00	GTH0000729
05.08	1.25	GTH0000730
05.09	1.00	GTH0000731
05.10	1.00	GTH0000732
05.11	1.00	GTH0000733
05.12	2.00	GTH0000658
05.13	2.00	GTH0000661
05.14	3.00	GTH0000719
05.15	1.50	GTH0000717
05.16	2.00	GTH0000747
05.17	1.00	GTH0000657
	26.25	
06.01	2.00	GTH0000735
06.02	1.00	GTH0000738
06.03	3.00	GTH0000739
06.04	1.50	GTH0000741
06.05	2.00	GTH0000743
06.06	2.00	GTH0000745
06.07	2.50	GTH0000778
06.08	2.00	GTH0000781
06.09	2.50	GTH0000782
06.10	1.50	GTH0000773
06.11	1.00	GTH0000774
06.12	1.50	GTH0000776
06.13	1.50	GTH0000779
06.14	2.50	GTH0000785
	26.50	
07.01	2.00	GTH0000712
07.02	1.00	GTH0000710
07.03	2.50	GTH0000757
07.04	2.00	GTH0000758
07.05	1.50	GTH0000759
07.06	1.50	GTH0000760
07.07	1.25	GTH0000761
07.08	2.00	GTH0000748
07.09	1.50	GTH0000749
07.10	2.00	GTH0000750
07.11	1.50	GTH0000755
07.12	1.00	GTH0000762
07.13	1.00	GTH0000711

QUESTION	VALUE	REFERENCE
07.14	1.00	GTH0000751
07.15	1.00	GTH0000752
07.16	2.00	GTH0000753
07.17	1.00	GTH0000754
07.18	1.00	GTH0000756
	26.75	
08.01	1.00	GTH0000715
08.02	1.00	GTH0000771
08.03	2.50	GTH0000765
08.04	2.00	GTH0000770
08.05	2.00	GTH0000764
08.06	1.00	GTH0000769
08.07	1.50	GTH0000709
08.08	2.00	GTH0000713
08.09	1.50	GTH0000714
08.10	1.50	GTH0000763
08.11	3.00	GTH0000768
08.12	2.00	GTH0000746
08.13	2.00	GTH0000767
08.14	1.50	GTH0000716
08.15	2.00	GTH0000766
	26.50	
	106.00	

DOCKET NO 259

MASTER

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: BROWNS FERRY 1, 2&3

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 88/03/23

EXAMINER: HOPPER, G

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
25.75 ~~26.25~~	24.85 ~~24.82~~	________	________	1.	PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
26.0 ~~26.50~~	25.12 ~~25.06~~	________	________	2.	PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.0 ~~25.75~~	24.15 ~~24.35~~	________	________	3.	INSTRUMENTS AND CONTROLS
26.75 ~~27.25~~	25.85 ~~25.77~~	________	________	4.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
103.5 ~~105.75~~		________ Final Grade	________%		Totals

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

____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

3. Use black ink or dark pencil <u>only</u> to facilitate legible reproductions.

4. Print your name in the blank provided on the cover sheet of the examination.

5. Fill in the date on the cover sheet of the examination (if necessary).

6. Use only the paper provided for answers.

7. Print your name in the upper right-hand corner of the first page of <u>each</u> section of the answer sheet.

8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a <u>new</u> page, write <u>only on one side</u> of the paper, and write "Last Page" on the last answer sheet.

9. Number each answer as to category and number, for example, 1.4, 6.3.

10. Skip at least <u>three</u> lines between each answer.

11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.

12. Use abbreviations only if they are commonly used in facility <u>literature</u>.

13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.

15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

16. If parts of the examination are not clear as to intent, ask questions of the <u>examiner</u> only.

17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

(2) Exam aids - figures, tables, etc.

(3) Answer pages including figures which are part of the answer.

b. Turn in your copy of the examination and all pages used to answer the examination questions.

c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 1.01 (1.50)

During startup you have established a stable 160-second period. By definition IS the reactor CRITICAL, SUBCRITICAL or SUPERCRITICAL? EXPLAIN the difference between subcritical and supercritical. ASSUME no startup sources are present. (1.5)

QUESTION 1.02 (3.00)

LIST the three (3) "thermal limits" observed during reactor operation and STATE the limiting condition for each.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.03 (1.50)

a. DOES the delayed neutron fraction INCREASE or DECREASE from the beginning of cycle (BOC) to the end of cycle (EOC)? (0.5)

b. LIST the two (2) major causes for the change in delayed neutron fraction from BOC to EOC. (1.0)

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.04 (1.00)

Reactor power is 60 on IRM range 2 with the MINIMUM permissible stable positive period allowed by procedure GOI-100-1. Heating power is determined to be 40 on IRM range 7. CALCULATE how long it will take for power to reach the point of adding heat if the period remains constant.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.05 (2.00)

SELECT the appropriate response for EACH of the following statements concerning Control Rod Worth:

a. (MORE/LESS) control rods would need to be pulled to make the reactor critical at 545 degrees F, as opposed to 140 degrees F.

b. An INCREASE in the Void Fraction will result in an (INCREASE/DECREASE) in individual control rod worth.

c. Control Rod Worth at End of Cycle would be (LESS/GREATER) than at the Beginning of Cycle.

d. Control Rod Worth will (INCREASE/DECREASE) with an INCREASE in moderator temperature.

e. Control Rod Worth will (INCREASE/DECREASE) as the adjacent control rods are withdrawn.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.06 (1.00)

Briefly EXPLAIN HOW a control rod withdrawal of one or two notches can result in a decrease in bundle power.

QUESTION 1.07 1.50 ~~(2.00)~~

For each of the following events, STATE which COEFFICIENT of reactivity (MODERATOR, VOID, DOPPLER) would act FIRST to change reactivity.

a. control rod drop at 25 percent power

b. SRV opening at 50 percent power

Deleted ~~c. loss of~~ shutdown cooling when removing decay heat

d. one recirc pump trips while at 50 percent power

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.08 (2.00)

Following a reactor scram from power, several control rods fail to insert to the full-in position. One hour after the scram, the Reactor is determined to be subcritical with a shutdown margin (SDM) of 0.22 % delta K/K.

a. If reactor coolant temperature and control rod positions remain constant during the next hour (from 1 - 2 hours after the scram), would SDM INCREASE, DECREASE, or REMAIN the SAME? Briefly EXPLAIN your answer.

b. During the next hour (2 - 3 hours after the scram) you notice reactor pressure is decreasing. WHAT effect would ONLY the pressure decrease have on SDM. Briefly EXPLAIN your answer.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.09 (1.00)

The reactor has been operating at 95 percent power for several days. An operator RAPIDLY reduces reactor power to 60 percent by reducing the speed of the recirculation pumps. During the next 2-3 MINUTES the operator notices that the reactor power slowly increases to 63 percent (with no operator action). EXPLAIN the cause of the power increase.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.10 (1.25)

As a reactor operator coming on shift, you are told that the previous shift performed a reactor shutdown and commenced a cooldown from 630 psig at 0630. It is now 0730 and you note that wide range reactor pressure is 200 psig. Your shift is to place the reactor in shutdown cooling.

a. HAS the previous shift exceeded the Technical Specification maximum allowable cooldown rate (YES or NO)? INCLUDE in your answer the Technical Specification Cooldown Limit and the assumptions and calculations used.) (0.75)

b. HOW many more degrees of cooldown are necessary before RHR can be unisolated for shutdown cooling? (INCLUDE your assumptions and calculations.) (0.5)

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.11 (1.00)

Concerning the Bypass Flow in the reactor core, STATE the two (2) most significant consequences that would occur if bypass flow were significantly reduced at full power.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.12 (1.00)

A centrifugal pump is operating at rated speed with a discharge head of 240 psig and power consumption of 0.5 MW. The speed of the pump is then decreased until the power consumption is 1/64 of its original value. CALCULATE the new discharge head. SHOW ALL WORK.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.13 (1.00)

A leak develops in the low pressure side of a flow transmitter. HOW will this effect the flow indication for the instrument? (CHOOSE one.)

a. The indicated delta pressure would decrease causing the indicated flow to decrease.

b. The indicated delta pressure would decrease causing the indicated flow to increase.

c. The indicated delta pressure would increase causing the indicated flow to decrease.

d. The indicated delta pressure would increase causing the indicated flow to increase.

QUESTION 1.14 (2.00)

Answer EACH ONE of the following statements TRUE or FALSE regarding reactivity coefficients.

a. An increase in flow through the reactor core will add negative reactivity by decreasing the void fraction and thus increasing reactor power.

b. As the burnable poison within a fuel bundle burns out, the VOID coefficient becomes more negative.

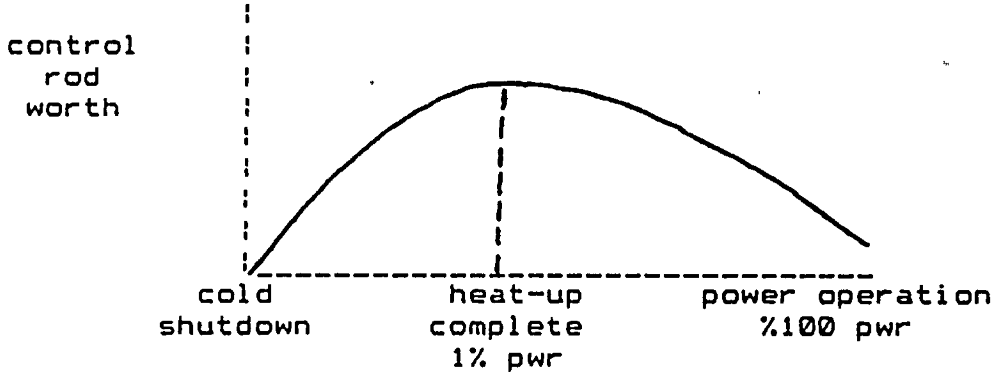
c. LATE in core life, the large reduction in fuel molecules and the decrease in moderator density during a plant HEAT-UP can lead to a positive reactivity addition.

d. As core age progresses, the DOPPLER coefficient becomes more negative due to plutonium-240 buildup.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.15 (2.00)

Answer EACH ONE of the following questions TRUE or FALSE concerning the graph of Contol Rod Worth During a Startup.



a. Control rod worth increases during heatup due to density decreases of the moderator which causes longer slowing down and thermal diffusion lengths, resulting in more thermal flux around a control blade.

b. Control rod worth decreases as power exceeds 1% due to the effects of rod shadowing. Withdrawal of rods increases the thermal diffusion length thereby increasing the flux around a control blade.

c. While heating-up, rod worth increase is due mainly to the effects of Bundle Coupling. Rod withdrawal couples fuel cells together making their effective size larger, resulting in increased leakage and a reduction in thermal flux.

d. Since control rods are worth more when the moderator is hot, fewer control rods must be withdrawn to go critical when the reactor is hot than when cold.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.16 (2.00)

The attached FIGURE (GTH-747) represents parameter changes for a plant transient on UNIT TWO. Use this figure and the following information to answer EACH of the questions below:

(1) Initial Power Level = 100 %
(2) Bypass Valves go to Full Open position
(3) No operator action is taken

a. The DECREASE in turbine steam flow. (POINT 4)

b. The INCREASE in power. (POINT 7)

c. The INCREASE in turbine steam flow. (POINT 5 and AREA 6)

d. The DECREASE in pressure. (POINT 2)

QUESTION 1.17 (1.00)

CALCULATE the equilibrium neutron count rate in a subcritical reactor after FOUR (4) generations given the following initial conditions:

Source = 100cps
Keff = .2
ASSUME: generation 0 consists of only source neutrons and equilibrium is achieved after four generations

(***** END OF CATEGORY 01 *****)

QUESTION 2.01 (2.00)

Answer EACH of the following concerning the SBLC System:

a. STATE two (2) reasons why the pump motors are interlocked such that only one motor can be run at a time.

b. WHAT is the purpose of the heat tracing that is provided on the system piping?

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.02 (2.00)

STATE whether the following statements concerning the Primary Containment Isolation System are TRUE or FALSE:

a. Most of the PCIS motor operated valves fail closed on loss of power to the valve.

b. The containment isolation reset switches on panel 9-5 must be operated to manually reset a RCIC turbine steam supply isolation.

c. Loss of RPS Bus A will NOT cause any PCIS isolation valves to close.

d. The TIP guide tube ball valve will isolate on a high radiation signal.

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.03 (2.50)

During startup, the Rod Worth Minimizer operator panel shows that control rods 22-31 and 30-39 are in "insert error" status.

a. DOES a rod block exist? EXPLAIN your answer. (1.5)

b. WHAT panel indication should tell you if a rod block is in force? (1.0)

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.04 (2.00)

WHAT four (4) conditions must exist before the D/G output breaker will close?

QUESTION 2.05 .50 ~~(1.00)~~

During your shift the Drywell Air System (DWAS) isolates. You verify a Group VI isolation has not occurred.

Deleted ~~a. NAME one o~~ther signal that could have caused the DWAS isolation.

b. WHAT air system valves close when the DWAS isolates?

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.06 (3.00)

For EACH of the following statements regarding the High Pressure Coolant Injection System, INDICATE whether the statement is TRUE or FALSE, and EXPLAIN your answer.

a. In the event low HPCI booster pump suction pressure is sensed during HPCI system operation, the turbine will trip, and the signal must be manually reset before the turbine will restart, even if initiation signals are still present.

b. Upon a HPCI system isolation due to low steam pressure, the system cannot restart until the pressure rises above the isolation setpoint and the isolation signal is reset.

c. If the HPCI turbine trips due to an overspeed condition, it will restart when the speed coasts down to less than 5000 rpm.

QUESTION 2.07 (3.00)

Concerning the CRD system:

a. WHAT are the normal values for CRD hydraulic system FLOW and DRIVE WATER DIFFERENTIAL PRESSURE? 1.0 (~~0.5~~)

b. WHAT percentage of CRD hydraulic system FLOW is supplied to the CRD cooling water header? .5 (~~1.0~~)

c. Immediately following a reactor scram the control rod full-in (green) lights on panel 9-5 are lit but there is no position readout displayed. EXPLAIN WHY this occurs and WHAT eventually happens that allows the control rod to settle into the "00" position. (1.5)

QUESTION 2.08 (1.50)

A Core Spray line breaks inside the shroud.

a. WILL the break cause an alarm in the control room (YES or NO)? (0.5)

b. HOW will the break affect core spray performance for that loop? (1.0)

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.09 (3.00)

For EACH of the following situations, STATE whether or not the indicated activity can occur. If the activity cannot occur, WHAT must change to allow it to occur?

a. The Refuel Bridge is over the reactor vessel and in motion toward the fuel pool with the fuel grapple loaded. All rods are inserted. The reactor mode switch position is changed from REFUEL to STARTUP. WILL the bridge continue to move?

b. The Refuel Platform is over the vessel. The frame mounted hoist is loaded. One rod is at position 30. CAN the load on the hoist be lowered into the vessel?

c. The Refuel Platform is over the vessel with the mode switch in REFUEL. The grapple is fully lowered and unloaded. CAN a control rod be withdrawn?

QUESTION 2.10 (2.00)

The RCIC system is in operation on your shift to demonstrate operability for Technical Specifications.

a. DESCRIBE what occurs to the RCIC system (components) if reactor water level exceeds 54 inches.

b. A low-low reactor water level condition occurs after the high level condition described in part a. above. DESCRIBE the operator actions required to permit the RCIC system to respond to this low-low level condition.

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.11 (2.50)

The Reactor Water Cleanup System is in operation with one pump and one filter demineralizer in service. A reactor startup and heatup is in progress with wide range reactor pressure indicating 400 psig. The RWCU dump valve is open, rejecting water to the main condenser to control reactor water level. Suddenly, the operator receives a RWCU low pump flow alarm and notes that system flow is 0 gpm and the previously running pump has stopped. Answer EACH of the following concerning the above situation:

a. LIST five (5) possible causes of the pump trip (other than pump failure). (1.0)

b. STATE whether the RWCU blowdown valve WILL or WILL NOT isolate CONCURRENTLY with any of the pump trips. (0.5)

c. In the above example, if the operator also notices that an RWCU system isolation has also occurred, STATE HOW the RWCU dump valve position at the time of the isolation can cause significant stress upon the RWCU system piping and components. (1.0)

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.12 (2.00)

For the Rod Block Monitor (RBM), answer EACH of the following:

a. WHAT adverse condition is the system designed to prevent?

b. When the Meter Function Switch on the Back Panel 9-14 Meter Section is in the "count" position, WHAT are the "units" of the indication on the meter and WHAT can be calculated by utilizing the indicated value?

(***** END OF CATEGORY 02 *****)

QUESTION 3.01 (2.50)

The plant is operating at 100% power and 100% core flow when the "A" flow converter output fails to zero. MATCH from column B the action that will exist for each trip function in Column A given the above conditions.

NOTE: REPSONSES MAY BE USED MORE THAN ONCE

COLUMN A	COLUMN B
a. "A" APRM Hi-Hi thermal	1. Rod Block
b. "B" APRM Hi-Hi thermal	2. Half Scram
c. "C" APRM Hi	3. Full Scram
d. "D" APRM Hi	4. None
e. "E" APRM Hi-Hi neutron	

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.02 (2.00)

LIST four (4) conditions that will initiate the annunciator "RPS ATU TROUBLE" on Panel 9-5.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.03 (2.50)

A reactor startup of Unit 2 is in progress. The plant is operating at 7% in the process of power ascension to 100% Rated Thermal Power. The Mode Switch has just been placed in "RUN" and the following equipment is out of service:

Condensate Pump A
CRD Pump b
APRM B Bypassed (FAILED DOWNSCALE)
RBCCW Pump C

A RED Hi-Hi/Inop light suddenly illuminates on the apron section of Panel 9-5 for IRM H. A check of Panel 9-14 determines a WHITE INOP light is illuminated for IRM H.

Answer EACH of the following with regards to the above situation:

a. LIST three (3) causes for the indications on IRM H. (1.5)

b. STATE what automatic trips should occur. JUSTIFY your response. (1.0)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.04 (1.25)

The Browns Ferry recirculation pump MG sets possess sixteen (16) different drive motor breaker trips. LIST five (5) of those trips which ALSO cause a scoop tube lock. (SETPOINTS NOT REQUIRED)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.05 (2.00)

You are in the process of preparing the Main Turbine for startup in accordance with OI-47.III.c. The following conditions exist:

Main Turbine is reset
VALVES CLOSED is selected
Warming rate indicator is at zero position
Load limit is set at 100%
FAST acceleration rate is selected

a. STATE the position for EACH of the following valves with the turbine in this condition.

(1) Main Stop Valves
(2) Control Valves
(3) CIVS Stop
(4) Intercept

b. You now select SHELL WARMING to prewarm the turbine by pressurization of the HP turbine. STATE the new position of the valves, specified in part "a" above, given this changed condition.

QUESTION 3.06 (1.50)

Answer EACH of the following given the below data for APRM Channel C:

LPRM Level:	A	B	C	D
No. of LPRMs assigned:	6	5	5	5
No. of LPRMs bypassed:	3	0	3	0

a. If APRM Channel C selector switch on the local (back) panel was placed to the COUNT position, STATE the expected meter reading. (0.5)

b. Based on the above information, STATE whether APRM Channel C is operable or inoperable. JUSTIFY your response. (1.0)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.07 (1.00)

Unit 2 is operating at 100% Rated Thermal Power with "A" feedwater level control selected. All other controls are in Normal/Automatic. A failure in a reactor vessel level narrow range instrument has occurred and resulted in the following related trips/indications:

REAC VESSEL WATER LEVEL LOW-LOW CHAN A ANNUNCIATOR - OFF
REAC VESSEL WATER LEVEL LOW-LOW CHAN B ANNUNCIATOR - OFF
"A" NR level indication reads at MINIMUM
"B" & "C" NR level indications read at MAXIMUM
REACTOR WTR LEVEL A ABNORMAL ANNUNCIATOR - ON
Feedwater flow is at ZERO
Two channels of ~~Level 8~~ have TRIPPED
One channel of ~~Level 3~~ has TRIPPED

STATE which NR level transmitter has failed and STATE in which direction it has failed (HIGH/LOW).

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.08 (3.00)

Answer EACH of the following with respect to the Rod Sequencer Control System:

a. The RSCS was developed for three different regions of rod withdrawal:

(1) 100% rod density to 50% rod density
(2) 50% rod density to preset power level
(3) Beyond preset power level

For EACH region above, STATE BOTH the design function of the RSCS AND the type of rod control in effect to accomplish this function. (1.5)

b. During a reactor startup under rod sequencer "A", all A12 and A34 rods are fully withdrawn then the Sequencer Mode Selector (SMS) and Rod Sequencer Selector (RSS) switches are placed in "Normal". LIST four (4) interlocks this action enables. (1.0)

c. STATE the effect on RSCS if its turbine generator 1st stage shell pressure input fails HIGH. (0.5)

QUESTION 3.09 (1.50)

Answer EACH of the following regarding the Reactor Protection System (RPS) for Unit 1:

a. STATE the alternate source of power to the RPS bus. (BE SPECIFIC! INCLUDE VOLTAGE AND BOARD NUMBER) (0.5)

b. LIST two (2) interlocks associated with this alternate power supply. (1.0)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.10 (1.50)

Answer EACH of the following with regard to the Source Range Monitors (SRMs):

a. STATE the two (2) permissives that must be satisfied before an operator may withdraw the SRMs without producing a rod block. (1.0)

b. DESCRIBE the indications available to the operator on Panels 9-5 and 9-12 that signify the SRMs may be withdrawn without causing a rod block. (0.5)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.11 (1.50)

Unit 3 is operating at 50% Rated Thermal Power with rods being pulled to establish the 100% rod pattern. The Rod Block Monitor (RBM) bypass joystick on the 9-5 panel is in "Normal". Periodically the following indications change on the 9-5 panel as various rods are selected and moved:

All "Detector Bypass" lights energize
Both RBM recorders go downscale

a. EXPLAIN the cause of these changes in indication (INCLUDE initiating event and component(s) affected).

b. EXPLAIN WHY this condition is normal and facility design permits continued operation.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.12 (3.00)

Answer EACH of the following with regard to the Standby Auxiliary Power System:

a. STATE the three (3) positions associated with the Diesel Generator Operational Mode Switch and DESCRIBE the function of each position. (1.5)

b. The Diesel Generators receive an auto start signal on Unit 2 low-low-low reactor water level. STATE the mode of voltage regulator operation that will be in effect. (0.5)

c. Subsequent to the above auto start signal, a "START FAILURE" alarm is received. STATE two (2) causes for this alarm. (SETPOINTS REQUIRED FOR FULL CREDIT) (1.0)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.13 (2.50)

Answer EACH of the following with regard to the 250v Unit and Plant DC power system:

a. LIST three (3) major types of loads supplied by this system. (0.75)

b. EXPLAIN how a reliable source of DC power is maintained to these loads. INCLUDE ALL NORMAL, ALTERNATE & BACKUP POWER SUPPLIES AND ASSOCIATED COMPONENTS! (1.0)

c. EXPLAIN why DC power is preferred for these types of load (other than for improved reliability). BE SPECIFIC! THREE RESPONSES REQUIRED FOR FULL CREDIT! (0.75)

(***** END OF CATEGORY 03 *****)

QUESTION 4.01 (2.00)

Entry into a HIGH RADIATION AREA is required. To complete the task, the operator will receive an estimated 70 mrem whole body dose. You have the following information on available operators. Time constraints will not permit authorization of an increase in administrative limits. NRC Form 4s are on file unless otherwise indicated.
STATE your REASONS for accepting or rejecting each operator for the job.

OPERATOR	1	2	3	4
SEX	male	male	female	male
AGE	29	30	24	20
Wk EXP	200 mrem	0 mrem	5 mrem	90 mrem
QTR EXP	1190 mrem	950 mrem	435 mrem	5 mrem
ANN EXP	2170 mrem	3990 mrem	750 mrem	2500 mrem
LIFE EXP	---	55370 mrem	2735 mrem	10050 mrem
Remarks	History Unavailable		3 months Pregnant- Signed Prenatal Document on File	

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.02 (1.00)

With Unit 2 operating at 90% power, an Off Gas Hydrogen [H-2] High alarm is received. Both Hydrogen analyzers indicate a Hydrogen level of ~ 5.3%. Which one of the following responses most accurately reflects the proper course of action you should take?

a. Change over to the alternate Recombiner.

b. Change over to the Alternate Off Gas Train.

c. Start an additional SJAE to assist in Condenser H-2 removal.

d. Manually Scram the reactor.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.03 (1.00)

WHICH one (1) of the following scenarios would result in the assumption that the Fuel Cladding Integrity Safety Limit of the Unit 1 Technical Specifications had been exceeded ?

a. Reactor power is at 42% RTP; the main turbine trips due to an EHC malfunction; the reactor SCRAMS on HIGH PRESSURE; the BPV's control pressure thereafter.

b. Reactor power is at 70% RTP; a steam leak to the Drywell occurs and Drywell pressure rises; the reactor SCRAMS at 1.85 psig; HPCI auto-actuation does not occur, but manual start is successful; the reactor is brought to a cold shutdown condition.

c. Reactor is in Start-Up, at 12% RTP; power is increased by rod pull; the reactor SCRAMS at 12.5% power, by APRM's; level and pressure are maintained by normal systems for the plant status.

d. The reactor is at 18% RTP; 1-1/2 BPV's are open in preparation for turbine warmup; controller failure reduces pressure to 875 psig; MSIV's close; reactor SCRAMS; level and pressure are maintained by normal systems for the plant status.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.04 (2.50)

The following parameter changes / annunciators are observed by the reactor operator:

RBCCW temperature Lower than normal
RBCCW Surge Tank HI Level alarm
(No other alarms present)

a. WHICH one (1) of the following malfunctions would most likely cause these indications: (1.0)

1. Raw Cooling Water leak in the RBCCW Heat Exchanger(s).

2. Reactor Coolant leak into RBCCW via NRHX.

3. Fuel Pool Cooling System leak from RBCCW.

4. RBCCW Makeup Valve (fill valve) leak

5. DWEDS Heat Exchanger leak into RBCCW.

b. LIST three (3) of the conditions/circumstances that will cause the isolation valve to non-essential equipment (MOV-48) to automatically close. (1.5)

NOTE: BE SPECIFIC AND INCLUDE SETPOINT VALUES

QUESTION 4.05 (2.00)

A Reactor Startup is in progress on UNIT 2 when the operating CRD Pump trips on motor overload. The Backup Pump is under repair and is expected to be operable within 45 minutes. The following initial conditions exist:

Reactor Power = 20 %
Reactor Pressure = 610 psig
Charging Water Pressure = 1490 psig (decreasing slowly)
1 ACCUM light on Full Core Display Illuminated

ANSWER EACH of the following questions TRUE or FALSE.

a. A Manual Scram is required if Reactor Pressure drops below 600 psig.

b. A Manual Scram is required if Charging Water Pressure cannot be maintained above 1410 psig.

c. A Manual Scram is required if 30 Control Rods receive High Temperature alarms with a Low CRD Water Pressure alarm.

d. A Manual Scram is required if the backup CRD Pump is started, Charging Water Pressure is 1500 psig, and a second ACCUM alarm comes in due to low pressure.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.06 (1.50)

Answer EACH of the following questions TRUE or FALSE concerning procedural limitations applied to squirrel cage induction and synchronous motors, 200 horsepower and larger:

a. The number of starts should be kept to a minimum since the life of a motor is affected by the number of starts.

b. A motor shall be limited to two (2) starts is succession, coasting to rest between starts, if the motor is initially at normal operating temperature.

c. Following a motor start at normal operating temperature, the motor should be allowed to cool for approximately 20 minutes while running at no load before an additional restart is attempted.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.07 (~~2.00~~)

Answer EACH of the following questions TRUE or FALSE.

a. The CO2 Fire Protection System shall be operable with a minimum of 50 % (.5 Tank) in storage units 1,2, and 3.

b. If CO2 fire protection is lost to a Cable Spreading Room, a continuous Fire Watch must be stationed until it is restored.

c. Reactor operation may continue with the High Pressure Fire Protection System inoperable, provided patrolling Fire Watches with portable fire equipment are available to patrol all areas hourly.

d. During Reactor operation, welding is permitted in the Cable Spreading Room, provided a continuous Fire Watch is stationed in the immediate vicinity where the work is in progress.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.08 (1.25)

Answer the following question regarding EPIP-2 (NUE) by FILLING IN THE BLANKS.

After an event is declared the ODS shall be notified within _____1______.

The SE/SED shall notify the NRC immediately or within ____2_____ by using the _____3_____.

Reanalysis of the current situation will be done by the ____4_____ at least every _____5_____ or more frequently if conditions warrant to determine if the NUE should be cancelled.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.09 (1.50)

An Operability Test is conducted on a safety-related system, following the installation of an approved modification. LIST three (3) separate criteria which would procedurally require that a TEST DEFICIENCY be initiated (documented).

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.10 (2.00)

Brown's Ferry Standard Practice 12.17, "Administrative Control for Plant Operation," establishes plant policy for the control of containment isolation and safety systems during an emergency.

a. STATE the evaluation which must be made prior to resetting a Primary Containment Isolation. 1.0 (~~0.5~~)

b. LIST the two (2) conditions which allow operators to override automatic operations of engineered safety features. (1.0)

NOTE: DO NOT CONFUSE THIS WITH THE GUIDANCE FOR MANUALLY SECURING AN ECCS SYSTEM.

QUESTION 4.11 (1.50)

Answer EACH of the following questions concerning EOI-1:

a. LIST the conditions under which SLC injection is mandatory. (0.5)

b. Permanent disabling of ADS is required when Reactor Shutdown is contingent upon SLC (ATWS condition), because core damage could occur. STATE two (2) methods of causing core damage if ADS is allowed to actuate. (1.0)

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.12 (2.00)

LIST eight (8) initial operator actions to be taken when Control Room abandonment is required as specified in Emergency Plans Manual Six (EPM-6).

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.13 (1.00)

LIST two (2) systems that require tagging prior to entry into the Primary Containment. INCLUDE in your answer the required status or position of the system.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.14 (2.00)

A single Recirculation Pump trips while operating at 100 % power in automatic control.

a. STATE the immediate action(s) that should be performed on the RUNNING PUMP.

b. EXPLAIN WHY the Running Pump speed must be reduced to < 50 % of rated speed prior to starting the idle pump.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.15 (1.00)

During Shell Warming of the High Pressure Turbine, the operator should maintain the 1st Stage Pressure between 60-100 psig in accordance with OI-47, since a Reactor Scram would occur if pressure became > 142 psig. EXPLAIN WHY this would initiate a Reactor Scram.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.16 (1.00)

Diesel Generator Fast Starts should be avoided during the time period of 15 minutes to 3 hours after Diesel shutdown, according to OI-82.

a. EXPLAIN the reason for this precaution.

b. Briefly DESCRIBE the method by which a Diesel Generator should be started during this time interval and from where it would be started.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.17 (2.00)

The following data was taken during recent testing of the Standby Liquid Control System:

Pump Flow Rates	A: 45gpm	B: 40gpm
Relief Valve Setting	A: 1375psig	B: 1325 psig
SLC Tank Level	4200 gal @ 13.0 % Boron concentration	
Solution Temperature	72 Deg F	

LIST any paramaters which do not meet Technical Specification requirements. INCLUDE, in your answer, any applicable Limiting Conditions for Operation.

NOTE: APPLICABLE TECHNICAL SPECIFICATIONS ATTACHED

(***** END OF CATEGORY 04 *****)
(************* END OF EXAMINATION ***************)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 1.01 (1.50)

supercritical [+0.5]

subcritical - reactor power is decreasing (OR neutrons per generation are decreasing) [+0.5]

supercritical - reactor power is increasing (OR neutrons per generation are increasing) [+0.5]

REFERENCE
BFNP: Reactor Theory, pp. 1-31 and p. 7-7.
Chapter 1. Objective 3.1.
Chapter 7. Objective 2.3.

3.5/3.5 3.9/3.9 4.1/4.1
292002K107 292008K107 292008K108 ...(KA'S)

ANSWER 1.02 (3.00)

1. LHGR - Linear Heat Generation Rate [+0.5]
(designed to limit the pin power at any node in the reactor to a value that) limits the fuel clad strain to less than one percent plastic strain. [+0.5]

2. APLHGR - Average Planer Linear Heat Generation Rate [+0.5]
(designed to limit average pin power at any node to a value such that following a design basis accident the) maximum fuel clad temperature will not exceed 2200 degrees F. [+0.5]

3. MCPR - Minimum Critical Power Ratio [+0.5]
(designed to limit the power of any fuel element to below the value that will) prevent any point in the bundle from experiencing the onset of transition boiling. [+0.5]

OTHER ANSWERS: LHGR - MFPLD or CMFLPD
APLHGR - MAPRAT or CMAPR .35 EACH
MCPR - CMFCP or MFLCPR

REFERENCE
BFNP: Heat Transfer and Fluid Flow, pp. 9-16 through 9-26.
Chapter 9, Objectives 2.3, 3.3, and 4.3.

2.8/3.6 2.8/3.6 2.8/3.6
293009K107 293009K111 293009K119 ...(KA'S)

ANSWER 1.03 (1.50)

a. decrease [+0.5]

b. The contribution to the delayed neutron population by U-235 decreases as the U-235 is burned out [+0.5] and the contribution from plutonium increases [+0.5], decreasing the delayed neutron fraction.

REFERENCE
BFNP: Reactor Theory, pp. 3-29
Chapter 3. Objective 4.6.

2.5/2.5
292003K104 ...(KA'S)

ANSWER 1.04 (1.00)

60 on range 2 is equal to 0.06 on range 7 [+0.25]

P(t) = P(o)e**-t/T [+0.25]

P(o) = 0.06, P(t) = 40, period = 60 seconds

t = 60 ln 40/0.06 [+0.25]

= 390 seconds or 6.5 minutes [+0.25]

REFERENCE
BFNP: Reactor Theory, pp. 3-17 and 3-19.
Chapter 3, Objective 3.2.
GOI-100-1, p. 13.

2.7/2.8
292003K108 ...(KA'S)

ANSWER 1.05 (2.00)

a. more
b. decrease
c. less
d. increase
e. increase

[+0.4 each]

REFERENCE
BFNP: Reactor Theory, pp. 5-9 through 5-16.
Chapter 5, Objective 2.4.

2.5/2.6
292005K109 ...(KA'S)

ANSWER 1.06 (1.00)

The steam bubbles generated by the withdrawal of a shallow rod increase the void fraction [+0.5], which adds negative reactivity, offsetting the positive reactivity effects of the rod withdrawal [+0.5].

REFERENCE
BFNP: Reactor Theory, p. 5-23.
Chapter 5, Objective 3.3.

3.1/3.2
292008K119 ...(KA'S)

ANSWER 1.07 1.50 ~~(2.00)~~

a. Doppler
b. void
~~c. moderator~~ Deleted
d. void

[+0.5 each]

REFERENCE
BFNP: Reactor Theory, pp. 7-2 through 7-23.
Chapter 7, Objective None.

3.3/3.3
292004K114 ...(KA'S)

ANSWER 1.08 (2.00)

a. increase [+0.5] due to Xe build in [+0.5]

b. The decrease in reactor pressure is directly related to a decrease in reactor coolant temperature [+0.25]. Due to the negative value of the moderator temperature coefficient, reactivity increases (adds positive reactivity with decreasing moderator temperature [+0.25]. Hence, SDM decreases [+0.5].

REFERENCE
BFNP: Reactor Theory, pp. 1-35, 4-5, and 6-9.
Chapter 1, Objective 4.1; Chapter 4, Objective 2.2; Chapter 6, Objective 2.3.d.

3.2/3.5 2.6/2.7
292002K110 292002K114 ...(KA'S)

ANSWER 1.09 (1.00)

Less feedwater heating will occur resulting in colder feedwater entering the vessel [+0.5] which will cause reactor power to increase (about 3 percent) from the positive reactivity addition (alpha m) [+0.5].

REFERENCE
BFNP: Heat Transfer and Fluid Flow, pp. 5-48.
Reactor Theory, pp. 7-18 and 7-19.
Chapter 7, Objective 8.1.

3.3/3.4 2.9/3.0 2.7/2.8
292008K120 292008K121 293005K105 ...(KA'S)

ANSWER 1.10 (1.25)

a. YES [+0.25], (the previous shift DID EXCEED the cooldown limit of) 100 degrees F/hr [+0.25]. 3368 or 384.8

Tsat for 630 psig = 494 degrees F;
Tsat for 200 psig = 388 degrees F;
cooldown rate = (494-388) degrees F/1 hour
= 106 degrees F/hr [+0.25]

b. 47 +/- 2 degrees F (of cooldown required) [+0.25]

Tsat for 200 psig = 388 degrees F;
Tsat for 105 psig = 341 degrees F;
(388-341) = 47 degrees F [+0.25]

ECF WILL BE APPLIED TO PART b.

REFERENCE
BFNP: Heat Transfer and Fluid Flow, pp. 3-1, 3-13.
Chapter 3, Objective 1.1
OPL171.044, RHR, pp. 14 and 16.
Objectives V.D.3, V.E.3, and V.E.4.
Technical Specifications, 3.6.A.

3.7/3.8 2.8/3.1
205000K402 293003K123 ...(KA'S)

ANSWER 1.11 (1.00)

1. (excessive voiding in bypass region resulting in) unreliable LPRM readings [+0.5]

2. inadequate cooling of LPRM detectors (resulting in premature LPRM detector failures) [+0.5]

REFERENCE
BFNP: Heat Transfer and Fluid Flow, pp. 8-49 through 8-52
Chapter 8, Objectives 9.4 and 9.5.

2.5/2.6 2.4/2.6
293008K132 293008K133 ...(KA'S)

ANSWER 1.12 (1.00)

use pump laws: power proportional to (speed)**3 [+0.4]
and head proportional to (speed)**2 [+0.4]

then: power decreased to 1/64 implies speed decrease by 1/4.

hence head decreased to (1/4)**2, which is 1/16
240 psig x (1/16) = 15 psig [+.2]

REFERENCE
BFNP: Heat Transfer and Fluid Flow, p. 6-96.
Chapter 6. Objective 7.11.

2.8/2.9
291004K105 ...(KA'S)

ANSWER 1.13 (1.00)

d

REFERENCE
BFNP: Heat Transfer and Fluid Flow, p. 6-68.
Chapter 6, Objective 6.1.

2.9/3.1 3.1/3.1
291002K104 291002K105 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 1.14 (2.00)

a. False
b. True
c. True
d. True (.5 each)

REFERENCE
G.E. Reactor Theory, Chpt. 4, LO 1.5,3.6,4.3,6.3, Chpt 7, LO 5.6
2.5/2.6 , 2.1/2.2 , 2.5/2.6 , 1.9/2.12
292004K102 292004K109 292004K111 292004K113 ...(KA'S)

ANSWER 1.15 (2.00)

a. True
b. False
c. False
d. False (.5 each)

REFERENCE
G.E. Reactor Theory, Chpt. 5, LO 2.5
2.5/2.6 , 2.6/2.9
292005K109 292005K112 ...(KA'S)

ANSWER 1.16 (2.00)

a. BPV'S open causing EHC to close Turbine CV's. (0.5)

b. Power increased due to lower feedwater temperature. (Less steam to the Turbine) (0.5)

c. (All BPV's are open at point 5.) (~~0.25~~) EHC follows increasing pressure by opening CV's. (~~0.25~~) (.50)

d. Pressure decreases due to BPV's opening. (0.5)

REFERENCE
BFNP: OPL 171.055 LO A
4.1/4.2 3.6/3.7 4.1/4.1
241000A203 241000K301 241000K610 ...(KA'S)

ANSWER 1.17 (1.00)

S/1-Keff 100/(1-.2) = 125

or

	fission	source	total
0	0	100	100
1	20	100	120
2	24	100	124
3	25	100	125
4	25	100	125

REFERENCE
G.E. Reactor Theory, Chpt 3, LO1.2,1.5
2.9/3.0 , 2.1/2.3
292003K101 292003K102 ...(KA'S)

ANSWER 2.01 (2.00)

a. 1. prevent too rapid of an injection rate (and subsequent power chugging) [+0.5]

2. prevent overpressurizing the SBLC system [+0.5]

b. to ensure the sodium pentaborate solution does not solidify in the pump suction lines and make the system inoperable [+1.0]

REFERENCE
BFNP: OPL171.039, SBLC, pp. 7, 12, and 15.
Objectives V.B, E, and F.

3.8/3.9 2.8/3.1 2.8/3.0
211000K403 211000K410 211000K502 ...(KA'S)

ANSWER 2.02 (2.00)

a. False (MOV's fail as-is)
b. False (separate reset switch for RCIC)
c. False ~~True~~ (both logic channels must deenergize)
d. False (only high D/W pressure or low RPV level)

[+0.5 each]

REFERENCE
BFNP: OPL171.017, PCIS, pp. 6, 17 and 18.
Objectives V.D. and V.E.

3.4/3.6 2.7/2.9 3.4/3.5 3.5/3.7
223002K107 223002K113 223002K406 223002K608 ...(KA'S)

ANSWER 2.03 (2.50)

a. No [+0.5]. It takes three insert errors to cause a rod block [+1.0]. (also accept "Yes [+0.5] if there is a third insert error [+1.0]")

b. If the Insert Rod Block Window is lit, an insert rod block is in force [+1.0]

REFERENCE
BFNP: OPL171.024, RWM, pp. 11 and 12.
Objective V.F.2, F.3, and F.4.

3.4/3.5
201006K401 ...(KA'S)

ANSWER 2.04 (2.00)

1. Diesel has started and is up to speed
2. All other supply breakers to the 4160-V board are open
3. No supply breaker overcurrent lock out exists
4. An undervoltage exists on the board

[+0.5 each]

REFERENCE
BFNP: OPL171.038, D/G's, pp. 10 and 11.
Objective V.E.

3.6/3.7 3.2/3.5
264000A405 264000K405 ...(KA'S)

ANSWER 2.05 ~~(1.00)~~ .50

Deleted ~~a. reactor~~ zone ventilation radiation signal [+0.5]
b. D/W air compressor suction valves (63,62) [+0.5]

REFERENCE
BFNP: OPL171.054, Control and Station Air Systems, p. 11.
Objective V.C.

3.3/3.2
295019A104 ...(KA'S)

ANSWER 2.06 (3.00)

a. False [+0.5] - once the low suction pressure signal is clear, the turbine will auto restart if the initiation signals are still present [+0.5].

b. False [+0.5] - the low steam pressure isolation signal does not seal in [+0.5].

c. True [+0.5] - the oil pressure will be restored when the turbine coasts down, thereby causing the stop valve to open [+0.5].

REFERENCE
BFNP: OPL171.042, HPCI, pp. 17, 25, and 26.
Objectives V.D.1, and V.D.2.

3.8/3.9 3.9/4.0 4.2/4.1 4.0/3.9
206000K401 206000K402 206000K403 206000K404 ...(KA'S)

ANSWER 2.07 (3.00)

a. 45 to 65 gpm (accept 0.25 to 0.33 gpm per CRD) [.5] ~~[+0.25]~~
260 psid (accept 250 to 270 psid) [.5] ~~[+0.25]~~

b. ~~100% (accept "all") 93% 52% or correct answer based upon assumptions made~~ [.5] ~~[+1.0]~~

c. Following a scram, but before the SDV is full, the control rod will be in the over travel-in position since there is still a large D/P across the piston. [+0.75]

After the SDV is full, there is no D/P across the piston and the control rod will settle into the "00" position. [+0.75]

REFERENCE
BFNP: OPL171.005, CRDH, pp. 9, 10, 24 through 29, and 40.
L.O. M, O, and S.

2.8/2.8 2.7/2.7 3.8/3.9 3.1/3.0
201001K110 201001K403 201001K406 201001K408 ...(KA'S)

b. If assume flow 45-65 gpm in part "a", ~6 gpm goes to stabilizing vlvs
⇒ 87-91% flow is correct $\left(\frac{\text{flow}-6}{\text{flow}}\right)$
If assume flow 100 gpm in part "a", ~20 gpm to mini flow
~4-6 gpm/RRP, ~1.8-2.0 gpm to RWCU pump seals & ~6 gpm to stabilizing vlvs
⇒ 58-62% flow is correct $\left(\frac{\text{flow}-20-12-4-6}{\text{flow}}\right)$

ANSWER 2.08 (1.50)

a. No [+0.5]. (If a core spray line breaks inside the shroud, the differential pressure indicating switch will detect reactor pressure inside the shroud as usual; therefore, no abnormal differential pressure will be indicated.)

b. The core spray loop will have Degraded performance ~~can perform a flooding function.~~ 1.0 ~~[+0.5] but~~ (its spray will not provide full core spray coverage) ~~[+0.5]~~.

REFERENCE
BFNP: OPL171.045. Core Spray, pp. 15 and 16.
Objective V.K.

2.8/3.0 3.0/3.2
209001K113 209001K404 ...(KA'S)

ANSWER 2.09 (3.00)

a. yes [+1.0]

b. no [+0.5]; must insert the rod [+0.5]

c. no [+0.5]; must raise the grapple fully or move refuel platform away from core [+0.5]

REFERENCE
BFNP: OPL171.053, Fuel Handling, pp. 8 and 9.
Objectives V.C and V.D.

3.1/3.7 3.1/3.7
234000A302 234000K502 ...(KA'S)

ANSWER 2.10 (2.00)

a. trip and throttle valve closes [+0.5]
min flow valve closes [+0.5]

b. Must manually open trip and throttle valve (71-9) from panel 9-3 by running motor generator to "close" position to relatch trip valve to trip solenoid. Then run motor operator to "open" position to open valve. [+1.0]

REFERENCE
BFNP: OPL171.040, RCIC.
Objective V.B.3.

3.5/3.5 3.3/3.3 3.8/3.7
217000G007 217000K102 217000K402 ...(KA'S)

ANSWER 2.11 (2.50)

a.
1. inlet isolation valve (FCV 69-1) not full open
2. inlet isolation valve (FCV 69-2) not full open
3. low pump flow
4. pump cooling water outlet high temperature
5. reactor return isolation valve FCV 69-12 fully closed
6. 480 V load shed logic
7. ANY isolation

[Any five @ +0.2 each]

b. the RWCU blowdown valve WILL NOT concurrently isolate [+0.5]

c. If the blowdown valve is open at the time of the RWCU system isolation, the system will rapidly depressurize from 400 psig to 5 psig (the blowdown valve's upstream isolation setpoint) [.5] the water in the piping will flash to steam ~~[+0.5]~~ in the high temperature portions of the system, shocking the system piping and components [+0.5].

REFERENCE
BFNP: OPL171.013, RWCU, pp. 10 and 11.
Objectives V.D, V.E, and V.F.

2.5/2.5 2.7/2.9 3.5/3.6 3.4/3.3
204000G007 204000K401 204000K402 204000K404 ...(KA'S)

ANSWER 2.12 (2.00)

a. local fuel damage (by generating a rod withdrawal block) [+1.0]

b. units = volts [+0.5], number of operable LPRM inputs can be calculated (by using 1 volt per operable input) [+0.5]

REFERENCE
BFNP: OPL171.035, RBM, pp. 4 and 28.
Objective V.K.

2.9/2.9 3.3/3.4 3.2/3.1
215002A402 215002G004 215002K102 ...(KA'S)

ANSWER 3.01 (2.50)

a. 2 (1 ok)

b. 4

c. 1

d. 4

e. 4 (0.5 each)

REFERENCE
BFNP: LP22. L.O. D

3.3/3.4 3.6/3.6 3.2/3.3
215005K116 215005K505 215005K607 ...(KA'S)

ANSWER 3.02 (2.00)

(1) Gross failure of a trip unit

(2) Card out of card file

(3) Calibration in progress

(4) Power supply failure (0.5 each)

REFERENCE
BFNP: LP 600, L.O. B.5

3.0/3.1 2.9/3.0
216000K118 216000K318 ...(KA'S)

ANSWER 3.03 (2.50)

a. (1) High voltage low

(2) Module unplugged (card pulled)

(3) Function switch not in operate (0.5 each)

b. None (0.5)

With the Mode Switch in "RUN", bypassing the companion APRM also bypasses IRM H. (IRM H's companion is APRM B) (0.5)

REFERENCE
BFNP: LP 20, L.O. C,G & H
LP 22, L.O. E & G
LP 28, L.O. K & L

3.9/3.9 3.9/4.0 3.9/4.0 3.7/3.8 3.7/3.7
4.0/4.0 3.2/3.4 3.5/3.7 3.7/3.6
215003K101 215003K106 215003K301 215003K305 215003K401
...(KA'S)

ANSWER 3.04 (1.25)

1. Lube oil temperature high (>210 deg F)

2. Lube oil pressure low (<30 psig w/ 6 sec T.D.)

3. MG set drive motor windings high temperature (>= 248 deg F)

4. MG set drive motor low voltage (1/2 credit for recirc bus lo voltage)

5. RPT breaker trip (0.25 each)

REFERENCE
BFNP: LP 8, L.O. I

3.4/3.4 3.1/3.1 3.6/3.6
202002A201 202002A205 202002G007 ...(KA'S)

ANSWER 3.05 (2.00)

a. (1) Closed
(2) Closed
(3) Open
(4) Closed (0.25 each)

b. (1) No. 2 bypass open (0.15)
Nos. 1,3,4 closed (0.10) or all closed
(2) Open (0.25)
(3) Closed (0.25)
(4) Closed (0.25)

REFERENCE
BFNP: LP 10, L.O. D
OI-47

3.4/3.5 3.1/3.2 2.7/2.7 2.8/2.8 2.9/2.9
245000A103 245000A302 245000A407 245000K108 245000K409
...(KA'S)

ANSWER 3.06 (1.50)

a. 75% (0.5)

b. Operable (0.5)

There are greater than 13 operable inputs on APRM C (0.5)

REFERENCE
BFNP: LP 21, L.O. A & B

3.6/3.6 3.6/3.4
215005G009 215005K104 ...(KA'S)

ANSWER 3.07 (1.00)

1. Level transmitter "A" (LIT 3-53)
2. Failed LOW (0.5 each)

REFERENCE
GGNS: OP-B21-501
BFNP: LP 3, L.O. J
LP 12, L.O. E.3
ARP 9-5: XA-55-5A-8, XA-55-5A-30, XA-55-5B-4, XA-55-5B-5

3.6/3.7 3.7/3.8 3.4/3.5 3.9/4.1
216000K112 216000K312 216000K313 216000K324 ...(KA'S)

ANSWER 3.08 (3.00)

a. (1) Prevents selection or movement of rods out of sequence (0.25)
Sequence Control (0.25)
(2) Restrict control rod movement of rods not in selected sequence and enforces ~~Prevents withdrawal errors within the sequence~~ GNC (0.125 each) ~~(0.25)~~
Group Notch Control (0.25)

(3) None (RSCS bypassed) (0.25)
None (0.25)

b. (1) Allows selection of any "B" sequence rod (0.25)

(2) Enables group notch control (GNC) logic (0.25)

(3) Bypasses the continuous withdraw mode of RMC (0.25)

(4) Prevents selction of any "A" sequence rod (0.25)

c. Bypasses all rod sequence control logic (0.5)

REFERENCE
BFNP: LP 25, L.O. A & I.1

3.3/3.4 3.6/3.7 3.3/3.4
201004K406 201004K407 201004K604 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 3.09 (1.50)

a. 480v Shutdown Board (1B) (via RPS regulating transformer) (0.5)

b. (1) Both RPS buses cannot be simultaneously fed from the alternate power source. (0.5)

(2) Prevent paralleling RPS MG set with alternate power source. (0.5)

REFERENCE
BFNP: LP 28, L.O. C

3.2/3.3 3.0/3.1 3.1/3.1
212000K201 212000K403 212000K404 ...(KA'S)

ANSWER 3.10 (1.50)

a. (1) SRM count rate > 100 cps

(2) IRMs are Range 3 or greater (0.5 each)

b. 9-5: Retract Permit Light - ON

9-12: Retract Permit Light - OFF (0.25 each)

REFERENCE
BFNP: LP 19, L.O. G & H

3.7/3.7 2.8/2.8 3.1/3.1 3.1/3.2 3.2/3.1
215004A106 215004A405 215004A406 215004K401 215004K503
...(KA'S)

ANSWER 3.11 (1.50)

(1) An edge rod is selected (0.5) which automatically bypasses the RBM (causing the above indications) (0.5).

(2) RBM is allowed to be bypassed (with power > 30%) since thermal limits can never be approached when an edge rod is withdrawn (0.5).

REFERENCE
BFNP: LP 35, L.O. C & I

3.0/3.1 3.1/3.0
215002A302 215002K106 ...(KA'S)

ANSWER 3.12 (3.00)

a. Single unit - sets VR for diesel supplying 4160v Shutdown Board as only source.

Units in parallel - sets VR for operating DG's in parallel

Parallel with system - sets VR to parallel with one of the 4160v unit boards via the 4160v shutdown buses

(0.25 for each mode, 0.25 for each function)

b. Automatically places VR in "single unit" mode (on any fast start signal) (0.5)

c. (1) Fast start relay not picked up within 1 sec after start signal

(2) Engine speed below 40 rpm 3 sec after start signal

(3) Engine speed above 40 rpm but below 100 rpm 4 sec after start signal

(any 2 @ 0.5 each)

REFERENCE
BFNP: LP 38, L.O. H

3.3/3.4 3.0/3.1 3.4/3.4 3.1/3.1 3.5/3.6
264000A301 264000A303 264000A304 264000G008 264000K407
...(KA'S)

ANSWER 3.13 1.75 (~~2.50~~)

a. (1) DC motor operated valves

(2) DC motor operated pumps

(3) Control power for ECCS

(4) Logic power for ECCS

or any examples of these. e.g. Control pwr: 480v SD boards, cooling tower swgr 4kV SD boards. Logic pwr: specific systems.

(any 3 @ 0.25 each)

b. The DC bus normally is supplied by a battery charger (0.25) powered from the 480v AC shutdown board (0.25).

Alternate power to the charger is from the 480v common board 1 (manual transfer only) (0.25).

Backup power is supplied by a (120 cell lead-acid) battery on a float charge (0.25).

c. (1) ~~Provides more constant pull on coils~~

(2) ~~Absence of hysterisis effects~~

(3) ~~Absence of eddy current losses~~ (0.25 each)

~~or any reasonable benefit of using DC power systems.~~

Deleted

REFERENCE
BFNP: LP 37, L.O. A,B & C

3.3/3.5 3.2/3.3 3.4/3.6 3.3/3.5
263000G004 263000G007 263000K101 263000K102 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 4.01 (2.00)

Operator 1 - Rejected - Would exceed 10 CFR 20 limit 1250 mrem/qtr with no Form 4 on file (0.5)
Operator 2 - Rejected - Would exceed 4000 mrem limit per year (0.5)
Operator 3 - Rejected - Would exceed pregnancy limit of 500 mrem (0.5)
Operator 4 - Accepted - Would not exceed quarterly limit of 1.25 rem whole body. 5(N-18) is limiting only when it is desired to exceed 1.25 rem/qtr whole body. (0.5)

REFERENCE
GPC: 60AC-HPX01-0, 10 CFR 20
EIH: SR-301, LP 300.3, LO #4
BFNP: RCI-1 LO B
3.3/3.8
294001K103 ...(KA'S)

ANSWER 4.02 (1.00)

d (1.0)

REFERENCE
BFNP: OI-66 L.O. I
3.3/3.6 3.8/4.1
271000G015 271000K404 ...(KA'S)

ANSWER 4.03 (1.00)

a

REFERENCE
EIH: U1 TS, Section 1.1.c
BFNP: U1 TS, Section 1.1.B , OPL174.728 LO 3
3.2/3.7
293009K118 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 4.04 (2.50)

a. 1 (1.0)

b. 1) Low Reactor Water Level (.1) <= -114.5 (~~0.2~~ .1) and (0.1) DG voltage applied to SD board(s) (0.2)

2) Drywell Pressure (.1) > 2.45 psig (~~0.2~~ .1) and (0.1) DG voltage applied to SD board(s) (0.2)

3) Low discharge header pressure < 57 psig. (0.5)
50 psig Header pressure

REFERENCE
BFNP: OI 70 LO A , AOI-70 , LP 171.047
3.8/4.1 3.3/3.4 2.9/3.2
295018AA10 295018AK30 295018G011 ...(KA'S)
Alternate answers for 1) & 2)
Accident signal (.2) AND (.2) Loss of Normal AC Power (.4)
Initiation of unit 1 & 2 480V Load shed Logic (.8)

ANSWER 4.05 (2.00)

a. TRUE (AOI) False (OI)
b. FALSE
c. FALSE
d. FALSE (0.5 each)

REFERENCE
BFNP: OI 85 LO H,I , 3-AOI-85-3
3.3/3.4 3.9/4.0 3.6/3.8 3.2/3.3
201001A201 201001G015 295022AK10 295022G011 ...(KA'S)

ANSWER 4.06 (1.50)

a. TRUE
b. FALSE
c. TRUE (0.5 each)

REFERENCE
BFNP: OSIL 28 LO K
3.3/3.6
294001K106 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 4.07 (~~2.00~~ 1.5)

~~a. TRUE~~ Deleted
b. TRUE
c. FALSE
d. FALSE (.5 each)

REFERENCE
BFNP: OPL174.728 LO 66.67,68 , TECHNICAL SPECIFICATIONS
3.4/3.6 3.8/4.0 3.2/4.1
286000G001 286000G011 286000K401 ...(KA'S)

ANSWER 4.08 (1.25)

1. 5 min
2. 1 hr
3. ENS - (red phone)
4. SE/SED
5. 2 hrs. (0.25 each)

REFERENCE
BFNP: EPIP-2 LO B,C,D
2.9/4.7
294001A116 ...(KA'S)

ANSWER 4.09 (1.50)

(1) System fails to operate.

(2) System operates in a suspected adverse manner.

(3) System operates outside of the limits of the documented acceptance criteria. (0.5 each)

REFERENCE
BFNP:(Standard Practice 10.9, L.O. "B") , PMI-17.1
3.9/4.5
294001K102 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 4.10 (2.00)

a. Ensure inadvertent transfer of significant amount of containment fluids will not occur. 1.0 (~~0.5~~)

b. Continued operation of the engineered safety features will result in an unsafe plant condition (with regard to either personnel or operability of safety features). (0.5)

The plant is in a stable condition (in which technical specifications clearly indicate that) operability of the engineered safety feature is no longer required. (0.5)

REFERENCE
BFNP: BF SP 12.17 LO A
3.8/4.0 3.6/3.8
223001G001 223001K102 ...(KA'S)

ANSWER 4.11 (1.50)

a. If the Reactor cannot be shutdown prior to Suppression Pool Temperature reaching 110 DEG F. (0.5)

b. ADS initiation may result in injection of cold (unborated) water from Low Pressure Injection Systems (0.5) and Boron Dilution (0.5).

REFERENCE
BFNP: EOI-1, OPL 171.057 LO B.6
4.4/4.7 3.7/3.9 4.0/4.2 4.2/4.4
295037EK10 295037G007 295037G011 ...(KA'S)

ANSWER 4.12 (2.00)

1. Manually Scram the Reactor
2. Trip the Recirc Pumps
3. Start the DG's
4. Check at least 2 EECW Pumps operating
5. If operating, Trip the Main Turbine, (continue Bypass Valve operation for as long as possible)
6. Announce Control Room Evacuation to all plant personnel.
7. Proceed to Backup Control Center (in the Shutdown Board Room.)
8. Place all MSRV Disconnects and Transfer Switches to Emergency.
9. Close all MSIV'S.

(any 8 @ .25 each)

REFERENCE
BFNP: EPMM-6 , OPL 174.711 EO 2
3.8/3.6 4.4/4.5 4.2/4.3
295016AA10 295016AK20 295016G010 ...(KA'S)

ANSWER 4.13 (1.00)

TIP (0.25) withdrawn (0.25)
Nitrogen Isolation Valves to Primary Containment (0.25) closed (0.25)
or CAD system isolated
or N_2 purge valves isolated

REFERENCE
BFNP: BF 14.9 LO A
3.2/3.7 3.2/3.4
294001K105 294001K114 ...(KA'S)

ANSWER 4.14 (2.00)

a. Place Recirculation Subpanel in Manual ~~(0.5)~~ (.25) and reduce speed to establish 100 % Loop Flow (45,200 gpm) ~~(0.5)~~ (.75).

b. Prevents excessive Jet Pump vibration. (1.0)

REFERENCE
BFNP: OI-68, LO E
3.4/3.4 3.9/3.5 3.1/3.5 3.6/3.7
202001A203 202001K411 202002A201 202002G014 ...(KA'S)

ANSWERS -- BROWNS FERRY 1, 2&3 -88/03/23-HOPPER, G

ANSWER 4.15 (1.00)

The Scram off of the Turbine Stop Valve position is armed when power is above 30 % (142 psig) as sensed by the 1st Stage Pressure (0.5). In Shell Warming, the Stop Valves are closed (0.5) (and a scram would occur).

REFERENCE
BFNP: OPL 171.010 LO D , OI-47
3.6/3.7 3.3/3.5 3.2/3.4 4.0/4.1
212000A212 212000K110 245000G001 245000K104 ...(KA'S)

ANSWER 4.16 (1.00)

a. Minimizes the possibility of damage to the Turbocharger Thrust Bearing. (0.5)

b. Manual Slow Starts (0.25) from the Engine Control Cabinet (0.25).

REFERENCE
BFNP: OI-82 LO A
3.7/3.7 3.7/4.2
264000A404 264000G001 ...(KA'S)

ANSWER 4.17 (2.00)

Relief Valve " B " setting of 1325 psig (<1350 psig) (0.5)
LCO - (3.4.B) Continued operation permitted provided that the component is returned to an operable condition within seven days. (0.5)

Sodium Pentaborate Solution LESS than minimum required by figure 3.4-1 (0.5)
LCO - (3.4.D) The Reactor shall be placed in a shutdown condition with all operable control rods fully inserted within 24 hours, (0.5) (unless condition can be corrected within specified time allowed.)

REFERENCE
BFNP: OPL 174.728 LO 71,20 , TECHNICAL SPECIFICATION 3.4
3.4/4.1 3.6/4.4
21000G005 21000G011 ...(KA'S)

QUESTION	VALUE	REFERENCE
01.01	1.50	GTH0000717
01.02	3.00	GTH0000719
01.03	1.50	GTH0000720
01.04	1.00	GTH0000721
01.05	2.00	GTH0000723
01.06	1.00	GTH0000724
01.07	2.00	GTH0000726
01.08	2.00	GTH0000727
01.09	1.00	GTH0000729
01.10	1.25	GTH0000730
01.11	1.00	GTH0000731
01.12	1.00	GTH0000732
01.13	1.00	GTH0000733
01.14	2.00	GTH0000658
01.15	2.00	GTH0000661
01.16	2.00	GTH0000747
01.17	1.00	GTH0000657
	26.25	
02.01	2.00	GTH0000734
02.02	2.00	GTH0000735
02.03	2.50	GTH0000736
02.04	2.00	GTH0000737
02.05	1.00	GTH0000738
02.06	3.00	GTH0000739
02.07	3.00	GTH0000740
02.08	1.50	GTH0000741
02.09	3.00	GTH0000742
02.10	2.00	GTH0000743
02.11	2.50	GTH0000744
02.12	2.00	GTH0000745
	26.50	
03.01	2.50	GTH0000778
03.02	2.00	GTH0000781
03.03	2.50	GTH0000782
03.04	1.25	GTH0000783
03.05	2.00	GTH0000772
03.06	1.50	GTH0000773
03.07	1.00	GTH0000774
03.08	3.00	GTH0000775
03.09	1.50	GTH0000776
03.10	1.50	GTH0000779
03.11	1.50	GTH0000780
03.12	3.00	GTH0000784
03.13	2.50	GTH0000785
	25.75	
04.01	2.00	GTH0000712

QUESTION	VALUE	REFERENCE
04.02	1.00	GTH0000710
04.03	1.00	GTH0000715
04.04	2.50	GTH0000757
04.05	2.00	GTH0000758
04.06	1.50	GTH0000760
04.07	2.00	GTH0000770
04.08	1.25	GTH0000761
04.09	1.50	GTH0000709
04.10	2.00	GTH0000713
04.11	1.50	GTH0000749
04.12	2.00	GTH0000750
04.13	1.00	GTH0000762
04.14	2.00	GTH0000753
04.15	1.00	GTH0000754
04.16	1.00	GTH0000756
04.17	2.00	GTH0000767
	27.25	
	105.75	

DOCKET NO 259

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
Browns Ferry Nuclear Plant
P. O. Box 2000
Decatur, Alabama 35601

MAR 30 1988

Mr. Ken Brockman, Chief
Operator Licensing Section
U.S. Nuclear Regulatory Commission, Region II
101 Marietta Street, NW
Atlanta, Georgia 30323

Dear Mr. Brockman:

In accordance with the provisions of NUREG-1021, "Operator Licensing Examiner Standards," Standard ES-201, enclosed are comments by the Browns Ferry Operator Training Group staff concerning the written examinations administered at Browns Ferry Nuclear Plant, March 23, 1988.

The enclosed comments are offered with the intent of providing assistance to the NRC examiners in establishing the appropriateness of the examination questions. Also, the comments serve to clarify and expand the answers on the NRC answer key as supported by TVA reference material.

With respect to any questions deleted, NRC is requested to consider allowing the examinee full credit for these questions in light of the time, effort and concentration required of the examinee.

These comments are respectfully submitted, and it is hoped the enclosed comments and proposed resolutions afford the examinees every opportunity to successfully pass the examination based upon the knowledges and skills required to safely operate the facility.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

H. P. Pomrehn
Site Director, BFN

Enclosure

D TE 3-18-85 TIME 1444 BROWNS FERRY - 1 SEQ. NO. 7

PERIODIC NSS CORE PERFORMANCE LOG

L CATION	1	2	3	4	5	6	7	8	9	10	11	12
A IAL REL PWR	0.58	1.02	1.04	1.03	1.05	1.08	1.09	1.13	1.13	1.15	1.02	0.69
R GION REL PWR	0.89	1.10	0.89	1.07	1.15	1.07	0.89	1.10	0.89			
R NG REL PWR	1.07	1.14	1.17	1.16	1.28	1.22	1.02	0.52				
A RM GAF	0.98	0.99	0.94	0.98	0.94	0.96						

R GION	1	2	3	4	5	6	7	8	9
M LCPR	0.716	0.821	0.717	0.789	0.726	0.789	0.717	0.819	0.714
OC	21-14	35-14	39-14	11-34	35-32	49-28	21-48	25-48	41-46
LON	0.1246	0.1195	0.1246	0.1222	0.1247	0.1222	0.1246	0.1197	0.1254
KF	1.34	1.52	1.34	1.44	1.35	1.44	1.34	[illegible]	[illegible]
M LPD	0.643	0.723	0.644	0.694	0.645	0.694	0.644	[illegible]	[illegible]
OC	21-14- 5	35-14- 5	39-14- 5	11-34-16	25-36-16	49-28-16	21-48- 5	[illegible]	[illegible]
KFL	1.75	1.97	1.75	1.89	1.75	1.89	1.75	1.97	1.75
M PRAT	0.629	0.706	0.630	0.689	0.636	0.689	0.630	0.707	0.629
OC	21-14- 5	35-14- 5	39-14- 5	13-32-16	25-36-19	47-30-16	21-48- 5	25-48- 5	39-48- 5
KFS	1.58	1.77	1.59	1.71	1.59	1.71	1.59	[illegible]	[illegible]

CMWT	2943.
PCT PWR	89.4
GMWE	977.3
CMFCP	0.821
CMFLPD	0.725
CMAPR	0.707
CMPF	1.[illegible]73
CAEQ	0.112
CAQA	0.129
CAVE	0.335
CAPD	43.558
CRD	0.002
CRSYM	2
PR	1001.88
DPC-M	18.62
DPC-C	24.52
RML	33.15
DHS	21.13
WFW	11.72
WD	35.62
WTSUB	105.31
WTHB	-1.00
WT	107.10
PCTWTR	104.5
WTFLAG	2.0
ITER	0.0
IREC	0.0
IEOL	0.0
IXYFLG	0.0

F ILED SENSORS 2

FAILED LPRM LIST

3209,D,2	3217,D,2	4817,D,5	5617,A,1
0825,B,2	2425,A,5	2425,B,5	2425,C,5
2425,D,5	3225,D,5	4825,A,1	1633,B,2
3233,B,1	4033,A,5	4033,B,1	4033,C,5
4033,D,5	5633,B,1	5633,C,1	5633,D,1
2441,B,2	2441,C,2	0849,A,5	0849,B,5
0849,C,5	0849,D,5	1649,B,1	2449,C,2
3249,D,2	4049,C,2	1657,B,5	1657,D,1

BASE CRIT CODE

2425,1 4033,1 0849,1

T E 12 MOST LIMITING BUNDLES

FOR MFLCPR				FOR MFLPD				FOR MAPRAT			
M LCPR	LOC	MCPR	CPRLIM	MFLPD	LOC	MRPD	RPDLIM	MAPRAT	LOC	MAPLHGR	LIMLHGR
0 821	35-14	1.534	1.260	0.725	25-48- 5	9.71	13.40	0.707	25-48- 5	8.49	12.00
0 821	25-14	1.534	1.260	0.724	35-48- 5	9.71	13.40	0.707	35-48- 5	8.48	12.00
0 819	25-48	1.538	1.260	0.723	35-14- 5	9.69	13.40	0.706	35-14- 5	8.47	12.00
0 819	35-48	1.539	1.260	0.722	25-14- 5	9.68	13.40	0.705	25-14- 5	8.46	12.00
0 805	37-12	1.565	1.260	0.694	11-34-16	9.30	13.40	0.699	25-12- 8	7.03	10.06
0 805	23-50	1.565	1.260	0.694	49-28-16	9.30	13.40	0.697	35-12- 8	7.06	[illegible]
0 805	37-50	1.565	1.260	0.694	11-28-16	9.29	13.40	0.696	35-50- 8	7.04	10.40
0 805	23-12	1.565	1.260	0.693	49-34-16	9.29	13.40	0.695	25-50- 8	7.07	10.18
0 790	29-14	1.595	1.260	0.687	13-32-16	9.20	13.40	0.689	13-32-16	8.27	12.00
0 789	31-14	1.596	1.260	0.687	47-30-16	9.20	13.40	0.689	47-30-16	8.27	12.00
0 789	11-34	1.596	1.260	0.686	13-30-16	9.19	13.40	0.688	13-30-16	8.26	12.00
0 789	31-48	1.597	1.260	0.686	47-32-16	9.19	13.40	0.688	47-32-16	8.26	12.00

T E NUMBER OF BUNDLES WITH MFLCPR GREATER THAN 1.0 = 0

T E NUMBER OF BUNDLES WITH MFLPD GREATER THAN 1.0 = 0

T E NUMBER OF BUNDLES WITH MAPRAT GREATER THAN 1.0 = 0

DATE 3-13-85 TIME 1444 BROWNS FERRY - 1 SEQ. NO. 7

PERIODIC NSS CORE PERFORMANCE LOG

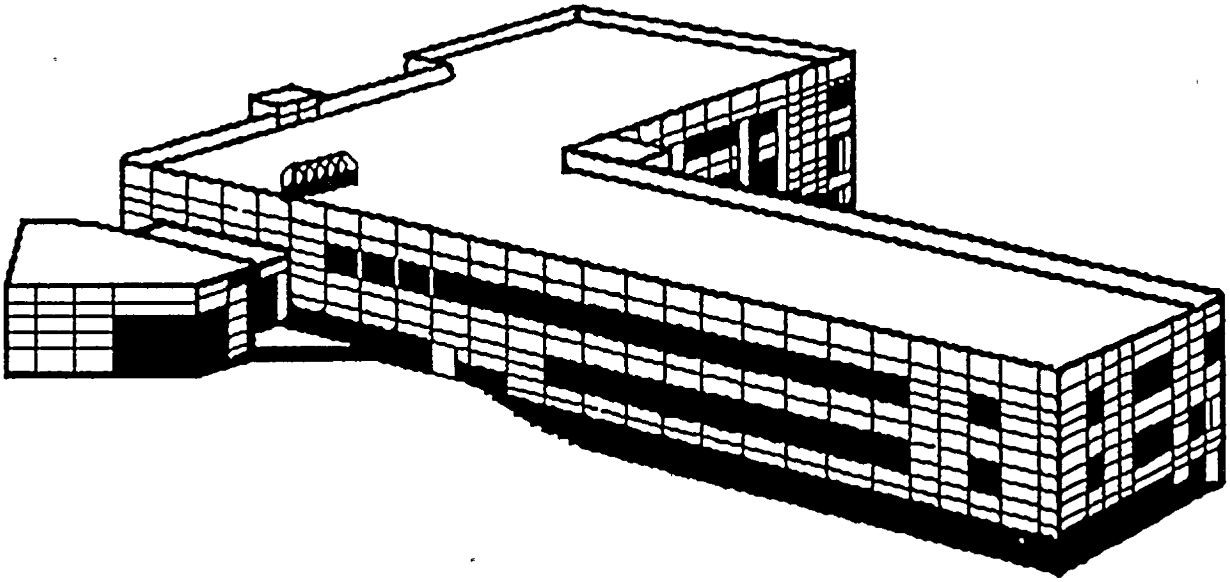
CONTROL ROD POSITIONS AND CALIBRATED LPRM READINGS

++=48

		02	06		10	14		18	22		26	30		34	38		42	46		50	54		58
59	D						30	46	++	38	++	++	39	++	++	39	++						
	C						34			43			45			43							
	B						28			43			43			41							
55	A					++	20	++	++	39	++	++	41	++	++	34	++	++					
51				32	++	++	43	++	++	49	++	++	50	++	++	45	46	[illegible]	41	++			
				37			51			64			62			59			50				
				33			56			70			67			60			52				
47			++	27	++	++	62	++	++	78	46	++	71	46	46	66	++	++	48	++	++		
43		++	++	41	++	++	46	++	++	48	46	++	49	++	++	47	++	++	45	++	++	37	++
				52			54			53			55			57			54			42	
				53			56			56			59			57			58			39	
39		++	++	57	++	++	62	46	++	55	++	++	59	++	++	59	++	++	66	++	++	33	++
35		++	++	46	++	++	51	++	++	50	++	++	50	++	++	47	++	++	51	++	++	38	++
				58			61			59			60			56			65			44	
				58			63			60			60			42			65			43	
31		++	++	65	++	++	66	++	++	61	++	++	49	++	++	57	++	++	68	++	++	41	++
27		++	++	46	++	++	46	++	++	51	++	++	47	++	++	51	++	++	47	++	++	38	++
				60			55			62			54			59			60			44	
				59			58			61			58			59			61			44	
23		++	++	67	++	++	66	++	++	58	++	++	60	++	++	55	++	++	69	++	++	40	++
19		++	++	38	++	++	45	++	++	53	++	++	48	++	++	48	++	++	45	++	++	31	++
				47			52			62			62			57			54			34	
				47			56			65			65			59			58			27	
15			++	47	++	++	66	++	++	72	++	++	67	++	++	66	++	++	60	++	++	21	
11	D				++	++	40	++	++	46	++	++	47	++	++	45	++	++	31	++			
	C						48			54			55			53			35				
	B						50			58			60			57			32				
07	A					++	48	++	++	64	++	++	69	++	++	59	++	++	27				
03								++	++		++	++		++	++		++						

OPERATOR SUBS VAL CR LOC
NON-SYM ROD LOC FIRST QUAD 1803 1811 2215 2615 2019 1823

B.F.N.



T.V.A.

BROWNS FERRY TRAINING AND VISITOR CENTER

DATE 3-18-85 TIME 1444 BROWNS FERRY - 1 SEQ. NO. 7

PERIODIC NSS CORE PERFORMANCE LOG

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12
AXIAL REL PWR	0.58	1.02	1.04	1.03	1.05	1.08	1.09	1.13	1.13	1.15	1.02	0.69
REGION REL PWR	0.89	1.10	0.89	1.07	1.15	1.07	0.89	1.10	0.89			
RING REL PWR	1.07	1.14	1.17	1.16	1.28	1.22	1.02	0.52				
APRM GAF	0.98	0.99	0.94	0.98	0.94	0.96						

REGION	1	2	3	4	5	6	7	8	9
MLCPR	0.716	0.821	0.717	0.789	0.726	0.789	0.717	0.819	0.714
LOC	21-14	35-14	39-14	11-34	35-32	49-28	21-48	25-48	41-46
LOW	0.1246	0.1195	0.1246	0.1222	0.1247	0.1222	0.1246	0.1197	0.1254
KF	1.34	1.52	1.34	1.44	1.35	1.44	1.34	1.52	1.32
MLPD	0.643	0.723	0.644	0.694	0.645	0.694	0.644	0.725	0.643
LOC	21-14- 5	35-14- 5	39-14- 5	11-34-16	25-30-16	49-28-16	21-48- 5	25-48- 5	39-48- 5
KFL	1.75	1.97	1.75	1.89	1.75	1.89	1.75	1.97	1.75
MPRAT	0.629	0.706	0.630	0.689	0.636	0.689	0.630	0.707	0.629
LOC	21-14- 5	35-14- 5	39-14- 5	13-32-16	25-30-19	47-30-16	21-48- 5	25-48- 5	39-48- 5
KFS	1.58	1.77	1.59	1.71	1.59	1.71	1.59	1.77	1.58

CMWT	2943.
PCT PWR	89.4
GMWE	977.3
CMFCP	0.821
CMFLPD	0.725
CMAPR	0.707
CMPF	1.973
CAEQ	0..112
CAOA	0.129
CAVF	0.335
CAPD	43.558
CRD	0.002
CRSYM	2.
PR	1001.88
DPC-M	18.62
DPC-C	24.52
RWL	33.15
DHS	21.13
WFW	11.72
WD	35.62
WTSUB	105.31
WTHB	-1.00
WT	107.10
PCTWTR	104.5
WTFLAG	2.0
ITER	0.0
IREC	0.0
IEOL	0.0
IXYFLG	0.0

FAILED SENSORS 2

FAILED LPRM LIST

3209,D,2 3217,D,2 4317,D,5 5617,A,1
0625,B,2 2425,A,5 2425,B,5 2425,C,5
2425,D,5 3225,D,5 4825,A,1 1633,B,2
3233,B,1 4033,A,5 4033,B,1 4033,C,5
4033,D,5 5633,B,1 5633,C,1 5633,D,1
2441,B,2 2441,C,2 0849,A,5 0849,B,5
0849,C,5 0849,D,5 1649,B,1 2449,C,2
3249,D,2 4049,C,2 1657,B,5 1657,D,1

BASE CRIT CODE .

2425,1 4033,1 0849,1

THE 12 MOST LIMITING BUNDLES

FOR MFLCPR				FOR MFLPD				FOR MAPRAT			
MFLCPR	LOC	MCPR	CPRLIM	MFLPD	LOC	MRPD	RPDLIM	MAPRAT	LOC	MAPLHGR	LIMLHGR
0.821	35-14	1.534	1.260	0.725	25-48- 5	9.71	13.40	0.707	25-48- 5	8.49	12.00
0.821	25-14	1.534	1.260	0.724	35-48- 5	9.71	13.40	0.707	35-48- 5	8.48	12.00
0.819	25-48	1.533	1.260	0.723	35-14- 5	9.69	13.40	0.706	35-14- 5	8.47	12.00
0.819	35-48	1.539	1.260	0.722	25-14- 5	9.68	13.40	0.705	25-14- 5	8.46	12.00
0.805	37-12	1.565	1.260	0.694	11-34-15	9.30	13.40	0.699	25-12- 8	7.03	10.06
0.805	23-50	1.555	1.260	0.694	47-28-16	9.30	13.40	0.697	35-12- 8	7.06	10.13
0.805	37-50	1.565	1.260	0.694	11-28-15	9.29	13.40	0.696	35-50- 8	7.04	10.10
0.803	23-12	1.555	1.260	0.693	47-34-16	9.29	13.40	0.695	25-50- 8	7.07	10.18
0.790	29-14	1.575	1.260	0.637	13-32-16	9.20	13.40	0.689	13-32-16	8.27	12.00
0.787	31-14	1.576	1.260	0.637	47-30-16	9.20	13.40	0.689	47-30-16	8.27	12.00
0.787	11-34	1.575	1.260	0.636	13-30-16	9.19	13.40	0.688	13-30-16	8.26	12.00
0.787	31-48	1.597	1.260	0.636	47-32-16	9.19	13.40	0.688	47-32-16	8.26	12.00

THE NUMBER OF BUNDLES WITH MFLCPR GREATER THAN 1.0 = 0
THE NUMBER OF BUNDLES WITH MFLPD GREATER THAN 1.0 = 0
THE NUMBER OF BUNDLES WITH MAPRAT GREATER THAN 1.0 = 0

```
DATE  3-1[illegible]-85        TIME     1444             BROWNS FERRY - 1                                     SEQ. NO.   7

                                     ***PERIODIC NSS CORE PERFORMANCE LOG***

                               CONTROL ROD POSITIONS AND CALIBRATED LPRM READINGS

      ++=48

 59 D                              30 46     ++  38 ++     ++  39 ++     ++  39 ++
    C                              34            43            45            43
    B                              28            43            43            41
 55 A                         ++   20 ++     ++  39 ++     ++  41 ++     ++  34 ++     ++

 51                  32 ++    ++   43 ++     ++  49 ++     ++  50 ++     ++  45 46     ++  41 ++
                     37            51            64            62            55            50
                     33            56            70            67            60            52
 47             ++   27 ++    ++   52 ++     ++  78 46     ++  71 46     46  66 ++     ++  48 ++     ++

 43      ++     ++   41 ++    ++   46 ++     ++  18 46     ++  49 ++     ++  47 ++     ++  45 ++     ++  37 ++
                     52            54            53            55            57            54            42
                     53            56            56            59            57            58            39
 39      ++     ++   57 ++    ++   62 46     ++  55 ++     ++  59 ++     ++  59 ++     ++  66 ++     ++  33 ++

 35      ++     ++   46 ++    ++   51 ++     ++  50 ++     ++  50 ++     ++  47 ++     ++  51 ++     ++  38 ++
                     58            61            59            60            56            65            44
                     58            63            60            60            42            65            43
 31      ++     ++   65 ++    ++   56 ++     ++  61 ++     ++  49 ++     ++  57 ++     ++  68 ++     ++  41 ++

 27      ++     ++   46 ++    ++   46 ++     ++  51 ++     ++  47 ++     ++  51 ++     ++  47 ++     ++  38 ++
                     60            55            52            54            59            60            44
                     59            56            61            53            59            61            44
 23      ++     ++   67 ++    ++   66 ++     ++  58 ++     ++  60 ++     ++  55 ++     ++  69 ++     ++  40 ++

 19      ++     ++   38 ++    ++   45 ++     ++  53 ++     ++  48 ++     ++  48 ++     ++  45 ++     ++  31 ++
                     47            52            52            62            57            54            34
                     47            56            65            65            59            58            27
 15             ++   47 ++    ++   63 ++     ++  72 ++     ++  57 ++     ++  66 ++     ++  60 ++     ++  21

 11 D                   ++    ++   40 ++     ++  46 ++     ++  47 ++     ++  45 ++     ++  31 ++
    C                              48            54            55            53            35
    B                              50            56            60            57            32
 07 A                         ++   13 ++     ++  64 ++     ++  69 ++     ++  59 ++     ++  27

 03                                   ++     ++        ++  ++        ++  ++        ++

         02     06     10     14     18     22     26     30     34     38     42     46     50     54     58

 OPERATOR SUBS VAL CR LOC
 NON-SYM ROD LOC FIRST QUAD  1803  1811  2215  2615  2619  1823
```

Question 5.14 (3.00)

Question 1.02 (3.00)

LIST the three (3) "thermal limits" observed during reactor operation and STATE the limiting condition for each. (i.e., what the thermal limit is there to protect against.)

Answer:

1. LHGR - Linear HEat Generation Rate (.5)
designed to limit the pin power at any node in the reactor to a value that limits the fuel clad strain to less than one percent plastic strain. (.5)

2. APLHGR - Average Planer Linear Heat Generation Rate (.5)
(designed to limit average pin power at any node to a value such that following a design basis accident the) maximum fuel clad temperature will not exceed 2200°F. (.5)

3. MCPR - Minimum Critical Power Ratio (.5)
(designed to limit the power of any fuel element to below the value that will) prevent any point in the bundle from experiencing the onset of transition boiling. (.5)

Reference:

BFNP: Heat Transfer and Fluid Flow, pp. 9-16 through 9-26.
Chapter 9, Objectives 2.3, 3.3, and 4.3.

TVA Comment:

The question does not elicit the detailed response of the answer key, specifically LHGR. The P-1 edit at BFN uses acronyms for these parameters and thses should be acceptable.
LHGR = MFLPD and CMFLPD
APLHGR = MAPRAT and CMAPR
MCPR - CMFCP and MFLCPR
(Reference chapter 9 and attached P-1 edit)

TVA Resolution:

Answer key should be changed to accept prevent >1% plastic strain vice. . . "pin power at any node" . . . Also key should reflect credit for acronym's if used for thermal limit designator.

1755Q

Reactor power is 60 on IRM range 2 with the MINIMUM permissible stable positive period allowed by procedure GOI-100-1. Heating power is determined to be 40 on IRM range 7. CALCULATE how long it will take for power to reach the point of adding heat if the period remains constant.

Answer:

60 on range 2 is equal to 0.06 on range 7	(.25)
P(t) = P(0)e**-t/T	(.25)
P(o) = 0.06, P(t) = 40, period = 60 seconds	(.25)
t = 60 In 40.0.06	(.25)
= 390 seconds or 6.5 minutes	(.25)

Reference:

BFNP: Reactor Theory, pp. 3-17 and 3-19
Chapter 3, Objective 3.2
GOI-100-1, p. 13

TVA Comment:

Answer key assumes that the reading on IRM range 7 is 40 on the 0-40 scale, and thus 40 on the range 8 (0-125 scale). But, heating range is normally reached mid range 7, so some may assume the question was giving POAH as 40/125 on range 7. This means that P(o) = .19/125 on range 7, instead of P(o) = .06/40 on range 7. Then, $t = T \ln \left(\frac{40/125}{.19/125}\right) = 321$ sec

Instead of $t = T \ln \left(\frac{40/40}{.06/40}\right) = 390$ sec.

"40/125" is a reasonable assumption also since 40/40 would result in full scale readings and scram trips. At BFNP, we commonly use the 0-125 scale on any range.

TVA Resolution:

Allow use of either 40/40 or 40/125
- Use 40/40 on Range 7 as POAH
  t = 390 sec
  <u>or</u>
- Use 40/125 on range 7 as POAH
  t = 321 sec.

1.04 (1.00)

5.02 (1.00)

TVA Comment:

Plant procedure GOI 100-1 does allow for reactor periods of < 60 seconds, but ≥ 30 seconds, although it is desirable to have a period of > 60 seconds. This question tests the application and understanding of Reactor Theory, therefore a candidate who elects to choose 60 seconds or 30 seconds as the minimum permissible period should receive credit. (REF GOI 100-1, p. 13)

TVA Resolution:

Expand the answer key to accept a response using 30 seconds as minimum period in addition to current answer key.

1749Q

Section III. Startup (Continued) INITIALS/TIME/DATE

A. Criticality (Continued)

****

CAUTION

DURING A HOT STARTUP FOLLOWING A REACTOR SCRAM AT HIGH POWER, THE CONDITIONS OF PEAK XENON WITH NO MODERATOR VOIDS COULD EXIST AT THE TIME OF STARTUP. UNDER THESE CONDITIONS, EXTREMELY HIGH ROD NOTCH WORTHS CAN BE ENCOUNTERED.

****

4. Upon approval of the shift engineer, start control rod withdrawal in accordance with OI-85. (R)_____/ /_____

NOTE: Shift all SRM and IRM recorders to fast speed prior to criticality and return to slow speed after initial period measurements are calculated.

NOTE: Within the approved control rod withdrawal sequence, it is possible to have a period less than 60 seconds. If a period less than 30 seconds is observed, insert rods until subcriticality is observed and contact the nuclear engineer and shift engineer before pulling any more rods. Periods less than 5 seconds are reportable to the NRC within 24 hours.

5. Observe the period meter when pulling rods and govern withdrawal rate to avoid having
* a period shorter than 60 seconds. (R)_____/ /_____

* NOTE: Reactor is critical when neutron flux rises on a constant (stable) period without further control rod movement.

6. When critical, record time, rod position, rod ~~notch~~, period, and reactor water temperature
* from recirculation loop A in daily journal. (R)_____/ /_____

NOTE: Measure period as follows:
For 10% power rise, multiply time of rise by 10.5.
For doubling time, multiply time of rise by 1.445.
For decade rise, divide time of rise by 2.3.

* For direct period measurement when on IRMs:
* a. Time 25 to 68 on black scale ranges
* b. Time 8 to 22 on red scale ranges

* Revision

5.16 (2.00)

1.16 (2.00)

Question:

The attached FIGURE (GTH-747 represents parameter changes for a plant transient on UNIT TWO. Use this figure and the following information to answer EACH of the questions below:

(1) Initial Power Level = 100%
(2) Bypass Valves go to Full Open position
(3) No operator action is taken

a. The DECREASE in turbine steam flow. (point 4)

b. The INCREASE in power. (point 7)

c. THe INCREASE in turbine steam flow. (point 5 and AREA 6)

d. The DECREASE in pressure. (point 2)

Answer:

a. BPV's open causing EHC to close Turbine CV's. (.5)

b. Power increased due to lower feedwater temperature. (Less steam to the Turbine) (.5)

c. All BPV's are open at point 5. (.25) EHC follows increasing pressure by opening CV's. (.25)

d. Pressure decreases due to BPV's opening. (.5)

Reference:

BFNP: OPL171.055 LO a
4.1/4.2 3.6/3.7 4.1/4.1

TVA Comment:

Part C; Since the question stated that the BPV's were full open due to operation of the BPV jack, requiring this in the answer should not be required.

TVA Resolution:

Accept for full credit (.5), EHC follows increasing pressure by opening CV's.

Question 2.02 (2.00)

Question 6.01 (2.00)

STATE whether the following statements concerning the Primary Containment Isolation System are TRUE or FALSE:

a. Most of the PCIS motor operated valves fail closed on loss of power to the valve.

b. The containment isolation reset switches on panel 9-5 must be operated to manually reset a RCIC turbine steam supply isolation.

c. Loss of RPS Bus A will NOT cause any PCIS isolation valves to close.

d. The TIP guide tube ball valve will isolate on a high radiation signal.

Answer:

a. False (MOV's fail as-is)
b. False (separate reset switch for RCIC)
c. True (both logic channels must deenergize)
d. False (only high D/W pressure or low RPV level) (0.5 each)

Reference:

BFNP: OPL171.017, PCIS, pp. 6, 17 and 18
Objectives V.D and V.E.

TVA Comment:

Part C — Answer key states True (both logic channels must de-energize). This is a true statement if RPS power is supplied to the respective logic channel. When the RPS bus power is lost, the relay loses potential even though the opposite PCIS channel relays have closed contacts.

OI-99 attachment 2 indicates valves that will isolate on RPS 'A' (inboard) or RPS 'B' (outboard) power loss.

TVA Resolution:

Answer key should be changed to reflect False as the correct response.

1752Q

RPS BUS A or B POWER TRANSFER

1. Transfer of power supply to either RPS Bus A or B may result in the following events:

VALVE	FUNCTION/SYSTEM	ACTION
FCV-32-62	Drywell Control Air Compressor suction	CLOSES
FCV-32-63	Drywell Control Air Compressor suction	CLOSES
FCV-64-17	Drywell/Suppression Chamber purge inlet	CLOSES
FCV-64-18	Drywell purge inlet inboard	CLOSES
FCV-64-19	Suppression Chamber purge inlet inboard	CLOSES
FCV-64-29	Drywell purge exhaust inboard	CLOSES
FCV-64-30	Drywell purge exhaust outboard	CLOSES
FCV-64-31	Drywell purge exhaust bypass to SGTS	CLOSES
FCV-64-32	Suppression Chamber purge exhaust inboard	CLOSES
FCV-64-33	Suppression Chamber purge exhaust outboard	CLOSES
FCV-64-34	Suppression Chamber purge exhaust bypass to SGTS	CLOSES
FCV-64-36	Drywell/Suppr Chbr purge exhaust to SGTS	CLOSES
FCV-64-139	Drywell to Suppr Chbr DP compressor suction	CLOSES
FCV-64-140	Drywell to Suppr Chbr DP compressor discharge	CLOSES
FCV-76-17	Drywell/Suppression Chamber nitrogen purge inlet	CLOSES
FCV-76-24	Drywell/Suppression Chamber nitrogen purge inlet	CLOSES
FCV-76-18	Drywell nitrogen purge inlet	CLOSES
FCV-76-19	Suppression Chamber nitrogen purge inlet	CLOSES
FCV-76-49	Containment Inerting System A sample	CLOSES
FCV-76-50	Containment Inerting System A sample	CLOSES

RPS BUS A or B POWER TRANSFER

1. Transfer of power supply to either RPS Bus A or B (Continued):

VALVE	FUNCTION/SYSTEM	ACTION
FCV-76-51	Containment Inerting System A sample	CLOSES
FCV-76-52	Containment Inerting System A sample	CLOSES
FCV-76-53	Containment Inerting System A sample	CLOSES
FCV-76-54	Containment Inerting System A sample	CLOSES
FCV-76-55	Containment Inerting System A sample	CLOSES
FCV-76-56	Containment Inerting System A sample	CLOSES
FCV-76-57	Containment Inerting System A sample	CLOSES
FCV-76-58	Containment Inerting System A sample	CLOSES
FCV-76-59	Containment Inerting System B sample	CLOSES
FCV-76-60	Containment Inerting System B sample	CLOSES
FCV-76-61	Containment Inerting System B sample	CLOSES
FCV-76-62	Containment Inerting System B sample	CLOSES
FCV-76-63	Containment Inerting System B sample	CLOSES
FCV-76-64	Containment Inerting System B sample	CLOSES
FCV-76-65	Containment Inerting System B sample	CLOSES
FCV-76-66	Containment Inerting System B sample	CLOSES
FCV-76-67	Containment Inerting System B sample	CLOSES
FCV-76-68	Containment Inerting System B sample	CLOSES
FCV-84-20	Drywell or Suppr Chbr exhaust to SGTS	CLOSES

RPS BUS A or B POWER TRANSFER

1. Transfer of power supply to either RPS Bus A or B (Continued):

VALVE	FUNCTION/SYSTEM	ACTION
FCV-90-254A	Drywell radiation monitoring sample	CLOSES
FCV-90-254B	Drywell radiation monitoring sample	CLOSES
FCV-90-255	Drywell radiation monitoring sample	CLOSES
FCV-90-257A	Drywell radiation monitoring sample	CLOSES
FCV-90-257B	Drywell radiation monitoring sample	CLOSES
FCO-64-13	Reactor Zone ventilation	CLOSES
FCO-64-14	Reactor Zone ventilation	CLOSES
FCO-64-40	Reactor Zone ventilation	CLOSES
FCO-64-41	Reactor Zone ventilation	CLOSES
FCO-64-42	Reactor Zone ventilation	CLOSES
FCO-64-43	Reactor Zone ventilation	CLOSES
FCO-64-5	Refuel Zone ventilation	CLOSES
FCO-64-6	Refuel Zone ventilation	CLOSES
FCO-64-9	Refuel Zone ventilation	CLOSES
FCO-64-10	Refuel Zone ventilation	CLOSES
FCO-64-44	Refuel Zone ventilation	OPENS
FCO-64-45	Refuel Zone ventilation	OPENS
	Reactor Zone supply and exhaust fans	TRIP
	Refuel Zone supply and exhaust fans	TRIP
	Standby Gas Treatment System	STARTS
	Control Bay Emergency Pressurization System A an B	STARTS
	Traversing Incore Probe System	AUTO RETRACT

RPS BUS A or B POWER TRANSFER

2. Transfer of power to RPS Bus A only may result in the following events in addition to those listed for RPS Bus A or B power transfer:

VALVE	FUNCTION/SYSTEM	ACTION
FCV-74-48	RHR shutdown cooling inboard suction	CLOSES
FCV-74-53	RHR System I inboard injection	CLOSES
FCV-74-102	RHR System HP flush/vent	CLOSES
FCV-74-103	RHR System LP flush/vent	CLOSES
FCV-75-57	Drain pump A inboard isolation	CLOSES
FCV-77-15A	Drywell equipment drain discharge	CLOSES
FCV-77-2A	Drywell floor drain discharge	CLOSES
FCV-69-1	RWCU inlet	CLOSES
FCV-69-2	RWCU inlet	CLOSES
FCV-69-12	RWCU outlet	CLOSES
FCV-1-14	MSIV AC control power	DE-ENERGIZES
FCV-1-26	MSIV AC control power	DE-ENERGIZES
FCV-1-37	MSIV AC control power	DE-ENERGIZES
FCV-1-51	MSIV AC control power	DE-ENERGIZES
FCV-1-55	Main Steam Line drain inboard	CLOSES
FCV-43-13	Recirc loop inboard sample	CLOSES

RPS BUS A or B POWER TRANSFER

3. Transfer of power to RPS Bus B only may result in the following events in addition to those listed for RPS Bus A or B power transfer:

VALVE	FUNCTION/SYSTEM	ACTION
FCV-74-47	RHR shutdown cooling outboard suction	CLOSES
FCV-74-67	RHR System II inboard injection	CLOSES
FCV-74-119	RHR System HP flush/vent	CLOSES
FCV-74-120	RHR System LP flush/vent	CLOSES
FCV-75-58	Drain pump A outboard isolation	CLOSES
FCV-77-15B	Drywell equipment drain discharge	CLOSES
FCV-77-2B	Drywell floor drain discharge	CLOSES
FCV-69-2	RWCU inlet	CLOSES
FCV-69-12	RWCU outlet	CLOSES
FCV-1-15	MSIV AC control power	DE-ENERGIZES
FCV-1-27	MSIV AC control power	DE-ENERGIZES
FCV-1-38	MSIV AC control power	DE-ENERGIZES
FCV-1-52	MSIV AC control power	DE-ENERGIZES
FCV-1-56	Main Steam Line drain outboard	CLOSES
FCV-43-14	Recirc loop outboard sample	CLOSES

LAST PAGE

IV. Abnormal Operations (Continued)

P. PRIMARY CONTAINMENT ISOLATIONS (1-8)

4. (Continued)

d. HPCI exhaust diaphragm pressure high (10 psig between rupture discs).
Refer to OI-73, Abnormal Section, for operator actions.

5. Group 5 - RCIC isolation is initiated by one or more of the following:

a. RCIC steamline space high temperature (200°).

b. RCIC steamline high flow (450" water ΔP or $\geq$ 150% after ~ 3-second time delay).

c. RCIC steamline low pressure (50 psig).

d. RCIC exhaust diaphragm pressure high (10 psig between rupture discs).
Refer to OI-71, Abnormal Section, for operator actions.

6. Group 6 - Ventilation systems isolation is initiated by one or more of the following:.

a. Reactor low level (+11 inches above instrument zero).

b. High drywell pressure (2.45 psig).

c. Reactor building high radiation (100 mr/hr).
Refer to OI-30, Abnormal Section, for operator actions.

1. Rx zone ventilation hi radiation 100mr/hr.
2. Refuel zone area hi radiation 100mr/hr.

7. Group 7 - Process line isolation is initiated by the following condition only.

a. The respective turbine steam supply valve not fully closed.
Refer to OI-64, Abnormal Section, for operator actions.

8. Group 8 - TIP isolation is initiated by the following:

a. High drywell pressure at 2.45 psig.

* b. Reactor vessel low water level at + 11 inches above
* instrument zero. Refer to GOI-100-9, Abnormal Section, for operator actions.

*Revision

Question 2.05 (1.00)

Question 6.02 (1.00)

During your shift the Drywell Air System (DWAS) isolates. You verify a Group VI isolation has not occurred.

a. Name one other signal that could have caused the DWAS isolation.

b. WHAT air system valves close when the DWAG isolates? (i.e., valves within DWAS that will closes when the system gets an isolation signal).

Answer:

a. reactor zone ventialtion radiation signal (.5)
b. D/W air compressor suction valves (63,62) (.5)

Reference:

BFNP: OPL171.054, Control and Station Air Systems, p. 11.
Objectives V.C.

TVA Comment:

Part A: The control air lesson plan does make it appear, due to outline format, that Reactor zone high radiation is not a group VI isolation signal. The PCIS lesson plan 171.017 page 12 and OI 64, page 24 indicates it is a group 6 PCIS isolation. The candidates know the isolation signals and this question confused them. The format of OPL171.054 is being corrected. The loss of control air on U1 & U2 will result in closure of the valves 62 & 63. (REF OI 32A, Section 3.0)

TVA Resolution:

Part A: This question caused a great deal of confusion. The question should be deleted with credit for the time spent addressing the response since this was a timed examination.

1752Q

Table 1
(continued)

Initiation Signals	Group 5	Valve Type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)
RCIC space hi temp. 200°F	RCIC turbine steam supply isolation valve (FCV 71-2)	MO Gate	Inside	AC	AC
RCIC steamline hi flow 150% (after a 3 second delay) RCIC steamline low press. 50 psig RCIC high pressure between rupture disc 10 psig	RCIC turbine steam supply isolation valve (FCV 71-3)	MO Gate	Outside	DC	DC

Initiation Signals	Group 6	Valve Type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)
Rx low level +11" Hi drywell press +2.45 psig Hi Rad Rx bldg ventilation ~~100~~ 72 mr/hr. PKH 1/30/87 Hi Rad refuel zone ~~100~~ 67 mr/hr PKH 1/30/87	Drywell nitrogen purge inlet isolation valves (FCV-76-18)	AO butterfly	Outside	Air/AC	Spring
	Suppression chamber nitrogen purge inlet isolation valves (FCV-76-19)	AO Butterfly	Outside	Air/AC	Spring
	Drywell main exhaust isolation valves (FCV-64-29 and 30)	AO Butterfly	Outside	Air/AC	Spring

NOTE: O & MR 294

Table 1
(continued)

Initiation Signals	Group 6	Valve Type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)
	Suppression chamber main exhaust isol. valves (FCV-64-32 and 33)	AO Butterfly	Outside	Air/AC	Spring
	Drywell/suppression chamber purge inlet (FCV-64-17)	AO Butterfly	Outside	Air/AC	Spring
	Drywell atmosphere purge inlet (FCV-64-18)	AO Butterfly	Outside	Air/AC	Spring
	Drywell hydrogen sample line valves analyzer A (FSV-76-49)	SO Gate	Inside	AC	Spring
	Drywell hydrogen sample line valves analyzer A (FSV-76-50)	SO Gate	Outside	AC	Spring
	Drywell oxygen sample line valves analyzer A (FSV-76-51)	SO Gate	Inside	AC	Spring
	Drywell oxygen sample line valves analyzer A (FSV-76-52)	SO Gate	Outside	AC	Spring
	Torus oxygen sample line valves analyzer A (FSV-76-53)	SO Gate	Inside	AC	Spring

Table 1
(continued)

Initiation Signals	Group 6	Valve Type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)
	Torus oxygen sample line valves analyzer A (FSV-76-54)	SO Gate	Outside	AC	Spring
	Torus hydrogen sample line valves analyzer A (FSV-76-55)	SO Gate	Inside	AC	Spring
	Torus hydrogen sample line valves analyzer A (FSV-76-56)	SO Gate	Outside	AC	Spring
	Sample return valves - analyzer A (FSV-76-57)	SO Gate	Inside	AC	Spring
	Sample return valves - Analyzer A (FSV-76-58)	SO Gate	Outside	AC	Spring
	Drywell hydrogen sample line valves - Analyzer B (FSV-76-59)	SO Gate	Inside	AC	Spring
	Drywell hydrogen sample line valves - Analyzer B (FSV-76-60)	SO Gate	Outside	AC	Spring
	Drywell oxygen sample line valves - Analyzer B (FSV-76-61)	SO Gate	Inside	AC	Spring

Table 1
(continued)

Initiation Signals	Group 6	Valve Type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)
	Drywell oxygen sample line valves - Analyzer B (FSV-76-62)	SO Gate	Outside	AC	Spring
	Torus oxygen sample line valves - Analyzer B (FSV-76-63)	SO Gate	Inside	AC	Spring
	Torus oxygen sample line valves - Analyzer B (FSV-76-64)	SO Gate	Outside	AC	Spring
	Torus hydrogen sample line valves - Analyzer B (FSV-76-65)	SO Gate	Inside	AC	Spring
	Torus hydrogen sample line valves - Analyzer B (FSV-76-66)	SO Gate	Outside	AC	Spring
	Sample return valves - Analyzer B (FSV-76-67)	SO Gate	Inside	AC	Spring
	Sample return valves - Analyzer B (FSV-76-68)	SO Gate	Outside	AC	Spring

Table 1
(continued)

Initiation Signals	Group 6	Valve Type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)
	Suppression chamber purge inlet (FCV-64-19)	AO Butterfly	Outside	Air/AC	Spring
	Drywell/ suppression chamber nitrogen purge inlet (FCV-76-17)	AO Butterfly	Outside	Air/AC	Spring
	Drywell exhaust valve bypass to standby gas treatment system (FCV-64-31)	AO Butterfly	Outside	Air/AC	Spring
	Suppression chamber exhaust valve bypass to standby gas treatment system (FCV-64-34)	AO Butterfly	Outside	Air/AC	Spring
	Drywell/ suppression chamber nitrogen purge inlet (FCV-76-24)	AO Butterfly	Outside	Air/AC	Spring
	System suction isolation valves to air compressors "A" and "B" (FCV-32-62, 63)	AO Valve	Outside	Air/AC	Spring

2.4 Plant Drawings (Continued)

2.4.4 45N1631-18, Wiring Diagram 120V AC/250V DC Valves and Misc. Connection Diagram

2.4.5 47A1366-32 - series, Valve Tabulation of Marker Tags

2.4.6 47B601-32 - series, Instrument Tabulation

2.4.7 1-47E610-32-2, Mechanical Control Diagram Control Air System

2.4.8 1-47E610-76-1, Mechanical Control Diagram Containment Inerting System

2.4.9 1-47E1847-6,10, Flow Diagram Control Air System

2.4.10 47W611-32-2, Mechanical Logic Diagram Drywell Air Compressor

2.5 Vendor Manuals

2.5.1 Ingersoll - Rand Instructions and Parts List model 2 Air Dryer (Form 1136B) Contract 75472 CVM #52

2.5.2 Ingersoll - Rand Instructions Finger Valve 1 through 3 horsepower Type 30 Compressors (Model 23 ANL and 235 HNL) (Form AP-0145) Contract 75472 CVM #52

3.0 PRECAUTIONS AND LIMITATIONS

3.1 DRYWELL CONTROL AIR COMPRESSOR SUCTION valves, 1-FCV-32-62 and 1-FCV-32-63, will close on any of the following Group VI isolation signals:

3.1.1 Low Reactor Water Level (+ 11 inches).

3.1.2 Drywell High Pressure (2.45 psig).

3.1.3 Reactor Building Ventilation Radiation High (72 mr/hr).

3.2 DRYWELL CONTROL AIR COMPRESSOR SUCTION valves, 1-FCV-32-62 and 1-FCV-32-63, will close on loss of Plant Control Air Supply.

3.3 The Drywell Control Air Compressors will trip on low oil level in the crankcase.

Question 2.07 (3.00)

Concerning the CRD system:

a. WHAT are the normal values for CRD hydraulic system FLOW and DRIVE WATER DIFFERENTIAL PRESSURE? (1.5)

b. WHAT percentage of CRD hydraulic system FLOW is supplied to the CRD cooling water header? (1.0)

c. Immediately following a reactor scram the control rod full-in (green) lights on panel 9-5 are lit but there is no position readout displayed. EXPLAIN WHY this occurs and WHAT eventually happens that allows the control rod to settle into the 00 position. (1.5)

Answer:

a. 45 to 65 gpm (accept 0.25 to 0.33 gpm per CRD)
260 psid (accept 250 to 270 psid)

b. 100% (accept "all")

c. Following a scram, but before the SDV is full, the control rod will be in the over travel-in position since there is still a large D/P across the piston.

After the SDV is full, there is no D/P across the piston and the control rod will settle into the 00 position.

Reference:

BFNP: OPL171.005, CRDH, pp. 9, 10, 24 through 29, and 40.
L.O. M, O, and S

TVA Comment:

a. The normal valve for CRD hydraulic system flow given in the answer key (45 to 65 gpm) is the flow to the drive and cooling water headers and is the Indicated system flow on panel 9-5. The total system flow however includes 4 to 6 gpm (REF BF 12.24, pg. 81 attached) to each of the Reactor Recirculation Pumps and 20 gpm (REF OPL171.005, pg. 18) pump minimum flow which are not seen by the flow indication. The total CRDH system flow can thus be as high as 97 gpm. An answer of 45 to 100 gpm should be accepted for full credit.

b. The percentage of flow which is directed to the cooling water header will vary based on the point used to calculate CRD hydraulic system flow in part a. However, even if the 45 to 65 gpm throught the flow element is used as the system flow, at least part of the flow through the stabilizing valves (~ 2 gpm) does not go through the cooling water header but goes through the exhaust header orificed check valve and lifts the 40D valve on some rods to relieve to the reactor. The flow through the cooling water header is something less than 100% and each answer should be evaluated individually based on the students assumptions.

Resolution:

a. Full credit should be given for:
45-65 gpm as stated in answer key or expand the answer key to accept 45 to 100 gpm.

b. change to reflect full credit for <100% flow

ATTACHMENT 6
(Continued)

Elevation 565 (Continued)

Unit 2

	2300-0700	0700-1500	1500-2300
Core Spray Sparger Break (SI-2)			
PdIS 75-28			
PdIS 75-56			
Ventilation (Rx Bldg 480-V Vent Bd 2B)			
Rx Zone Supply Fan A (OFF, SLOW, FAST)			
Rx Zone Supply Fan B (OFF, SLOW, FAST)			
Refuel Zone Supply Fan A (OFF, SLOW, FAST)			
Refuel Zone Supply Fan B (OFF, SLOW, FAST)			
Reactor Recirc Pump			
A Seal Water Flow (4-6 gpm)			
B Seal Water Flow (4-6 gpm)			
Drywell A/C Suction filter 2 min Blowdown (32-304)	(I)		
Rx Water Level (SI-2)			
LIS-3-52			
LIS-3-62			
Secondary Containment Doors (closed)			
No. 236 Unit 2 to Air Lock (NW)	(I)	(I)	(I)
No. 238 Unit 2 Inside Eqpt Lock (NW)	(I)	(I)	(I)
No. 237 Unit 2 Outside Eqpt Lock (NW)	(I)	(I)	(I)
No. 240 Unit 2 to Elev Shaft (SW)	(I)	(I)	(I)
No. 244 Unit 2 to Air Lock (NE)	(I)	(I)	(I)
No. 242 Unit 2 to Elev Shaft (SE)	(I)	(I)	(I)
CRD HCU manual valve check (visual) (Monday)	(I)		

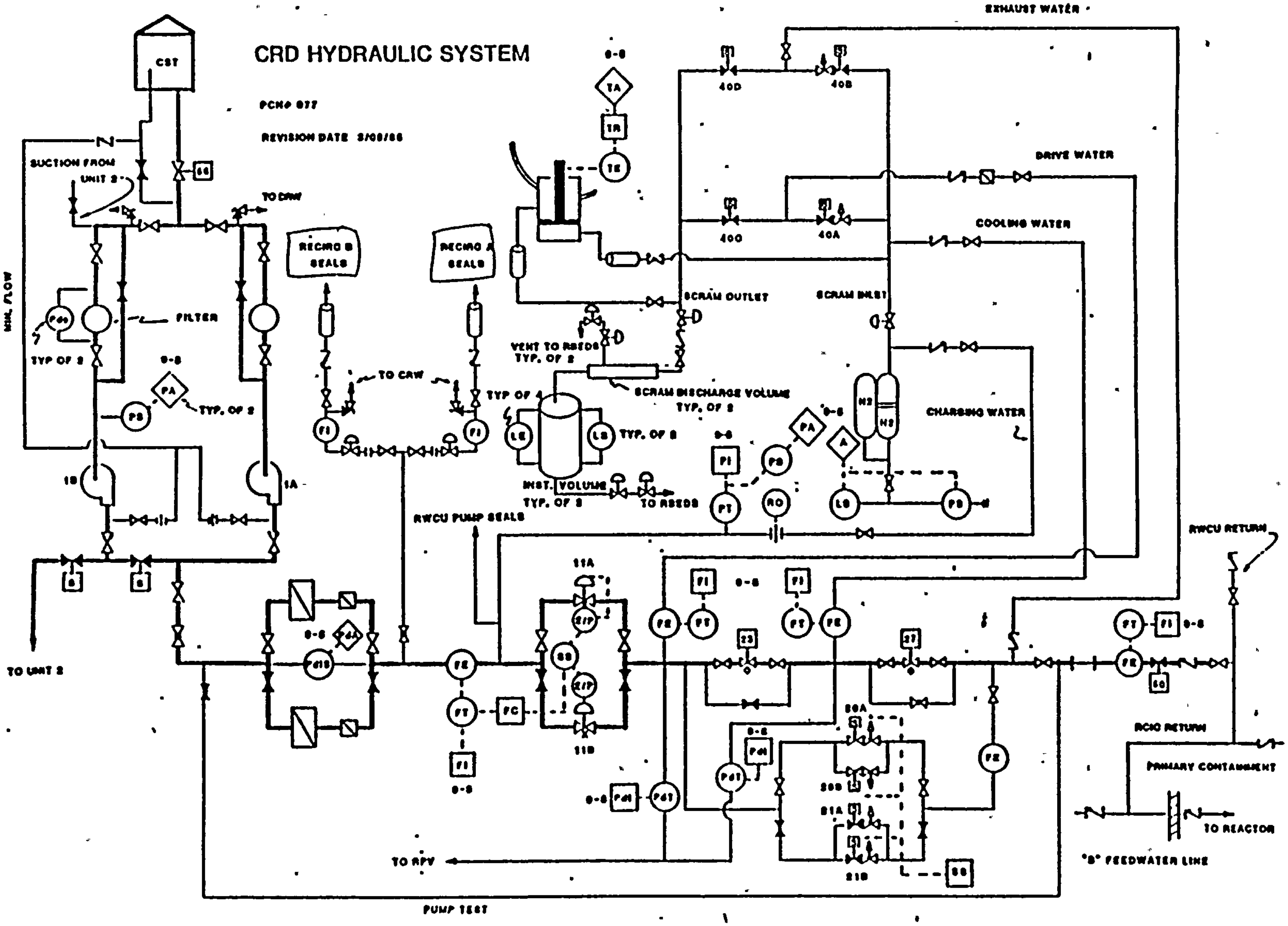


FIGURE 3 CONTROL ROD DRIVE HYDRAULIC SYSTEM (TP-3)

Lesson Outline	Instructor Notes
(b) The flow control valve maintains a constant flow of 55-65 GPM through the system. (That is, the flow control valve will have to open further as reactor pressure increases in order to maintain the required system flow.)	
(c) If flow stays constant in the system, the pressure drop across the drive and cooling water pressure control valves will stay constant regardless of reactor pressure.	
(d) As a result, the drive and cooling water pressure control valves will require adjusting only once (upon system startup) and will not require constant adjustment during a startup or shutdown.	
(6) System return line (Figure 6)	TP-6
(a) Flow path	
Returns water from the CRD hydraulic system to reactor.	NOTE: GE SIL No Supplement 2
i. Via cooling water pressure control valves and cooling water supply lines up through drives and into the vessel. Additionally 1-2 gpm is distributed through the exhaust header orificed check valve. This flow unseats the 40D valve at ~ 3 psid and flows to P-over area into vessel. During control rod movement when the selected HCU 40D or 40B valve opens the drive water flows to the exhaust water header. This flow is distributed through the other HCU's 40D valves. This occurs because the 40D valves require only 3 psid to unseat as opposed to 20 psid for the cooling water header flow path.	

Question 2.08 (1.50)

Question 6.04 (1.50)

A Core Spray line breaks inside the shroud.

a. WILL the break cause an alarm in the control room (YES or NO)? (.5)

b. HOW will the break affect core spary performance for that loop? (1.0)

Answer:

a. No (.5) (If a core spray line breaks inside the shroud, the differential pressure indicating switch will detect reactor pressure inside the shroud as usual; therefore, no abnormal differential pressure will be indicated.)

b. The core spray loop can perform a flooding function (.5) but its spray will not provide full core spray coverage (.5)

Reference:

BFNP: OPL171.045, Core Spray, pp. 15 and 16
Objective V.K.

TVA Comment:

The answer key requires "flooding function" for full credit; however, an answer which addresses lost of spray function should receive full credit since break was inside shroud as given is question.

TVA Resolution:

Accept lost of spray cooling function for full credit.

1752Q

Question 6.07 (2.50)

Question 3.1 (2.50)

The plant is operating at 100% power and 100% core flow when the "A" flow converter output fails to zero. MATCH from Column B the action that will exist for each trip function in Column A given the above conditions.

NOTE: REPSONSES MAY BE USED MORE THAN ONCE

COLUMN A	COLUMN B
a. "A" APRM Hi-Hi thermal	1. Rod Block
b. "B" APRM Hi-Hi thermal	2. Half Scram
c. "C" APRM Hi	3. Full Scram
d. "D" APRM Hi	4. None
e. "E" APRM Hi-Hi neutron	

Answer:

a. 2

b. 4

c. 1

d. 4

e. 4

Reference:

BFNP: LP 22, L.O.D

TVA Comment:

Clarification received by several candidates resulted in no credit. The clarification given: conditions in Column 'A' existed in addition to the flow converter 'A' failure. An additional answer should be developed that addresses the question in this context. Therefore a candidate who successfully answers based upon this clarification will not be jeopardized.

TVA Resolution:

Expand the answer key to accept for full credit a correct response to the question taken from the concept of Column 'A' existing in addition to the flow converter failure.

1753Q

Question 3.05 (2.00)

You are in the process of preparing the Main Turbine for startup in accordance with OI-47.III.c. The following conditions exist:

Main Turbine is reset
VALVES CLOSED is selected
Warming rate indicator is at zero position
Load limit is set at 100%
FAST acceleration rate is selected

a. STATE the position for EACH of the following valves with the turbine in this condition.

(1) Main Stop Valves
(2) Control Valves
(3) CIVS Stop
(4) Intercept

b. You now select SHELL WARMING to prewarm the turbine by pressurization of the HP turbine. STATE the new position of the valves, specified in part "a" above, given this changed condition.

Answer:

a. (1) Closed
(2) Closed
(3) Open
(4) Closed

b. (1) No. 2 bypass open
Nos. 1,3,4 closed
(2) Open
(3) Closed
(4) Closed

Reference:

BFNP: LP 10, L.O.D
OI-47

TVA Comment:

Part B: With the initial conditions stated, i.e "warming rate at zero," the number 2 stop valve internal pilot (bypass) will remain closed until the warming rate potentiometer is increased.

The warming rate potentiometer must be at low speed stop (zero position) procedurally and mechanically to select shell or chest warming.

TVA Resolution:

Part B: Change answer key to accept number 2 stop valve internal pilot (bypass) valve closed and stop valves closed should be accepted for full credit.

III. Operating Instructions (Continued)

C. Preparation for Startup (Continued)

1. To reset main turbine (Continued)

b. Depress the master reset pushbutton switch (HS-47-67B) until the emergency trip system TRIPPED light goes out (approximately 3 to 5 seconds).

c. The mechanical trip valve and the vacuum trip will also light their RESET lamps.

d. Observe the following:

1) The No. 2 stop valve is held closed.

2) The No. 1, 3, and 4 main stop valves are held closed by their respective test solenoid valves until the No. 2 main stop valve reaches its full open position.

3) The control valves are held closed.

4) The intercept valves are held closed.

5) The intermediate stop valves will open.

2. To prewarm, by pressurization HP turbine.

* NOTE: When the first stage bowl temperature is < 250°F,
* prewarming by pressurization is necessary. This is to be
* done as the reactor temperature increases into the heating power range.

a. Check the following permissives met:

1) Turbine reset.

2) VALVES CLOSED selected.

—>3) Warming rate indicator must be at zero position.

b. Set load limit to 100%.

c. Select FAST acceleration rate.

NOTE: Prior to performing the next step, close the following valves: FCV-1-121, -129, and -137 (LP STM SUPPLY TO RFPTs).

*Revision for pagination

III. <u>Operating Instructions</u> (Continued)

C. Preparation for Startup (Continued)

2. To prewarm, by pressurization HP turbine. (Continued)

d. Open/check open steam leads drain FCV-6-109.

e. Select SHELL WARMING and observe:

1) Shell warming light comes ON.

2) Intercept valves remain CLOSED.

3) Intermediate stop valves go CLOSED.

4) Control valves fully OPEN.

5) Main stop valve <u>No. 2 servo current is at zero</u>.

f. Press INCREASE button until pressure starts to build up in the high pressure turbine.

<u>NOTE</u>: In the event the turbine should roll off turning gear, the governor will limit turbine speed to 100 rpm by closing the control valves.

<u>NOTE</u>: If turbine rolls off turning gear, decrease flow to zero, wait until zero speed on the turbine then place turbine back on turning gear and repeat the above step as necessary. * * *

g. Monitor high pressure turbine exhaust pressure to maintain 60-100 psig.

<u>NOTE</u>: Monitor computer point A345(U-1&3) D345 (U-2) continuously to maintain turbine 1st stage pressure ~60-100 psig. Reactor scram may result when in shell warming with stop valves closed and turbine 1st stage pressure $\geq$ 142 psig. *

<u>NOTE</u>: The first stage bowl metal temperature differential is limited to 75°F.

<u>NOTE</u>: The temperature rise on the inner first stage bowl metal should not exceed 150°F/hr.

h. Keep differential expansion within limits.

i. Keep HP shell temperature 250-280°F and steam chest temperature 280°F.

*Revision for pagination

III. Operating Instructions (Continued)

C. Preparation for Startup (Continued)

2. To prewarm, by pressurization HP turbine. (Continued)

j. Continue to warm for length of time indicated by Figure 47-2.

→ k. Upon completion of warming, zero flow and select OFF.

NOTE: The control valves will now close and the intermediate valves will open.

l. Open all drain valves.

3. Valve chest warming.

a. Check the following permissives met:

1) Turbine reset.

2) VALVES CLOSED selected.

→ 3) Warming rate indicator must be set at zero.

b. Select CHEST WARMING mode.

NOTE: The control, intercept, and main stop valves should be closed.

c. Slowly increase flow through the No. 2 MSV to establish the required warming rate.

NOTE: The warming rate should be regulated in such a way as to remain within the control valve chest metal temperature differential limits given on Figure 47-1.

d. After steam chest pressure and temperature are at rated and the differential expansion is normal, terminate chest warming by pushing OFF button.

NOTE: The turbine should be rolled within 2-3 hours after completion of the prewarming operations so that unnecessary cooling is avoided.

*Revision for pagination

Question 3.08 (3.00)

Answer EACH of the following with respect to the Rod Sequencer Control System:

a. The RSCS was developed for three different regions of rod withdrawal:

(1) 100% rod density to 50% rod density
(2) 50% rod density to preset power level
(3) Beyond preset power level

For EACH region above STATE BOTH the design function of the RSCS AND the type of rod control in effect to accomplish this fucntion. (1.5)

b. During a reactor startup under rod sequencer "A" all A12 and A34 rods are fully withdrawn then the Sequencer Mode Selector (SMS) and Rod Sequencer Seclector (RSS) switches are placed in "Normal". LIST four (4) interlocks this action enables. (1.0)

c. State the effect on RSCS if its turbine generator 1st stage shell pressure input fails HIGH. (.5)

Answer:

a. (1) Prevents selection or movement of rods out of sequence (.25)
Sequence Control (.25)

(2) Prevents withdrawal errors within the sequence (.25)
Group Notch Control (.25)

(3) None (RSCS bypassed) (.25)
None (.25)

b. (1) Allows selection of any "B" sequence rod (.25)

(2) Enables group notch control (GNC) logic (.25)

(3) Bypasses the continuous withdraw mode of RMC (.25)

(4) Prevents selection of any "A" sequence rod (.25)

c. Bypasses all rod sequence control logic (.25)

Reference:

BFNP: LP 25, L.O. A & I.1

Question 3.08 (3.00) continued

TVA Comment:

Part a. The purpose (Design Function) of the RSCS system is to restrict control rod movement in the startup and low power ranges. This limits peak full enthalpy to <280 calories/gm upon the postulated rod drop accident. (REF: OPL171.025, p. 3). In any region in which RSCS is enforcing, selection of rods not in the required sequence is prevented. In region from 100% Rod density to 50% Rod density, only one RSCS group A_{12} or A_{34} (B_{12} or B_{34} if starting up using B sequence rods) can be selected. In region from 50% rod density to preset power level only the rods in opposite sequence (B_{12} and B_{34}) can be selected. This region also enforces Group Notch Logic on the individual RSCS groups. Group Notch Logic will keep all rods within a RSCS group within one notch of the other rods in the group (i.e. RSCS does not enforce group insert or withdraw limits)

TVA Resolution:

Part a (100% RD to 50% RD): Restrict control rod movement of rods not in selected sequence.

: Sequence control

Part a (50% and Preset power level): Restrict control rod movement of rods not in selected sequence and enforces group notch control.

: Group Notch Control

1753Q

Question 3.13 (2.50)

Question 6.14 (2.50)

Answer EACH of the following with regard to the 250V Unit and Plant DC power system:

a. LIST three (3) major types of loads supplied by this system. (.75)

b. EXPLAIN how a reliable source of DC power is maintained to these loads. INCLUDE ALL NORMAL, ALTERNATE & BACKUP POWER SUPPLIES AND ASSOCIATED COMPONENTS. (1.0)

c. EXPLAIN why DC power is preferred for these types of load (other thatn for imporived reliability).
BE SPECIFIC. THREE RESPONSES REQUIRED FOR FULL CREDIT. (.75)

Answer:

a. (1) DC motor operated valves

(2) DC motor operted pumps

(3) Control power for ECCS

(4) Logic power for ECCS (any 3 @ .25 ea)

b. The DC bus normally is supplied by a battery charger (.25) powered from the 480V AC shutdown board (.25)

Alternate power to the charger is from the 480V common board 1 (manual transfer only) (.25)

Backup power is supplied by a (120 cell lead-acid) battery on a float charge (.25)

c. (1) Provides more constant pull on coils
(2) Absence of hysterisis effects
(3) Absence of eddy current losses (.25 each)

Reference:

BFNP: LP 37, L.O. A, B & C

TVA Comment:

Part a 3 major types of loads: The objective does state motive power for D.C. powered pumps and motor operated valves. The control and logic power for ECCS. The candidates responses could be more specific. The logic and control provided by the D.C. system is supplied to more than ECCS.

Part b Reliability of DC is subjective and all Normal, Alternate and Backup power and associated components. This is objective based and relative straight forward. But consider the bold print DC power INCLUDE ALL NORMAL, ALTERNTE AND BACKUP POWER SUPPLIES AND ASSOCIATED COMPONENTS. Considering D.C. only:

(1) normal battery charger
(2) alternate battery charger
(3) the battery itself

Considering A.C.: Both the normal and alternate battery chargers (manual transfer between two) have normal AC from 480V shutdown board with manual transfer to alternate AC from 480V common board.

Part c Not objective based, more a plant design consideration than a concern of an oprator.

TVA Resolution:

Part a Anser key should be expaned to reflect credit given for responses stating control power and logic power. Control power may be specified by boards ex: 480V shutdown boards, cooling tower switch gear, 4Kv shutdown boards. Logic power may specify systems. The candidates response should be analyzed and credit given for valid response.

Part b Candidates should receive credit for a response that addresses the question from the D.C. application. The answer key should be (expanded to reflect a correct response for:
Normal - normal battery cahrger
Alternate - alternate battery charger
Backup (lead acid) battery

Part c Delete the question or conversely analyze the responses for validity and credit respectively.

Question 7.03

Question 4.04

The following parameter changes / annunciators are observed by the reactor operator:

RBCCW temperature Lower than normal
RBCCW Surge Tank HI Level alarm
(No other alarms present)

a. WHICH one (1) of the following malfunctions would most likely cause these indications: (1.0)

1. Raw Cooling Water leak in the RBCCW Heat Exchanger(s).
2. Reactor Coolant leak into RBCCW via NRHX.
3. Fuel Pool Cooling System leak from RBCCW.
4. RBCCW Makeup Valve (fiol valve) leak.
5. DWEDS Heat Exchanger leak into RBCCW.

b. LIST three (3) of the conditions/circumstances that will cause the isolation valve to non-essential equipment (MOV-48) to automatically close. (1.5)

NOTE: BE SPECIFIC AND INCLUDE SETPOINT VALUES

Answer:

a. 1 (1.0)

b. 1) Low Reactor Water Level <= -114.5 (0.2) and 90.1) DG voltage applied to SD board(s) (0.2)
   2) Drywell Pressure > 2.45 psig (0.2) and (0.1) DG voltage applied to SD board(s) (0.2)
   3) Low discharge header pressure < 57 psig. (0.5)

Reference:

BFNP: OI 70 LO A, AOI-70, LP 171.047
3.8/4.1 3.3/3.4 2.9/3.2

TVA Comment:

(B) OI-70 list the signals as loss of normal AC power in conjunction with an accident signal or 57 psig header pressure. This should be the correct answer.
LP 171-047 List Signals as
(1) Initiation of Unit 1 and 2 480 volt load shed logic.
(2) Low RBCCW supply header pressure (57 psig)

TVA Resolution:

(B) Accept answer in "B" above as correct for full credit if only two responses are given.)

1757Q

Question 4.13 (1.00)

Question 7.12 (1.00)

LIST two (2) systems that require tagging prior to entry into the Primary Containment. INCLUDE in your answer the required status or position of the system.

Answer:

TIP (.25) withdrawn (.25)
Nitrogen Isolation Valves to Primary Containment (.25) closed (.25)

Reference:

BFNP: BF 14.9 LO A
3.2/3.7 3.2/3.4

Comment:

Per BF 14.9 the Tips are to be withdrawn and tagged also the Nitrogen Isolation Valves to Primary Containment are to be closed and tagged (76-539, 76-541, 76-24, 84-37 and 84-38) as can be seen the Nitrogen systems tagged are Sy. 76 and Sy. 84. Sy. 76 valves are the purge and makeup nitrogen valves. Sy. 84 is the CAD valves.

TVA Resolution:

Accept for full credit any two of the following:

1. Tips withdrawn and tagged
2. CAD system isolated and tagged (system 84)
3. Nitrogen isolated and tagged (system 76)

Question 7.16

Question 4.14

A single Recirculation Pump trips while operating at 100% power in automatic control.

a. STATE the immediate action(s) that should be performed on the RUNNING PUMPL.

b. EXPLAIN WHY the Running Pump speed must be reduced to <50% of rated speed prior to starting the idle pump.

Answer:

a. Place Recirculation Subpanel in Manual (.5) and reduce speed to establish 100% loop flow (45,200 gpm) (.5)

b. Prevents excessive Jet Pump vibration.

Reference

TVA Comment

a. OI 68 does say place recirculation subpanel in manual and reduce speed to establish 100% loop flow; however, the same thing could be done with pump in auto using the master controller. The purpose is to reduce flow to within required limits. The method is not fixed as long as actions taken result in 100% loop flow.

TVA Resolution

Revise answer key to only require reduction to 100% loop flow for full credit. Re-assign point value 75% pump flow 25% sub-panel manual.

1757Q

Question 8.03 (2.50)

STATE whether a Radiation Work Permit (RWP) is "REQUIRED" or NOT REQUIRED" for EACH of the situation given below:

a. An employee will need to work in an area having airborne radioactivity of 15% MPC.

b. Work will be done in a designated "RADIATION AREA".

c. Work is to be done in an area with 1500 DPM/100 cm2 loose surface contamination.

d. A radiological survey inside a Contamination Zone will be performed while standing outside the Zone.

e. Trash and procetive clothing will be removed from a Contamination Zone while standing outside the Contamination Zone on the stepoff pad.

Answer:

a. not required
b. not required
c. required
d. not required
e. not required (.5 each)

TVA Comment:

Operations does not do surveys to determine how an area will be zoned and operations does not have equipment to determine airborne or contamination areas. This is done by RADCON. The people had to assume that numbers given in Part 'A' and 'C' would require the area be so zoned in order to get correct answer.

TVA Resolution:

Don't require people to know from memory limits for zones that they don't have equipment to check and are not responsible for doing.

1758Q

Question 8.09 (2.50)

Unit 1 Technical Specifications specify for REACTIVITY CONTROL . . .

"A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive condition during the operating cycle . . ."

LIST the three (3) conditions/assumptions which are used to verify this "Reactivity Margin" (Adequate Shutdown Margin).

Answer:

(1) Highest worth rod (.25) fully withdrawn (.25)
(2) Xenon free core
(3) Cold core (68°F)

Reference:

EIH: U2 TS, 1.0 "SDM"
BFNP: U1 TS, 3.3/4.3A, OPL174.728 LO 9

TVA Comment:

The wording of the question can be misleading as to what response is required. Since question stated "most reactive", it is possible to assume this to mean "cold and xenon free."

TVA Resolution:

Accept as another response for full credit:

1. Strongest control rod fully withdrawn
2. All other operable control rod fully inserted
3. .38% $\Delta K/K$ margin

1758Q

Question 8.10 (1.50)

STATE the six (6) ITEMS to be recorded in the daily journal when the Reactor is declared "critical" during a Reactor startup in accordance with GP 100-1.

Answer:

Time
Rod Group
Rod Number
Rod Notch
Period
Recirc Loop Temperature

Reference:

BFNP: OPL171.174.724, LO 6

TVA Comment:

Rod group is not listed in GOI 100-1 as one of the items to be recorded when declare Reactor critical.

TVA Resolution:

Change key to delete 'Rod Group' from answer key and accept (5) five responses for full credit.

1758Q

Section III. Startup (Continued) INITIALS/TIME/DATE

A. Criticality (Continued)

****

CAUTION

DURING A HOT STARTUP FOLLOWING A REACTOR SCRAM AT HIGH POWER, THE CONDITIONS OF PEAK XENON WITH NO MODERATOR VOIDS COULD EXIST AT THE TIME OF STARTUP. UNDER THESE CONDITIONS, EXTREMELY HIGH ROD NOTCH WORTHS CAN BE ENCOUNTERED.

****

4. Upon approval of the shift engineer, start control rod withdrawal in accordance with OI-85. (R)_____/ /_____

NOTE: Shift all SRM and IRM recorders to fast speed prior to criticality and return to slow speed after initial period measurements are calculated.

NOTE: Within the approved control rod withdrawal sequence, it is possible to have a period less than 60 seconds. If a period less than 30 seconds is observed, insert rods until subcriticality is observed and contact the nuclear engineer and shift engineer before pulling any more rods. Periods less than 5 seconds are reportable to the NRC within 24 hours.

5. Observe the period meter when pulling rods and govern withdrawal rate to avoid having
* a period shorter than 60 seconds. (R)_____/ /_____

* NOTE: Reactor is critical when neutron flux rises on a constant (stable) period without further control rod movement.

6. When critical, record time, rod position, rod notch, period, and reactor water temperature
* from recirculation loop A in daily journal. (R)_____/ /_____

NOTE: Measure period as follows:
For 10% power rise, multiply time of rise by 10.5.
For doubling time, multiply time of rise by 1.445.
For decade rise, divide time of rise by 2.3.

* For direct period measurement when on IRMs:
* a. Time 25 to 68 on black scale ranges
* b. Time 8 to 22 on red scale ranges

* Revision

Question 8.12 (2.00)

DESCRIBE the four (4) standards (i.e. symbols/colors) used in marking TEMPORARY ALTERATIONS on plant drawings and WHAT they mean.

Answer:

1. Green - information deleted (by the TEMPALT)
2. Red - information added (by the TEMPALT)
3. Red circle - surrounds area affected (by the TEMPLAT)
4. TACF # - number assigned to track the TEMPLAT and is placed beside the red circle.

Reference:

BFNP: PMI-8.1, L.O. F

TVA Comment:

PMI 8.1 have been revised (revision attached) and changed the color requirements or were intended to be changed. The new revision has you circle affected area in yellow in one part, but says place TACF # beside Red circle in another part. The new revision was placed in Required Reading so some people may respond using new colors..

TVA Resolution:

Accept for full credit: Yellow or red for color on area to be circled.

1758Q

5.2 (Continued)

i. (Continued)

Green - information to be deleted by temporary alteration.

Red - information to be added by temporary alteration.

Yellow- circle the area affected by temporary alteration.

TACF# - to be written beside red circle (the originator is responsible for placing this number on the drawings after the SE has assigned a TACF number).

j. DNE Drafting Services is responsible for updating the as-constructed plant drawings affected by TACFs. The drawings should be revised and distributed in accordance with Standard Practice BF-2.5.

k. In the event the SE deems it necessary for the temporary alteration to be placed more quickly than the above procedure will allow (but the condition is not an emergency), the SE can direct the originator to mark up the SE's office copy and the affected control room(s) copies of the as-constructed drawings. (The standards for marking drawings listed in Paragraph 5.2.3.i above will be used). When this is done, the STA will verify the accuracy of the drawing changes made by the originator. It should be stressed that this is not the normal procedure to follow, but when and if it is followed, the originator is still responsible for taking a copy of the marked up drawings to DNE Drafting Services for update and distribution in accordance with Standard Practice BF-2.5 within 24 hours.

After installation, the SE's clerk will make two copies of the TACF. One copy will be mailed to the Technical Services System Engineering section. The other copy will be mailed to Planning and Scheduling.

5.2 (Continued)

g. On the TACF, under the section, "Effects, Limitation(s), and/or Actions," the originator should briefly describe the effect(s) of the temporary alteration. A detailed explanation is required where the temporary alteration has an effect on the system or other systems that may jeopardize the safe and continued operation of the plant. Note any limitation and/or action required during the period that the temporary alteration may exist. Explicit information shall be noted for situations that may require immediate operator action. List any requirements which must be completed prior to removal of temporary alterations such as approval of DCR, clearance of nonconforming item, etc. If a system or component cannot be made operable with the temporary alteration in place, it shall be so stated in this section of the TACF.

h. On the TACF on the lines, Tests That Will Be Performed To Prove Operability After TACF Installation and Tests That Will Be Performed To Prove Operability After TACF Removal, the orginator shall list all tests which will be done after installation and after removal of the temporary alteration. These tests should be written and performed (a) to assure system integrity and (b) to provide for evaluating the performance of the alteration <u>before</u> system operation. If the originator believes that testing is not required, he should provide a brief justification of why testing is not required.

i. The originator is responsible for supplying two sets of marked up drawings with the TACF. These drawings will be stamped "For Information Only". These drawings will show the configuration of the affected equipment after installation of the temporary alteration. One set of drawings will be used by DNE Drafting Services as a reference for marking up the original drawings. One will remain with the original TACF in the TACF file as a reference copy. If time permits, the TACF originator should work with the DNE Drafting Services to determine which drawings need to be marked up and included with the TACF file. The Shift Engineer's controlled copy of the as-constructed drawings and the affected unit control room's as-constructed drawings, shall be marked before the system is declared operable. The following standards will be used in marking drawings:

ENCLOSURE 4

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee:	Tennessee Valley Authority
Facility Licensee Docket No.:	50-259, 50-260, and 50-296
Facility Licensee No.:	DPR-33, DPR-52, and DPR-68
Operating Tests administered at:	Browns Ferry Nuclear Plant
Operating Tests Given On:	May 2-12, 1988

During the conduct of the simulator portion of the operating tests identified above, the following apparent performance and/or human factors discrepancies were observed:

1. The CRD system modeling is inaccurate in that cooling water header d/p does not go below 30 psid during normal system operation, while it should indicate approximately 20 psid.

2. Feedwater system modeling interacts with recirculation pump operation such that a recirculation pump runback proceeds down to approximately 52% flow while the plants' procedural runback specification is approximately 75% flow.

3. Plant procedure allows 3 element level control at 10% power and the simulator does not respond adequately at such a low power. This may be the result of a procedural or modeling problem. Candidates appeared unfamiliar with the simulator's response and devoted significant time to investigative efforts during the examination.

4. The RBCCW system displayed a modeling problem in that when a slow degradation in flow to approximately 90% by the partial closure of the return from the drywell valve was simulated, an inappropriately rapid response (2 to 3 seconds) in the temperature increase was indicated.

5. The simulator's copy of OI-68, did not have attachment C to which the candidate had been procedurally referenced. This may be a result of the many recent procedural changes. Checks should be made on all recent OI changes, and more care should be given to ensure proper referencing is maintained when such changes are made.

6. While at 75% power one MSIV was simulated to close. The simulator modeling gave a serious EHC problem which resulted in severe pressure transients of such duration and magnitude as to lead the operator to manually scram the reactor.

7. The remote closure of the suction from the suppression pool to the RHR pumps was poorly modeled such as to allow continued pump operation.

8. The simulator self-initiated three events that were programed for use later in the scenario.