UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

ENCLOSURE 1

EXAMINATION REPORT - 50 321/OL-89-03

Facility Licensee: Georgia Power Company
P. O. Box 256
Birmingham, AL 35201

Facility Name: E. I. Hatch Nuclear Plant

Facility Docket Nos.: 50-321 and 50-366

Facility License Nos.: DRP-57 and NPF-5

Examinations were administered at the E. I. Hatch Nuclear Plant near Vidalia, Georgia.

Chief Examiner: Curt Rapp 1/3/90
Curt W. Rapp Date Signed

Approved By: John F. Munro 1/3/90
John F. Munro, Chief Date Signed
Operator Licensing Section 1
Division of Reactor Safety

Summary:

Examinations were administered on October 9-13, 1989.

Written examinations and operating tests were administered to 2 SRO applicants and 11 RO applicants. All applicants passed these examinations.

REPORT DETAILS

1. Facility Employees Contacted:

   *S. Grantham, Operations Training Superintendent
   *M. Crosby, Operations Classroom Training Supervisor
   *S. Beck, Senior Simulator Instructor
   *S. Tipps, Nuclear Safety and Compliance Manager
   *L. Sumner, Assistant General Manager - Plant Operations
   *H. Nix, General Manager - Nuclear Plant
   *T. Moore, Assistant General Manager - Plant Support
   *C. Coggin, Training and Emergency Preparedness Manager
   *O. Fraser, QA Site Manager
   *J. Lewis, Operations Manager

   *Attended Exit Meeting

2. Examiners:

   *C. Rapp, Region II
   E. Rau, Region III
   C. Tyner, EG&G
   M. Spenser, EG&G
   B. Holbrook, Region II (Observer)
   E. Lea, Region II (Observer)
   J. Hammer, Region III (Observer)

   *Chief Examiner

3. Pre-Examination Review

   On September 27-29, 1989, members of the facility Training staff reviewed the written examination at the Region II office. This review was conducted to improve examination technical accuracy and ensure all test items were correct and concise prior to examination administration. The facility also used this review to challenge several test items on the examination. These test items are fully supported by the Knowledges and Abilities Catalog developed from the INPO generic Job/Task Analysis (JTA) database. Facility training objectives are not the sole source for determining test item inclusion. The pre-examination review is not intended as an open opportunity to tailor or customize examination content.

4. Post-Examination Review

   Following examination administration, the facility was allowed to review the written examination. A copy of the examination is included as Enclosure 3. The facility submitted five (5) comments for NRC review.

These comments are included in this report as Enclosure 3. The NRC resolution is included as Enclosure 4. This is significant improvement over the last examination for which 25 comments were submitted. However, four (4) comments were due to inaccurate, incomplete, or inconsistent facility submitted material for examination development. The facility should reevaluate current training material for additional weaknesses. Comments made during the pre-examination review will be taken into consideration but may not be used in the administered examination.

SRO MASTER
COPY

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
REGION 2

FACILITY: E. I. Hatch 1 & 2

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 89/10/09

CANDIDATE: ____________________

INSTRUCTIONS TO CANDIDATE:

Use answer sheet for the answers and write on one side only. Use paper provided for continuation of answer. 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%. The examination will have a time limit of four (4) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
47.25 ~~48.75~~	58.51 ~~58.56~~	________	________	5. EMERGENCY AND ABNORMAL PLANT EVOLUTIONS (60%)
33.50 ~~34.50~~	41.49 ~~41.44~~	________	________	6. PLANT SYSTEMS AND PLANT-WIDE GENERIC RESPONSIBILITIES (40%)
80.75 ~~83.25~~		________ 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. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done AFTER you complete the examination.

3. 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.

4. Use black ink or dark pencil only to facilitate legible reproductions.

5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet.

6. If you need additional space to answer a specific question, use a separate sheet of the paper provided. DO NOT WRITE ON THE BACK SIDE OF THE EXAMINATION ANSWER SHEET.

7. Print your name in the upper right-hand corner of the answer sheet.

8. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.

9. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

10. The point value for each question is indicated in parentheses after the question. The amount of blank space on an examination question page is NOT an indication of the depth of answer required.

11. Show all calculations, methods, or assumptions used to obtain an answer.

12. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK. NOTE: partial credit will NOT be given on multiple choice questions.

13. Proportional grading will be applied. Any additional wrong information that is provided will count against you. For example, if a question is worth one point and asks for four responses, each of which is worth 0.25 points. If you give five responses, each of your responses will be worth 0.20 points. If one of your five responses is incorrect, 0.20 will be deducted and your total credit for that question will be 0.80 instead of 1.00 even though you have four correct answers.

14. If the intent of a question is unclear, ask questions of the examiner ONLY.

15. When turning in your examination, assemble the completed answer sheet with examination aids and any additional paper. In addition, turn in all scrap paper. Keep your copy of the examination.

16. To pass the examination, you must achieve an overall grade of 80% or greater and at least 70% in each category.

17. There is a time limit of four (4) hours for completion of the examination.

18. When you are done and have turned in your examination, leave the examination area as defined by the examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 5.01 (1.00)

Given the following conditions:

- A manual scram has been inserted on BOTH RPS channels
- All blue scram lights are lit
- All control rods have failed to insert

Which one of the following correctly explains why the control rods failed to insert?

a) Hydraulic lock in the scram discharge volume

b) Failure of the scram valves to open

c) Blocked scram air header

d) Failure on one (1) RPS channel

QUESTION 5.02 (1.00)

Which one of the following is correct concerning the Heat Capacity Level and Temperature Limits?

a) The Heat Capacity Level Limit can be complied with for all suppression poll levels by reducing reactor pressure.

b) The Heat Capacity Level Limit requires Emergency Depressurization be accomplished before the downcomer vents are uncovered.

c) The Heat Capacity Level Limit is less restrictive than the Heat Capacity Temperature Limit when suppression pool level is less than 146 inches.

d) The Heat Capacity Temperature Limit is allowed to be increased at suppression pool levels higher than 146 inches.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.03 (1.50)

In accordance with Path 3, one of the four (4) pressure control methods are alternate pressure control systems such as HPCI or RCIC. State the other three (3) pressure control methods in Path 3.

QUESTION 5.04 (0.50)

State the reason the EOPs may direct the concurrent execution of 34AB or 34AR procedures.

QUESTION 5.05 (1.50)

State three (3) methods for entry to the top of any flowchart.

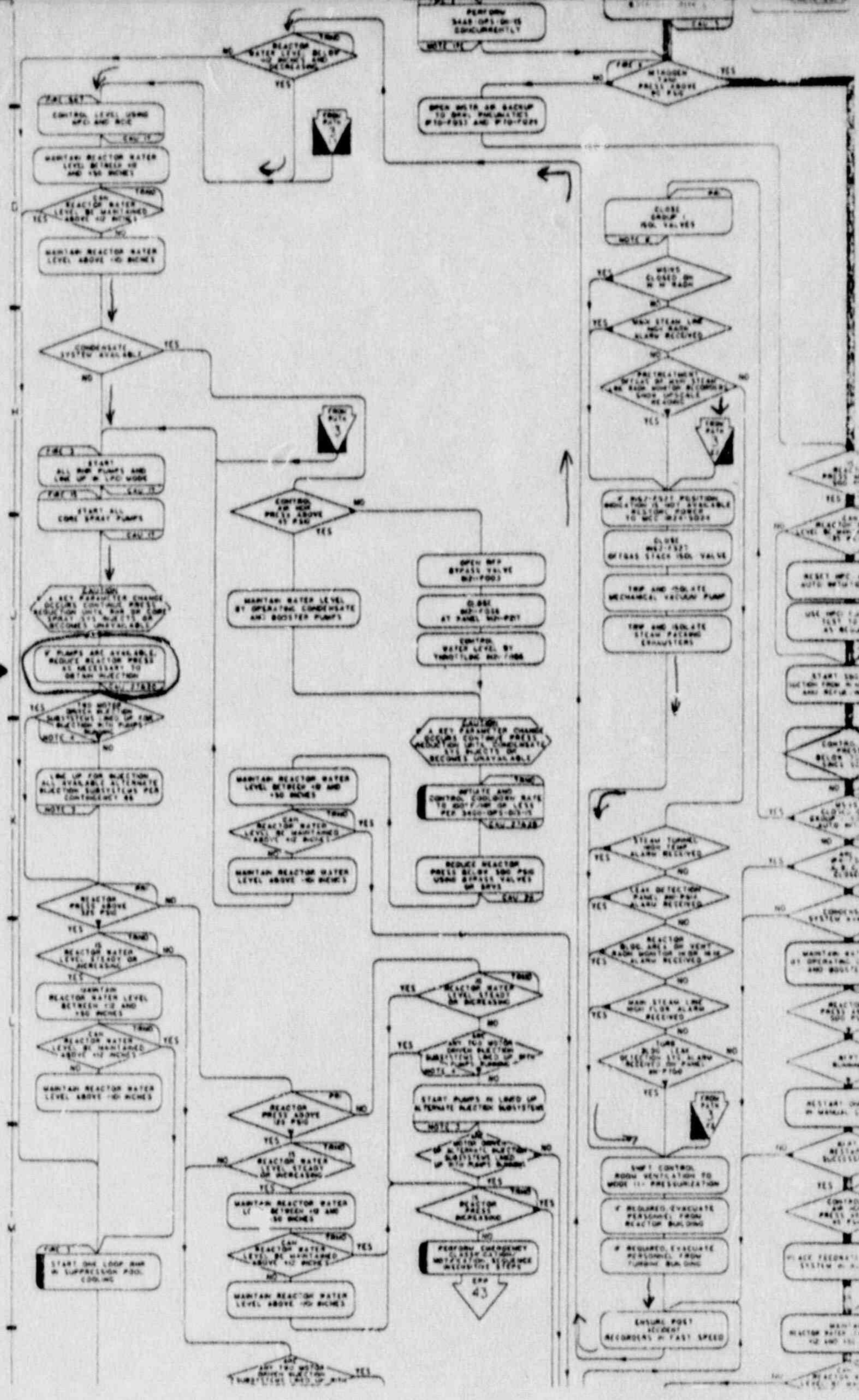
QUESTION 5.06 (~~1.00~~) 0.50

Given the following conditions:

- A LOCA is in progress on Unit 1
- All MSIVs have closed on High Radiation
- The Shift Supervisor has transitioned from Path 3 E2 to Path 4 H3

Referring to the attached Path 4 flowchart, state the ~~two (2) bases~~ basis for reducing reactor pressure at step J0.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)


CONDENSATE SYSTEM AVAILABLE
START ALL CORE SPRAY PUMPS
START PUMPS IN LINED UP ALTERNATE INJECTION SUBSYSTEM
EPP 43
CLOSE GROUP I ISOL VALVES
TRIP AND ISOLATE MECHANICAL VACUUM PUMP
TRIP AND ISOLATE STEAM PACKING EXHAUSTERS
ENSURE POST ACCIDENT RECORDERS IN FAST SPEED

0 1 2 3

QUESTION 5.07 (1.00)

In accordance with 31EO-EOP-001-2S, Emergency Operating Procedure Inside Control Room, RHR Service Water can be used as a source of makeup water to the reactor vessel. Which one of the following correctly describes how the RHR Service Water system would be aligned to inject?

a) Open the RHRSW cross tie valves from the control room

b) Open two manual valves in the RHR heat exchanger room

c) Connect a spool piece between RHRSW and the RHR system

d) Connect a fire hose from RHRSW to the RHR system

QUESTION 5.08 (1.50)

State the three (3) conditions which require a manual reactor scram in accordance with 34AB-OPS-020-2S, Loss of Instrument Air System

QUESTION 5.09 (1.00)

Which one of the following evolutions CANNOT be performed from the Remote Shutdown Panel?

a) Start a RHR Service Water pump

b) Start and inject Core Spray

c) Operate Safety Relief valves

d) Initiate Suppression Pool Cooling

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.10 (1.00)

Which one of the following is NOT an indication of jet pump failure?

a) Reduction in core flow

b) Core plate dp decreases

c) Recirculation loop flows are not equal

d) Individual jet pump differential pressures decrease

QUESTION 5.11 (1.00)

Given the following conditions:

- ADS has initiated on the 120 second ADS timer timing out
- Drywell pressure is 6 psig and reactor water level is -120
- Core spray and RHR pumps are running

Which one of the following actions will NOT result in closing of the ADS valves?

a) Secure RHR pumps

b) Reset the 120 second ADS timer

c) Place ADS inhibit switches in inhibit

d) Deenergize Auto Blowdown control power

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.12 (1.00)

Which one of the following is the approximate water level above TAF if reactor water level indicates -100 on R604A & R604B?

a) 75 inches

b) 65 inches

c) 50 inches

d) 35 inches

QUESTION 5.13 (1.00)

Which one of the following is the correct SLC tank level which corresponds to the HOT SHUTDOWN BORON WEIGHT?

a) 32%

b) 40%

c) 50%

d) 66%

QUESTION 5.14 (1.00)

Which one of the following is the MINIMUM number of SRVs required to be open for Emergency Depressurization to ensure peak clad temperature does not exceed 2200 F?

a) 1

b) 3

c) 5

d) 7

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.15 (1.00)

Given the following conditions:

- Reactor water level is -265
- Core cooling is by steam cooling through a single SRV.

Which one of the following is the MINIMUM reactor pressure which assures sufficient steam flow for adequate core cooling for the given conditions?

a) 500 psig

b) 600 psig

c) 700 psig

d) 800 psig

QUESTION 5.16 (1.00)

The reactor has been flooded due to a loss of all level indication following a reactor scram. Two (2) hours after the scram, level indication is restored.

Referring to Figure 2.12, which one of the following is the MAXIMUM amount of time injection can be terminated to bring water level on scale before injection must be recommenced?

a) 5 min

b) 10 min

c) 15 min

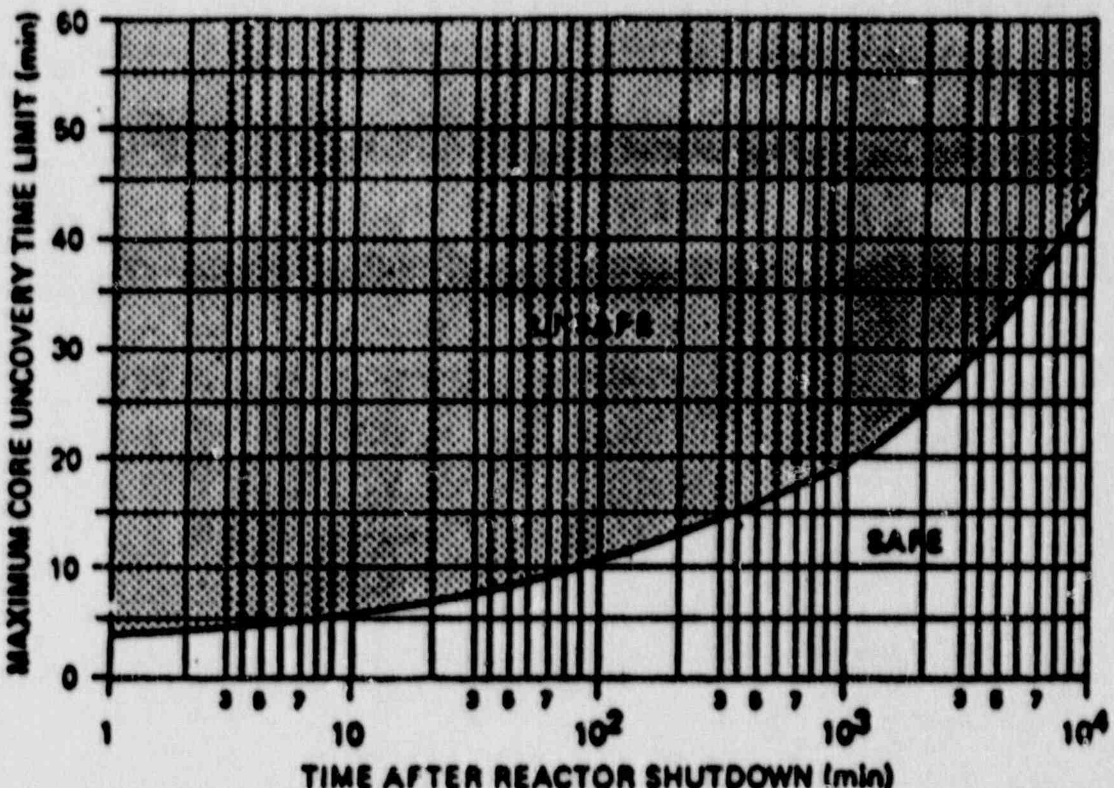
d) 50 min

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

MAXIMUM CORE UNCOVERY TIME LIMIT

TIME AFTER REACTOR SHUTDOWN	MAXIMUM ALLOWABLE CORE UNCOVERY TIME
[illegible] MIN.	3 MIN. 30 SEC.
5 MIN.	4 MIN. 49 SEC.
[illegible]0 MIN.	5 MIN. 33 SEC.
[illegible]5 MIN.	6 MIN. 6 SEC.
20 MIN.	6 MIN. 34 SEC.
30 MIN.	7 MIN. 25 SEC.
40 MIN.	8 MIN. 7 SEC.
50 MIN.	8 MIN. 45 SEC.
60 MIN.	9 MIN. 8 SEC.
80 MIN.	[illegible]0 MIN. 3 SEC.
[illegible]00 MIN.	[illegible]0 MIN. 58 SEC.
600 MIN.	[illegible]6 MIN. 54 SEC.
[illegible]000 MIN.	[illegible]9 MIN. 9 SEC.
3000 MIN.	27 MIN. 2 SEC.
6000 MIN.	35 MIN. 6 SEC.

NOTE: MAY USE SPDS EMERGENCY DISPLAYS IN PLACE OF THIS GRAPH



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MAXIMUM CORE UNCOVERY TIME LIMIT
UNIT 2

QUESTION 5.17 (1.00)

In accordance with the Emergency Operating Procedures, which one of the following would NOT allow water level reduction to be terminated during an ATWS?

a) Reactor power is less than 3%

b) Suppression pool temperature is less than 110 F

c) Drywell pressure is less than 1.85 psig

d) SRVs are open or cycling

QUESTION 5.18 (1.00)

Which one of the following is the injection rate that corresponds to approximately 8% power?

a) 900 gpm

b) 1700 gpm

c) 2000 gpm

d) 4250 gpm

QUESTION 5.19 (1.00)

Which one of the following is the reactor water level which requires closure of the MSIVs in accordance with the Emergency Operating Procedures?

a) +58 inches

b) +75 inches

c) +92 inches

d) +100 inches

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.20 (1.00)

Which one of the following correctly explains why lowering water level during an ATWS will reduce power?

a) Reduce natural circulation through the core

b) Reduce pressure in the core

c) Reduce differential pressure between downcomer and core

d) Reduce the subcooling of water entering the core

QUESTION 5.21 (1.00)

Which one of the following is the power supply to Unit 2 ARI system?

a) Essential Cabinet 2A

b) Essential Cabinet 2B

c) Diesel Generator Battery 2A

d) Diesel Generator Battery 2B

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.22 (1.00)

Which one of the following requires Emergency Depressurization in accordance with 34AB-EOP-049-2S, Radioactivity Release Control.

a) Release in excess of 1000 mr/hr with a primary system discharging outside the primary and secondary containments.

b) Release in excess of 1000 mr/hr with a primary system discharging into the primary and secondary containments.

c) Release in excess of 1000 mr/hr with a primary system discharging into the primary and secondary containment and one area exceeds Maximum Safe Operating Level.

d) Release which requires a General Emergency to be declared with a primary system discharging to secondary containment.

QUESTION 5.23 (1.00)

Given the following conditions:

- 60% power
- RBCCW Heat Exchanger outlet temperature increasing
- RBCCW Surge Tank level increasing
- RBCCW radiation level increasing
- All other plant parameters are normal

Which one of the following would correctly explain these indications ?

a) Service Water leak in the RBCCW Heat Exchanger.

b) Reactor Coolant leak in the NRHX.

c) RBCCW Surge Tank Level Controller failure.

d) RBCCW Heat Exchanger PSW Outlet valve closed.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.24 (1.00)

Which one of the following would result in a violation of a safety limit in accordance with Unit 1 Technical Specifications?

a) Water level decreases to -150 inches following a scram then is restored to +56.5 inches by RCIC.

b) Main turbine and both feed pumps trip at +58 inches while operating at 20% power and a scram occurs on low level.

c) EHC pressure regulator fails causing a reactor scram on MSIV closure while operating at 20% power.

d) A Group I isolation occurs on MSL High Radiation and a reactor scram occurs on MSIVs Not Full Open while operating at 100% power.

QUESTION 5.25 (1.00)

During an ATWS, several actions are taken to mitigate heat addition to the suppression pool. Which one of the following is correct concerning implementation of these actions?

a) If power is at 100% then immediate tripping of the recirculation pumps from their present speed is required to reduce heat added to the suppression pool.

b) If SLC initiation fails to inject boron, level reduction should be delayed until alternate boron injection is started.

c) Boron injection is required prior to tripping the recirculation pumps when greater than 30% power to enhance the dispersal of boron through the core and expedite reactor shutdown.

d) Level reduction is commenced if power is greater than 3%, suppression pool temperature is greater than 110 F and an SRV is cycling.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.26 (1.00)

Which one of the following is the cause for the loss of condenser vacuum during a loss of Instrument and Service air?

a) Reduced steam condensation

b) An increase in Hotwell level

c) Buildup of noncondensible gases

d) Loss of turbine gland seals

QUESTION 5.27 (1.00)

Which one of the following correctly describes the operation of Unit 1 RCIC from the Remote Shutdown Panel?

a) All automatic and manual turbine trips are operable.

b) The automatic isolation features are operable.

c) The Steam Supply Valve F045 High Reactor Water Level closure is operable.

d) The sequencing of the steam supply valves to the turbine is operable.

QUESTION 5.28 (1.00)

Which one of the following requires a manual scram in accordance with 34AB-OPS-040-2S, Loss of CRD?

a) 2 accumulator trouble lights are received.

b) Neither CRD pump can be immediately restarted.

c) Any CRD temperature exceeds 400 F.

d) 9 control rods exceed 250 F.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.29 (1.00)

Given the following conditions:

- Unit 2 is at 80% power
- Vital AC Power is supplied from ESS 600V Bus 2C.

Which one of the following is correct concerning the unrecoverable loss of ESS 600V Bus 2C?

a) Automatically transfer to the battery through the Static Inverter.

b) Manually transferred to the Static Inverter by depressing the "Alternate" pushbutton.

c) The reactor must be manually scrammed.

d) The recirculation pumps must be manually reduced to minimum speed.

QUESTION 5.30 (1.00)

In accordance with SOFI PATH 1, reactor power is to be maintained greater than 8% during an ATWS when RPV flooding is required and boron injection has commenced.

Which one of the following correctly describes the reason for maintaining reactor power greater than 8%?

a) Provides adequate steam flow to adequately cool uncovered portions of the core.

b) Corresponds to a level above TAF where natural circulation stops.

c) Allows higher injection flow rates to be achieved without causing large power excursions.

d) Ensures that APRMs are sufficiently on scale to yield an accurate power level indication.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.31 (1.00)

Given the following conditions:

- Unit 2 at 80% Power
- Unit 2 in control of DG 1B
- DG 1B surveillance in progress
- DG 1B at rated speed and voltage ready to be synchronized.

Which one of the following statements is correct if SUT 2D deenergizes ?

a) 4160V Bus 2F will automatically be energized by DG 1B twelve seconds after the loss of SUT 2D.

b) 4160V Bus 2F cannot be energized by DG 1B in the "Test" mode given these conditions.

c) 4160V Bus 2F will be energized from SUT 2C when the operator takes the alternate supply breaker control switch to the closed position.

d) 4160V Bus 2F will be energized when the operator resets the LOSP Lockout Relay, turns on the synchroscope, and takes the DG 1B output breaker control switch to the closed position.

QUESTION 5.32 (1.00)

Which one of the following correctly describes a consequence of operating in the UNSAFE region of the Heat Capacity Level Limit upon ADS actuation ?

a) Result in containment failure due to excessive temperature and unstable steam condensation.

b) Cause excessive hydrodynamic stress on the Torus.

c) Result in inadequate core cooling due to insufficient driving head through the SRVs.

d) Result in containment failure due to exceeding the Primary Containment Pressure Limit.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.33 (1.00)

Which one of the following conditions requires a manual scram in accordance with 34-AB-OPS-018-0S, Loss of Secondary Containment Integrity and Secondary Containment Control?

a) Any area temperature, differential temperature, radiation level, or floor drain sump/area water level exceeds maximum normal level.

b) A primary system is discharging into an area and cannot be isolated and prior to the maximum safe operating level being reached.

c) A primary system is discharging into an area and cannot be isolated, and prior to any maximum normal level being exceeded.

d) Any area temperature, differential temperature, or radiation level exceeds 3 times its maximum normal level.

QUESTION 5.34 (1.00)

Which one of the following conditions requires a manual scram in accordance with 34-AB-OPS-058-2S, Reactor Power Instabilities?

a) Core Flow is at 43% and reactor power is just above the 80% Load Line.

b) Core Flow is at 38% and reactor power is just below the 100% Nominal Flow Control Line.

c) Bandwidth oscillations of 11% peak to peak exist on APRM A.

d) Bandwidth oscillations of 7% peak to peak exist on three LPRMs.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.35 (1.00)

Given the following conditions:

- Unit 2 at 90% power
- An operator opens a Bypass Valve using the Bypass Jack

Which one of the following statements correctly describes plant response?

a) Reactor power remains constant and generator load decreases.

b) Reactor power decreases and generator load remains constant.

c) Reactor power increases and generator load increases.

d) Reactor power decreases and generator load decreases.

QUESTION 5.36 (2.00)

State ALL immediate operator actions PRIOR to evacuating the Control Room in accordance with 34AB-OPS-055-2S, Control Room Evacuation - Unit Shut Down.

QUESTION 5.37 (1.00)

Which one of the following correctly describes the safety significance of a failed jet pump?

a) Invalid APRM Flow-biased scram setpoints

b) Increased blowdown area

c) Unbalanced power distribution

d) Loose parts damage to the core

QUESTION 5.38 (1.00)

Which one of the following Off Gas System hydrogen concentrations is the minimum percentage requiring entry into a Limiting Condition for Operation?

a) 2%

b) 3%

c) 4%

d) 5%

QUESTION 5.39 (1.00)

In accordance with 34AB-OPS-024-2S, RPIS FAILURE, which one of the following is the correct action for a loss of RPIS?

a) A manual reactor scram

b) Commence a normal reactor shutdown

c) Commence a fast reactor shutdown

d) Suspend all rod motion

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.40 (1.00)

Given the following conditions:

- 10/09/89 Time: 1600
- Unit 2 at 100% power
- DG 2A has been declared Inoperable for 14 hours
- Technical Specification 3.8.1.1 Action [b] being performed

During performance of DG 1B surveillance, the Speed Setting knob was broken. Present setting of the speed control is unknown.

Which one of the following statements is correct ?

a) Start DG 2C within 1 hour and gradually load it to approximately 1850 kw for greater than 1 hour.

b) Restore three diesel generators to OPERABLE by 10/12/89 Time: 1600 or be in HOT SHUTDOWN within the next 12 hours.

c) Tech Spec 3.8.1.1 Action [b] applies. DG 1B is operable.

d) Restore DG 2A and 1B to OPERABLE by 10/12/89 Time: 0200 or be in HOT SHUTDOWN within the next 12 hours.

QUESTION 5.41 (1.00)

Which one of the following would NOT result in entry into the Primary Containment Control Guidelines?

a) Drywell temperature of 140 F

b) Suppression Pool level of 144 inches

c) Drywell pressure of 1.65 psig

d) Suppression pool temperature of 105 F

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.42 (2.00)

State four (4) responsibilities the Emergency Director CANNOT delegate during implementation of 10AC-MGR-006-OS, Hatch Emergency Plan.

QUESTION 5.43 (1.00)

~~Which one of the following is NOT a positive indication that SLC solution is being injected?~~

~~a) Pump running light ON~~

~~b) Pump discharge pressure INCREASING~~

~~c) SLC storage tank level DECREASING~~

~~d) Neutron flux level DECREASING~~

Deleted

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.44 (1.00)

Given the following conditions:

- Unit 2 at 80% power
- POWDEX SYSTEM TROUBLE alarm lit
- CONTAMINATED FEEDWATER alarm lit
- Feedwater conductivity 2.1 umhos/cm
- CST level increasing

Which one of the following is the correct immediate action in accordance with 34-AB-OPS-028-2N, Condenser Tube Leaks, for the given conditions?

a) Reduce load as necessary until the affected Hotwell Water Box can be isolated.

b) Conduct a Fast Reactor Shutdown in accordance with 34-GO-OPS-014-2S, Fast Reactor Shutdown.

c) Scram the reactor and trip the reactor feed pumps, condensate booster pumps and condensate pumps.

d) Reduce power below 30% and be in HOT STANDBY within 24 hours if the leak cannot be isolated.

QUESTION 5.45 (1.75)

Match each of the items in COLUMN B with those items in COLUMN A to which they apply. Each item in COLUMN B may be used more than once.

COLUMN A	COLUMN B
A. Technical Support Center	1) Primary on-site communications center during emergencies.
B. Operations Support Center	2) Placed in standby in an ALERT
C. Emergency Operations Facility	3) Provides Fire Brigade personnel when needed.
	4) Provides Dose Assessment functions
	5) Activated during an ALERT
	6) Provides direction for all on-site activities in a GENERAL EMERGENCY.

(***** END OF CATEGORY 5 *****)

QUESTION 6.01 (1.00)

Given the following conditions:

- Unit 2 startup in progress
- All plant parameters are normal
- RWM Group 12 control rods being withdrawn
- A RWM Notch Error alarm is received

Which one of the following failures would cause the RWM Notch Error alarm?

a) RBM

b) RSCS

c) RWM

d) RMCS

QUESTION 6.02 (1.00)

Which one of the following correctly describes RSCS rod position input between 100% and 50% rod densities?

a) Rod Movement control switch and settle bus

b) Full-in and full-out reed switches

c) Position 00 and 48 reed switches

d) Plant computer

QUESTION 6.03 (1.00)

The following annunciator is received in the control room: "SBLC LOSS OF CONTINUITY TO SQUIB VALVE". State two (2) unique control room indications available to verify this alarm.

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.04 (1.00)

Which one of the following electrical buses supplies power to the Standby Liquid Control system squib valves?

a) R24-S011, R24-S012

b) R24-S009, R24-S010

c) R24-S041, R24-S042

d) R24-S016, R24-S017

QUESTION 6.05 (1.00)

Which one of the following is the correct fire pump discharge header pressure which will cause automatic starting of the electric fire pump?

a) 110 psig

b) 100 psig

c) 90 psig

d) 80 psig

QUESTION 6.06 (1.00)

RCIC is operating a rated flow during a full flow test surveillance when a spurious RCIC initiation signal is received. State ALL RCIC valves which realign and their position.

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.07 (1.00)

Which one of the following is the Rod Worth Minimizer designed to minimize the consequences of?

a) Control rod drop

b) Continuous control rod withdrawal error

c) Continuous control rod insert error

d) Control rod sequence error

QUESTION 6.08 (1.00)

The Rod Block Monitor system designed to prevent fuel damage during high level power operation for a specific event. Which one of the following is this event?

a) Control rod drop

b) Continuous control rod withdrawal error

c) Continuous control rod insert error

d) Control rod sequence error

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.09 (1.00)

~~Given the following conditions:~~

~~- Unit 1 is operating at 40% RTP~~
~~- APRM 'A' fails DOWNSCALE~~
~~- APRM 'A' has NOT been bypassed~~

Deleted

~~Which one of the following correctly describes the response of RBM A for the given conditions?~~

~~a) Generates a rod withdrawal block~~

~~b) Automatically shifts to an alternate reference APRM~~

~~c) Generates a rod select block~~

~~d) Automatically bypasses rod block outputs~~

QUESTION 6.10 (1.00)

The reference leg for "B" emergency range level (D003) ruptures and is isolated by the excess flow check valve.

Which one of the following is correctly describes the indication of "B" emergency range level and narrow range reactor pressure to this event?

a) "B" emergency range level upscale; narrow range reactor pressure upscale

b) "B" emergency range level downscale; narrow range reactor pressure upscale

c) "B" emergency range level upscale; narrow range reactor pressure downscale

d) "B" emergency range level downscale; narrow range reactor pressure downscale

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.11 (1.00)

Which one of the following authorizes bypassing of the Rod Worth Minimizer?

a) Senior Nuclear Engineer

b) Operations Supervisor

c) Shift Technical Advisor

d) Any SRO licensed individual

QUESTION 6.12 (1.00)

HPCI is in a normal standby alignment when both divisions of Leak Detection associated with HPCI are deenergized. Which one of the following correctly describes the HPCI valves which will automatically close?

a) F042, F002

b) F002, F003

c) F042, F041

d) F059, F003

QUESTION 6.13 (1.00)

The HPCI system has an interlock to prevent draining the suppression pool to the CST. Which one of the following statements correctly describes this interlock?

a) F041 closes if F004 is opened

b) F042 closes if F004 is opened

c) F004 closes if F042 or F041 are opened

d) F004 closes if F042 and F041 are opened

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.14 (1.00)

Which one of the following is the MAXIMUM allowable internal design pressure of the primary containment system?

a) 56 psig

b) 62 psig

c) 66 psig

d) 72 psig

QUESTION 6.15 (1.50)

Match the system response in column A with the corresponding Reactor Water Level setpoint in Column B. Each response may be used more than once.

COLUMN A	COLUMN B
A. RECIRC RUNBACK DURING RFP LOSS	1. 47
B. STANDBY GAS STARTS	2. 32
C. MSIVs CLOSE	3. 12
D. DRYWELL PNEUMATICS ISOLATE	4. 10
E. RHR SHUTDOWN COOLING ISOLATES	5. -47
F. ADS 13 MIN TIMER STARTS	6. -75
	7. -113

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.16 (1.00)

Which one of the following is the correct condenser vacuum setpoint which will cause the main turbine bypass valves to trip close ?

a) 5.0 inches Hg

b) 7.0 inches Hg

c) 13.0 inches Hg

d) 22.3 inches Hg

QUESTION 6.17 (1.00)

Which one of the following is the generator load limitation for single reactor feed pump operation in accordance with 34GO-OPS-005-2, Power Changes?

a) 55%

b) 60%

c) 65%

d) 70%

QUESTION 6.18 (1.00)

Which one of the following conditions will cause a control rod to drift?

a) Failed open CRD flow control valve

b) Excessive cooling water pressure

c) Low scram accumulator pressure

d) Failed open stabilizing valves

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.19 (1.00)

Which one of the following is the purpose of the EOC-RPT?

a) Provide a redundant means of reducing power during ATWS

b) Improve thermal margins during turbine trip/load reject

c) Prevent exceeding thermal limits during recirc flow controller failure

d) Prevent exceeding thermal limits during loss of feedwater heating

QUESTION 6.20 (1.00)

The "GENERATOR PROTECTION CIRCUIT ENERGIZED" annunciator is received in the control room.

Which one of the following correctly describes the unit condition indicated by this annunciator?

a) Main Generator will trip on reverse power in 30 sec

b) Main Generator will trip in 2 min unless stator water is restored

c) Main Generator will trip in 3.5 min unless stator amps are reduced to less than 4500 amps

d) Main Generator field breaker has tripped

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.21 (1.00)

The output of the EHC pressure regulator fails HIGH causing a bypass valve to open.

Which one of the following will cause the bypass valve to close?

a) Decrease Bypass Jack

b) Decrease Load Limit Pot

c) Decrease Pressure Setpoint

d) Decrease Maximum Combined Flow Pot

QUESTION 6.22 (1.50)

The unit is at 100% power with feedwater in automatic three element control. Match the Instrument Failure in Column A with the expected plant response in Column B. The responses in Column B may be used more than once.

COLUMN A	COLUMN B
A. Feed flow detector fails low	1. Reactor level increases to turbine trip
B. Feed flow detector fails high	2. Reactor level decreases below scram setpoint
C. Steam flow detector fails low	3. Reactor Level stabilizes 6"- 10" below normal
D. Steam flow detector fails high	4. Reactor level stabilizes 6" - 10" above normal
E. Selected level transmitter fails low	
F. Selected level transmitter fails high	

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.23 (1.00)

Which one of the following corresponds to the MAXIMUM acceptable time for the emergency diesels to start and come to rated speed and voltage?

a) 8 sec

b) 10 sec

c) 12 sec

d) 15 sec

QUESTION 6.24 (1.00)

The emergency diesel generators have started on high drywell pressure and loss of power and have tied to their respective buses.

Which one of the following would cause a diesel generator to trip for these conditions?

a) Generator Reverse Power

b) Crankcase High pressure

c) Jacket water pressure low

d) Engine Overspeed

QUESTION 6.25 (1.00)

Which one of the following is the ALTERNATE power supply to the Unit 2 control room NSSS annunciators?

a) 125 vdc Bus 2A

b) 125 vdc Bus 2B

c) Instrument Bus 2A

d) Instrument Bus 2B

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.26 (1.00)

The following annunciator is received in the control room: "SPEED CONTROL A SIGNAL FAILURE".

Which one of the following correctly describes the response of Recirc MG A for this alarm?

a) Drive Motor breaker trips

b) Generator Field breaker trips

c) Runback to Speed Limiter # 2

d) Scoop tube lock up

QUESTION 6.27 (1.00)

Given the following conditions:

- Unit 1 at 100% RTP
- "B" RWCU in operation and "A" RWCU pump suction is isolated
- "B" RWCU filter/demineralizer and pump are taken out of service
- "B" RWCU pump suction isolation valve is CLOSED
- "B" RWCU pump is allowed to cool to less than 130 F
- "A" RWCU pump suction is unisolated
- A Group 5 PCIS isolation occurs

Which one of the following is the cause for the Group 5 PCIS isolation?

a) High NRHX outlet temperature

b) High RWCU room temperature

c) Low reactor water level

d) High differential flow

QUESTION 6.28 (1.00)

Which one of the following conditions will NOT result in a shutdown of the Standby Gas Treatment System with a valid initiation signal present ?

a) High temperature 225 degrees F in the Charcoal Bed.

b) Overloads in the Local Control Panel.

c) High temperature 225 degrees F at the Heater outlet.

d) High temperature 190 Degrees F at the Heater inlet.

QUESTION 6.29 (1.00)

Which one of the following is NOT required for PRIMARY CONTAINMENT INTEGRITY in accordance with Unit 2 Technical Specifications?

a) All equipment hatches closed and sealed

b) Reactor coolant leakage rates within limits

c) SBGT system is OPERABLE

d) Containment airlock is OPERABLE

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.30 (1.00)

Given the following conditions:

- Unit 2 at 100% power
- APRM 'A' in BYPASS
- IRM 'C' is placed in BYPASS

Which one of the following statements correctly describes the Technical Specification requirements for the given conditions?

a) Technical Specification 3.3.1.a

b) Technical Specification 3.3.1.b. ACTION 3

c) Technical Specification 3.3.5.b

d) Technical Specification 3.0.3

QUESTION 6.31 (1.00)

Given the following conditions:

- A fire alarm sounds over the PA system
- The shift fire brigade has donned turnout gear and SCBA

Which one of the following correctly describes the next action the shift fire brigade should take?

a) Report to the Fire Brigade Leader

b) Report to the Shift Supervisor

c) Report to the Fire Equipment Building

d) Report to the Security Building

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.32 (1.00)

A 26 year old operator will be performing work in a 200 mrem radiation field. His lifetime exposure is 10 rem and his current quarter dose is .75 rem. He has a current NRC form 4 on file and has received authorization to exceed administrative limits. Which one of the following is the maximum number of hours this individual can work in this area?

a) 3.25

b) 8.25

c) 11.25

d) 13.25

QUESTION 6.33 (1.50)

a. State the quarterly radiation exposure limit for each of the following in accordance with 10CFR20:

1) Whole body

2) Hands and forearms, feet and ankles

3) Skin of the whole body.

b. The limit for whole body dose may be exceeded if specific conditions are complied with. Answer the following concerning the maximum limits allowed:

1) State the maximum quarterly whole body limit.

2) State ALL conditions that must be met.

(***** END OF CATEGORY 6 *****)
(********** END OF EXAMINATION **********)

ANSWER 5.01 (1.00)

a)

REFERENCE

LT-IH-20107-03 Objective 7

295015K201 ..(KA's)

ANSWER 5.02 (1.00)

b)

REFERENCE

LT-IH-20113-00 Objective 1,2

295026K301 295026K206 295026K102 ..(KA's)

ANSWER 5.03 (1.50)

[0.25] each:
1. Lo-Lo Set
2. Turbine BPVs
3. SRVs

REFERENCE

LT-IH-20104-00 Objective 10

295025A102 295025A103 ..(KA's)

ANSWER 5.04 (0.50)

To address problems with support systems (prevent equipment damage)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-20101-02 Objective 11

295024G012 ..(KA's)

ANSWER 5.05 (1.50)

[0.50] each:
1. Any scram
2. Key parameter change
3. Directed by EPM (End Path Manual) (Reactivity Control Procedure)

REFERENCE

LT-IH-20102-03 Objective 7
LT-IH-20101-02 Objective 8

295024KG01 ..(KA's)

ANSWER 5.06 (~~1.00~~) 0.50

[0.50] each:
1. conserve (restore) RPV inventory
~~2. limit radioactive release~~

REFERENCE

LT-IH-20104-03 Objective 25
LT-IH-20105-03 Objective 25
LT-IH-20114-00 Objective 1

295031G012 295038G012 295024G012 ..(KA's)

ANSWER 5.07 (1.00)

a)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-03401-00 Objective 13d

295031A108 ..(KA's)

ANSWER 5.08 (1.50)

[0.50] each:
1. Four or more control rods drift in

2. "SCRAM VALVE PILOT HDR HI/LO PRESS" alarm coincident with "CRD HYD HIGH TEMP" alarm

3. scram pilot air header less than 50 psig (on the local indicator)

REFERENCE

34AB-OPS-020-2S
LT-ST-03501-00 Objective 16

295019G010 ..(KA's)

ANSWER 5.09 (1.00)

b)

REFERENCE

LT-ST-05201-00 Objective3,4

295016G006 ..(KA's)

ANSWER 5.10 (1.00)

b)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-00401-00 B.7.c.

295001A205 202001A201 295001K306 ..(KA's)

ANSWER 5.11 (1.00)

a)

REFERENCE

LT-ST-03801-00 Objective12

218000K403 295024K208 295024A108 ..(KA's)

ANSWER 5.12 (1.00)

b)

REFERENCE

LT-IH-20012-01 Objective 1

295031A201 ..(KA's)

ANSWER 5.13 (1.00)

a)

REFERENCE

LT-IH-20113-00 Objective2

295037A203 ..(KA's)

ANSWER 5.14 (1.00)

b)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-20113-00 Objective2

295031G007 ..(KA's)

ANSWER 5.15 (1.00)

C

REFERENCE

LT-IH-20113-00 Objective 2

295031G007 ..(KA's)

ANSWER 5.16 (1.00)

b)

REFERENCE

LT-IH-20113-00 Objective2

295031G012 ..(KA's)

ANSWER 5.17 (1.00)

d)

REFERENCE

LT-IH-20107-03 Objective 16

295037A202 295037A204 295037A201 ..(KA's)

ANSWER 5.18 (1.00)

b)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-2Ø114-ØØ Objective 2

295Ø37A2Ø2 295Ø37A2Ø4 295Ø37A2Ø1 ..(KA's)

ANSWER 5.19 (1.ØØ)

d)

REFERENCE

LT-IH-2Ø114-ØØ Objective 2

295ØØ8A1Ø3 ..(KA's)

ANSWER 5.2Ø (1.ØØ)

~~b)~~ a)

REFERENCE

LT-IH-2Ø113-ØØ Objective 1,2

295Ø37K1Ø2 295Ø37K2Ø9 295Ø37K3Ø3 ..(KA's)

ANSWER 5.21 (1.ØØ)

c)

REFERENCE

LT-IH-ØØ1Ø1-ØØ Change Notice 89-ØØ7Ø Objective 28

2Ø1ØØ1K2Ø5 295ØØ4K3Ø3 ..(KA's)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

ANSWER 5.22 (1.00)

a

REFERENCE

34-AB-EOP-049-06-02
LT-IH-20115-00 Objective 6

295038K304 295033G010 ..(KA's)

ANSWER 5.23 (1.00)

b)

REFERENCE

LT-IH-00901, Objective 19
34SO-P42-001-2S

295018G011 ..(KA's)

ANSWER 5.24 (1.00)

d)

REFERENCE

TS 3.7.A.1 Amendment 165

295037G012 295037K301 ..(KA's)

ANSWER 5.25 (1.00)

d)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

SOFI Flowchart 1: Content and Use
LT-IH-20107-03 Objective 11,15,16

295037G012 295037K301 ..(KA's)

ANSWER 5.26 (1.00)

c)

REFERENCE

34AB-OPS-020-2S
LT-IH-02501-00 Objective 13.b, 18.c

271000K107 295002K306 ..(KA's)

ANSWER 5.27 (1.00)

a

REFERENCE

LT-IH-03901-00 Objective 12

217000K404 295016A107 ..(KA's)

ANSWER 5.28 (1.00)

d)

REFERENCE

34-AB-OPS-040-2S

295022K301 295022G011 ..(KA's)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

ANSWER 5.29 (1.00)

c)

REFERENCE

34-AB-OPS-15-2S
LT-IH-02703 Objective 21

295003G010 295003K203 295003K301 295003K305 ..(KA's)

ANSWER 5.30 (1.00)

b)

REFERENCE

LT-IH-20107-03

295037A202 295037G007 295037K303 ..(KA's)

ANSWER 5.31 (1.00)

b)

REFERENCE

LT-IH-00742-05 Objective 8
LT-IH-02702-00

295003A102 295003K202 ..(KA's)

ANSWER 5.32 (1.00)

a)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-20113-00 Objective 1

295029G007 295029K301 295029K101 ..(KA's)

ANSWER 5.33 (1.00)

b)

REFERENCE

34AB-EOP-018-0S Objective 10

295033G010 295033K302 ..(KA's)

ANSWER 5.34 (1.00)

c)

REFERENCE

LT-IH-20201-01 Objective 19

295006G010 ..(KA's)

ANSWER 5.35 (1.00)

a)

REFERENCE

LT-IH-01901-00 Objective 20

241000K101 295000K102 ..(KA's)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

ANSWER 5.36 (2.00)

[0.50] each:
Close: (Inboard and Outboard) MSIVs
Steamline Drains (Inboard and Outboard) Isolation Valves
Reactor Water Sample (Inboard and Outboard) Isolation Valves

Open : RFP Bypass Valve [2N21-F113]

REFERENCE

34AB-OPS-055-2S

295016G010 ..(KA's)

ANSWER 5.37 (1.00)

b)

REFERENCE

LT-IH-00401-00 Objective 25

202001A201 202001K601 202001K106 ..(KA's)

ANSWER 5.38 (1.00)

c)

REFERENCE

TS 3/4 3.11.2.6

271000G005 ..(KA's)

ANSWER 5.39 (1.00)

d)

REFERENCE

34AB-OPS-24-2S

214000G014 ..(KA's)

ANSWER 5.40 (1.00)

a

REFERENCE

LT-IH-02801 Objective 6
UNIT 2 TS 3.8.1.1

264000G011 ..(KA's)

ANSWER 5.41 (1.00)

c)

REFERENCE

LT-IH-20111-06-01
31EO-EOP-001

264000G011 ..(KA's)

ANSWER 5.42 (2.00)

Four ~~Six~~ of the following at (0.50) each:
1. Decision to notify offsite emergency response agencies
2. Decision to recommend protective actions to offsite agencies
3. Declaration of emergency classification
4. Authorization for plant personnel to exceed 10CFR20 radiation exposure limits
5. Authorization for use of KI during a declared emergency
6. Decision to downgrade an emergency classification
7. Decision to terminate the emergency classification
8. DEcision to request Federal assistance
9. DEcision to order evacuation of nonessential personnel from the site at an Alert or higher classification

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

10AC-MGR-006-OS

264000G011 ..(KA's)

~~ANSWER 5.43 (1.00)~~

~~a)~~

~~REFERENCE~~ Deleted Rebuttal

~~LT-IH-01101 Objective 16.b.~~

ANSWER 5.44 (1.00)

c)

REFERENCE

34-AB-OPS-028-2N

256000A215 256000G014 256000G015 ..(KA's)

ANSWER 5.45 (1.75)

A. 1,5,6
B. 3,5
C. 2,4 [0.25 each]

REFERENCE

10AC-MGR-006-0S

294001A116 ..(KA's)

(***** END OF CATEGORY 5 *****)

ANSWER 6.01 (1.00)

b)

REFERENCE

LT-IH-050402-02 Objective 9

201004K406 201004A305 ..(KA's)

ANSWER 6.02 (1.00)

b)

REFERENCE

LT-IH-050402-02 Objective 6

201004K405 201004K103 ..(KA's)

ANSWER 6.03 (1.00)

[0.50] each:
1. amber squib continuity lights
2. continuity meters

REFERENCE

LT-ST-01101-00 Objective 14

211000K404 ..(KA's)

ANSWER 6.04 (1.00)

a)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-ST-01101-00 Objective6a

211000K202 ..(KA's)

ANSWER 6.05 (1.00)

a)

REFERENCE

LT-IH-03601-01 Objective5

286000K403 ..(KA's)

ANSWER 6.06 (1.00)

F013, Inboard Discharge (.25) OPENS (.25)
F022, CST Test Isolation (.25) CLOSES (.25)

REFERENCE

LT-IH-03901-00 Objective 9.b.

217000A201 ..(KA's)

ANSWER 6.07 (1.00)

a)

REFERENCE

LT-IH-05403-00 Objective 1
TS BASES 3.3.6.1

201006G004 ..(KA's)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

ANSWER 6.08 (1.00)

b)

REFERENCE

TS BASES Pg # 3.3-14
LT-IH-05403-00 Objective 1

215002G004 ..(KA's)

~~ANSWER 6.09 (1.00)~~

~~d)~~

~~REFERENCE~~

Deleted

~~LT-IH-01203-00 Objective 15~~

~~215002A203 ..(KA's)~~

ANSWER 6.10 (1.00)

c)

REFERENCE

LT-IH-04404-00 Objective 15

216000K324 216000K325 ..(KA's)

ANSWER 6.11 (1.00)

b)

REFERENCE

LT-IH-05403-00 Objective 8

201006G001 ..(KA's)

ANSWER 6.12 (1.00)

b)

REFERENCE

LT-ST-00501-00 Oojective 21c

206000K610 ..(KA's)

ANSWER 6.13 (1.00)

d)

REFERENCE

LT-ST-005001-02

206000K417 ..(KA's)

ANSWER 6.14 (1.00)

b)

REFERENCE

LT-IH-01301-0C Objective 1

223001G005 ..(KA's)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

ANSWER 6.15 (1.50)

[0.25] each:
A. 2
B. 5
C. 7
D. 4
E. 4
F. 7

REFERENCE

LT-IH-01301-00 Objective 6a
LT-IH-04404-00 Objective 1

216000K102	218000K103	239001K401	259001K109	212000K102
261000A210	205000K102	202001K115	217000K102	..(KA's)

ANSWER 6.16 (1.00)

b)

REFERENCE

LT-IH-02501-00 L012

241000K605 ..(KA's)

ANSWER 6.17 (1.00)

a)

REFERENCE

LT-IH-00201-00 Objective 12

259001G010 ..(KA's)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

ANSWER 6.18 (1.00)

b)

REFERENCE

LT-IH-00401-00 Objective 12

201001G010 ..(KA's)

ANSWER 6.19 (1.00)

b)

REFERENCE

LT-IH-00401-00 Objective 12

202001K505 ..(KA's)

ANSWER 6.20 (1.00)

c)

REFERENCE

LT-ST-02301-00 Objective 14

245000K605 ..(KA's)

ANSWER 6.21 (1.00)

d)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-01901-00 Objective 8

241000A114 241000A115 ..(KA's)

ANSWER 6.22 (1.50)

[0.25] each:
A. 1
B. 2
C. 3
D. 4
E. ~~2~~ 1
F ~~1~~ 2

REFERENCE

LT-IH-00202-00 Objective 19

259000A203 259002A201 259002A202 ..(KA's)

ANSWER 6.23 (1.00)

c)

REFERENCE

LT-IH-02801-00 Objective 15

264000A302 ..(KA's)

ANSWER 6.24 (1.00)

d)

REFERENCE

LT-IH-02801-00 Objective 1

264000K402 ..(KA's)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

ANSWER 6.25 (1.00)

a)

REFERENCE

34AB-OPS-053-2S

263000K201 ..(KA's)

ANSWER 6.26 (1.00)

d)

REFERENCE

LT-IH-00401-00 LO 17

2020001A20 ..(KA's)

ANSWER 6.27 (1.00)

d)

REFERENCE

LT-IH-00301-00 Objective 7e
LER 89-001

204000K404 ..(KA's)

ANSWER 6.28 (1.00)

d

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-03001-00 Objective 7c

288000A301 261000A205 ..(KA's)

ANSWER 6.29 (1.00)

c)

REFERENCE

Unit 2 TS Definitions

ANSWER 6.30 (1.00)

a)

REFERENCE

LER 88-001

215005G005 ..(KA's)

ANSWER 6.31 (1.00)

a)

REFERENCE

40AC-ENG-008-05

294001K116 ..(KA's)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

ANSWER 6.32 (1.00)

c)

REFERENCE

10CFR20
LT-IH-3008

294001K103 ..(KA's)

ANSWER 6.33 (1.50)

[0.25] each:
a. 1) 1.25 rem
   2) 18.75 rem
   3) 7.5 rem

b. 1) 3 rem
   2) Exposure must not exceed 5[N-18].
      NRC Form 4 completed or previous history known.

REFERENCE

RADIATION EXPOSURE LIMITS, 60AC-HPX-001-0S
10CFR20

294001K103 ..(KA's)

(***** END OF CATEGORY 6 *****)
(********** END OF EXAMINATION **********)

QUESTION	VALUE	REFERENCE
5.01	1.00	9000793
5.02	1.00	9000794
5.03	1.50	9000795
5.04	0.50	9000797
5.05	1.50	9000798
5.06	1.00	9000806
5.07	1.00	9000820
5.08	1.50	9000823
5.09	1.00	9000825
5.10	1.00	9000827
5.11	1.00	9000841
5.12	1.00	9000861
5.13	1.00	9000862
5.14	1.00	9000863
5.15	1.00	9000864
5.16	1.00	9000865
5.17	1.00	9000866
5.18	1.00	9000867
5.19	1.00	9000868
5.20	1.00	9000869
5.21	1.00	9000870
5.22	1.00	9000799
5.23	1.00	9000843
5.24	1.00	9000845
5.25	1.00	9000846
5.26	1.00	9000847
5.27	1.00	9000848
5.28	1.00	9000851
5.29	1.00	9000852
5.30	1.00	9000853
5.31	1.00	9000854
5.32	1.00	9000856
5.33	1.00	9000857
5.34	1.00	9000858
5.35	1.00	9000859
5.36	2.00	9000855
5.37	1.00	9000796
5.38	1.00	9000807
5.39	1.00	9000860
5.40	1.00	9000800
5.41	1.00	9000801
5.42	2.00	9000802
5.43	1.00	9000808
5.44	1.00	9000850
5.45	1.75	9000803

	48.75	
6.01	1.00	9000804
6.02	1.00	9000805
6.03	1.00	9000811
6.04	1.00	9000812
6.05	1.00	9000813
6.06	1.00	9000814
6.07	1.00	9000815

QUESTION	VALUE	REFERENCE
6.08	1.00	9000816
6.09	1.00	9000817
6.10	1.00	9000818
6.11	1.00	9000819
6.12	1.00	9000821
6.13	1.00	9000822
6.14	1.00	9000824
6.15	1.50	9000826
6.16	1.00	9000828
6.17	1.00	9000831
6.18	1.00	9000832
6.19	1.00	9000833
6.20	1.00	9000834
6.21	1.00	9000835
6.22	1.50	9000836
6.23	1.00	9000837
6.24	1.00	9000838
6.25	1.00	9000839
6.26	1.00	9000840
6.27	1.00	9000842
6.28	1.00	9000849
6.29	1.00	9000809
6.30	1.00	9000810
6.31	1.00	9000829
6.32	1.00	9000830
6.33	1.50	9000844

	34.50	

	83.25	

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Examination

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U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION
REGION 2

FACILITY: E. I. Hatch 1 & 2

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 89/10/09

CANDIDATE: ______________

INSTRUCTIONS TO CANDIDATE:

Use answer sheet for the answers and write on one side only. Use paper provided for continuation of answer. 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%. The examination will have a time limit of four (4) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE		CATEGORY
31.00 ~~32.50~~	39.74 ~~40.37~~	________	________	2.	EMERGENCY AND ABNORMAL PLANT EVOLUTIONS (40%)
47.00 ~~48.00~~	60.23 ~~59.63~~	________	________	3.	PLANT SYSTEMS AND PLANT-WIDE GENERIC RESPONSIBILITIES (60%)
78.00 ~~80.50~~		________ 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. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done AFTER you complete the examination.

3. 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.

4. Use black ink or dark pencil only to facilitate legible reproductions.

5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet.

6. If you need additional space to answer a specific question, use a separate sheet of the paper provided. DO NOT WRITE ON THE BACK SIDE OF THE EXAMINATION ANSWER SHEET.

7. Print your name in the upper right-hand corner of the answer sheet.

8. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.

9. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

10. The point value for each question is indicated in parentheses after the question. The amount of blank space on an examination question page is NOT an indication of the depth of answer required.

11. Show all calculations, methods, or assumptions used to obtain an answer.

12. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK. NOTE: partial credit will NOT be given on multiple choice questions.

13. Proportional grading will be applied. Any additional wrong information that is provided will count against you. For example, if a question is worth one point and asks for four responses, each of which is worth 0.25 points. If you give five responses, each of your responses will be worth 0.20 points. If one of your five responses is incorrect, 0.20 will be deducted and your total credit for that question will be 0.80 instead of 1.00 even though you have four correct answers.

14. If the intent of a question is unclear, ask questions of the examiner ONLY.

15. When turning in your examination, assemble the completed answer sheet with examination aids and any additional paper. In addition, turn in all scrap paper. Keep your copy of the examination.

16. To pass the examination, you must achieve an overall grade of 80% or greater and at least 70% in each category.

17. There is a time limit of four (4) hours for completion of the examination.

18. When you are done and have turned in your examination, leave the examination area as defined by the examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 2.01 (~~1.00~~) 0.50

Given the following conditions:

- A LOCA is in progress on Unit 1
- All MSIVs have closed on High Radiation
- The Shift Supervisor has transitioned from Path 3 E2 to Path 4 H3

Referring to the attached Path 4 flowchart, state the ~~two (2) bases~~ basis for reducing reactor pressure at step J0.

QUESTION 2.02 (1.00)

In accordance with 31EO-EOP-001-2S, Emergency Operating Procedure Inside Control Room, RHR Service Water can be used as a source of makeup water to the reactor vessel. Which one of the following correctly describes how the RHR Service Water system would be aligned to inject?

a) Open the RHRSW cross tie valves from the control room

b) Open two manual valves in the RHR heat exchanger room

c) Connect a spool piece between RHRSW and the RHR system

d) Connect a fire hose from RHRSW to the RHR system

QUESTION 2.03 (1.50)

State the three (3) conditions which require a manual reactor scram in accordance with 34AB-OPS-020-2S, Loss of Instrument Air System

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.04 (1.00)

Which one of the following evolutions CANNOT be performed from the Remote Shutdown Panel?

a) Start a RHR Service Water pump

b) Start and inject Core Spray

c) Operate Safety Relief valves

d) Initiate Suppression Pool Cooling

QUESTION 2.05 (1.00)

Which one of the following is NOT an indication of jet pump failure?

a) Reduction in core flow

b) Core plate dp decreases

c) Recirculation loop flows are not equal

d) Individual jet pump differential pressures decrease

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.06 (1.00)

Given the following conditions:

- ADS has initiated on the 120 second ADS timer timing out
- Drywell pressure is 6 psig and reactor water level is -120
- Core spray and RHR pumps are running

Which one of the following actions will NOT result in closing of the ADS valves?

a) Secure RHR pumps

b) Reset the 120 second ADS timer

c) Place ADS inhibit switches in inhibit

d) Deenergize Auto Blowdown control power

QUESTION 2.07 (1.00)

Which one of the following is the approximate water level above TAF if reactor water level indicates -100 on R604A & R604B?

a) 75 inches

b) 65 inches

c) 50 inches

d) 35 inches

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.08 (1.00)

Which one of the following is the correct SLC tank level which corresponds to the HOT SHUTDOWN BORON WEIGHT?

a) 32%

b) 40%

c) 50%

d) 66%

QUESTION 2.09 (1.00)

Which one of the following is the MINIMUM number of SRVs required to be open for Emergency Depressurization to ensure peak clad temperature does not exceed 2200 F?

a) 1

b) 3

c) 5

d) 7

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.10 (1.00)

Given the following conditions:

- Reactor water level is -265
- Core cooling is by steam cooling through a single SRV.

Which one of the following is the MINIMUM reactor pressure which assures sufficient steam flow for adequate core cooling for the given conditions?

a) 500 psig

b) 600 psig

c) 700 psig

d) 800 psig

QUESTION 2.11 (1.00)

The reactor has been flooded due to a loss of all level indication following a reactor scram. Two (2) hours after the scram, level indication is restored.

Referring to Figure 2.12, which one of the following is the MAXIMUM amount of time injection can be terminated to bring water level on scale before injection must be recommenced?

a) 5 min

b) 10 min

c) 15 min

d) 20 min

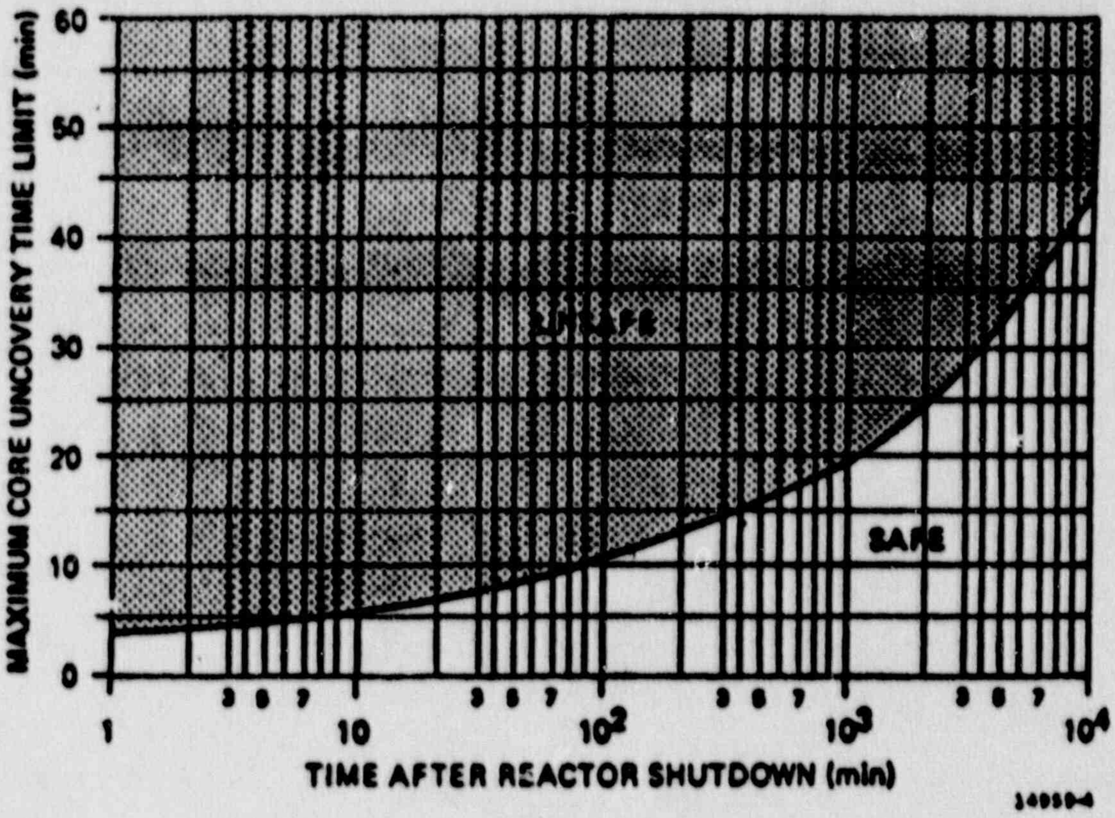
(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

[illegible]

MAXIMUM CORE UNCOVERY TIME LIMIT [illegible]

TIME AFTER REACTOR SHUTDOWN		MAXIMUM ALLOWABLE CORE UNCOVERY TIME			
[illegible]	MIN.	3	MIN.	30	SEC.
5	MIN.	4	MIN.	49	SEC.
0	MIN.	5	MIN.	33	SEC.
5	MIN.	6	MIN.	6	SEC.
20	MIN.	6	MIN.	34	SEC.
30	MIN.	7	MIN.	25	SEC.
40	MIN.	8	MIN.	[illegible]	SEC.
50	MIN.	8	MIN.	45	SEC.
60	MIN.	9	MIN.	8	SEC.
80	MIN.	0	MIN.	3	SEC.
00	MIN.	0	MIN.	50	SEC.
600	MIN.	6	MIN.	54	SEC.
000	MIN.	9	MIN.	9	SEC.
3000	MIN.	27	MIN.	2	SEC.
6000	MIN.	35	MIN.	6	SEC.

NOTE: MAY USE SPDS EMERGENCY DISPLAYS IN PLACE OF THIS GRAPH



MAXIMUM CORE UNCOVERY TIME LIMIT
UNIT 2

QUESTION 2.12 (1.00)

In accordance with the Emergency Operating Procedures, which one of the following would NOT allow water level reduction to be terminated during an ATWS?

a) Reactor power is less than 3%

b) Suppression pool temperature is less than 110 F

c) Drywell pressure is less than 1.85 psig

d) SRVs are open or cycling

QUESTION 2.13 (1.00)

Which one of the following is the injection rate that corresponds to approximately 8% power?

a) 900 gpm

b) 1700 gpm

c) 2000 gpm

d) 4250 gpm

QUESTION 2.14 (1.00)

Which one of the following is the reactor water level which requires closure of the MSIVs in accordance with the Emergency Operating Procedures?

a) +58 inches

b) +75 inches

c) +92 inches

d) +100 inches

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.15 (1.00)

Which one of the following correctly explains why lowering water level during an ATWS will reduce power?

a) Reduce natural circulation through the core

b) Reduce pressure in the core

c) Reduce differential pressure between downcomer and core

d) Reduce the subcooling of water entering the core

QUESTION 2.16 (1.00)

Which one of the following is the power supply to Unit 2 ARI system?

a) Essential Cabinet 2A

b) Essential Cabinet 2B

c) Diesel Generator Battery 2A

d) Diesel Generator Battery 2B

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.17 (1.00)

Given the following conditions:

- 60% power
- RBCCW Heat Exchanger outlet temperature increasing
- RBCCW Surge Tank level increasing
- RBCCW radiation level increasing
- All other plant parameters are normal

Which one of the following would correctly explain these indications ?

a) Service Water leak in the RBCCW Heat Exchanger.

b) Reactor Coolant leak in the NRHX.

c) RBCCW Surge Tank Level Controller failure.

d) RBCCW Heat Exchanger PSW Outlet valve closed.

QUESTION 2.18 (1.00)

Which one of the following would result in a violation of a safety limit in accordance with Unit 1 Technical Specifications?

a) Water level decreases to -150 inches following a scram then is restored to +56.5 inches by RCIC.

b) Main turbine and both feed pumps trip at +58 inches while operating at 20% power and a scram occurs on low level.

c) EHC pressure regulator fails causing a reactor scram on MSIV closure while operating at 20% power.

d) A Group I isolation occurs on MSL High Radiation and a reactor scram occurs on MSIVs Not Full Open while operating at 100% power.

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.19 (1.00)

During an ATWS, several actions are taken to mitigate heat addition to the suppression pool. Which one of the following is correct concerning implementation of these actions?

a) If power is at 100% then immediate tripping of the recirculation pumps from their present speed is required to reduce heat added to the suppression pool.

b) If SLC initiation fails to inject boron, level reduction should be delayed until alternate boron injection is started.

c) Boron injection is required prior to tripping the recirculation pumps when greater than 30% power to enhance the dispersal of boron through the core and expedite reactor shutdown.

d) Level reduction is commenced if power is greater than 3%, suppression pool temperature is greater than 110 F and an SRV is cycling.

QUESTION 2.20 (1.00)

Which one of the following requires a manual scram in accordance with 34AB-OPS-040-2S, Loss of CRD?

a) 2 accumulator trouble lights are received.

b) Neither CRD pump can be immediately restarted.

c) Any CRD temperature exceeds 400 F.

d) 9 control rods exceed 250 F.

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.21 (1.00)

Given the following conditions:

- Unit 2 is at 80% power
- Vital AC Power is supplied from ESS 600V Bus 2C.

Which one of the following is correct concerning the unrecoverable loss of ESS 600V Bus 2C?

a) Automatically transfer to the battery through the Static Inverter.

b) Manually transferred to the Static Inverter by depressing the "Alternate" pushbutton.

c) The reactor must be manually scrammed.

d) The recirculation pumps must be manually reduced to minimum speed.

QUESTION 2.22 (1.00)

In accordance with SOFI PATH 1, reactor power is to be maintained greater than 8% during an ATWS when RPV flooding is required and boron injection has commenced.

Which one of the following correctly describes the reason for maintaining reactor power greater than 8%?

a) Provides adequate steam flow to adequately cool uncovered portions of the core.

b) Corresponds to a level above TAF where natural circulation stops.

c) Allows higher injection flow rates to be achieved without causing large power excursions.

d) Ensures that APRMs are sufficiently on scale to yield an accurate power level indication.

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.23 (1.00)

Given the following conditions:

- Unit 2 at 80% Power
- Unit 2 in control of DG 1B
- DG 1B surveillance in progress
- DG 1B at rated speed and voltage ready to be synchronized.

Which one of the following statements is correct if SUT 2D deenergizes ?

a) 4160V Bus 2F will automatically be energized by DG 1B twelve seconds after the loss of SUT 2D.

b) 4160V Bus 2F cannot be energized by DG 1B in the "Test" mode given these conditions.

c) 4160V Bus 2F will be energized from SUT 2C when the operator takes the alternate supply breaker control switch to the closed position.

d) 4160V Bus 2F will be energized when the operator resets the LOSP Lockout Relay, turns on the synchroscope, and takes the DG 1B output breaker control switch to the closed position.

QUESTION 2.24 (1.00)

Which one of the following correctly describes a consequence of operating in the UNSAFE region of the Heat Capacity Level Limit upon ADS actuation ?

a) Result in containment failure due to excessive temperature and unstable steam condensation.

b) Cause excessive hydrodynamic stress on the Torus.

c) Result in inadequate core cooling due to insufficient driving head through the SRVs.

d) Result in containment failure due to exceeding the Primary Containment Pressure Limit.

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.25 (1.00)

Which one of the following conditions requires a manual scram in accordance with 34-AB-OPS-018-0S, Loss of Secondary Containment Integrity and Secondary Containment Control?

a) Any area temperature, differential temperature, radiation level, or floor drain sump/area water level exceeds maximum normal level.

b) A primary system is discharging into an area and cannot be isolated and prior to the maximum safe operating level being reached.

c) A primary system is discharging into an area and cannot be isolated, and prior to any maximum normal level being exceeded.

d) Any area temperature, differential temperature, or radiation level exceeds 3 times its maximum normal level.

QUESTION 2.26 (1.00)

Which one of the following conditions requires a manual scram in accordance with 34-AB-OPS-058-2S, Reactor Power Instabilities?

a) Core Flow is at 43% and reactor power is just above the 80% Load Line.

b) Core Flow is at 38% and reactor power is just below the 100% Nominal Flow Control Line.

c) Bandwidth oscillations of 11% peak to peak exist on APRM A.

d) Bandwidth oscillations of 7% peak to peak exist on three LPRMs.

QUESTION 2.27 (2.00)

State ALL immediate operator actions PRIOR to evacuating the Control Room in accordance with 34AB-OPS-055-2S, Control Room Evacuation - Unit Shut Down.

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.28 (1.00)

Which one of the following Off Gas System hydrogen concentrations is the minimum percentage requiring entry into a Limiting Condition for Operation?

a) 2%

b) 3%

c) 4%

d) 5%

QUESTION 2.29 (1.00)

In accordance with 34AB-OPS-024-2S, RPIS FAILURE, which one of the following is the correct action for a loss of RPIS?

a) A manual reactor scram

b) Commence a normal reactor shutdown

c) Commence a fast reactor shutdown

d) Suspend all rod motion

QUESTION 2.30 (1.00)

~~Which one of the following is NOT a positive indication that SLC solution is being injected?~~

~~a) Pump running light ON~~

Deleted

~~b) Pump discharge pressure INCREASING~~

~~c) SLC storage tank level DECREASING~~

~~d) Neutron flux level DECREASING~~

QUESTION 2.31 (1.00)

Given the following conditions:

- Unit 2 at 80% power
- POWDEX SYSTEM TROUBLE alarm lit
- CONTAMINATED FEEDWATER alarm lit
- Feedwater conductivity 2.1 umhos/cm
- CST level increasing

Which one of the following is the correct immediate action in accordance with 34-AB-OPS-028-2N, Condenser Tube Leaks, for the given conditions?

a) Reduce load as necessary until the affected Hotwell Water Box can be isolated.

b) Conduct a Fast Reactor Shutdown in accordance with 34-GO-OPS-014-2S, Fast Reactor Shutdown.

c) Scram the reactor and trip the reactor feed pumps, condensate booster pumps and condensate pumps.

d) Reduce power below 30% and be in HOT STANDBY within 24 hours if the leak cannot be isolated.

(***** END OF CATEGORY 2 *****)

QUESTION 3.01 (1.00)

Which one of the following is the cause for the loss of condenser vacuum during a loss of Instrument and Service air?

a) Reduced steam condensation

b) An increase in Hotwell level

c) Buildup of noncondensible gases

d) Loss of turbine gland seals

QUESTION 3.02 (1.00)

Which one of the following correctly describes the operation of Unit 1 RCIC from the Remote Shutdown Panel?

a) All automatic and manual turbine trips are operable.

b) The automatic isolation features are operable.

c) The Steam Supply Valve F045 High Reactor Water Level closure is operable.

d) The sequencing of the steam supply valves to the turbine is operable.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.03 (1.00)

Given the following conditions:

- Unit 2 at 90% power
- An operator opens a Bypass Valve using the Bypass Jack

Which one of the following statements correctly describes plant response?

a) Reactor power remains constant and generator load decreases.

b) Reactor power decreases and generator load remains constant.

c) Reactor power increases and generator load increases.

d) Reactor power decreases and generator load decreases.

QUESTION 3.04 (1.00)

Given the following conditions:

- Unit 2 startup in progress
- All plant parameters are normal
- RWM Group 12 control rods being withdrawn
- A RWM Notch Error alarm is received

Which one of the following failures would cause the RWM Notch Error alarm?

a) RBM

b) RSCS

c) RWM

d) RMCS

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.05 (1.00)

Which one of the following correctly describes rod position input to RSCS between 100% and 50% rod densities?

a) Rod Movement control switch and settle bus

b) Full-in and full-out reed switches

c) Position 00 and 48 reed switches

d) Plant computer

QUESTION 3.06 (1.00)

Which one of the following is the Technical Specification setpoint for the SRM Hi Flux Rod Block?

a) 1X10E4 cps

b) 5X10E4 cps

c) 1X10E5 cps

d) 5X10E5 cps

QUESTION 3.07 (1.00)

The following annunciator is received in the control room: "SBLC LOSS OF CONTINUITY TO SQUIB VALVE". State two (2) unique control room indications available to verify this alarm.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.08 (1.00)

Which one of the following electrical buses supplies power to the Standby Liquid Control system squib valves?

a) R24-S011, R24-S012

b) R24-S009, R24-S010

c) R24-S041, R24-S042

d) R24-S016, R24-S017

QUESTION 3.09 (0.75)

Given the following conditions:

- A complete Loss of RBCCW in progress

Match the automatic action in COLUMN A with the corresponding temperature in COLUMN B for the given conditions. A response may be used more than once.

COLUMN A	COLUMN B
A. RWCU pump trips	1. 90 F
B. RWCU isolates	2. 140 F
C. RECIRC MG trips	3. 210 F
	4. 226 F

QUESTION 3.10 (1.00)

Unit 2 is conducting a complete core off-load when the Fuel Pool Cooling pump fails. State two (2) other systems which can be used to cool the fuel pool.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.11 (1.00)

Which one of the following is the correct fire pump discharge header pressure which will cause automatic starting of the electric fire pump?

a) 110 psig

b) 100 psig

c) 90 psig

d) 80 psig

QUESTION 3.12 (1.00)

RCIC is operating a rated flow during a full flow test surveillance when a spurious RCIC initiation signal is received. State ALL RCIC valves which realign and their position.

QUESTION 3.13 (1.00)

Which one of the following is the Rod Worth Minimizer designed to minimize the consequences of?

a) Control rod drop

b) Continuous control rod withdrawal error

c) Continuous control rod insert error

d) Control rod sequence error

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.14 (1.00)

The Rod Block Monitor system designed to prevent fuel damage during high level power operation during a specific event. Which one of the following is this event?

a) Control rod drop

b) Continuous control rod withdrawal error

c) Continuous control rod insert error

d) Control rod sequence error

QUESTION 3.15 (1.00)

Given the following conditions:

- Unit 1 is operating at 40% RTP
- APRM 'A' fails DOWNSCALE
- APRM 'A' has NOT been bypassed

Deleted

Which one of the following correctly describes the response of RBM A for the given conditions?

a) Generates a rod withdrawal block

b) Automatically shifts to an alternate reference APRM

c) Generates a rod select block

d) Automatically bypasses rod block outputs

QUESTION 3.16 (1.00)

The reference leg for "B" emergency range level (D003) ruptures and is isolated by its excess flow check valve.

Which one of the following is correctly describes the indication of "B" emergency range level and narrow range reactor pressure to this event?

a) "B" emergency range level upscale; narrow range reactor pressure upscale

b) "B" emergency range level downscale; narrow range reactor pressure upscale

c) "B" emergency range level upscale; narrow range reactor pressure downscale

d) "B" emergency range level downscale; narrow range reactor pressure downscale

QUESTION 3.17 (1.50)

Match the function in Column A with the Off Gas system components in Column B. Each response in Column B may be used more than once.

COLUMN A	COLUMN B
A. Decay of short-lived radioisotopes	1) Preheater
B. Minimizes possibility of explosive gas	2) Reheater
C. Minimizes moisture entry into catalytic recombiners	3) Glycol System
D. Cools off gas mixture to condense moisture	4) After filter
E. Reduces humidity prior to entry into the adsorbers	5) Holdup Volume
F. Prevents charcoal fines from entering the main stack	6) Recombiner

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.18 (1.00)

Which one of the following authorizes bypassing of the Rod Worth Minimizer?

a) Senior Nuclear Engineer

b) Operations Supervisor

c) Shift Technical Advisor

d) Any SRO licensed individual

QUESTION 3.19 (1.00)

HPCI is in a normal standby alignment when both divisions of Leak Detection associated with HPCI are deenergized. Which one of the following correctly describes the HPCI valves which will automatically close?

a) F042, F002

b) F002, F003

c) F042, F041

d) F059, F003

QUESTION 3.20 (1.00)

The HPCI system has an interlock to prevent draining the suppression pool to the CST. Which one of the following statements correctly describes this interlock?

a) F041 closes if F004 is opened

b) F042 closes if F004 is opened

c) F004 closes if F042 or F041 are opened

d) F004 closes if F042 and F041 are opened

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.21 (1.00)

Given the following conditions:

- Unit 1 is at 75% power
- Turbine stop valve testing is in progress
- The operator begins testing MSV # 4 before MSV # 3 is fully open

Which one of the following is correctly describes the response of the unit for the given conditions?

a) half scram

b) turbine runback

c) full scram

d) recirculation runback

QUESTION 3.22 (1.00)

Which one of the following is the reactor pressure which will cause the RHR Shutdown Cooling suction isolation valves to close?

a) 125 psig

b) 145 psig

c) 165 psig

d) 175 psig

QUESTION 3.23 (1.00)

Which one of the following is the MAXIMUM allowable internal design pressure of the primary containment system?

a) 56 psig

b) 62 psig

c) 66 psig

d) 72 psig

QUESTION 3.24 (0.50)

Complete the following in accordance with Unit 1 Technical Specifications:

The reactor coolant system integrity safety limit is __(a)__ psig at all times when irradiated fuel is in the reactor vessel and is __(b)__ psig when RHR is the Shutdown cooling mode

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.25 (1.50)

Match the system response in column A with the corresponding Reactor Water Level setpoint in Column B. Each response may be used more than once.

COLUMN A	COLUMN B
A. RECIRC RUNBACK DURING RFP LOSS	1. 47
B. STANDBY GAS STARTS	2. 32
C. MSIVs CLOSE	3. 12
D. DRYWELL PNEUMATICS ISOLATE	4. 10
E. RHR SHUTDOWN COOLING ISOLATES	5. -47
F. ADS 13 MIN TIMER STARTS	6. -75
	7. -113

QUESTION 3.26 (1.00)

The Intermediate Range monitors are reading 10 on range 9. Which one of the following corresponds to the percent full power?

a) 2%

b) 4%

c) 6%

d) 8%

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.27 (0.75)

Complete the following statement:

The MSIV Leakage Control System takes a suction between the inboard MSIVS and outboard MSIVS and between the __(a)__ and the __(b)__. The system then exhausts to the __(c)__.

QUESTION 3.28 (1.00)

Which one of the following is the correct condenser vacuum setpoint which will cause the main turbine bypass valves to trip close ?

a) 5.0 inches Hg

b) 7.0 inches Hg

c) 13.0 inches Hg

d) 22.3 inches Hg

QUESTION 3.29 (1.00)

Which one of the following is the generator load limitation for single reactor feed pump operation in accordance with 34GO-OPS-005-2, Power Changes?

a) 55%

b) 60%

c) 65%

d) 70%

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.30 (1.00)

Which one of the following conditions will cause a control rod to drift?

a) Failed open CRD flow control valve

b) Excessive cooling water pressure

c) Low scram accumulator pressure

d) Failed open stabilizing valves

QUESTION 3.31 (1.00)

Which one of the following is the purpose of the EOC-RPT?

a) Provide a redundant means of reducing power during ATWS

b) Improve thermal margins during turbine trip/load reject

c) Prevent exceeding thermal limits during recirc flow controller failure

d) Prevent exceeding thermal limits during loss of feedwater heating

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.32 (1.00)

Given the following conditions:

- The motor area temperature element for Recirc pump A is out of service.
- Recirculation Pump A trips due to a relay malfunction and is restarted.
- The pump runs for 20 minutes and trips again.

Which one of the following correctly corresponds to the minimum waiting period required before the pump may be restarted?

a) Restart immediately

b) Restart in 15 min

c) Restart in 45 min

d) Restart in 60 min

QUESTION 3.33 (1.00)

The "GENERATOR PROTECTION CIRCUIT ENERGIZED" annunciator is received in the control room.

Which one of the following correctly describes the unit condition indicated by this annunciator?

a) Main Generator will trip on reverse power in 30 sec

b) Main Generator will trip in 2 min unless stator water is restored

c) Main Generator will trip in 3.5 min unless stator amps are reduced to less than 4500 amps

d) Main Generator field breaker has tripped

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.34 (1.00)

The output of the EHC pressure regulator fails HIGH causing a bypass valve to open.

Which one of the following will cause the bypass valve to close?

a) Decrease Bypass Jack

b) Decrease Load Limit Pot

c) Decrease Pressure Setpoint

d) Decrease Maximum Combined Flow Pot

QUESTION 3.35 (1.50)

The unit is at 100% power with feedwater in automatic three element control. Match the Instrument Failure in Column A with the expected plant response in Column B. The responses in Column B may be used more than once.

COLUMN A	COLUMN B
A. Feed flow detector fails low	1. Reactor level increases to turbine trip
B. Feed flow detector fails high	2. Reactor level decreases below scram setpoint
C. Steam flow detector fails low	3. Reactor Level stabilizes 6"- 10" below normal
D. Steam flow detector fails high	4. Reactor level stabilizes 6" - 10" above normal
E. Selected level transmitter fails low	
F. Selected level transmitter fails high	

QUESTION 3.36 (1.00)

Which one of the following corresponds to the MAXIMUM acceptable time for the emergency diesels to start and come to rated speed and voltage?

a) 8 sec

b) 10 sec

c) 12 sec

d) 15 sec

QUESTION 3.37 (1.00)

The emergency diesel generators have started on high drywell pressure and loss of power and have tied to their respective buses.

Which one of the following would cause a diesel generator to trip for these conditions?

a) Generator Reverse Power

b) Crankcase High pressure

c) Jacket water pressure low

d) Engine Overspeed

QUESTION 3.38 (1.00)

Which one of the following is the ALTERNATE power supply to the Unit 2 control room NSSS annunciators?

a) 125 vdc Bus 2A

b) 125 vdc Bus 2B

c) Instrument Bus 2A

d) Instrument Bus 2B

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.39 (1.00)

34SO-C11-005-2/1S, Control Rod Drive Hydraulic system, cautions a CRD pump should not be started with a recirculation loop isolated.

Which one of the following will this caution prevent?

a) Thermal shock to the recirc pump

b) Overpressurizing the recirc loop

c) CRD pump overcurrent trip

d) CRD pump runout

QUESTION 3.40 (1.00)

The following annunciator is received in the control room: "SPEED CONTROL A SIGNAL FAILURE".

Which one of the following correctly describes the response of Recirc MG A for this alarm?

a) Drive Motor breaker trips

b) Generator Field breaker trips

c) Runback to Speed Limiter # 2

d) Scoop tube lock up

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.41 (1.00)

Given the following conditions:

- Unit 1 at 100% RTP
- "B" RWCU in operation and "A" RWCU pump suction is isolated
- "B" RWCU filter/demineralizer and pump are taken out of service
- "B" RWCU pump suction isolation valve is CLOSED
- "B" RWCU pump is allowed to cool to less than 130 F
- "A" RWCU pump suction is unisolated
- A Group 5 PCIS isolation occurs

Which one of the following is the cause for the Group 5 PCIS isolation?

a) High NRHX outlet temperature

b) High RWCU room temperature

c) Low reactor water level

d) High differential flow

QUESTION 3.42 (1.00)

Which one of the following conditions will NOT result in a shutdown of the Standby Gas Treatment System with a valid initiation signal present ?

a) High temperature 225 degrees F in the Charcoal Bed.

b) Overloads in the Local Control Panel.

c) High temperature 225 degrees F at the Heater outlet.

d) High temperature 190 Degrees F at the Heater inlet.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.43 (1.00)

Which one of the following is NOT required for PRIMARY CONTAINMENT INTEGRITY in accordance with Unit 2 Technical Specifications?

a) All equipment hatches closed and sealed

b) Reactor coolant leakage rates within limits

c) SBGT system is OPERABLE

d) Containment airlock is OPERABLE

QUESTION 3.44 (1.00)

Given the following conditions:

- Unit 2 at 100% power
- APRM 'A' in BYPASS
- IRM 'C' is placed in BYPASS

Which one of the following statements correctly describes the Technical Specification requirements for the given conditions?

a) Technical Specification 3.3.1.a

b) Technical Specification 3.3.1.b. ACTION 3

c) Technical Specification 3.3.5.b

d) Technical Specification 3.0.3

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.45 (1.00)

Given the following conditions:

- A fire alarm sounds over the PA system
- The shift fire brigade has donned turnout gear and SCBA

Which one of the following correctly describes the next action the shift fire brigade should take?

a) Report to the Fire Brigade Leader

b) Report to the Shift Supervisor

c) Report to the Fire Equipment Building

d) Report to the Security Building

QUESTION 3.46 (1.00)

A 26 year old operator will be performing work in a 200 mrem radiation field. His lifetime exposure is 10 rem and his current quarter dose is .75 rem. He has a current NRC form 4 on file and has received authorization to exceed administrative limits. Which one of the following is the maximum number of hours this individual can work in this area?

a) 3.25

b) 8.25

c) 11.25

d) 13.25

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.47 (1.50)

a. State the quarterly radiation exposure limit for each of the following in accordance with 10CFR20:

1) Whole body

2) Hands and forearms, feet and ankles

3) Skin of the whole body.

b. The limit for whole body dose may be exceeded if specific conditions are complied with. Answer the following concerning the maximum limits allowed:

1) State the maximum quarterly whole body limit.

2) State ALL conditions that must be met.

(***** FND OF CATEGORY 3 *****)
(********** END OF EXAMINATION **********)

ANSWER 2.01 (~~1.00~~) 0.50

[0.50] each:
1. conserve (restore) RPV inventory
~~2. limit radioactive release~~

REFERENCE

LT-IH-20104-03 Objective 25
LT-IH-20105-03 Objective 25
LT-IH-20114-00 Objective 1

295038G012 295031G012 295024G012 ..(KA's)

ANSWER 2.02 (1.00)

a)

REFERENCE

LT-IH-03401-00 Objective 13d

295031A108 ..(KA's)

ANSWER 2.03 (1.50)

[0.50] each:
1. Four or more control rods drift in

2. "SCRAM VALVE PILOT HDR HI/LO PRESS" alarm coincident with "CRD HYD HIGH TEMP" alarm

3. scram pilot air header less than 50 psig (on the local indicator)

REFERENCE

34AB-OPS-020-2S
LT-ST-03501-00 Objective 16

295019G010 ..(KA's)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

ANSWER 2.04 (1.00)

b)

REFERENCE

LT-ST-05201-00 Objective3,4

295016G006 ..(KA's)

ANSWER 2.05 (1.00)

b)

REFERENCE

LT-IH-00401-00 B.7.c.

295001K306 202001A201 295001A205 ..(KA's)

ANSWER 2.06 (1.00)

a)

REFERENCE

LT-ST-03801-00 Objective12

218000K403 295024A108 295024K208 ..(KA's)

ANSWER 2.07 (1.00)

b)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-20012-01 Objective 1

295031A201 ..(KA's)

ANSWER 2.08 (1.00)

a)

REFERENCE

LT-IH-20113-00 Objective2

295037A203 ..(KA's)

ANSWER 2.09 (1.00)

b)

REFERENCE

LT-IH-20113-00 Objective2

295031G007 ..(KA's)

ANSWER 2.10 (1.00)

C

REFERENCE

LT-IH-20113-00 Objective 2

295031G007 ..(KA's)

ANSWER 2.11 (1.00)

b)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-20113-00 Objective2

295031G012 ..(KA's)

ANSWER 2.12 (1.00)

d)

REFERENCE

LT-IH-20107-03 Objective 16

295037A204 295037A201 295037A202 ..(KA's)

ANSWER 2.13 (1.00)

b)

REFERENCE

LT-IH-20114-00 Objective 2

295037A202 295037A204 295037A201 ..(KA's)

ANSWER 2.14 (1.00)

d)

REFERENCE

LT-IH-20114-00 Objective 2

295008A103 ..(KA's)

ANSWER 2.15 (1.00)

~~b)~~ ~~a? or c?~~
a)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-20113-00 Objective 1,2

295037K102 295037K209 295037K303 ..(KA's)

ANSWER 2.16 (1.00)

c)

REFERENCE

LT-IH-00101-00 Change Notice 89-0070 Objective 28

295004K303 201001K205 ..(KA's)

ANSWER 2.17 (1.00)

b)

REFERENCE

LT-IH-00901, Objective 19
34SO-P42-001-2S

295018G011 ..(KA's)

ANSWER 2.18 (1.00)

d)

REFERENCE

TS 3.7.A.1 Amendment 165

295037K301 295037G012 ..(KA's)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

ANSWER 2.19 (1.00)

d)

REFERENCE

SOFI Flowchart 1: Content and Use
LT-IH-20107-03 Objective 11,15,16

295037K301 295037G012 ..(KA's)

ANSWER 2.20 (1.00)

d)

REFERENCE

34-AB-OPS-040-2S

295022G011 295022K301 ..(KA's)

ANSWER 2.21 (1.00)

c)

REFERENCE

34-AB-OPS-15-2S
LT-IH-02703 Objective 21

295003K305 295003K301 295003K203 295003G010 ..(KA's)

ANSWER 2.22 (1.00)

b)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-20107-03

295037G007 295037A202 295037K303 ..(KA's)

ANSWER 2.23 (1.00)

b)

REFERENCE

LT-IH-00742-05 Objective 8
LT-IH-02702-00

295003A102 295003K202 ..(KA's)

ANSWER 2.24 (1.00)

a)

REFERENCE

LT-IH-20113-00 Objective 1

295029K301 295029K101 295029G007 ..(KA's)

ANSWER 2.25 (1.00)

b)

REFERENCE

34AB-EOP-018-0S Objective 10

295033G010 295033K302 ..(KA's)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

ANSWER 2.26 (1.00)

c)

REFERENCE

LT-IH-20201-01 Objective 19

295006G010 ..(KA's)

ANSWER 2.27 (2.00)

[0.50] each:
Close: (Inboard and Outboard) MSIVs
Steamline Drains (Inboard and Outboard) Isolation Valves
Reactor Water Sample (Inboard and Outboard) Isolation Valves

GROUP I ISOLATION VALVES

Open : RFP Bypass Valve [2N21-F113]

REFERENCE

34AB-OPS-055-2S

295016G010 ..(KA's)

ANSWER 2.28 (1.00)

c)

REFERENCE

TS 3/4 3.11.2.6

271000G005 ..(KA's)

ANSWER 2.29 (1.00)

d)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

REFERENCE

34AB-OPS-24-2S

214000G014 ..(KA's)

~~ANSWER 2.30 (1.00)~~

~~a)~~

Deleted

~~REFERENCE~~

~~LT-IH-01101 Objective 16.b.~~

ANSWER 2.31 (1.00)

c)

REFERENCE

34-AB-OPS-028-2N

256000A215 256000G014 256000G015 ..(KA's)

(***** END OF CATEGORY 2 *****)

ANSWER 3.01 (1.00)

c)

REFERENCE

34AB-OPS-020-2S
LT-IH-02501-00 Objective 13.b, 18.c

271000K107 295002K306 ..(KA's)

ANSWER 3.02 (1.00)

a

REFERENCE

LT-IH-03901-00 Objective 12

217000K404 295016A107 ..(KA's)

ANSWER 3.03 (1.00)

a)

REFERENCE

LT-IH-01901-00 Objective 20

295000K102 241000K101 ..(KA's)

ANSWER 3.04 (1.00)

b)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-050402-02 Objective 9

201004A305 201004K406 ..(KA's)

ANSWER 3.05 (1.00)

b)

REFERENCE

LT-IH-050402-02 Objective 6

201004K405 201004K103 ..(KA's)

ANSWER 3.06 (1.00)

c)

REFERENCE

LT-IH-01201-00 Objective 9.b.

215004A105 ..(KA's)

ANSWER 3.07 (1.00)

[0.50] each:
1. amber squib continuity lights
2. continuity meters

REFERENCE

LT-ST-01101-00 Objective 14

211000K404 ..(KA's)

ANSWER 3.08 (1.00)

a)

REFERENCE

LT-ST-01101-00 Objective6a

211000K202 ..(KA's)

ANSWER 3.09 (0.75)

[0.25] each:
A. 2
B. 2
C. 3

REFERENCE

LT-IH-00901-00 Objective 17, 18

204000K104 202001K107 ..(KA's)

ANSWER 3.10 (1.00)

1. RHR (.5)
2. Unit 1 FPCC (.5)

REFERENCE

LT-IH-04501-01 Objective 3

233000K102 ..(KA's)

ANSWER 3.11 (1.00)

a)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-03601-01 Objective5

286000K403 ..(KA's)

ANSWER 3.12 (1.00)

F013, Inboard Discharge (.25) OPENS (.25)
F022, CST Test Isolation (.25) CLOSES (.25)

REFERENCE

LT-IH-03901-00 Objective 9.b.

217000A201 ..(KA's)

ANSWER 3.13 (1.00)

a)

REFERENCE

LT-IH-05403-00 Objective 1
TS BASES 3.3.6.1

201006G004 ..(KA's)

ANSWER 3.14 (1.00)

b)

REFERENCE

TS BASES Pg # 3.3-14
LT-IH-05403-00 Objective 1

215002G004 ..(KA's)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

ANSWER 3.15 (1.00)

d)

Deleted

REFERENCE

LT-1H-01203-00 Objective 15

215002A203 ..(KA's)

ANSWER 3.16 (1.00)

c)

REFERENCE

LT-1H-04404-00 Objective 15

216000K324 216000K325 ..(KA's)

ANSWER 3.17 (1.50)

[0.25] each:
A. 5
B. 6
C. 1
D. 3
E. 2
F. 4

REFERENCE

LT-1H-03101-00 Objective 6

2710006007 ..(KA's)

ANSWER 3.18 (1.00)

b)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-05403-00 Objective 8

201006G001 ..(KA's)

ANSWER 3.19 (1.00)

b)

REFERENCE

LT-ST-00501-00 Objective 21c

206000K610 ..(KA's)

ANSWER 3.20 (1.00)

d)

REFERENCE

LT-ST-005001-02

206000K417 ..(KA's)

ANSWER 3.21 (1.00)

a)

REFERENCE

LT-IH-01001-00

212000K110 ..(KA's)

ANSWER 3.22 (1.00)

b)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-ST-00701-00 Objective 7d

205000K402 ..(KA's)

ANSWER 3.23 (1.00)

b)

REFERENCE

LT-IH-01301-00 Objective 1

223001G005 ..(KA's)

ANSWER 3.24 (0.50)

[0.25] each:
a. 1325
b. 162

REFERENCE

Hatch TS 1.2.A.1 & 1.2.A.2
LT-IH-04401-00 Objective 9a

290002G005 ..(KA's)

ANSWER 3.25 (1.50)

[0.25] each:
A. 2
B. 5
C. 7
D. 4
E. 4
F. 7

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-01301-00 Objective 6a
LT-IH-04404-00 Objective 1

216000K102	218000K103	259001K109	239001K401	205000K102
261000A210	212000K102	202001K115	217000K102	..(KA's)

ANSWER 3.26 (1.00)

b) ~~c)~~

REFERENCE

LT-IH-01202-00 Objective 5

215003G009 215003A407 ..(KA's)

ANSWER 3.27 (0.75)

[0.25] each:
a. outboard MSIVS
b. main turbine (stop valves)
c. torus (area)

REFERENCE

LT-IH-04901-00

239003K101 ..(KA's)

ANSWER 3.28 (1.00)

b)

REFERENCE

LT-IH-02501-00 L012

241000K605 ..(KA's)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

ANSWER 3.29 (1.00)

a)

REFERENCE

LT-IH-00201-00 Objective 12

259001G010 ..(KA's)

ANSWER 3.30 (1.00)

b)

REFERENCE

LT-IH-00401-00 Objective 12

201001G010 ..(KA's)

ANSWER 3.31 (1.00)

b)

REFERENCE

LT-IH-00401-00 Objective 12

202001K505 ..(KA's)

ANSWER 3.32 (1.00)

a)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-I- 0401-00 Objective 1
3480-L31-001-28

202001G010 ..(KA's)

ANSWER 3.33 (1.00)

c)

REFERENCE

LT-ST-02301-00 Objective 14

245000K605 ..(KA's)

ANSWER 3.34 (1.00)

d)

REFERENCE

LT-IH-01901-00 Objective 8

241000A115 241000A114 ..(KA's)

ANSWER 3.35 (1.50)

[0.25] each:
A. 1
B. 2
C. 3
D. 4
E. ~~2~~ 1
F ~~1~~ 2

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-00202-00 Objective 19

259000A203 259002A202 259002A201 ..(KA's)

ANSWER 3.36 (1.00)

c)

REFERENCE

LT-IH-02801-00 Objective 15

264000A302 ..(KA's)

ANSWER 3.37 (1.00)

d)

REFERENCE

LT-IH-02801-00 Objective 1

264000K402 ..(KA's)

ANSWER 3.38 (1.00)

a)

REFERENCE

34AB-OPS-053-2S

263000K201 ..(KA's)

ANSWER 3.39 (1.00)

b)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

LT-IH-00101-00 Objective 14

201001G010 ..(KA's)

ANSWER 3.40 (1.00)

d)

REFERENCE

LT-IH-00401-00 LO 17

2020001A20 ..(KA's)

ANSWER 3.41 (1.00)

d)

REFERENCE

LT-IH-00301-00 Objective 7e
LER 89-001

204000K404 ..(KA's)

ANSWER 3.42 (1.00)

d

REFERENCE

LT-IH-03001-00 Objective 7c

261000A205 288000A301 ..(KA's)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

ANSWER 3.43 (1.00)

c)

REFERENCE

Unit 2 TS Definitions

ANSWER 3.44 (1.00)

a)

REFERENCE

LER 88-001

215005G005 ..(KA's)

ANSWER 3.45 (1.00)

a)

REFERENCE

40AC-ENG-008-05

294001K116 ..(KA's)

ANSWER 3.46 (1.00)

c)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

10CFR20
LT-IH-3008

294001K103 ..(KA's)

ANSWER 3.47 (1.50)

[0.25] each:
a. 1) 1.25 rem
2) 18.75 rem
3) 7.5 rem

b. 1) 3 rem
2) Exposure must not exceed 5[N-18].
NRC Form 4 completed or previous history known.

REFERENCE

RADIATION EXPOSURE LIMITS, 60AC-HPX-001-0S
10CFR20

294001K103 ..(KA's)

(***** END OF CATEGORY 3 *****)
(********** END OF EXAMINATION **********)

QUESTION	VALUE	REFERENCE
2.01	1.00	9000873
2.02	1.00	9000891
2.03	1.50	9000896
2.04	1.00	9000899
2.05	1.00	9000902
2.06	1.00	9000919
2.07	1.00	9000939
2.08	1.00	9000940
2.09	1.00	9000941
2.10	1.00	9000942
2.11	1.00	9000943
2.12	1.00	9000944
2.13	1.00	9000945
2.14	1.00	9000946
2.15	1.00	9000947
2.16	1.00	9000948
2.17	1.00	9000921
2.18	1.00	9000923
2.19	1.00	9000924
2.20	1.00	9000929
2.21	1.00	9000930
2.22	1.00	9000931
2.23	1.00	9000932
2.24	1.00	9000934
2.25	1.00	9000935
2.26	1.00	9000936
2.27	2.00	9000933
2.28	1.00	9000874
2.29	1.00	9000938
2.30	1.00	9000875
2.31	1.00	9000928

	32.50	
3.01	1.00	9000925
3.02	1.00	9000926
3.03	1.00	9000937
3.04	1.00	9000871
3.05	1.00	9000872
3.06	1.00	9000878
3.07	1.00	9000879
3.08	1.00	9000880
3.09	0.75	9000881
3.10	1.00	9000882
3.11	1.00	9000883
3.12	1.00	9000884
3.13	1.00	9000885
3.14	1.00	9000886
3.15	1.00	9000887
3.16	1.00	9000888
3.17	1.50	9000889
3.18	1.00	9000890
3.19	1.00	9000892
3.20	1.00	9000893
3.21	1.00	9000894

QUESTION	VALUE	REFERENCE
3.22	1.00	9000895
3.23	1.00	9000897
3.24	0.50	9000898
3.25	1.50	9000900
3.26	1.00	9000901
3.27	0.75	9000903
3.28	1.00	9000904
3.29	1.00	9000907
3.30	1.00	9000908
3.31	1.00	9000909
3.32	1.00	9000910
3.33	1.00	9000911
3.34	1.00	9000912
3.35	1.50	9000913
3.36	1.00	9000914
3.37	1.00	9000915
3.38	1.00	9000916
3.39	1.00	9000917
3.40	1.00	9000918
3.41	1.00	9000920
3.42	1.00	9000927
3.43	1.00	9000876
3.44	1.00	9000877
3.45	1.00	9000905
3.46	1.00	9000906
3.47	1.50	9000922

	48.00	

	80.50	

ENCLOSURE 3

PLANT E. I. HATCH
Reactor Operator and Senior Operator
Written License Examination Utility Comments
October 9, 1989

QUESTION 2.01/5.06

Question was not contained in the exam at the time of the Pre Exam Review. Stem of question states that a LOCA is in progress. If this is true the operator should have entered path five not path four as indicated in question.

The answer given in answer key appears to be answering the question, "Under what conditions may the RPV Cooldown Limit of 100°F per hour be exceeded?", not the question given. The actual reason for the step J0 on path four is to reduce reactor pressure to below the shut off head of the Low Pressure Injection Systems to allow for reactor water level restoration. The rad protection discussed in the answer key was actually accomplished by steps other than J0.

The required answer is not supported by referenced objectives.

RECOMMENDATION: Delete question.

REFERENCE: 31EO-EOP-001-2S, Path Three and Path Four

QUESTION 2.05/5.10

Question previously identified as problem question on Pre Exam Review. Question asks which one of the following is not an indication of Jet Pump failure, but there is no correct answer. Answer given on answer key is "B" (core plate delta pressure decreases) which is an indication you expect to see on a Jet Pump failure. As core flow decreases, core plate delta pressure will decrease.

RECOMMENDATION: Delete question due to no correct answer being given.

REFERENCE: 34SV-SUV-023-2S, LT-IH-00401-00
HNP-2-FSAR 7.6-71

QUESTION 2.15/5.20

Question was previously identified as a problem question on our June 12, 1988 exam and was not contained in the present exam at the time of the Pre Exam Review. Question asks which of the following correctly explains why lowering water level during an ATWS will reduce power. (Answer given on key was "B".)

a. Reduce natural circulation through the core.
b. Reduce pressure in the core
c. Reduce differential pressure between downcomer and core.
d. Reduce the subcooling of water entering the core.

Plant Hatch training material objectives referenced on answer key ARE NOT ASSOCIATED with level power control.

Hatch material LT-IH-20107 refers to A & C when describing level power control.

Answer given in answer key is incorrect.

RECOMMENDATION: Delete question due to A & C being possible correct answers and answer given in answer key being incorrect.

REFERENCE: LT-IH-20107, Plant Hatch PSTG, page 347 -> 351

QUESTION 2.30/5.43

Question was not contained in the exam at the time of the Pre Exam Review. Question asks which of the following is NOT a positive indication that SLC solution is being injected. (Answer given on key was "A".)

a. Pump running light on
b. Pump discharge press increasing
c. SLC storage tank level decreasing
d. Neutron flux level decreasing

A, B, C and D are correct answers because individually neither is a positive indication that SBLC solution is being injected.

If SBLC pump is running the pump running light will be illuminated, but this does not mean that SBLC is injecting. Pump discharge pressure increasing does not indicate SBLC is being injected unless discharge pressure is greater than reactor pressure and a flow path exists. SBLC storage tank level decreasing may be indication that a line has ruptured or that a drain valve was left open. Neutron flux level decreasing may be indication of reactor water level being lowered or of control rod insertion.

One student questioned an exam proctor as to whether SBLC Pump discharge pressure was greater than reactor pressure but proctor did not write his response on board or verbalize his response to other candidates.

RECOMMENDATION: Delete question due to more than one possible answer.

REFERENCE: LT-ST-01101-01

QUESTION 3.15/6.09

APRM referenced in question was changed from "C" to "A" after the Pre Exam Review was conducted. The question asks about the response of rod bock monitor "A" if Unit One was at 40% power and the "A" APRM fails downscale. There is no correct answer listed. The "A" RBM is not affected by "A" APRM failure. The reference APRM for RBM "A" is "C" and its backup APRM is "E". Stem of question should have referred to "C" APRM failure to make question a valid question.

RECOMMENDATION: Delete question due to no correct answer.

REFERENCE: Elementary Diagram H-27571

Georgia Power Company
Post Office Box 439
Baxley, Georgia 31513
Telephone 912 367-7781
912 537-9444

Georgia Power

Edwin I. Hatch Nuclear Plant

October 11, 1989

PLANT E. I. HATCH
NRC Exam Comments
Rtype: A029
Log: LR-GM-018-1089

Mr. Charles Casto
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street
Suite 2900
Atlanta, Georgia 30323

Dear Mr. Casto:

As previously discussed with Curt Rapp, our comments on the written examination administered October 9, 1989 are attached.

If you have any further questions, please call Mr. Curtis Coggin at 912/367-7851, extension 2896.

Harvey Nix

H. C. Nix
General Manager Nuclear Plant

HCN/pjc

attachments

OCT 13 1989

INSTRUCTOR ACTIVITY	OUTLINE OF INSTRUCTION
L3-M5	5. If condensate is the only high pressure system available, reactor pressure will be reduced if needed and condensate will be used to control RPV level
Point out decision steps on chart	6. RFPT's running or available
	a. Determination of RFPT's running
	1. Check RFPT RPM at panel 2H11 - P650 (1H11 - P650)
	2. Check RFP discharge pressure at panel 2H11 - P650 (1H11 - P650)
	b. Determination of RFPT's available
EO 24	1. Able to start and inject through a normal injection line-up with at least one RFP
	c. If RFPT's available, direction given to control level with RFPT's
	d. If RFPT's are not available, use of HPCI and RCIC is directed
Use SOFI Path 3	E. RPV Level Recovery Steps
TRANS 20104-12 TRANS 20104-13 EO 25 Walk through the steps of the RPV Level Control Section. Use various plant conditions to get through all steps	1. Review the different flowpaths and steps of the RPV Level Recovery Steps a. Use TRANS 20104-12 and 20104-13 as guides 2. Specific information
Use specific information as the steps are reached to answer any questions students may have and to show the direction SOFI Path 3 is leading towards	a. EPP 36 - level control and cooldown with MSIV's closed b. EPP 34 - level control with HPCI/RCIC also addresses cooldown; assumes MSIV's open c. EPP 30 - same as EPP 34 except assumes MSIV's closed

d. EEP 35 - level control by CRD/RCIC. Also addresses cooldown.

e. EPP 33 - level control with condensate. Also addresses cooldown.

f. EPP 32 - level control with condensate and feed. Also addresses cooldown.

g. Path 4, H3 - Actions to prevent radioactivity release and control RPV level

h. Path 4, M3 - Actions for steam leak outside primary containment and to control RPV level

i. Path 4, H1 - Control level above -101" with low pressure injection systems

j. Path 4, G1 - Control level with HPCI/RCIC

k. Steps taken if control air pressure less than 45 psig (M4)- align condensate flow around RFP's and through motor operated valve F110

e. Reducing Forced Circulation

Show steps tripping recirc pumps on SOFI Path 1. LOCATIONS: A0, D3, D4

1) If reactor power is above 3% or cannot be determined during an ATWS, recirc pumps will be run back and tripped.

Tripping recirc pumps without first reducing their speed could result in sufficient level swell to trip the main turbine.

a) If below 3% power, the power reduction due to the loss of forced circulation is insignificant. Flow is maintained to promote boron mixing should boron injection be required.

f. Reducing Natural Circulation

EO 13

1) Natural circulation is reduced by lowering RPV level.

a) Natural circulation is a function of the heights of the water columns inside and outside the shroud. Lowering the heights of these water columns reduces the natural circulation driving head.

2) The steps taken to lower RPV level are part of a series of steps known as "Level/Power Control Steps."

C. Level/Power Control

EO 14

Show students the location of the Level/Power Control Steps on SOFI Path 1.

1. Purpose - control reactor power during an ATWS until sufficient boron can be injected to shutdown the reactor.

a. Located on SOFI Path 1 in Column 2, Rows G through N and Column 3, Rows P and R.

2. As mentioned earlier, excessive heat addition to the suppression pool during an ATWS will require emergency depressurization of the RPV if the HCTL is exceeded.

ENABLING OBJECTIVES:

Upon completion of this lesson, the student will be able to:

1. State the purpose of the Path Selection Steps of SOFI Path 1.
2. Given several placards, identify the placard containing the scram action steps.
3. Given several placards, identify the placard containing the feedwater control actions steps.
4. Identify the six parameters checked to determine proper SOFI Path selection in SOFI Path 1.
5. State the action taken if a key parameter in the Path Selection Steps changes.
6. Given SOFI Path 1 and plant conditions, select the proper SOFI Path for use.
7. State the two possible causes of a failure to scram.
8. Identify the nine different methods used to insert control rods in SOFI Path 1.
9. Locate the reactor scram and control rod insertion sequence insensitive steps on SOFI Path 1.
10. State the actions taken if, during use of SOFI Path 1, all control rods are inserted to beyond position 02.
11. State the plant condition requiring boron injection.
12. Explain why ADS is inhibited when boron is injected.
13. State how natural circulation is reduced during an ATWS.
14. State the purpose of level/power control during an ATWS.
15. Identify the three criteria that together determine if Level/Power Control Steps are needed to reduce the energy release to the primary containment.

2) Note also that ADS is inhibited when boron is injected.

EO 12

a) ADS depressurization could result in injection by low pressure ECCS systems, causing a large power excursion.

1) Cold water.

2) Void collapse.

3) Boron dilution.

6. Reducing Core Flow

a. A final method of lowering reactor power and energy addition to the containment is to reduce core flow.

A-0

1) Reducing core flow causes void formation with the subsequent reduction in thermal neutron flux.

b. Core flow is composed of two types of flow.

1) Forced circulation.

2) Natural circulation.

c. Both forced and natural circulation can be reduced to lower reactor power.

d. Note - there is a drawback to lowering core flow during an ATWS.

1) Without core flow, boron injected into the RPV will tend to remain in the vessel lower head instead of being distributed throughout the fuel bearing region. This is an important factor which had to be considered when the ATWS steps of the EPG's were written.

2.15/5.20

Power ↓ & prevent swell

e. Reducing Forced Circulation

Show steps tripping recirc pumps on SOFI Path 1. LOCATIONS: A0, D3, D4

1) If reactor power is above 3% or cannot be determined during an ATWS, recirc pumps will be run back and tripped.

Tripping recirc pumps without first reducing their speed could result in sufficient level swell to trip the main turbine.

A-0

a) If below 3% power, the power reduction due to the loss of forced circulation is insignificant. Flow is maintained to promote boron mixing should boron injection be required.

f. Reducing Natural Circulation

EO 13

1) **Natural circulation is reduced by lowering RPV level.**

C-2

a) Natural circulation is a function of the heights of the water columns inside and outside the shroud. Lowering the heights of these water columns reduces the natural circulation driving head.

2) The steps taken to lower RPV level are part of a series of steps known as "Level/Power Control Steps."

C. Level/Power Control

EO 14

Show students the location of the Level/Power Control Steps on SOFI Path 1.

1. Purpose - control reactor power during an ATWS until sufficient boron can be injected to shutdown the reactor.

a. Located on SOFI Path 1 in Column 2, Rows G through N and Column 3, Rows P and R.

2. As mentioned earlier, excessive heat addition to the suppression pool during an ATWS will require emergency depressurization of the RPV if the HCTL is exceeded.

3. Because boron injection rate is fixed, there is a need for some method of lowering reactor power (which lowers the heat addition to the suppression pool) until enough boron is injected to shutdown the reactor.

TRANS 20107-08

4. **Different Approaches Considered**

   a. Maintain a low flow condition throughout the ATWS - reactor power is low but so is boron mixing in the core.

      1) Result is a slow heat addition to the suppression pool but a high suppression pool temperature because it takes so long for the boron to get into the core and shut down the reactor.

   b. Maintain a high flow condition throughout the ATWS - boron mixing is very good but reactor power is high and a high suppression pool temperature again results.

Transparency 20107-08 does not show approach c.

   c. Combination of low flow and high flow approaches.

      1) Boron injection is commenced (already done earlier in the procedure).

If, for some reason boron injection was delayed (i.e., failure of SLC system), level would still be lowered. Efforts would then be made to get boron in as soon as possible.

      2) RPV level is lowered, reducing power by lowering natural circulation. Lower natural circulation also reduces the amount of boron mixing in the core. Suppression pool heatup is slow.

3) When Hot Shutdown Boron Weight has been injected, RPV level is raised. Increased natural circulation causes virtually simultaneous boron mixing in the core and the reactor shuts down.

4) This is the approach used in the Level/Power Control Steps.

5) This Level/Power Control approach results in the lowest possible suppression pool temperature.

TRANS 20107-09

5. When Level/Power Control is Required

a. If energy addition to the primary containment is minimal, level/power control is not needed.

1) Natural circulation is maintained to mix the boron in the core and shut down the reactor relatively quickly.

Point out the criteria decision steps.

b. If energy addition to the primary containment is substantial, level/power control IS needed.

LOCATION: G2, H2
EO 15
Reactor power 3% is an easily monitorable power level approximating decay heat. The containment can handle decay heat. 110°F is the suppression pool temperature LCO. SRV open is a path for energy to the containment. High drywell pressure signifies a leak from the RPV to the containment.

c. Three criteria together determine whether level/power control steps should be used. All three must be met.

1) Reactor power is above 3% or cannot be determined.

2) Suppression pool temperature is above 110°F.

3) SRV's open or cycling OR drywell pressure is above 1.85 psig.

2.30 / 5.43

The actuation of the Standby Liquid Control System when the requirements have been met will follow the sequence of steps listed below:

1. Turn the keylock switch to either the START A OR the START B position. At this the selected pump will start and both squib valves will fire. If the selected pump does not start then the keylock switch is to be turned to the other position in order to start the other pump. At this the Reactor Water Cleanup System is isolated by its outboard isolation valve closing.

2. There are several monitor indicators that can be used to verify that the system is operating properly. First, the red light for the selected pump will come on to indicate that it is running. Then the squib loss of continuity alarm will occur and the amber lights will go off to indicate that the squibs have fired. The discharge pressure will begin to rise and the storage tank level will begin to decrease. Also, the neutron level will begin to decrease. The discharge pressure, tank level and the neutron level are positive indications that the Standby Liquid Control solution is being injected into the reactor core.

The actuation of the system is by manual action and thus, the termination is also by manual action. Once injection has started, injection must continue until directed by Emergency Operating Procedures to terminate injection. At this point, the pumps are turned off by turning the key lock switch to the STOP position.

If SBLC System is inoperable, CRD, RWCU, RCIC, or HPCI may be used as directed by the Emergency Operating Procedures (flowcharts) to inject Boron to shut down the Reactor.

2.30/5.43

What Control Room indications are used to determine that SBLC is injecting?

b. Monitor indicators

1) Pump running light on.

2) Squib loss of continuity alarm and amber lights off.

3) Discharge pressure greater than reactor pressure.

4) Storage tank level begins to decrease.

5) Neutron level decreasing.

6) Note: 3, 4, & 5 above are positive indications that the solution is being injected.

TAKES ALL THREE SImultAneous

EO 12

3. Termination

a. Requirement:

1) When all of the solution is injected <u>or</u>,

2) When directed by Plant Hatch Emergency Operating Procedures.

b. Action: Turn key lock switch to STOP position.

EO 12

4. Local Initiation

a. All operations for local initiation are performed at the SBLC tank and equipment area.

b. Five squib valves by installing jumpers from H21-P011 to squib valve junction box.

A differential pressure transmitter indicates core plate pressure drop by measuring the pressure difference between the core inlet plenum and the space just above the core support assembly. The instrument sensing line used to determine the pressure in the core inlet plenum is the same line used for injection of the standby liquid from the SLCS. An instrument sensing line is provided for measuring pressure above the core support assembly. The differential pressure across the core plate is indicated and recorded in the MCR.

A differential pressure transmitter indicates the jet pump pressure head by measuring the difference between the pressure above the core and the pressure below the core plate.

This instrumentation permits the determination of total core flow in two ways. The first method is the readout of the summed flow measurements from all the jet pumps. The second method includes the use of jet pump prototype performance data, jet pump differential pressures, and core plate differential pressure. A temporary correlation can also be made to define core flow as a function of reactor operating power level and the readout of the head developed by the jet pump. This correlation is of a temporary nature, because it changes with a fixed core arrangement over a period of time as a result of crud buildup on the fuel. The MCR flowrate readouts of the specially calibrated jet pumps can be used to cross check the flowrate readouts of all the other jet pumps. A discrepancy in the cross-checks is reason enough to check local flow indications.

7.6.7.2.3.4 Reactor Vessel Pressure. Pressure switches, indicators, and transmitters detect reactor vessel internal pressure from the same instrument lines used for measuring reactor vessel water level.

The following list shows the subsections in which the reactor vessel pressure measuring instruments are discussed:

A. Pressure transmitters/trip units for initiating scram are discussed in section 7.2.

B. Pressure transmitters/trip units used for high-pressure coolant injection (HPCI), core spray (CS), low-pressure coolant injection (LPCI), and automatic depressurization system (ADS) are discussed in subsection 7.3.1.

C. Pressure transmitters and recorders used for feedwater control are discussed in subsection 7.7.3.

c) 50°F difference reactor coolant to loop (Unit 2 only)

4) Discuss Technical Specifications requirements.

3. Abnormal Operations

a. Loss of Recirc Pump A or B

1) Conditions

a) Annunciator Alarms

(1) DRIVE MOTOR A/B TRIP

(2) RECIRC LOOP A/B OUT OF SERVICE

(3) GENERATOR A/B LOCKOUT

b) Recirc Pump differential pressure on affected loop decreases to zero.

c) Loop flow in affected loop decreases to zero then increases slightly as reverse flow is established.

d) Total core flow decreases then stabilizes at a lower value.

e) Reactor Core Plate differential pressure decreases then stabilizes at a lower value.

2.05/5.10

f) Reactor power decreases.

2) There are no automatic actions associated with this condition.

I.05/5.10

GEORGIA POWER COMPANY PLANT E.I. HATCH	DOCUMENT TYPE: SURVEILLANCE PROCEDURE		PAGE 1 OF 23
DOCUMENT TITLE: JET PUMP INTEGRITY		DOCUMENT NUMBER: 34SV-SUV-023-2S	REVISION NO: 0
EXPIRATION DATE: N/A	APPROVALS: DEPARTMENT MANAGER J G Betsill DATE 1/12/89		EFFECTIVE DATE: NR 2-1-89
P	GEN/PLT/PLT.SUP.MGR	DATE	

1.0 OBJECTIVE

This procedure provides instructions for determining the operability of the jet pumps. This procedure satisfies all of the requirements of Unit 2 Technical Specifications, Section 4.4.1.2.

TABLE OF CONTENTS

TABLE OF CONTENTS

2.0 APPLICABILITY

2.1 This procedure applies to all Unit 2 jet pumps and both recirculation pumps.

2.2 The jet pump operability sections of this procedure shall be performed during operational conditions 1 and 2 on the following frequencies:

2.2.1 Anytime prior to exceeding 25% of rated thermal power.

2.2.2 At least once per 24 hours.

2.2.3 Following recirculation pump restarts.

2.2.4 Following any unexpected or unexplained change in core flow, jet pump loop flow, recirculation pump flow or core plate differential pressure.

2.3 In order to ensure that the Vax minicomputer software is functioning properly, jet pump operability by the Vax computer and by manual calculation must be compared at least once per cycle.

2.4 In order to maintain an accurate data base, the Updating the Jet Pump and Recirc. System Data Base section of this procedure must be performed. This may be performed at the discretion of the STA Supervisor but must be performed at least during each outage where Core Alterations are performed.

3.0 REFERENCES

3.1 40AC-REG-001-0S, Technical Specifications Surveillance Program

3.2 Technical Specifications, Unit 2, Section 3/4.1.2

3.3 FSAR, Unit 2, Figure 4.4-2

3.4 H-26003, Reactor Recirculation System P&ID
thru
H-26005

3.5 H-26001, Nuclear Boiler System P&ID, Sheets 2 and 3
and
H-26189

3.6 H-27475, Jet Pump Instrumentation Elementary Diagram
and
H-27476

3.7 H-27580, Power Range Neutron Monitoring System Elementary Drawing, Sheet 42

3.8 GE Service Information Letter 330, BWR/4 Jet Pump Beam Cracks

3.9 NUREG/CR-3052, Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure

3.10 User Manuals for Jet Pump Integrity, Calculation Program, Data Storage Program and Plotting/Least Squares Fit Program.

4.0 <u>REQUIREMENTS</u>

4.1 PERSONNEL REQUIREMENTS

One Shift Technical Advisor

4.2 MATERIAL AND EQUIPMENT

VAX Minicomputer

4.3 SPECIAL REQUIREMENTS

4.3.1 The Shift Technical Advisor (STA) will direct the test, perform procedure steps, as necessary, and sign off procedure sections completed.

4.3.2 Each person initialing this procedure will print his or her name and write his or her initials in the spaces provided at the end of the procedure.

4.3.3 Independent verification, as defined in 10AC-MGR-003-0S, Preparation and Control of Procedures, will be required for portions of this procedure as indicated by a VERIF or VERIF BY sign-off blank.

4.3.4 <u>IF</u> the Vax computer is in operation, this test must be performed using the Jet Pump Operability By the Vax computer section of this procedure. <u>IF</u> the Vax computer is not in operation, the Jet Pump Operability By Manual Calculation Section must be used.

4.3.5 <u>IF</u> one or more jet pumps are found to be outside the acceptable limits, it may be necessary to increase pump speed to above 60% of rated speed and to repeat the measurements before declaring a jet pump inoperable. Refer to Unit 2 Technical Specifications.

5.0 <u>PRECAUTIONS/LIMITATIONS</u>

5.1 PRECAUTIONS

5.1.1 Observe safety rules outlined in the E.I. Hatch Nuclear Plant Safety Standards.

5.1.2 Improper instrument calibration can severely affect the data base and cause unnecessary failures of the test.

5.1.3 IF jet pump and recirculation system data are NOT taken consistently from the same instruments, the accuracy of the test can be affected. When performing the operability test always use the same instruments as those used to obtain the data base.

5.1.4 The density correction factor is used to account for inaccuracies of D/P instruments at other than rated temperatures. Since D/P instruments measure volumetric flow rates, a mass correction is needed to maintain D/P to mass flow rate correlation.

5.1.5 At power operation above approximately 20% of rated, there is a constant addition to core flow from natural circulation. At less than 20% of rated power, the natural circulation component falls off to zero at zero steam flow. The natural circulation correction provides the magnitude of this component at various powers below 20% of rated. With this factor added back to indicated flows the data can be compared to normally expected characteristics derived from operating data.

5.2 LIMITATIONS

N/A - Not applicable to this procedure

6.0 PREREQUISITES

The Reactor Recirculation System is in operation with one or both pumps running.

7.0 PROCEDURE

7.1 DATA

Record data as per Attachment 1.

7.2 JET PUMP OPERABILITY BY THE VAX COMPUTER

7.2.1 If available, perform the Jet Pump Operability Test using the Unit Two Vax Computer.

7.2.1.1 Log on the "SAFETY" Account of the Vax Computer and type JP_CALC.

7.2.1.2 Using the data on Attachment 1 and key numbers (in parenthesis) from the previous step, run the 34SV-SUV-023 calculation program.

7.2.1.3 Attach a copy of the input and output data from the Vax Computer prior to forwarding this procedure to Document Control.

7.3 JET PUMP OPERABILITY BY MANUAL CALCULATION

7.3.1 If the Vax Computer is unavailable, perform the Jet Pump Operability Test by manual calculations.

7.3.1.1 Using the data from Attachment 1 and the graphs in Attachment 3 determine the Density Correction Factor and the Natural Circulation Correction Factor. Record these values on Attachment 2.

7.3.1.2 Perform the Jet Pump Operability Test manually by completing Attachment 2.

7.4 UPDATING THE JET PUMP AND RECIRC SYSTEM DATA BASE

> **NOTE**
>
> A temporary file is created each time the calculation program is run. That file contains that day's recorded and calculated data. This information is put into storage by running the data storage program.

> **NOTE**
>
> New graphs are generated from the data base by the plotting/least squares fit program. The graphs generated show the allowable limits for the plotted parameters. The normal expected operating range is determined by a least squares fit of the operating data $\pm$ 2 standard deviations.

7.4.1 Run the Data Storage Program for Unit 2 on the Vax until the data from all of the temporary storage files has been stored.

7.4.2 Run the Plotting/Least Squares Fit Program on the Vax.

7.4.3 Enter the new least squares fit coefficients into the Vax computer file for use by the calculation program.

7.4.4 Forward the new graphs to Document Control for revision per AC-ADM-06-0383N, Forms Control, and 10AC-MGR-003-0S, Preparation and Control of Procedures.

7.5 RESULTS

7.5.1 The STA Supervisor will review the procedure data for completeness and indicate concurrence with the test Acceptable/Non-Acceptable determination by signing Attachments 1 and/or 2 and the Retrieval Code Sheet.

7.5.2 The STA Supervisor will forward the appropriate documents, with all sign-offs complete, to Document Control for retention in accordance with 20AC-ADM-002-0S, Plant Records Management.

Is the unit in single loop operation (SLO)? _____ yes _____ no

IF the unit is in SLO, which loop is operating? _____ A _____ B
At Panel 2H11-P602 and 2H11-P603 or the Process Computer, record the following data:

(1) Core Thermal Power, OD3 or NI _____ %
(2) Core Flow, OD3 or 2B21-R613 Black Pen __________ Mlb/hr
(3) Core Plate D/P, OD3 or 2B21-R613 Red Pen X 0.3 PSID/% __________ PSID
(4) Recirculation Suction Temperature, 2B31-R650A or B __________ °F
(5) Recirculation Pump A Driving Flow, 2B31-R617 __________ KGPM
(6) Recirculation Pump B Driving Flow, 2B31-R613 __________ KGPM
(7) Loop A Jet Pump Flow, 2B21-R611A __________ Mlb/hr
(8) Loop B Jet Pump Flow, 2B21-R611B __________ Mlb/hr
(9) Recirculation Pump A Speed, 2B31-R621A __________ %
(10) Recirculation Pump B Speed, 2B31-R621B __________ %

NOTE

Turbulence in the jet pump diffuser causes the differential pressure signal to be noisy when the pump is in operation. The proper method for recording differential pressure is to take the average of the high and low readings. However, noise is the most positive indication that the jet pump is operating.

At Panel 2H11-P619, record the following jet pump differential pressures:

(16) Jet Pump #1, 2B21-R608B __________ PSID
(17) Jet Pump #2, 2B21-R608D __________ PSID
(18) Jet Pump #3, 2B21-R608F __________ PSID
(19) Jet Pump #4, 2B21-R608H __________ PSID
(20) Jet Pump #5, 2B21-R608K __________ PSID
(21) Jet Pump #6, 2B21-R608M __________ PSID
(22) Jet Pump #7, 2B21-R608P __________ PSID
(23) Jet Pump #8, 2B21-R608S __________ PSID
(24) Jet Pump #9, 2B21-R608U __________ PSID
(25) Jet Pump #10, 2B21-R608W __________ PSID
(26) Jet Pump #11, 2B21-R608A __________ PSID
(27) Jet Pump #12, 2B21-R608C __________ PSID
(28) Jet Pump #13, 2B21-R608E __________ PSID
(29) Jet Pump #14, 2B21-R608G __________ PSID
(30) Jet Pump #15, 2B21-R608J __________ PSID
(31) Jet Pump #16, 2B21-R608L __________ PSID
(32) Jet Pump #17, 2B21-R608N __________ PSID
(33) Jet Pump #18, 2B21-R608R __________ PSID
(34) Jet Pump #19, 2B21-R608T __________ PSID
(35) Jet Pump #20, 2B21-R608V __________ PSID

TITLE: JET PUMP INTEGRITY DATA

INITIALS

If in Single Loop Operation (SLO) confirm that Core Plate ΔP bandwidth is less than or equal to 1.5 psid. If Core Plate ΔP bandwidth is greater than 1.5 psid in SLO, reduce recirc pump speed until ΔP bandwidth is less than 1.5 psid and notify Engineering. If not in SLO, "N/A" this step. ______

Attach a copy of the input and output data from the Vax computer prior to forwarding this procedure to Document Control. Mark this step N/A if performing manual calculations. ______

Upon completion of this procedure, the responsible STA will record the test results and notify the Shift Supervisor that the procedure has been completed. The Shift Supervisor shall be advised of any unsatisfactory conditions.

Results: (all values within acceptance criteria)

( ) Acceptable ( ) Non-Acceptable

Non-Acceptable Conditions

____

____

Comments/Corrective Actions

____

____

Completed and/or verified by:

______________ / ______ / ______
Print Name Initials Date

______________ / ______ / ______
Print Name Initials Date

______________ / ______ / ______
Print Name Initials Date

The STA Supervisor will review the procedure data for completeness and indicate concurrence with the test acceptable/non-acceptable determination by signing below.

Results Reviewed By: ______________ ______
STA Supervisor Date

The STA Supervisor will forward this procedure, with all sign-offs complete, to Document Control for retention in accordance with 20AC-ADM-002-0S, Plant Records Management.

INITIALS

Determine the density correction factor using Recirculation Suction Temperature, key #(4), from Form ENG-0371. Record the density correction factor here, key # (11) ________. ______

VERIF ______

Determine the natural circulation correction factor using core thermal power, key #(1), from Form ENG-0371. IF core thermal power exceeds 20% of rated, use zero. Record the natural circulation correction factor here, key #(12) ________. ______

VERIF ______

> **NOTE**
>
> The following steps need only be performed on operable loops. IF the plant is in SLO, an N/A may be placed in the sign-off blanks for the steps applying to the inactive loop.

> **NOTE**
>
> Use of the word "PLOT" in the following steps does not imply that a point must be drawn on the form or that the form must be sent to Document Control. Simply locate the point on the graph and confirm that it falls within the applicable limits.

Plot recirculation pump A driving flow, key#(5) and recirculation pump A speed, key#(9), on Form ENG-0372. ______

______ VERIF

Confirm that the plot of recirculation pump A driving flow and recirculation pump A speed is within ± 5% (limit lines) of the normal range as determined by Form ENG-0372. ______

Plot the recirculation pump B driving flow, key #(6), and recirculation pump B speed key #(10) on Form ENG-0373. ______

______ VERIF

Confirm that the plot of recirculation pump B driving flow and recirculation pump B speed is within ± 5% (limit lines) of the normal range as determined by Form ENG-0373. ______

INITIALS

Determine the normalized loop A jet pump flow by multiplying the indicated loop A jet pump flow by the density correction factor and adding one half of the natural circulation factor for two loop operation or one times the natural circulation factor for single loop operation.

(__________ x __________) + (_____) (__________) = __________

loop A jet pump flow, indicated Key # (7)	density correction factor Key # (11)	1/2 or 1	natural circulation factor Key # (12)	loop A jet pump flow normalized Key # (14)

____

__________ VERIF

Plot the normalized loop A jet pump flow, key #(14), and recirculation pump A speed, key #(9), on Form ENG-0374 (two loop ops) or Form ENG-0378 (single loop ops).

____

__________ VERIF

Confirm that the plot of normalized loop A jet pump flow vs. recirculation pump A speed is within ± 5% (limit lines) of the normal range as determined by Form ENG-0374 or Form ENG-0378.

____

Determine the normalized loop B jet pump flow by multiplying the indicated loop B jet pump flow by the density correction factor and adding one half of the natural circulation factor for two loop operation or one times the natural circulation factor for single loop operation.

(__________ x __________) + (_____) (__________) = __________

loop B jet pump flow, indicated Key # (8)	density correction factor Key # (11)	1/2 or 1	natural circulation factor Key # (12)	loop B jet pump flow normalized Key # (15)

____

__________ VERIF

Plot the normalized loop B jet pump flow, key #(15), and recirculation pump B speed, key #(10), on Form ENG-0375 (two loop ops) or Form ENG-0379 (single loop ops).

____

__________ VERIF

Confirm that the plot of normalized loop B jet pump flow vs. recirculation pump B speed is within ± 5% (limit lines) of the normal range as determined by Form ENG-0375 or Form ENG-0379.

____

TITLE: JET PUMP INTEGRITY MANUAL CALCULATIONS

INITIALS

NOTE

IF both the plots of loop jet pump flow vs. recirculation pump speed and recirculation pump driving flow vs. recirculation pump speed for each loop, are within limits, all jet pumps may be considered operable and test may be signed off as satisfactory.

IF one of the plots mentioned in the previous test is out of limits continue with the next step, otherwise, proceed to the Results section.

Determine the normalized jet pump differential pressures by multiplying the indicated jet pump differential pressures, by the density correction factor. Complete the following table.

Jet Pump number	Jet Pump D/P indicated Key # next to blank	Density correction factor Key #(11)	Jet Pump D/P normalized, Key # next to blank	INITIALS	VERIF
1	((16)______	x______)	= (36)______	______	______
2	((17)______	x______)	= (37)______	______	______
3	((18)______	x______)	= (38)______	______	______
4	((19)______	x______)	= (39)______	______	______
5	((20)______	x______)	= (40)______	______	______
6	((21)______	x______)	= (41)______	______	______
7	((22)______	x______)	= (42)______	______	______
8	((23)______	x______)	= (43)______	______	______
9	((24)______	x______)	= (44)______	______	______
10	((25)______	x______)	= (45)______	______	______

TITLE: JET PUMP INTEGRITY MANUAL CALCULATIONS

INITIALS

Jet Pump number		Jet Pump D/P indicated Key # next to blank		Density correction factor Key #(11)		Jet Pump D/P normalized, Key # next to blank	INITIALS	
11	(	(26)________	x	________)	=	(46)________	________	________
12	(	(27)________	x	________)	=	(47)________	________	________
13	(	(28)________	x	________)	=	(48)________	________	________
14	(	(29)________	x	________)	=	(49)________	________	________
15	(	(30)________	x	________)	=	(50)________	________	________
16	(	(31)________	x	________)	=	(51)________	________	________
17	(	(32)________	x	________)	=	(52)________	________	________
18	(	(33)________	x	________)	=	(53)________	________	________
19	(	(34)________	x	________)	=	(54)________	________	________
20	(	(35)________	x	________)	=	(55)________	________	________

Plot the individual normalized jet pump differential pressures for loop B, keys (36) through (45), on Form ENG-0376 (two loop ops) or Form ENG-0380 (single loop ops). ________

________ VERIF

Confirm that the plot of normalized jet pump differential pressures is within ± 20% (limit lines) of the normal range of average jet pump differential pressures as determined by Form ENG-0376 or Form ENG-0380. ________

Plot the individual normalized jet pump differential pressures for loop A, keys (46) through (55), on Form ENG-0377 (two loop ops) or Form ENG-0381 (single loop ops). ________

________ VERIF

Confirm that the plot of normalized jet pump differential pressures is within ± 20% (limit lines) of the normal range of average jet pump differential pressures as determined by Form ENG-0377 or Form ENG-0381. ________

TITLE: JET PUMP INTEGRITY MANUAL CALCULATIONS

RESULTS

Upon completion of this procedure, the responsible STA will record the test results and notify the Shift Supervisor that the procedure has been completed. The Shift Supervisor shall be advised of any unsatisfactory conditions.

Results: (all values within acceptance criteria)

( ) Acceptable ( ) Non-Acceptable

Non-Acceptable Conditions

____

____

Comments/Corrective Actions

____

____

Completed and/or verified by:

____________________ / ________ / ________
Print Name Initials Date

____________________ / ________ / ________
Print Name Initials Date

____________________ / ________ / ________
Print Name Initials Date

The STA Supervisor will review the procedure data for completeness and indicate concurrence with the test acceptable/non-acceptable determination by signing below.

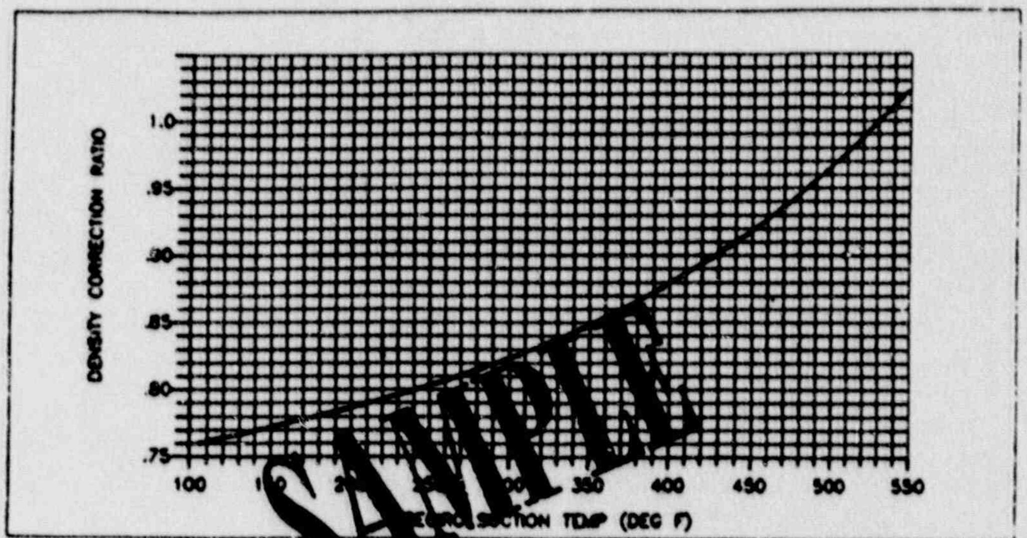
Results Reviewed By: ____________________ __________
STA Supervisor Date

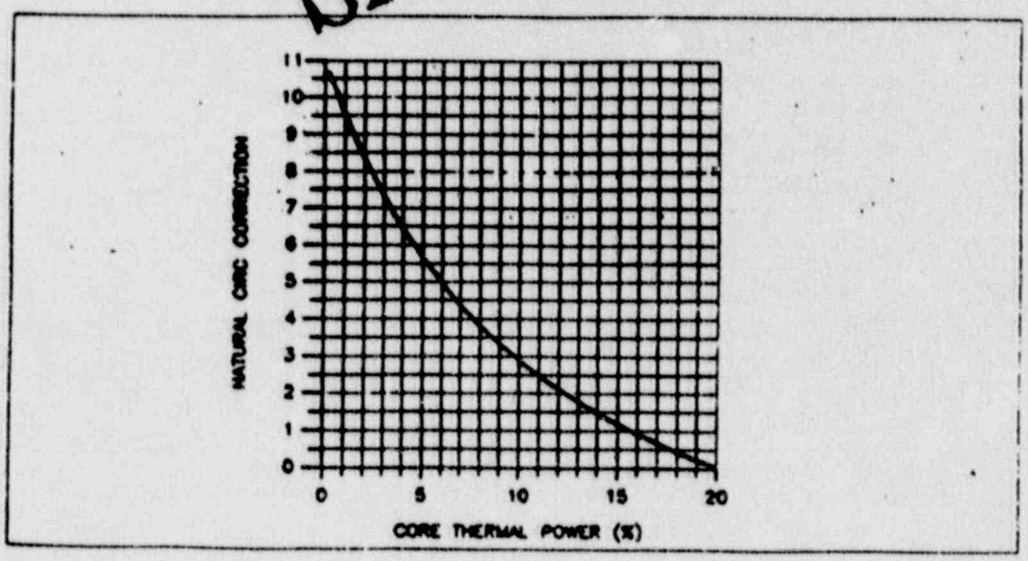
The STA Supervisor will forward this procedure, with all sign-offs complete, to Document Control for retention in accordance with 20AC-ADM-002-0S, Plant Records Management.

TITLE: DENSITY CORRECTION AND NATURAL CIRCULATION FACTORS

(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY
PLANT E.I. HATCH | PAGE 1 OF 1
FORM TITLE:
DENSITY CORRECTION AND NATURAL CIRCULATION FACTORS





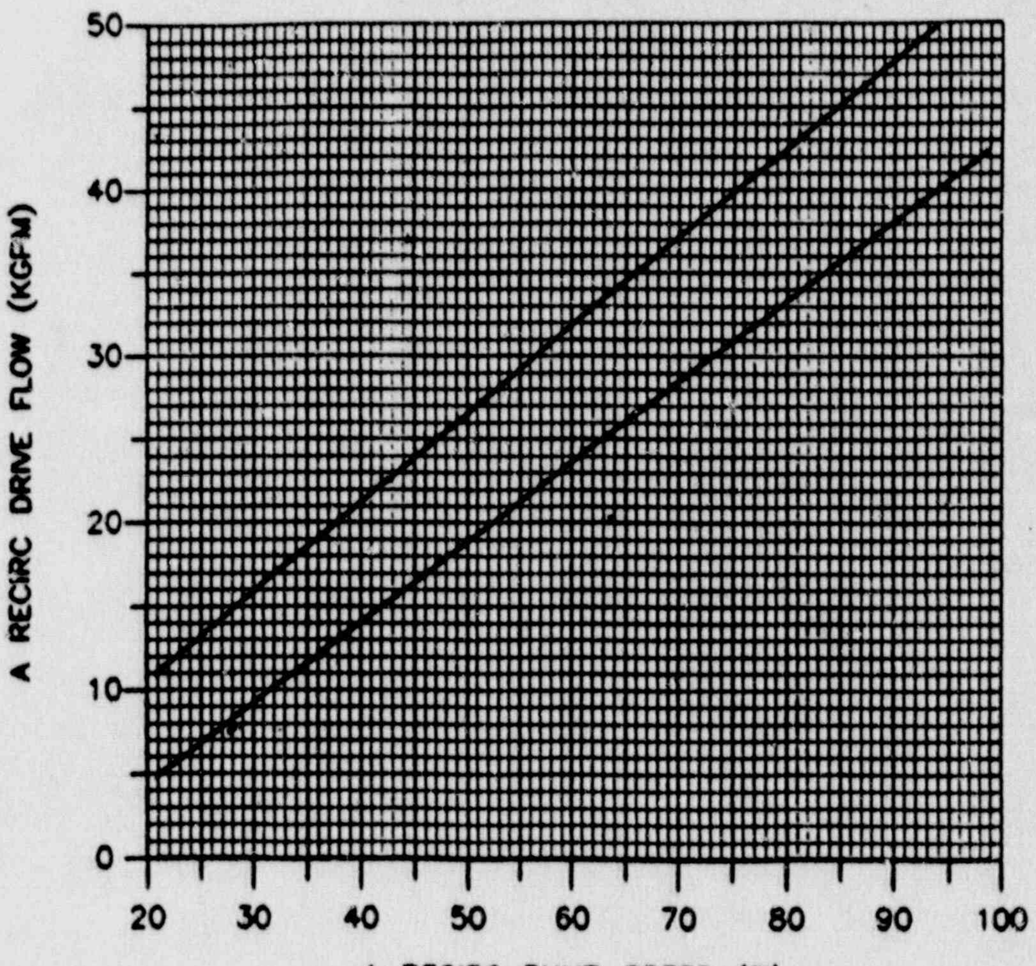
DOCUMENT TITLE:	DOCUMENT NUMBER:	REVISION NO:
JET PUMP INTEGRITY	34SV-SUV-023-2S	0
ATTACHMENT 4		ATTACHMENT PAGE:
TITLE: RECIRCULATION PUMP A DRIVING FLOW vs. SPEED		1 OF 1

(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY
PLANT E.I. HATCH
PAGE 1 OF 1
FORM TITLE:
RECIRCULATION PUMP A DRIVING FLOW vs. SPEED

UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88
(TWO LOOP OPERATION)



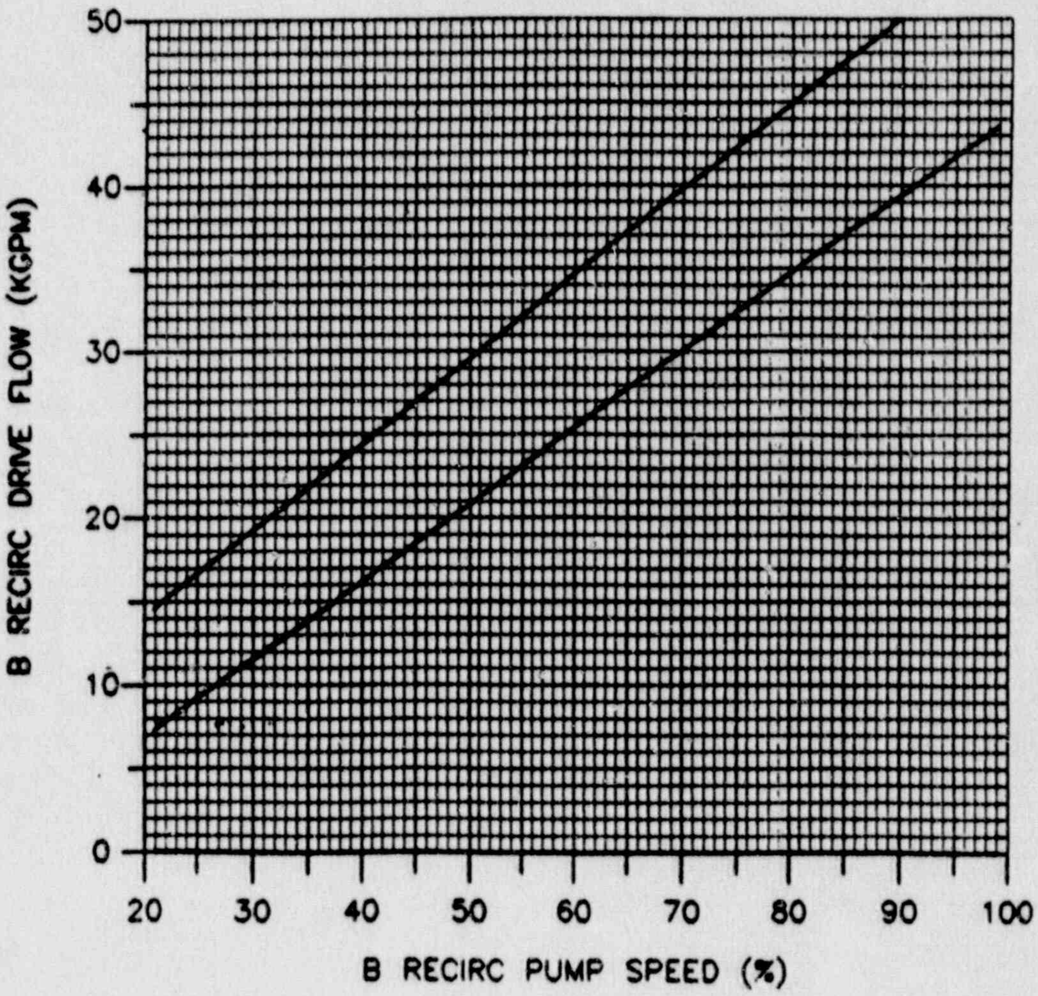
TITLE: RECIRCULATION PUMP B DRIVING FLOW vs. SPEED

(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY
PLANT E.I. HATCH
PAGE 1 OF 1
FORM TITLE:
RECIRCULATION PUMP B DRIVING FLOW vs. SPEED

UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88
(TWO LOOP OPERATION)



DOCUMENT TITLE:	DOCUMENT NUMBER:	REVISION NO:
JET PUMP INTEGRITY	34SV-SUV-023-2S	0

ATTACHMENT 6 — ATTACHMENT PAGE: 1 OF 1

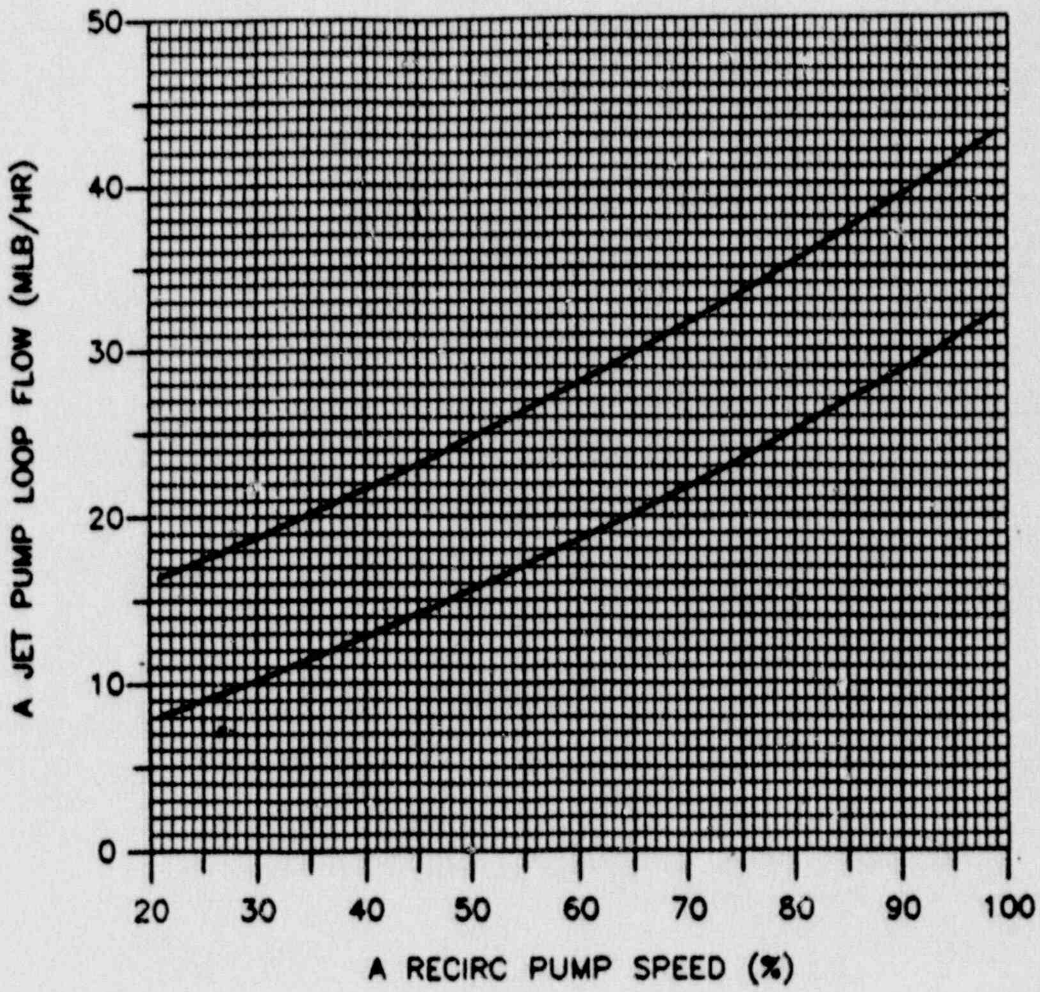
TITLE: LOOP A JET PUMP FLOW vs. RECIRCULATION PUMP A SPEED

(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY
PLANT E.I. HATCH
PAGE 1 OF 1
FORM TITLE:
LOOP A JET PUMP FLOW vs. RECIRCULATION PUMP A SPEED

UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88
(TWO LOOP OPERATION)



DOCUMENT TITLE: JET PUMP INTEGRITY	DOCUMENT NUMBER: 34SV-SUV-023-2S	REVISION NO: 0
ATTACHMENT 7		ATTACHMENT PAGE: 1 OF 1

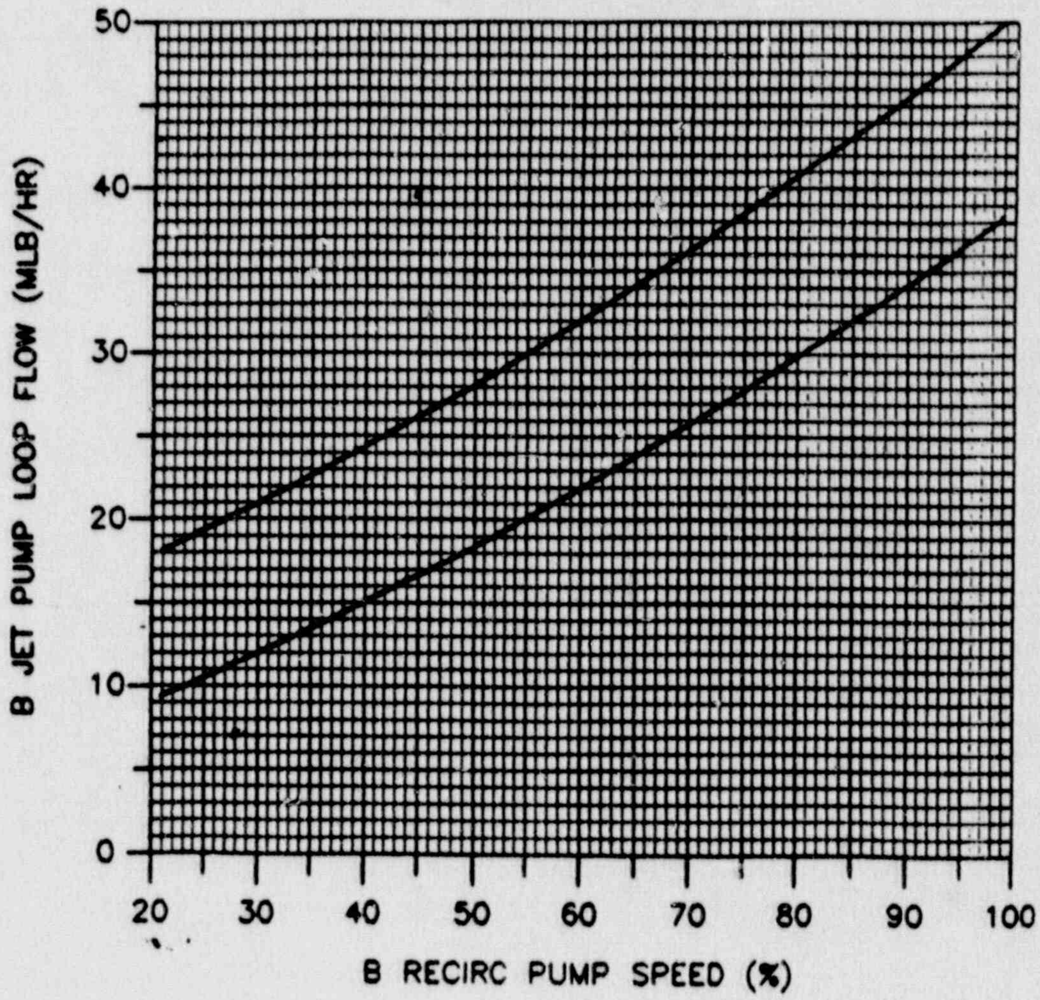
TITLE: LOOP B JET PUMP FLOW vs. RECIRCULATION PUMP B SPEED

(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY PLANT E.I. HATCH	PAGE 1 OF 1
FORM TITLE: LOOP B JET PUMP FLOW vs. RECIRCULATION PUMP B SPEED	

UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88
(TWO LOOP OPERATION)



ENG-0375 Rev. 1

G16.30

34SV-SUV-023-2S

DOCUMENT TITLE:	DOCUMENT NUMBER:	REVISION NO:
JET PUMP INTEGRITY	34SV-SUV-023-2S	0

ATTACHMENT 8

ATTACHMENT PAGE: 1 OF 1

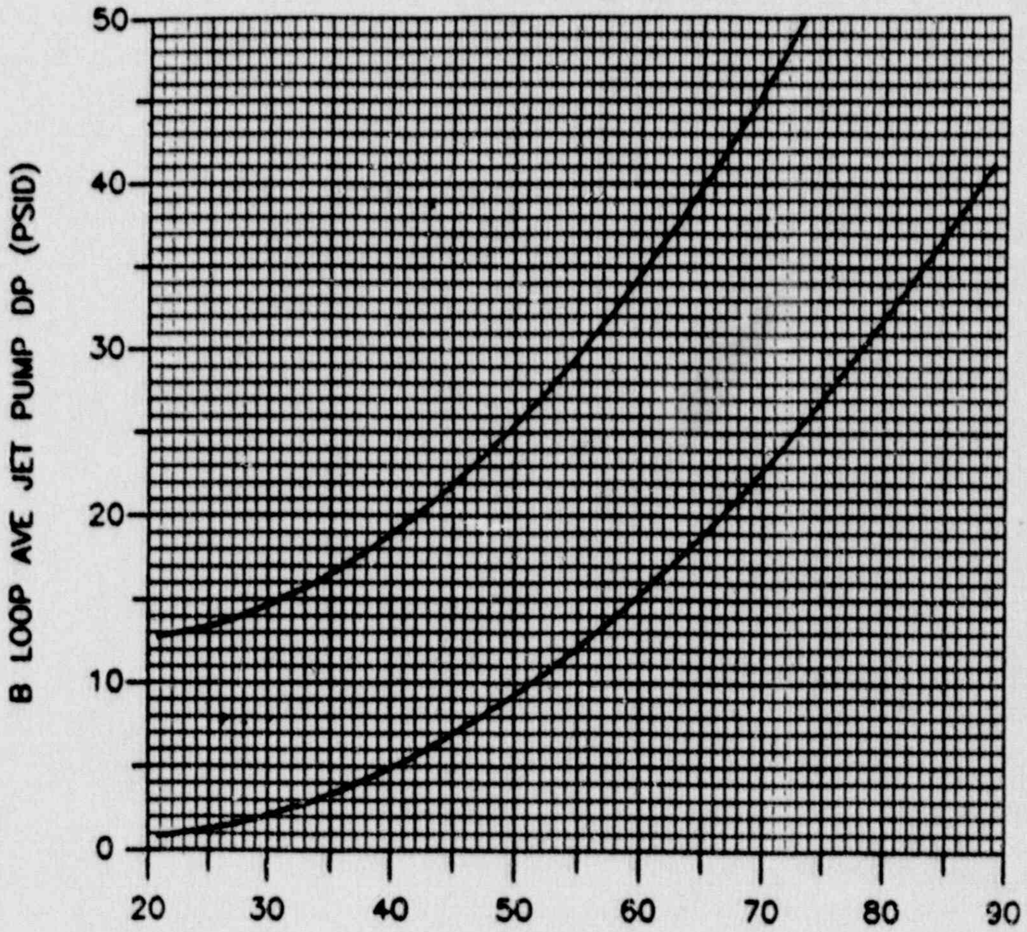
TITLE: LOOP B AVERAGE JET PUMP D/P vs. CORE FLOW

(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY
PLANT E.I. HATCH
PAGE 1 OF 1
FORM TITLE:
LOOP B AVERAGE JET PUMP D/P vs. CORE FLOW

UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88
(TWO LOOP OPERATION)



ENG-0376 Rev. 1

G16.30

34SV-SUV-023-2S

ATTACHMENT 9 | ATTACHMENT PAGE: 1 OF 1

TITLE: LOOP A AVERAGE JET PUMP D/P vs. CORE FLOW

(TYPICAL - USE LATEST REVISION)

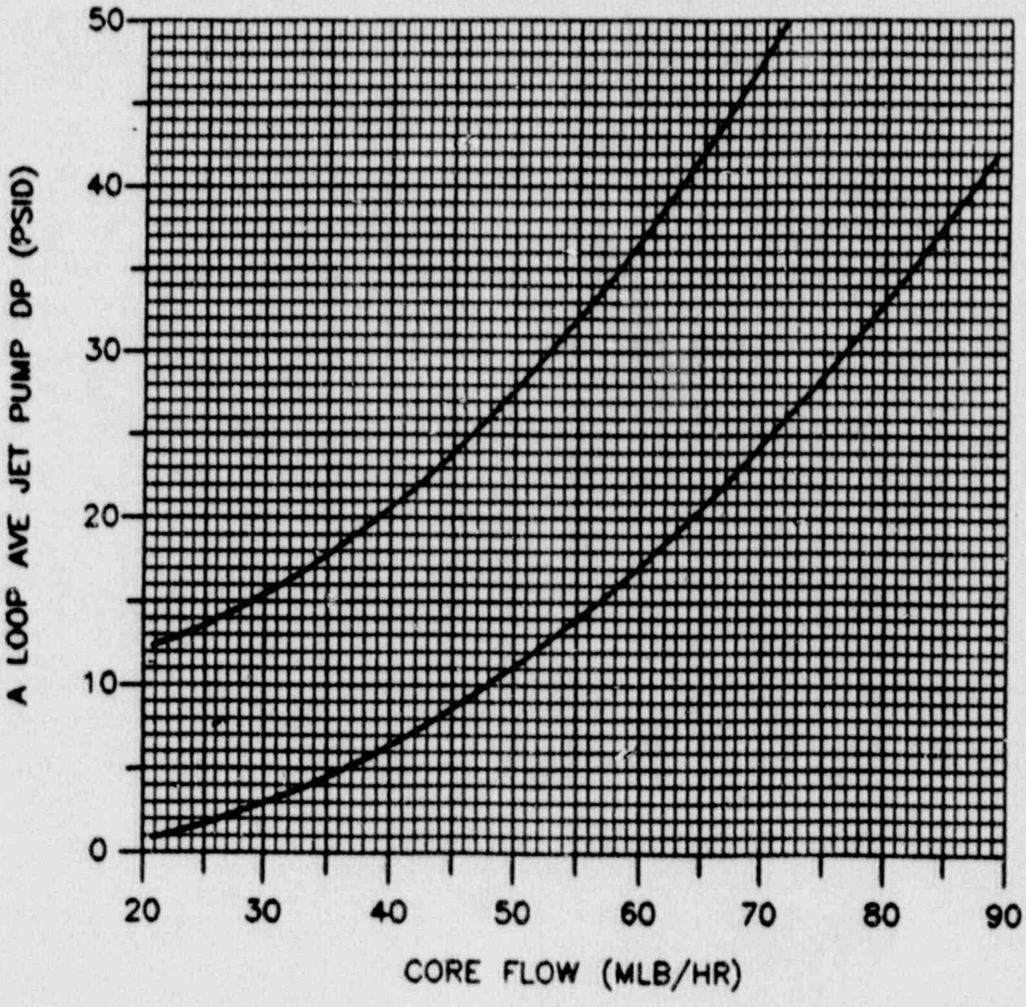
GEORGIA POWER COMPANY
PLANT E.I. HATCH

PAGE 1 OF 1

FORM TITLE:
LOOP A AVERAGE JET PUMP D/P vs. CORE FLOW

UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88
(TWO LOOP OPERATION)



DOCUMENT TITLE:	DOCUMENT NUMBER:	REVISION NO:
JET PUMP INTEGRITY	34SV-SUV-023-2S	0

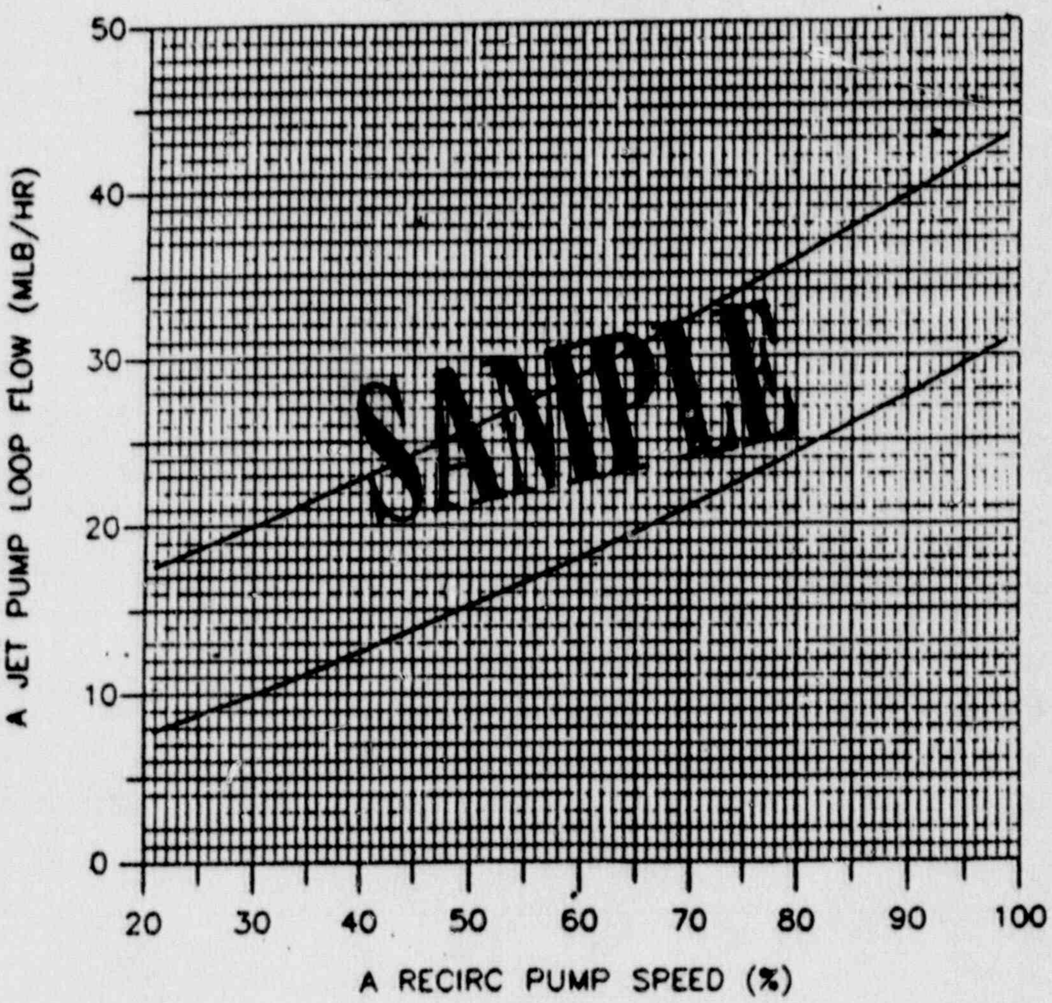
ATTACHMENT 10 — ATTACHMENT PAGE: 1 OF 1

TITLE: SINGLE LOOP A JET PUMP FLOW vs. RECIRCULATION PUMP A SPEED

(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY
PLANT E.I. HATCH
PAGE 1 OF 1
FORM TITLE:
SINGLE LOOP A JET PUMP FLOW vs. RECIRCULATION PUMP A SPEED

UNIT 2
DATA FROM CYCLE 07
THRU 5-AUG-87
(SINGLE LOOP OPERATION)



ENG-0378 Rev. 0 G16.30 34SV-SUV-023-2S

ATTACHMENT 11

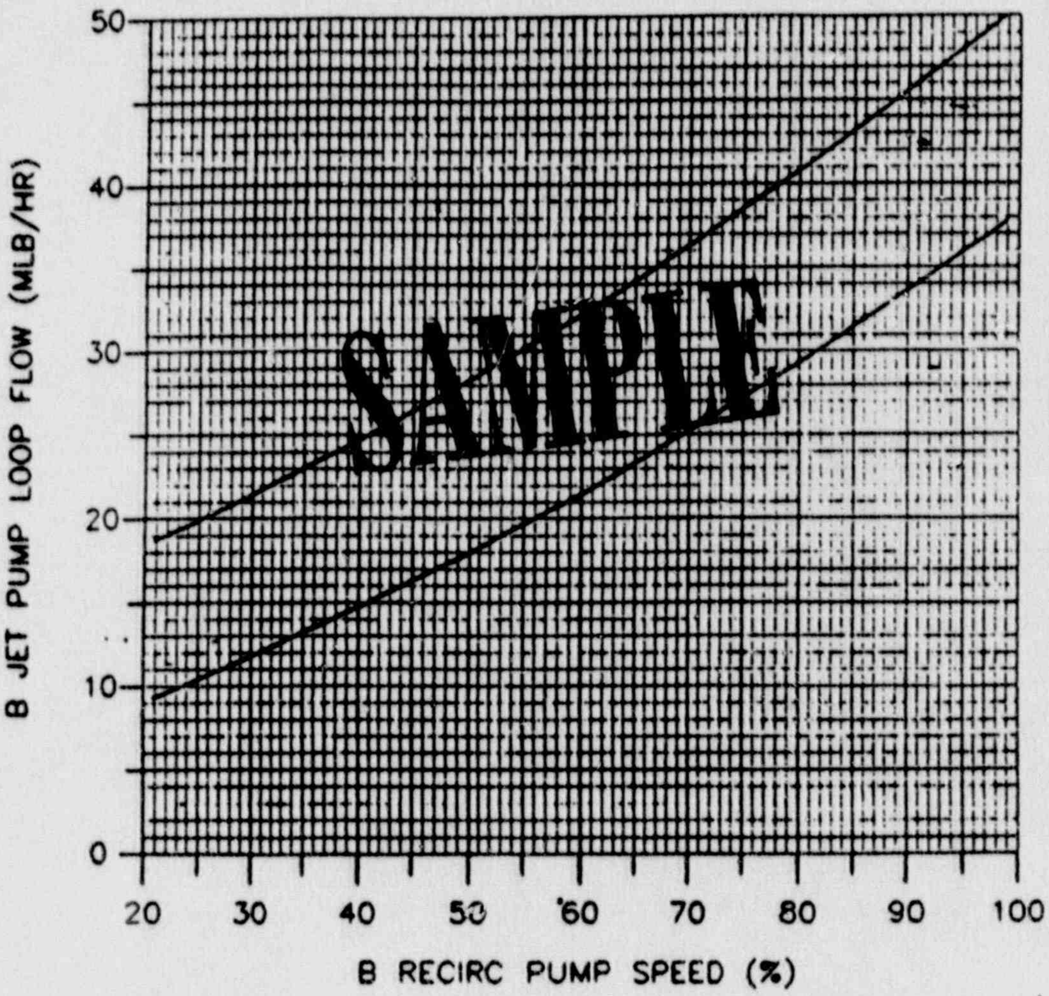
ATTACHMENT PAGE: 1 OF 1

TITLE: SINGLE LOOP B JET PUMP FLOW vs. RECIRCULATION PUMP B SPEED

(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY
PLANT E.I. HATCH
PAGE 1 OF 1
FORM TITLE:
SINGLE LOOP B JET PUMP FLOW vs. RECIRCULATION PUMP B SPEED

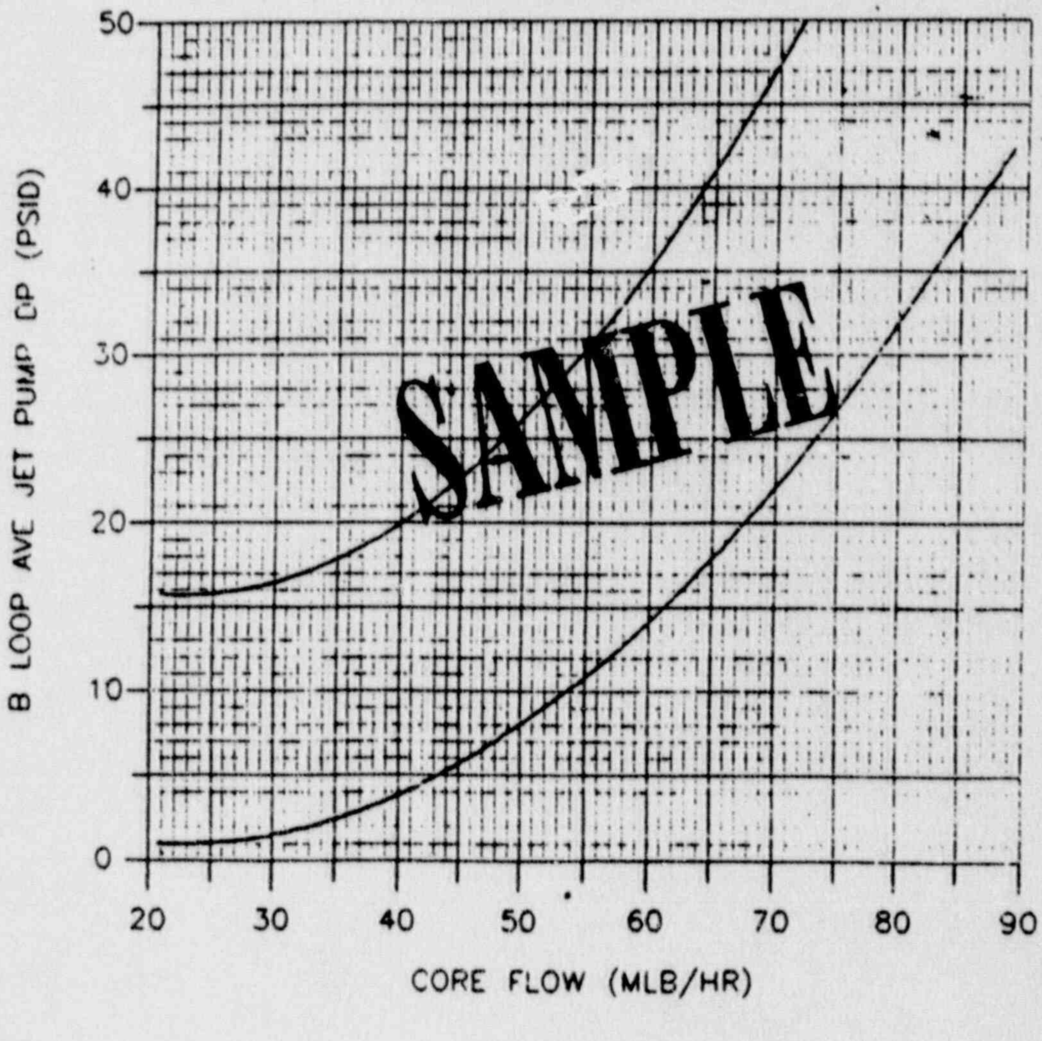
UNIT 2
DATA FROM CYCLE 07
THRU 5-AUG-87
(SINGLE LOOP OPERATION)



(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY PLANT E.I. HATCH	PAGE 1 OF 1
FORM TITLE: SINGLE LOOP B AVERAGE JET PUMP D/P vs. CORE FLOW	

UNIT 2
DATA FROM CYCLE 07
THRU 5-AUG-87
(SINGLE LOOP OPERATION)



ENG-0380 Rev. 0 G16.30 34SV-SUV-023-2S

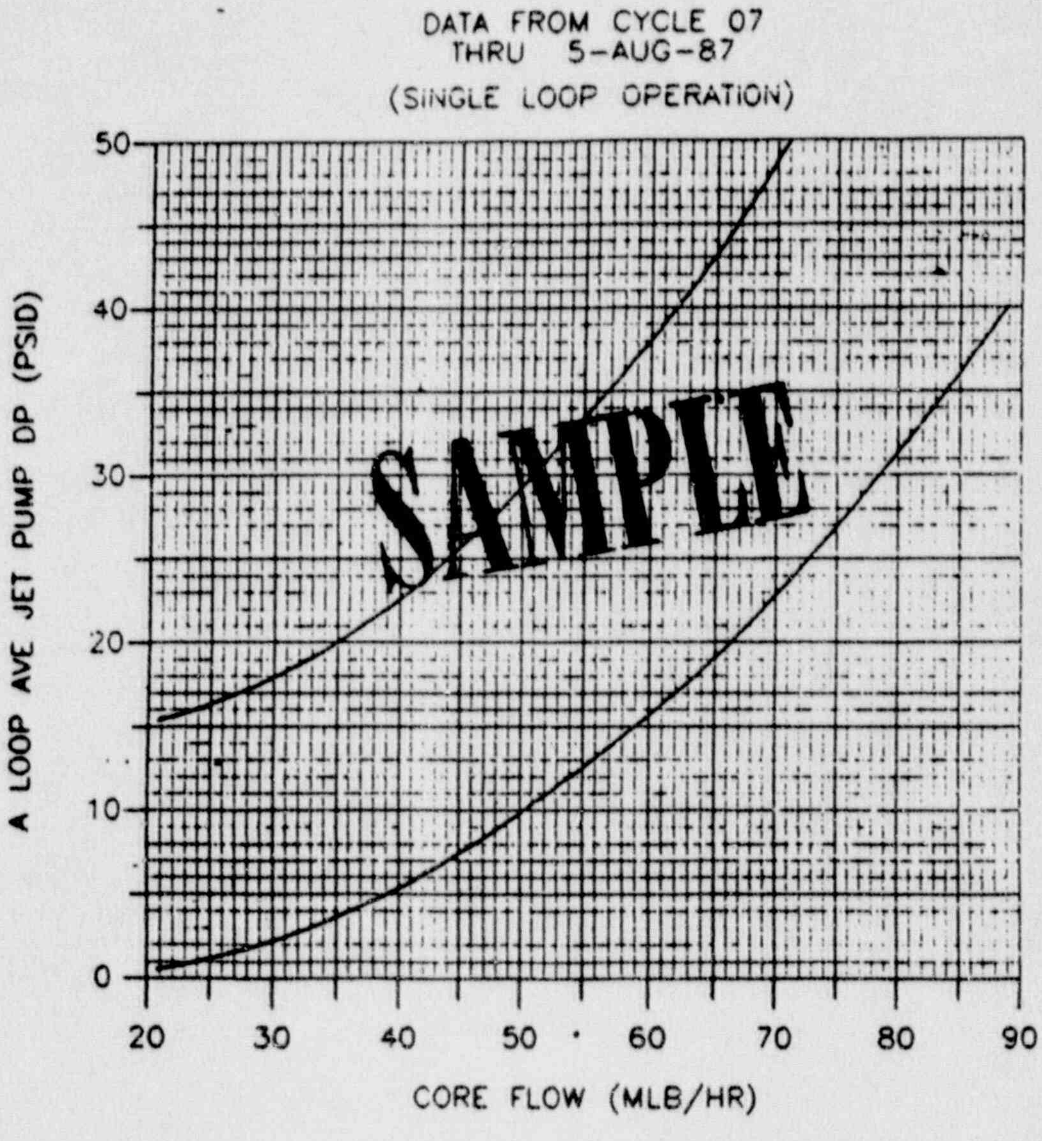
ATTACHMENT 13

ATTACHMENT PAGE: 1 OF 1

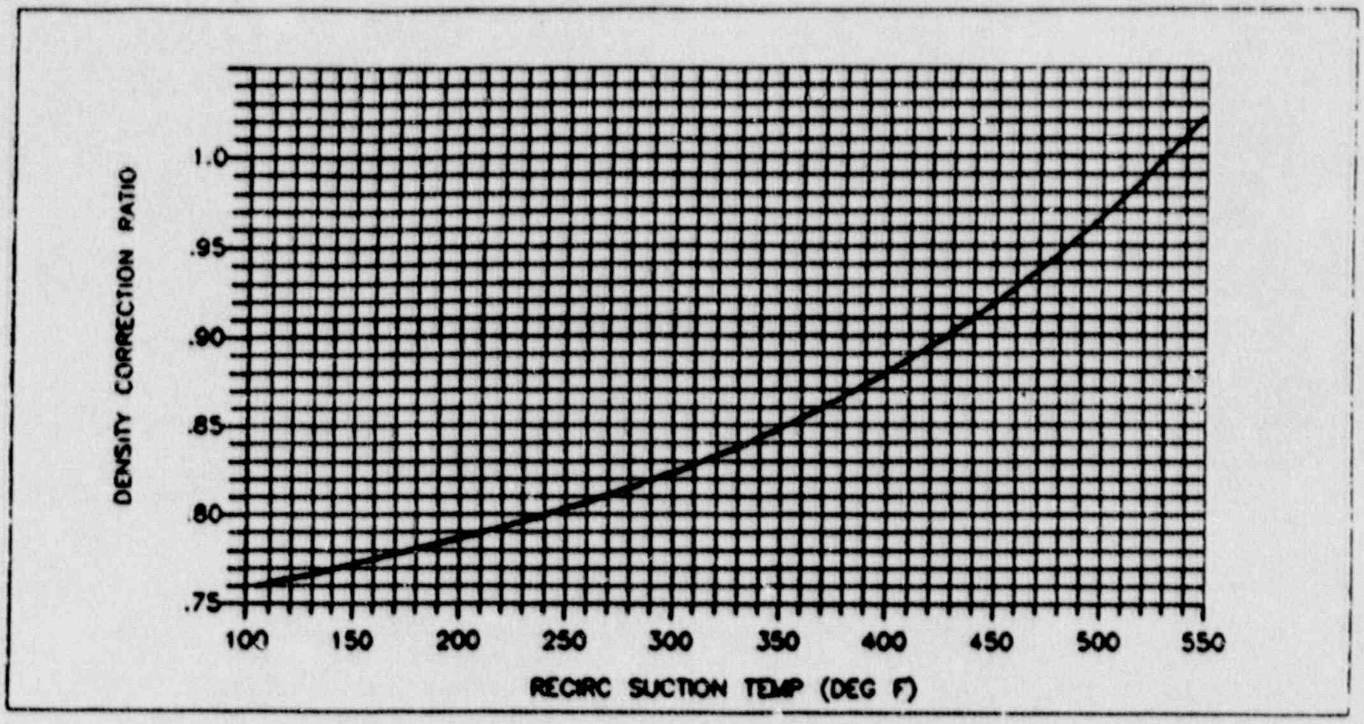
TITLE: SINGLE LOOP A AVERAGE JET PUMP D/P vs. CORE FLOW

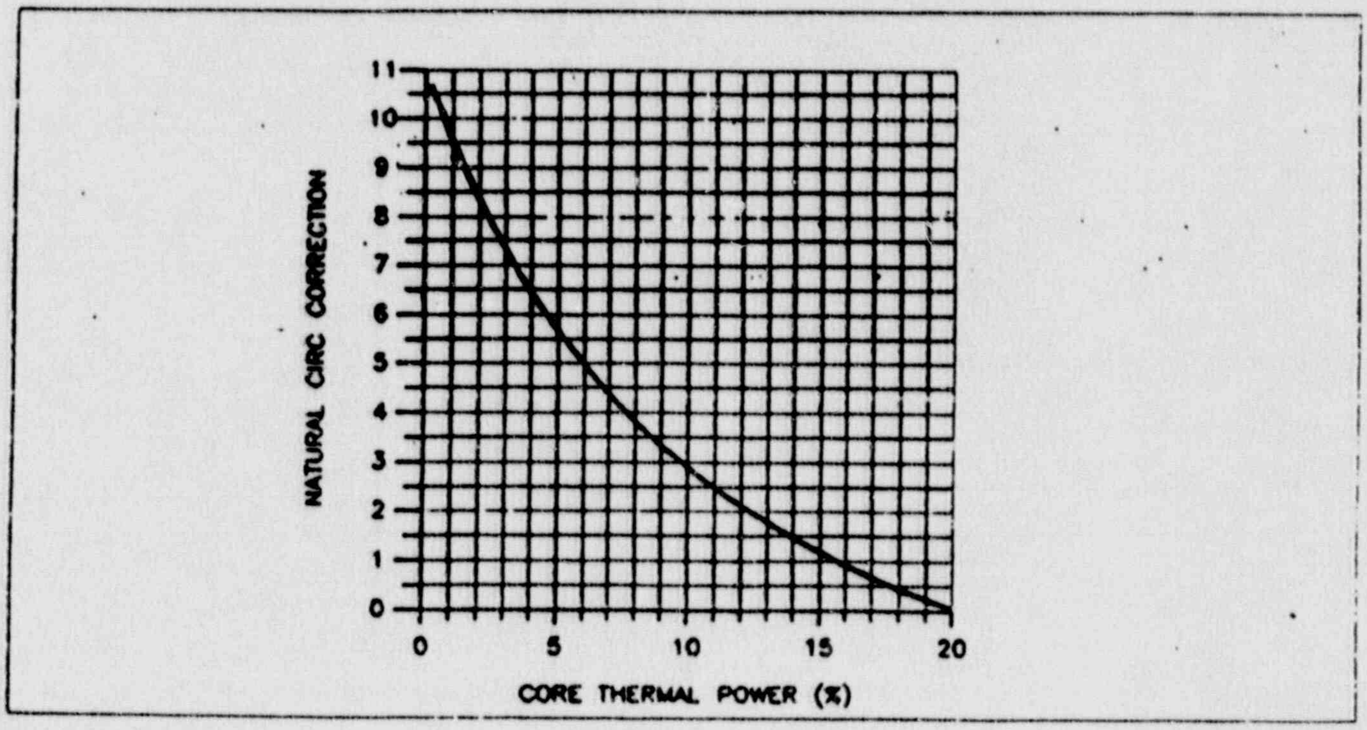
(TYPICAL - USE LATEST REVISION)

GEORGIA POWER COMPANY
PLANT E.I. HATCH
PAGE 1 OF 1
FORM TITLE:
SINGLE LOOP A AVERAGE JET PUMP D/P vs. CORE FLOW



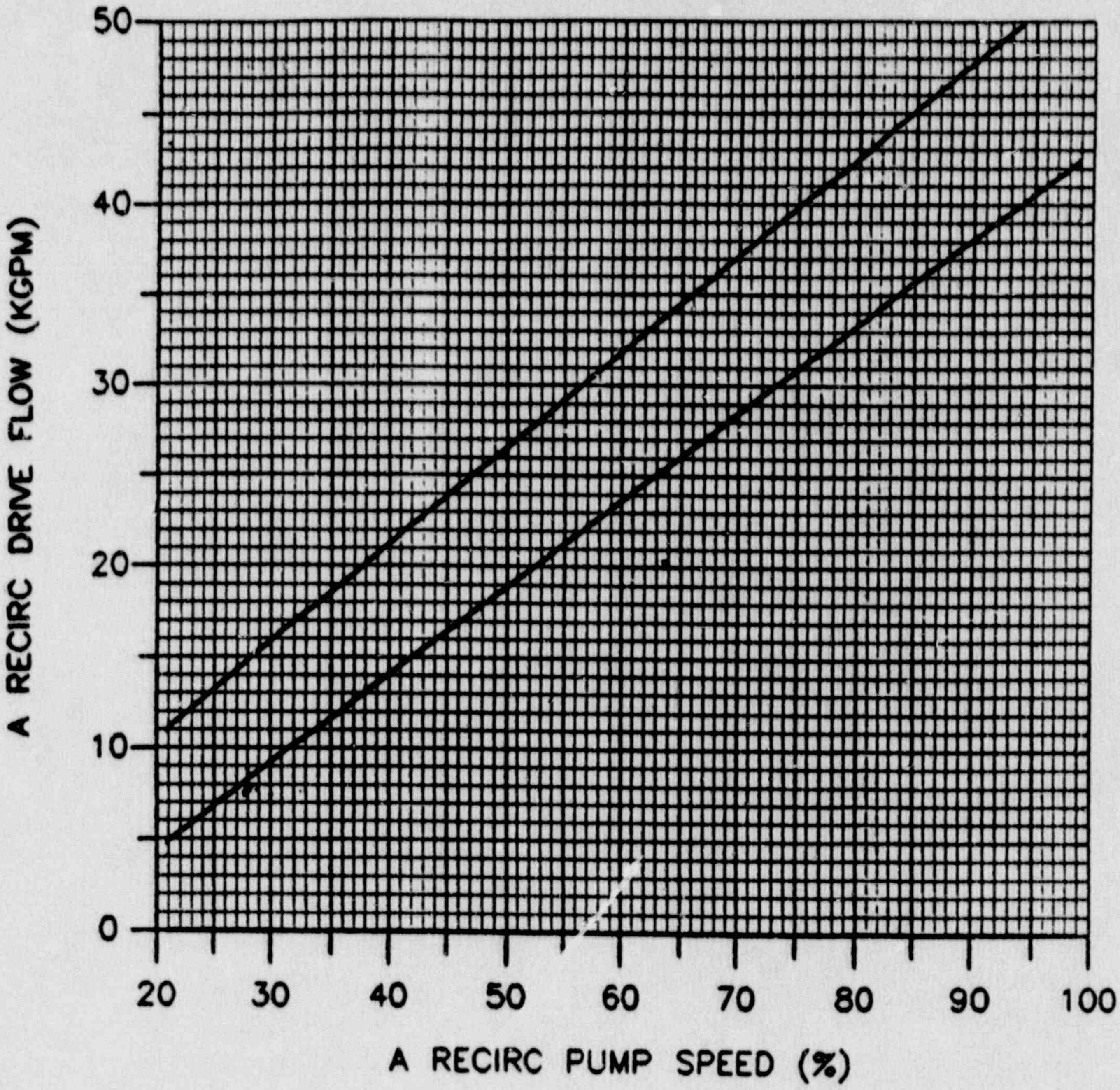
FORM TITLE:
DENSITY CORRECTION AND NATURAL CIRCULATION FACTORS





UNIT 2

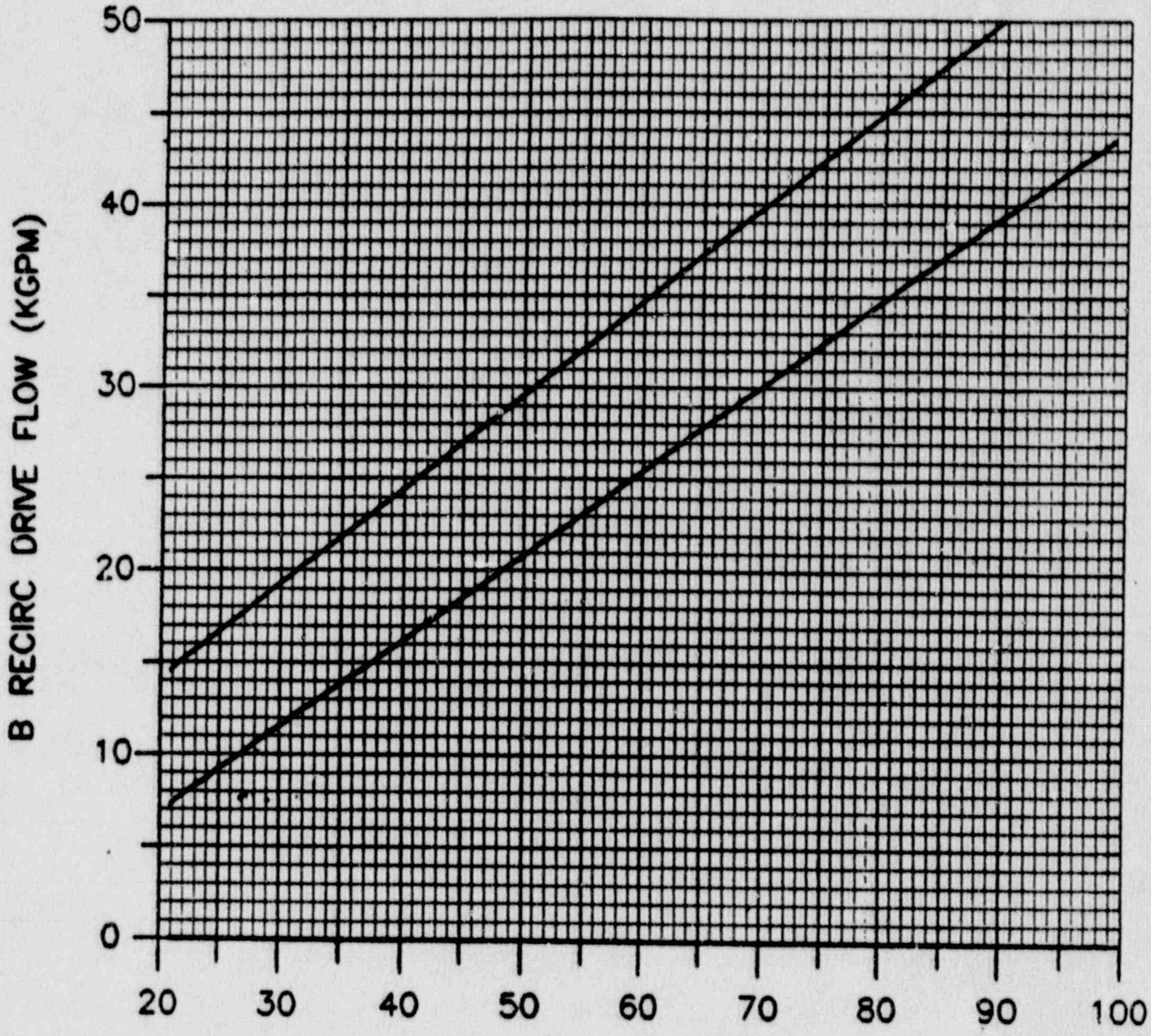
DATA FROM CYCLE 07
THRU 13-JAN-88
(TWO LOOP OPERATION)



FORM TITLE:
RECIRCULATION PUMP B DRIVING FLOW vs. SPEED

UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88
(TWO LOOP OPERATION)

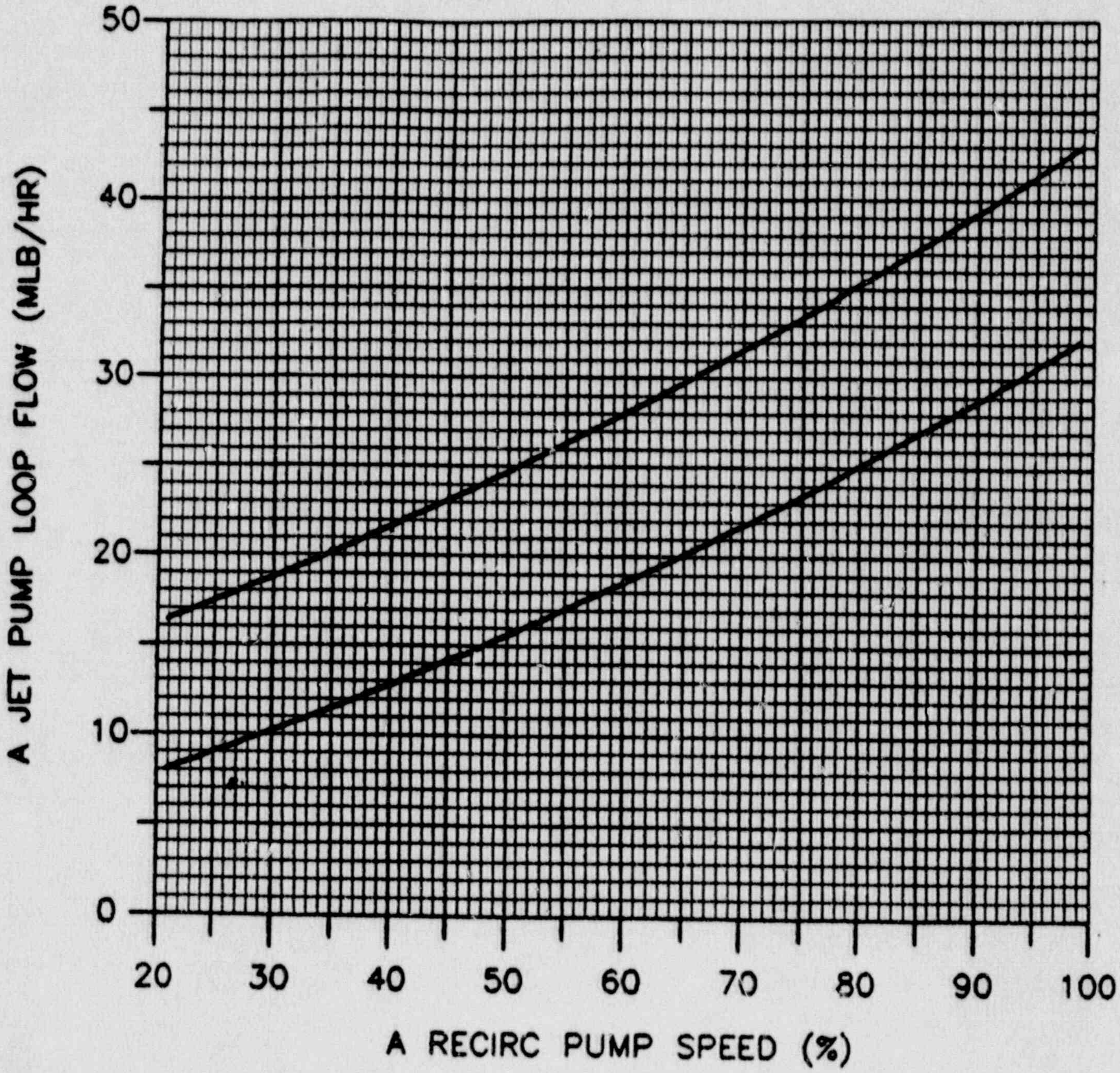


FORM TITLE:
LOOP A JET PUMP FLOW vs. RECIRCULATION PUMP A SPEED

UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88

(TWO LOOP OPERATION)

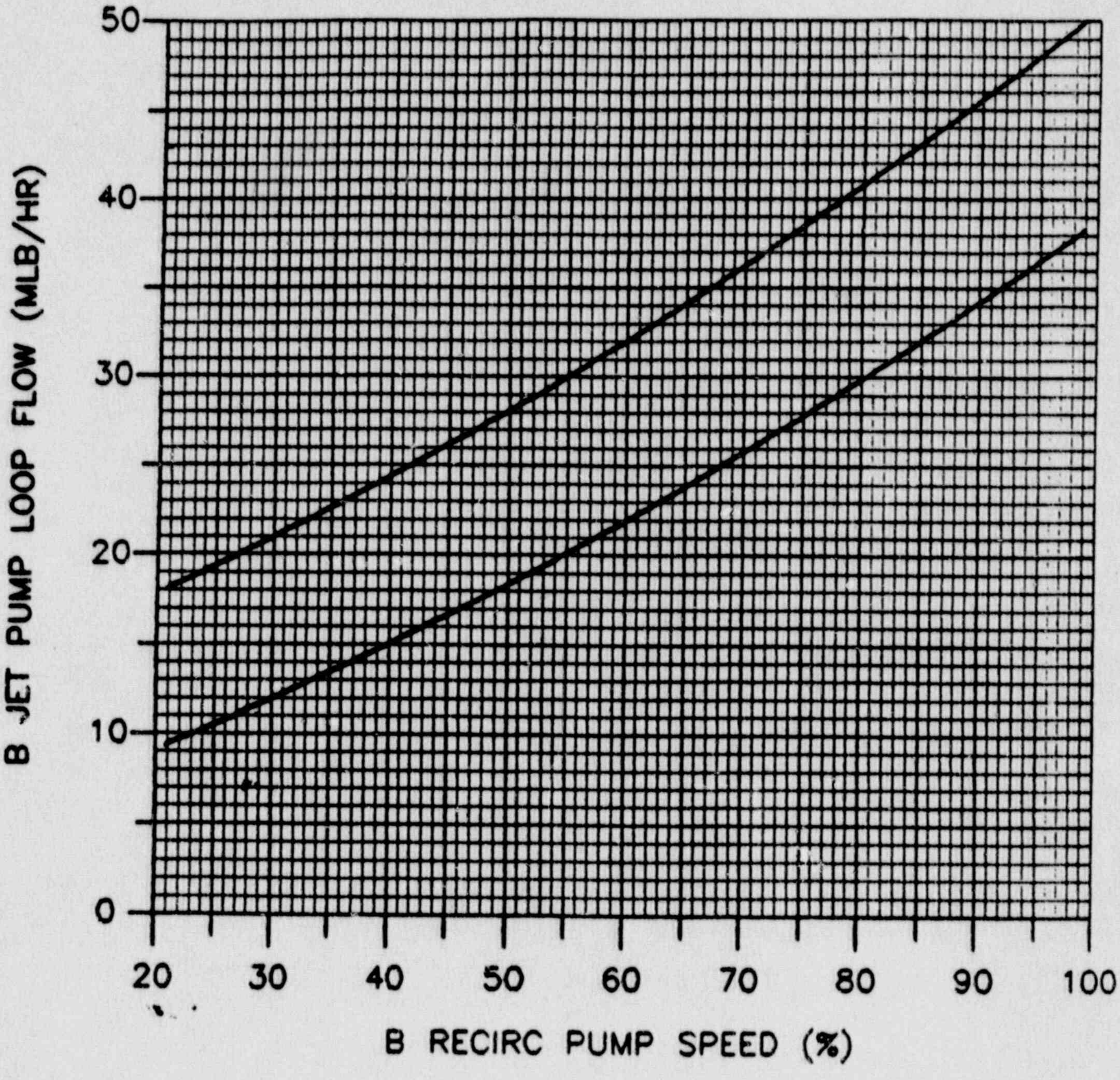


FORM TITLE:
LOOP B JET PUMP FLOW vs. RECIRCULATION PUMP B SPEED

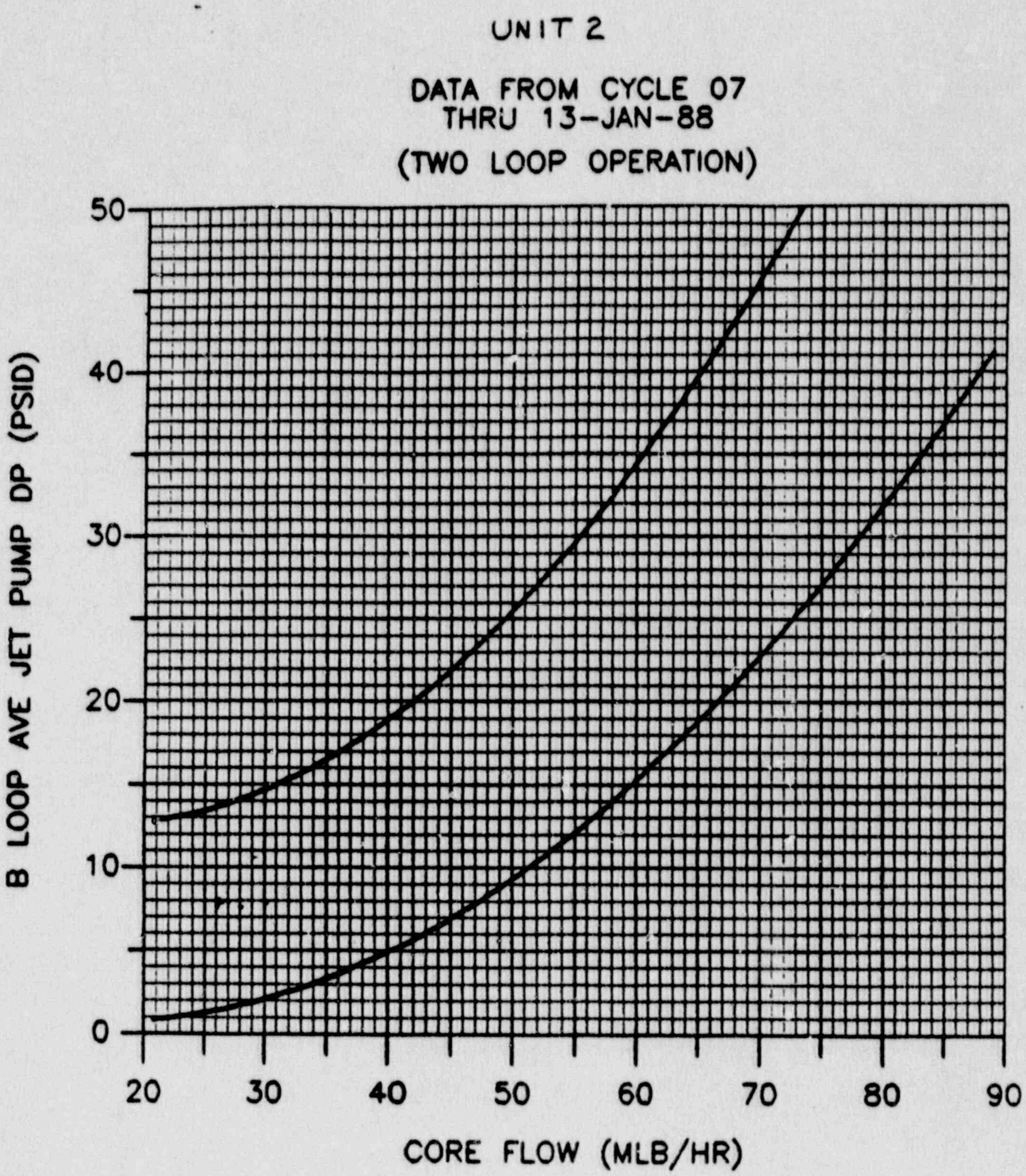
UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88

(TWO LOOP OPERATION)



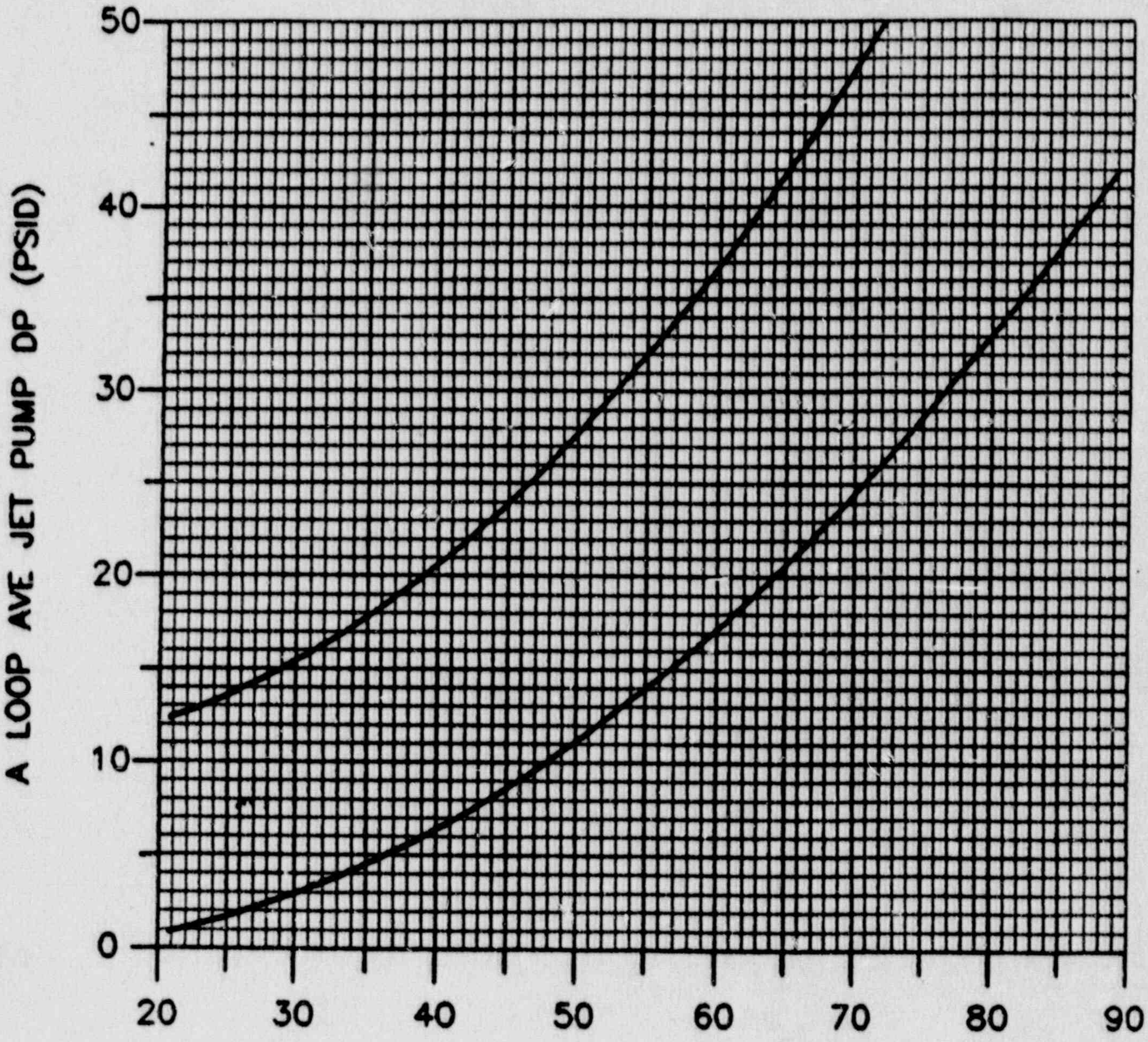
FORM TITLE:
LOOP B AVERAGE JET PUMP D/P vs. CORE FLOW



FORM TITLE:
LOOP A AVERAGE JET PUMP D/P vs. CORE FLOW

UNIT 2

DATA FROM CYCLE 07
THRU 13-JAN-88
(TWO LOOP OPERATION)

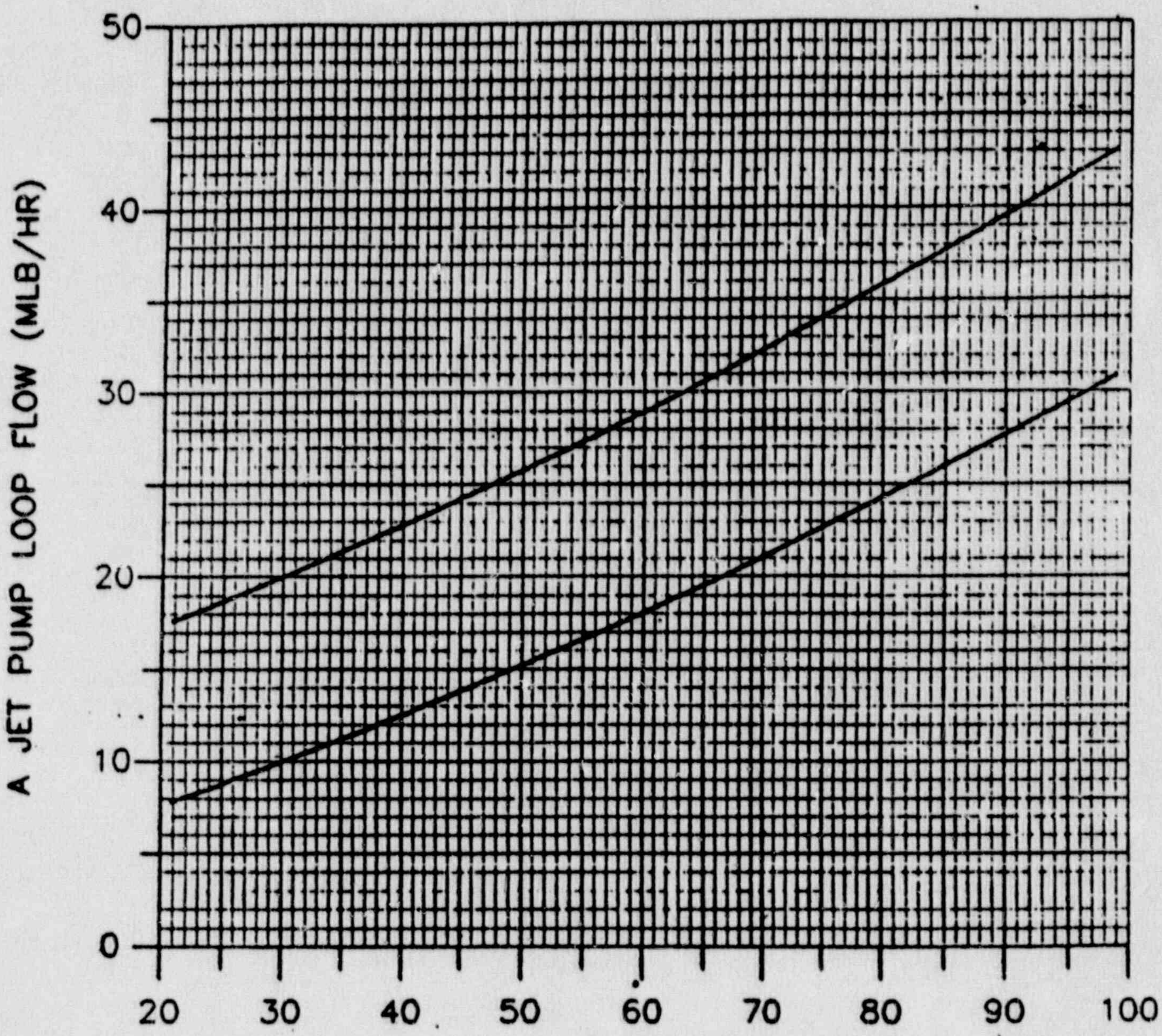


FORM TITLE:
SINGLE LOOP A JET PUMP FLOW vs. RECIRCULATION PUMP A SPEED

UNIT 2

DATA FROM CYCLE 07
THRU 5-AUG-87

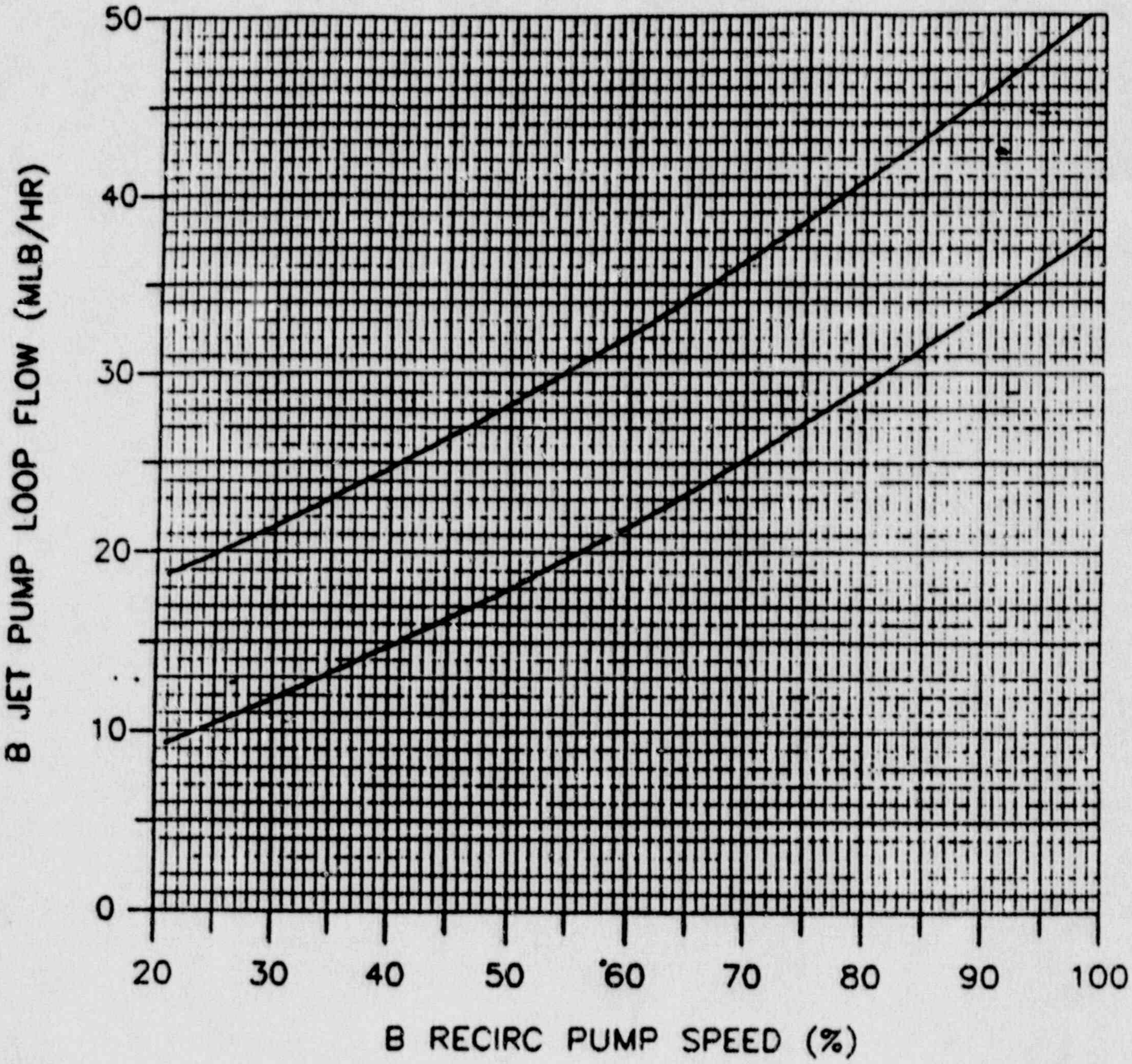
(SINGLE LOOP OPERATION)



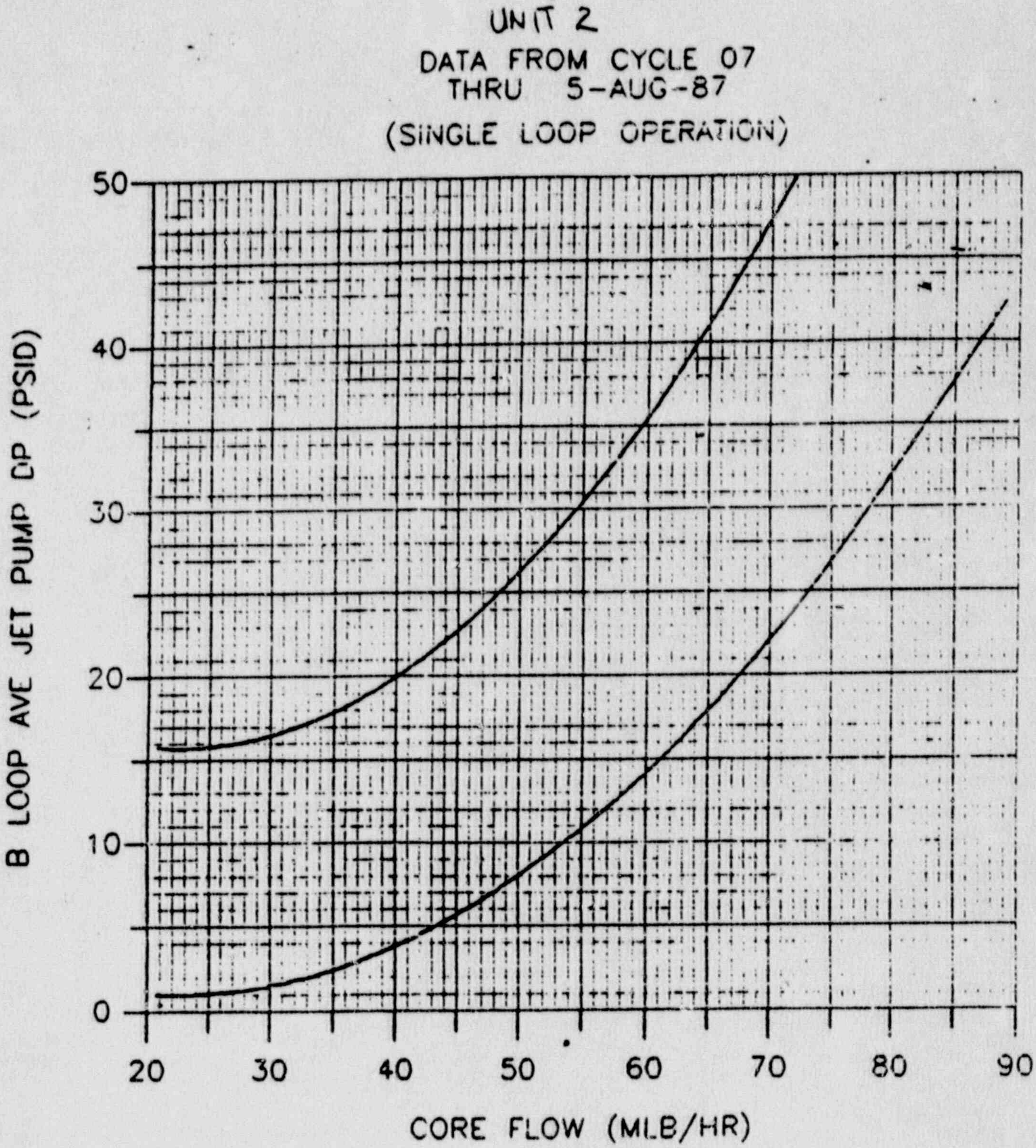
UNIT 2

DATA FROM CYCLE 07
THRU 5-AUG-87

(SINGLE LOOP OPERATION)



FORM TITLE:
SINGLE LOOP B AVERAGE JET PUMP D/P vs. CORE FLOW



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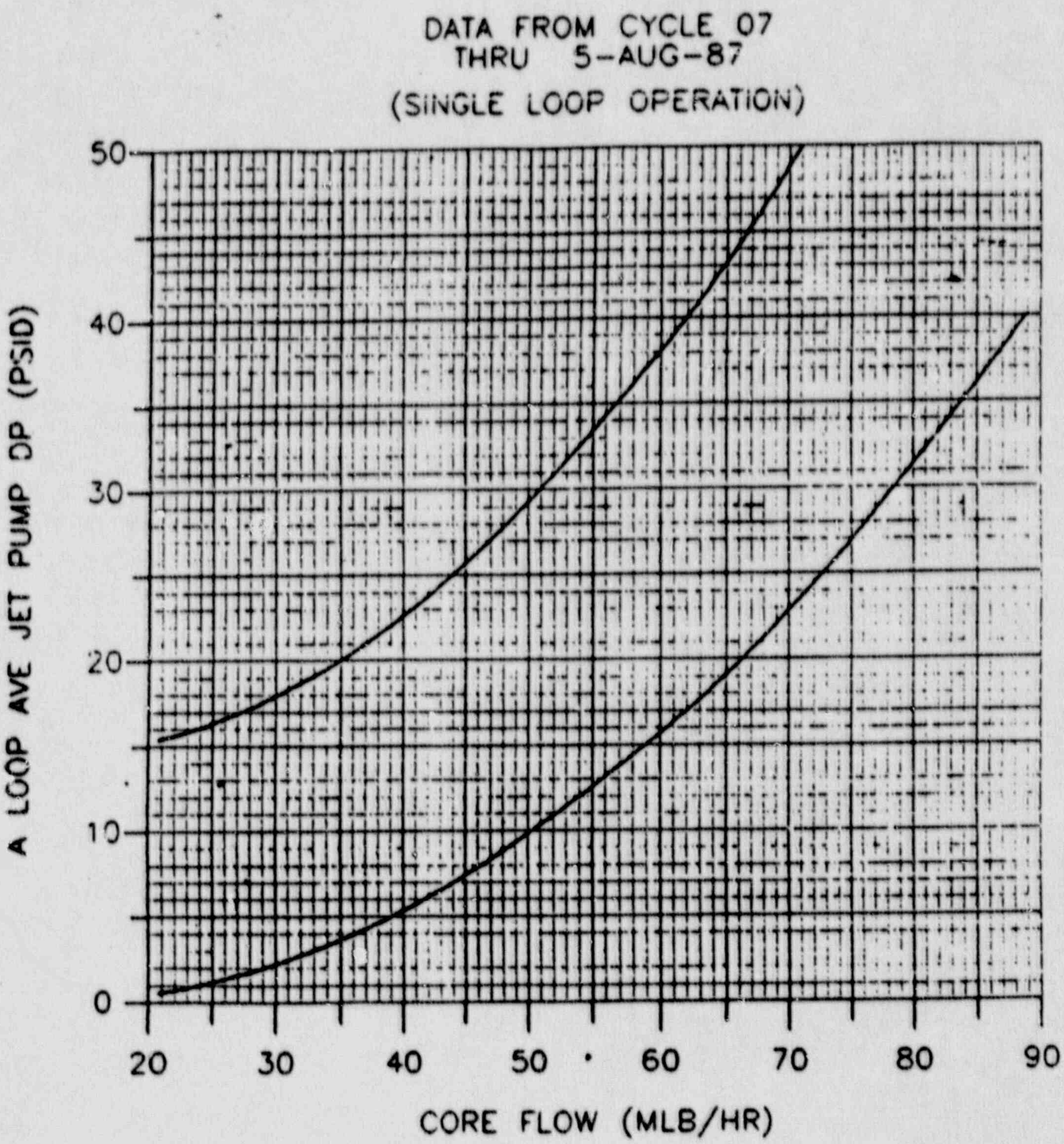
SINGLE LOOP A AVERAGE JET PUMP D/P vs. CORE FLOW



Recirculation pumps are tripped in Section RC/Q before Contingency 7 is entered. **However, substantial natural circulation flow can still exist through internal flowpaths, proportional to the difference in density between the water inside and outside the shroud and the difference in height between them as seen in Figure 17.1.2. The operator has little control over fluid densities, but can affect the driving head by adjusting the height differential. As shown in Figure 17.1.3, core flow, and hence reactor power, is a function of RPV water level.** Contingency 7 utilizes this relationship by lowering RPV water level to reduce reactor power, and thus reduce the heat input to the suppression pool.

If flow is decreased sufficiently to reduce power to the desired level, boron mixing efficiency is so low that the boron essentially stagnates in the lower plenum. Once the required amount of boron has been injected, core flow must be increased to rapidly distribute the boron throughout the core. This may be accomplished by raising RPV water level until natural circulation is reestablished. (Recirculation pumps cannot be used because of the low RPV water level interlock, which is designed to prevent cavitation). Test data indicates boron mixing will occur virtually simultaneously with the water level increases.

The RPV water level is decreased below the elevation capable of supporting natural circulation through the steam separators, as shown in Figure 17.1.2, so only flowpaths internal to the shroud will remain. Under these conditions, extrapolation of existing computer codes indicate reactor power will stabilize at 8%. This is the "Reactor Flow Stagnation Power" referred to in Contingency 7. Contingency 7 may be exited to normal operating procedures, to Contingency 5 if Alternate Shutdown Cooling is required, or to Contingency 6 if RPV flooding is required.

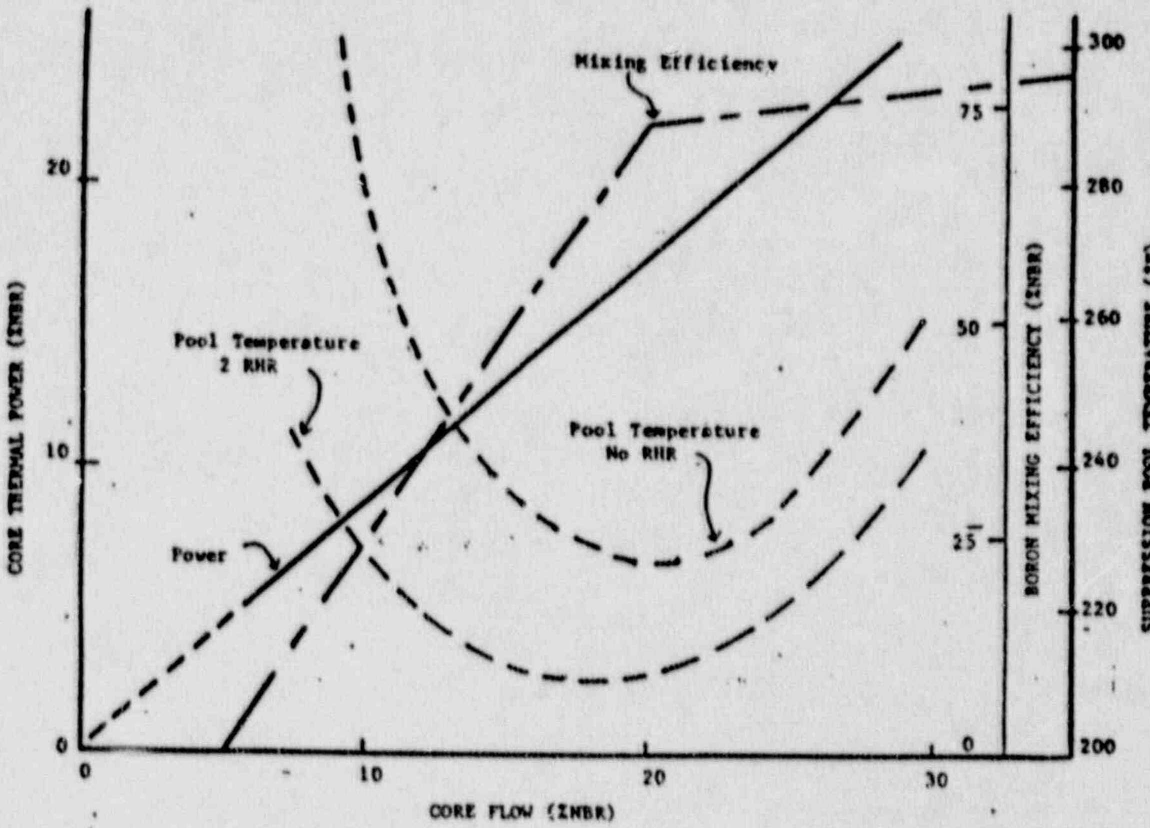


FIGURE 17.1.1

Core Flow vs. Power and Suppression Pool Temperature

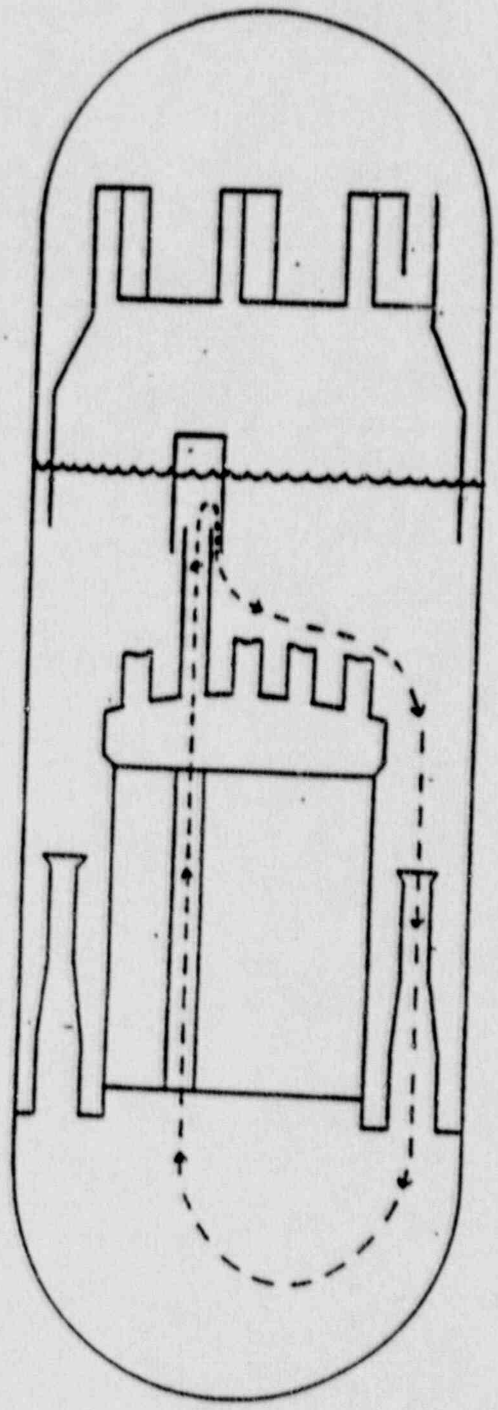


FIGURE 17.1.2

Natural Circulation Flowpath

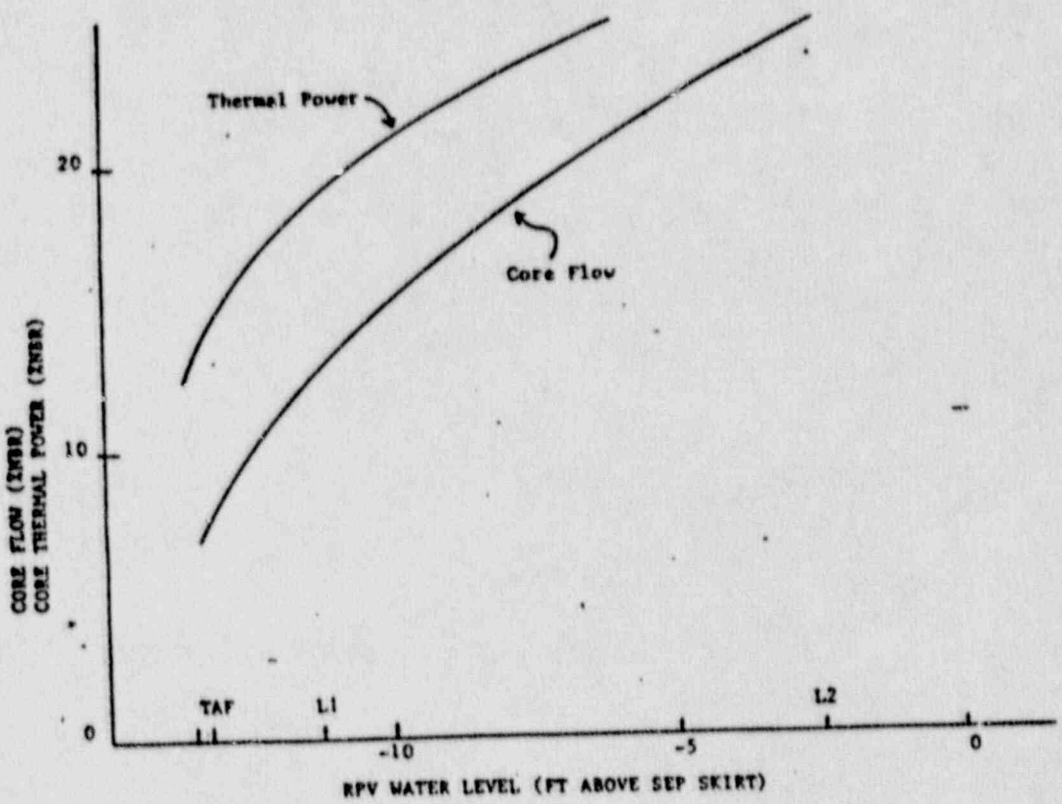


Figure 17.1.3

Core Flow vs. Power

ENCLOSURE 4

NRC Resolution of Facility Comments

RO/SRO Examinations

Question 2.01/5.06

NRC Resolution:

Facility comment noted. The stem of the question states that a LOCA is in progress; NOT that a LOCA signal has been generated. An operator could enter Path 4 during a LOCA if drywell pressure was less than 1.85 psig and RPV water level was above -101 inches. Since the given conditions clearly stated the operators transitioned from Path 3 to Path 4, no confusion between Path 4 and Path 5 should exist. However, since the given conditions also stated all MSIVs have closed and actions prior to step JO apparently mitigate radioactive release, the answer key will be modified to require RPV level restoration as the full credit answer and the question point value adjusted. This question is clearly supported by referenced facility enabling objectives and KAs.

Question 2.05/5.10

NRC Resolution:

Facility comment not accepted. As supporting documentation for this comment, the facility supplied the following items: (1) Page 7.6-71 from the HNP-2 FASR, (2) Jet Pump Integrity surveillance procedure, and (3) a single page from a training lesson plan on the effect of a recirculation pump trip. Item (1) describes how individual jet pump delta pressure is determined and the methods of determining total core flow based on jet pump delta pressure. It does not address the response of core plate delta pressure on a single jet pump failure. Item (2) requires core plate delta pressure to be recorded but is not used to determine jet pump integrity. Item (3) describes effects of a recirculation pump trip. A recirculation pump trip would result in a larger flow decrease, and consequently a larger decrease in core plate delta pressure, than would a failed jet pump. Such a large decrease in core plate delta pressure would be indicated on control room instrumentation, but would not be indicated for a failed jet pump. Since the stem of the question specifically asks for indication of jet pump failure, core plate delta pressure would NOT be an indication of jet pump failure. Additionally, when this malfunction was used during simulator examination validation, the simulator operator stated there are observable in changes core flow and power but no observable changes in core plate delta pressure. No change to the examination is warranted. This question was taken directly from facility supplied material for examination development. Core plate delta pressure was not identified as an indication of jet pump failure.

Question 2.15/5.20

NRC Resolution:

Facility comment accepted. Since either (a) or (c) are correct answer, the questions will be deleted and the point value adjusted. This knowledge is covered by Enabling Objective 13 and is supported by referenced KA 295037K303 (4.1/4.5). As noted by the facility, the lesson plan and objectives cited on the examination are incorrect but do not affect the validity of the question. This question is appropriate to NRC licensing examination and may be used on future examiantions.

Question 2.30/5.43

NRC Resolution:

Facility comment accepted. This question was taken directly from facility supplied material for examination development. Additional material provided as supporting documentation for this comment state different indications than in the facility supplied material. Also, a hand-written notation on the supporting documentation clarifies that all three conditions are required simultaneously for indication of positive injection. This clarification does not exist in facility supplied material. Based on information in the additional material, there is no correct answer. The question will be deleted and the point value adjusted. It is not standard practice to inform all candidates when only one (1) candidate out of eleven (11) requests clarification.

Question 3.15/6.09

NRC Resolution:

Facility comment accepted. The APRM was not specified in the examination that was pre-reviewed and was added after discussion with the facility reviews. Confusion between RBM and APRM channels and incomplete facility supplied material caused the incorrect APRM to be specified in the question. Since APRM A has no effect on RBM A, the question will be deleted and the point value adjusted. The facility should consider correcting the training material to include APRM assignments to RBM channels.

ENCLOSURE 5

SIMULATION FACILITY REPORT

Facility Licensee: Georgia Power Company

Facility Docket Nos.: 50-321 and 50-366

Operating Tests Administered On: October 10 - 12, 1989

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed

1. SRVs did not cycle as would be expected following a reactor scram with the MSIVs closed. This problem was previously identified as a modelling error with RPV level shrink following a reactor scram.

2. Feedpump turbine vibration is not modelled to indicate on the data point recorder. The candidates could not use approved plant procedures to determine feedpump turbine vibration during this abnormal event.

3. When transferring RPS power, RWCU isolation valve F004 was de-energized in the open position. When RPS power was transferred, F004 closed causing RWCU pump to trip. This impacted use of normal plant procedure and componded the evolution.

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