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

ENCLOSURE 1

EXAMINATION REPORT - 50-259/OL-90-02

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

Facility Name: Browns Ferry Nuclear Plant

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

Facility License No.: DPR-33, DPR-52, and DPR-68

Initial examinations were administered at Browns Ferry Nuclear Plant near Decatur Alabama.

Chief Examiner: George T. Hopper — 7/19/90 Date Signed

Approved By: Charles A. Casto, Chief
Operator Licensing Section 2
Division of Reactor Safety — 7/23/90 Date Signed

SUMMARY:

Examinations were administered June 25-29, 1990.

Written examinations and operating tests were administered to one (1) RO applicant and nine (9) SRO applicants. All applicants passed these examinations.

REPORT DETAILS

1. Examiners:

   *G. Hopper, NRC - Region II
   M. Daniels, Sonalysts
   R. Miller, Sonalysts

   *Chief Examiner

2. Facility Personnel at Exit Meeting:

   L. Durham, Manager, Nuclear Training
   N. Kazanas, Vice President Nuclear Assur. and Services
   T. Dexter, Browns Ferry Training Manager
   E. Howard, Lead Instructor

3. Pre-examination Review

   The written examinations were reviewed at the NRC - Region II office by facility representatives prior to examination administration. This review minimized the number of questions asked by the candidates during the exam and also helped to minimize the number of post-exam changes.

4. Exit Meeting:

   At the conclusion of the site visit, examiners met with representatives of the plant staff to discuss administration of the examination and problems noted.

   The accommodations made by the facility for administering the written exam were adequate, however, the window cleaners using high pressure hoses provided an unsuitable noisy distraction during the test. Efforts should be made in the future to ensure that candidates are given a suitable environment, free from any outside disturbances, during the written examination.

   The examiners made the following observations concerning your training program:

   a. Several candidates held different concepts of the term "Subcritical" as it applies to implementation of step RC/Q-4 of EOI-I. This step directs injection of boron if suppression pool temperature reaches 110 Deg F and the Reactor is not "Subcritical". Answers given, in reference to neutron monitoring, varied from " < 3 percent on the APRMs" to "when power reached range 6 on the IRMs". It is recommended that all operators utilize a consistent set of parameters to determine subcriticality.

b. Several candidates experienced difficulty in reading or utilizing prints (P&IDs) to answer questions.

c. All candidates for this exam were not trained on the Main Steam Line Radiation Monitors installed in Unit 2 (NUMACS) which differ significantly from those found in the simulator.

d. Candidates were observed to initially rely on the One Rod Out Permissive light for verification of all rods in, rather than utilize the Full Core Display or Process Computer (OD-7) as an equally acceptable means of verification. When this light was disabled, several candidates mistakenly reported that all rods were not inserted when, in fact, they were. This led to delays in recovery actions during some of the scenarios.

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

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

ENCLOSURE 2

MASTER

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

FACILITY: Browns Ferry 1, 2, & 3

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 90/06/25

CANDIDATE: ________________________

INSTRUCTIONS TO CANDIDATE:

Points for each question are indicated in parentheses after the question. To pass this examination, you must achieve an overall grade of at least 80%. Examination papers will be picked up four and one half (4 1/2) hours after the examination starts.

NUMBER QUESTIONS	TOTAL POINTS	CANDIDATE'S POINTS	CANDIDATE'S OVERALL GRADE (%)
96	100.00		

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

7. You may write your answers on the examination question page or on a separate sheet of paper. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

8. If you write your answers on the examination question page and you need more space to answer a specific question, use a separate sheet of the paper provided and insert it directly after the specific question. DO NOT WRITE ON THE BACK SIDE OF THE EXAMINATION QUESTION PAGE.

9. Print your name in the upper right-hand corner of the first page of answer sheets whether you use the examination question pages or separate sheets of paper. Initial each of the following answer pages.

10. 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.

11. If you are using separate sheets, number each answer and skip at least 3 lines between answers to allow space for grading.

12. Write "Last Page" on the last answer sheet.

13. 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.

14. 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.

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

16. 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.

17. Proportional grading will be applied. Any additional wrong information that is provided may 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, and 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 got the four correct answers.

18. If the intent of a question is unclear, ask questions of the examiner only.

19. When turning in your examination, assemble the completed examination with examination questions, examination aids and answer sheets. In addition, turn in all scrap paper.

20. To pass the examination, you must achieve an overall grade of 80% or greater.

21. There is a time limit of (4 1/2) hours for completion of the examination. (or some other time if less than the full examination is taken.)

22. 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: 001 (1.00)

The following conditions exist on Unit 2:

Reactor scrammed from 100% power due to low level [loss of all reactor feed pumps].

All control rods are fully inserted.
RCIC and HPCI failed to operate.
Reactor water level is -150 inches.
No injection system or alternate injection is available.

Which ONE of the following describes the operator actions required?

a. Enter C2, Emergency Rx Depressurization, and attempt to restore a low pressure injection system to operation in accordance with C1, Alternate Level Control.

b. Enter C3, Steam Cooling and attempt to restore an injection system to operation in accordance with C1, Alternate Level Control.

c. Enter C4. Reactor Flooding Pressure, and when pressure is below the Minimum Alternate Flooding Pressure, return to RC/L Level Control.

d. No action is required until reactor water level reaches the top of the active fuel. Then enter C7, Core Cooling Without Level Restoration.

QUESTION: 002 (1.00)

The following plant conditions exist:
Unit 2 has scrammed from rated conditions.
Twelve control rods failed to insert.
A condition exists requiring Emergency Depressurization.
Injection has been terminated and prevented.
All six ADS valves have been manually opened.

Which ONE of the following describes the appropriate point at which injection to the vessel can be re-established?

a. Reactor pressure is <190 psig.

b. Reactor pressure is <450 psig.

c. Reactor water level decreases to TAF.

d. Reactor water level decreased to -150 inches.

QUESTION: 003 (1.00)

Which ONE of the following describes the purpose of the Drywell Spray Initiation Limit Curve?

a. Assures the prevention of equipment failure due to unstable steam condensation during an ADS blowdown.

b. Assures that actuation of ADS will not result in damage to the pool or any submerged structure within the suppression pool.

c. Assures that any steam released in the drywell will be directed under water in the suppression pool.

d. Assures that the drywell will not collapse or otherwise fail due to negative pressure.

QUESTION: 004 (1.00)

With the Reactor at 94% rated thermal power and Recirc Flow at 90%, a malfunction of the EHC system results in a slow increase in reactor pressure and power until the reactor scrams on High Neutron Flux at 118%.

Which ONE of the following statements is correct concerning the above situation?

a. The Thermal Power Safety Limit has been exceeded.

b. The Power Transient Safety Limit has been exceeded.

c. The Reactor Vessel Water Level Safety Limit has been exceeded.

d. The Reactor Coolant System Integrity Safety Limit has been exceeded.

QUESTION: 005 (1.00)

Unit 2 is operating at 80% power when a steam leak in the tunnel results in a Group 1 Isolation. The reactor scrams due to MSIV closure. ONE SRV with a normal setpoint of 1125 psig fails to open, and peak reactor pressure reaches 1150 psig.

Which ONE of the following statements is correct concerning the above situation?

a. The Thermal Power Safety Limit has been exceeded.

b. The Power Transient Safety Limit has been exceeded.

c. The Reactor Coolant System Integrity Safety Limit has been exceeded.

d. No safety limit has been exceeded.

QUESTION: 006 (1.00)

Unit 2 has experienced a failure to scram from 100% power. The following plant parameters and conditions exist:

Reactor power : 7%
Suppression pool temperature : 118 F
Rx. pressure : 920 psig
SLC tank level : 40%
MSIV's : open
Rx. water level : -75 inches.

Which ONE of the following describes the conditions necessary to start a cooldown:

a. Suppression pool temperatures < 110 F.

b. Reactor power < 3%.

c. Reactor subcritical.

d. SLC injected to a level of 6%.

QUESTION: 007 (1.00)

While operating at power, a transient occurs. The MSIVs close due to low water level. The reactor automatically scrams, however, many rods fail to insert. HPCI fails, but RCIC slowly recovers level. Suppression pool cooling is placed in service. Pressure control is established on the SRVs. The following plant conditions exist:

Reactor power = 5%
Reactor pressure = 1000 psig
Reactor level = -45
Suppression pool temperature = 100 Deg F [Slowly increasing]
Suppression pool level = +1"

Which ONE of the following actions should be performed?

a. Reactor water level should be deliberately lowered to control reactor power.

b. SLC injection should be initiated.

c. MSIVs should be opened to re-establish the main condenser as a heat sink.

d. Emergency depressurization should be conducted.

QUESTION: 008 (1.00)

A loss of coolant accident occurs from normal operating conditions. All emergency systems [PRS, PCIS, ECCS] respond normally. Use the following parameters to answer the question below.

Reactor pressure = 100 psig
Reactor level = -20"
Drywell pressure = 26 psig
Drywell Temperature = 180 Deg. F
Suppression pool level = 18.5 ft.
Suppression pool temperature = 165 Deg. F
Suppression chamber temperature 160 Deg. F
Suppression chamber pressure = 26 psig

Which ONE of the following actions is appropriate?

a. Initiate Drywell and Suppression pool sprays.

b. Initiate Drywell spray, but not Suppression Pool spray.

c. Initiate Suppression pool sprays, but not Drywell sprays.

d. Do not initiate Drywell or Suppression pool sprays.

QUESTION: 009 (1.00)

Which ONE of the following describes the purpose of the Heat Capacity Temperature Limit?

a. Assures the prevention of equipment failure due to unstable steam condensation during an ADS blowdown.

b. Assures that actuation of ADS will not result in damage to the pool or any submerged structure within the suppression pool.

c. Assures that any steam released in the drywell will be directed under water in the suppression pool.

d. Assures that the drywell will not collapse or otherwise fail due to negative pressure.

QUESTION: 010 (1.00)

The following plant conditions exist on Unit 2:

Reactor scrammed
Control Room evacuated
Plant cooldown required using SRV's

Suppression Pool Cooling is required during the cooldown. Which ONE of the following is the location from which RHR pump 2C would be started.

a. The Backup Control Panel.

b. 4KV Shutdown Board B .

c. 4KV Shutdown Board C.

d. Locally at the pump.

QUESTION: 011 (1.00)

A Unit 2 start-up is in progress, at 20% power, with the turbine rolling at 1800 rpm. The RFP "A" control signal is lost, and the MGU is locked. For the given conditions, which ONE of the following describes the response of the "A" RFP if the MGU lock-out were reset [Hydraulic Jack switch is on]:

a. Speed would increase to the MGU High Speed Stop.

b. Speed would decrease to the MSC LSS.

c. Speed would be controlled by the master FWLCS controller.

d. Speed would remain the same, controlled by the MSC.

QUESTION: 012 (1.00)

EOI-1 Reactor Control, is being executed following a scram due to a turbine trip at high power. During the initial phase of the transient, pressure increased to the SRV lift setpoint, and ONE of the SRV, sticks open. Suppression pool temperature has reached 95 Deg. F. Which ONE of the following actions should take place?

a. Re-enter EOI-1 at the beginning.

b. Renter EOI-1 at the beginning and enter EOI-2.

c. Continue in EOI-1 and enter EOI-2.

d. Continue in EOI-1.

QUESTION: 013 (1.00)

A reactor start-up and heat-up is in progress on Unit 2. Reactor pressure is 520 psig and the 1B CRD pump is out of service for bearing replacement. The following alarms/indicators are received on Unit 2:

PA-85-1, CRD pump A suct press low
2A CRD pump breaker trips
CRD drive water HDR diff press. is 175 psid
TA-85-1, Control Rod Drive Temp High

Which ONE of the following describes the action to be taken:

a. Insert a manual scram.

b. When the second accumulator light comes in, manually scram.

c. If charging water pressure is < 1410 psig, manually scram.

d. If CRD system not restored in 1 hour, manually scram.

QUESTION: 014 (1.00)

Unit 2 is operating at 55% power when the RBCCW essential loop isolation valve [70-47] closes and will not respond to an open signal. Which ONE of the following describes the required action?

a. If temperature limits are exceeded on A or B recirc pump, Manually scram the Rx and trip the associated recirc pump.

b. Manually scram the Rx and initiate a cooldown at 90 deg. F/hour.

c. Reduce Rx power in an attempt to reduce drywell and recirc pump temperatures.

d. If drywell temperature exceeds 160 deg. F, enter EOI-1 and EOI-2.

QUESTION: 015 (1.00)

With Unit 2 operating at power and Unit 1 in cold shutdown, an event occurs causing the SOS to decide that the control room must be evacuated. All required immediate actions are taken. Which ONE of the following describes the condition of Unit 2?

a. Hot standby

b. Hot shutdown

c. Cold shutdown

d. Operating at reduced power

QUESTION: 016 (1.00)

Which ONE of the following describes the purpose of the Suppression Pool Load Limit Curve?

a. Assures the prevention of equipment failure due to unstable steam condensation during an ADS blowdown.

b. Assures that actuation of ADS will not result in damage to the pool or any submerged structure within the suppression pool.

c. Assures that any steam released in the drywell will be directed under water in the suppression pool.

d. Assures that the drywell will not collapse or otherwise fail due to negative pressure.

QUESTION: 017 (1.00)

The following plant condition exist:

Unit 2 scrammed
MSIV's closed
Suppression chamber pressure 3 psig.
Suppression Pool Temp 200 Deg. F

Which ONE of the following states the maximum allowable RHR pump flow? [Refer to EOI Charts]

a. 3500 gpm

b. 5000 gpm

c. 8500 gpm

d. 11000 gpm

QUESTION: 018 (1.00)

The reactor is operating at 70% rated thermal power and 70% rated core flow. Which ONE of the following describes how you would decrease reactor power to 60% of rated, while maintaining core flow constant [70%]?

a. Insert control rods.

b. Withdraw control rods and decrease recirculation pump speed.

c. Insert control rods and decrease recirculation pump speed.

d. Insert control rods and increase recirculation pump speed.

QUESTION: 019 (1.00)

Following a reactor scram, reactor water level is restored with the use of HPCI and RCIC. If Rx level was allowed to increase to +54", which ONE of the following describes how HPCI and RCIC would respond?

a. Both would trip, if Rx level then decreased to -51.5", HPCI would auto initiate but RCIC would not.

b. Both would trip and neither would auto initiate if RX level later decreased to -51.5".

c. Both would trip, if Rx level then decreased to -51.5", both would auto initiate.

d. Both would trip, if Rx level then decreased to -51.5", RCIC would auto initiate but HPCI would not.

QUESTION: 020 (1.00)

Which ONE of the following explains why the CRD system flow controller [FIC 85-11] could indicate flow off-scale high [>100 gpm] with the flow control valves [FCV's 85-11A and 11B] both indicating closed.

a. Due to a rupture of the differential pressure detector bellows of the flow element [FE 85-11].

b. Due to drive water header flow directed to the CRD under piston area sensed by 85-11, as a result of a scram, but supplied through FCV's 85-11A and 11B.

c. Due to charging water header flow to the CRD under-piston area sensed by 85-11, as a result of a scram, but not supplied through FCV's 85-11A and 11B.

d. Due to a stem/disc seperation on Charging Water Throttle Valve resulting in CRD pump runout.

QUESTION: 021 (1.00)

If manual rod insertion Per RC/Q is required, choose the correct statement describing which interlocks, if any, are bypassed and how each bypass is accomplished.

a. RWM by use of the Full In/Full Out bypass switches; RSCS by use of the Emergency In position of HS 85-47.

b. RSCS by use of the Full In/Full Out bypass switches; and RWM by use of Manual Bypass switch.

c. RSCS by Mode Switch in Refuel and the Emergency In position of HS85-47; RWM by use of Manual Bypass switch.

d. Place Mode Switch in Refuel to allow selection of any rod; use the Emergency In position of HS 85-47 to bypass RSCS and RWM.

QUESTION: 022 (1.00)

Unit 2 is operating at 100% power with the following initial conditions:

Turbine load set	100%
Pressure set	920 psig
Max. combine flow set	100 [125% steam flow]
Load limit set	100%
Recirc flow control	Master manual

An electrical fault causes the load reject relay to pick-up. Which ONE of the following correctly describes the plant's response to this transient? Assume no operator action.

a. Reactor pressure will increase due to rapid closure of the Turbine Control Valves resulting in a high pressure reactor scram.

b. Control oil pressure will decrease due to rapid closure of the Turbine Control Valves resulting in a low control oil pressure scram.

c. Reactor power will increase due to rapid closure of the Turbine Control Valves resulting in a high flux reactor scram.

d. The Turbine Control Valves will rapidly close resulting in the Bypass Valves opening to control Pressure with the Plant eventually stabilizing at 60% power.

QUESTION: 023 (1.00)

A plant heatup is in progess from cold shutdown to rated pressure. During the heatup actual reactor water level remains constant. The Emergency System Range Level Indicators (LI-3-58A and LI-3-58B) will decrease and come on scale due to which ONE of the following?

a. Due to reference leg heat up.

b. Due to variable leg heat up.

c. Due to increased downcomer subcooling.

d. Due to increased core flow.

QUESTION: 024 (1.00)

During an ATWS SLC was injected per the EOI's. When would SLC injection be terminated?

a. All control rods are inserted to 00.

b. Suppression pool temperature is less than 110 deg. F.

c. SLC tank level has decreased to 18%.

d. Indicated Reactor Power < 3%.

QUESTION: 025 (1.00)

Assume the plant is operating with the Bypass Valves open. If an SRV suddenly fails 100% open, how would the Turbine Bypass Valves respond? [Select ONE]

a. All Bypass Valves would close, and throttle pressure would decrease.

b. No change in Bypass Valve position of throttle pressure.

c. Bypass Valves would close down in response to the decrease in throttle pressure.

d. All Bypass Valves would close and throttle pressure would remain constant.

QUESTION: 026 (1.00)

Which ONE of the following is the purpose of the recirculation pump 75% limiter?

a. Prevent caviatation of the Recirculation Pumps.

b. Prevent overspeed trip of the remaining Feedpump.

c. Prevent a low reactor water level scram.

d. Prevent Jetpump vibration.

QUESTION: 027 (1.00)

With Unit 2 at 100% power, Unit 1 and 3 defueled, a loss of 500KV system occurs, followed a minute later by a loss of the 161KV system. All Diesel Generators start and are operating properly. After 30 minutes, Unit 2 is shutdown, with pressure and level under control. Which ONE of the following describes the requirements for paralleling D/G's to establish the main condenser as a heat sink.

a. Conditions for paralleling D/G's are not met.

b. D/G's B and 3EB should be paralleled.

c. D/G's A and 3EA should be paralleled.

d. D/G's A and B should be paralleled.

QUESTION: 028 (1.00)

Due to an inadvertent Group 6 Isolation, the SOS has determined either the plant control air system or the CAD system must be aligned to the drywell control air system. Which ONE of the following describes the concerns that must be addressed when either system is lined up?

a. Plant control air could cause a drywell pressure increase. CAD could cause an increase of drywell pressure and oxygen concentration.

b. Both plant control air and CAD could cause drywell pressure and oxygen concentration to increase.

c. Plant control air could cause an increase of drywell pressure and oxygen concentration. CAD could cause an increase of drywell pressu

d. The only concerns for either plant control air or CAD would be an increase of drywell pressure.

QUESTION: 029 (1.00)

Unit 2 is operating at 100% power, when RPS M/G set 2B trips. Five minutes later, the RPS bus is re-energized, from the alternate power supply. If the Reactor Mode Switch is placed in Shutdown before the Half Scram is reset, which ONE of the following RPS scram functions would cause the reactor to scram?

a. Mode switch in S/D

b. APRM Hi-Hi neutron flux

c. APRM Hi-Hi thermal

d. IRM Hi-Hi

QUESTION: 030 (1.00)

Which ONE of the following conditions will result in a trip of the Diesel Generator following an automatic start due to Hi Drywell Pressure?

a. Low oil pressure of 20 psig

b. Reverse power

c. High differential current

d. Low cooling water pressure of 20 psig

QUESTION: 031 (1.00)

Unit 2 is at approximately 10% power, with preparations in progress to roll the main turbine. The unit operator observes the following:

All bypass valves open
Rx pressure is decreasing
Rx power is decreasing
Rx level is increasing

Which ONE of the following describes the action to be taken.

a. Place EHC pumps in pull to lock.

b. Scram the reactor and break condenser vacuum.

c. Scram the reactor and close the MSIV's

d. Use max. combined flow to close the bypass valves.

QUESTION: 032 (1.00)

Which ONE of the following condition[s] is required to automatically bypass SRM rod blocks during a startup ?

a. SRM rod blocks are automatically bypassed ONLY when the Mode Switch in the RUN position.

b. SRM rod blocks are automatically bypassed when all IRM range switche are on range 8 or above, or the Mode Switch is in the RUN position.

c. SRM rod blocks will be automatically bypassed when all IRM switchs are on range 2 and the SRMs are reading less than 100 cps.

d. SRM rod blocks can only be bypassed when the Unit Operator has fully withdrawn all SRMs after verifying correct IRM over lap.

QUESTION: 033 (1.00)

Unit 2 is operating at 80 % rated thermal power when an accident signal occurs. Offsite power is available. Which ONE of the following describes the Unit 2 Core Spray system response?

a. Only the 2B and 2D Core Spray pumps start.

b. All Unit 2 Core Spray pumps start after 7 seconds.

c. 2B Core Spray pump starts, in 7 seconds, 2C Core Spray pump starts in 14 seconds.

d. 2B and 2D Core Spray pumps start in 7 seconds: 2A and 2C Core Spray pumps start in 14 seconsds.

QUESTION: 034 (1.00)

During a Rx startup operating temperature and pressure was reached at 0100 hrs and the Mode Switch was placed in Run at 0345 hrs. Which ONE of the following describes the requirements for satisfying primary containment limitations?

a. Nitrogen inerting must be completed by 0100 hrs the next day and delta P control must be established by 0345 hrs the next day.

b. Both inerting and delta P control must be completed by 0100 the next day.

c. Nitrogen inerting must be completed by 0345 hrs the next day and delta P control must be established by 0100 hrs the next day.

d. Both inerting and delta P control must be completed within 24 hours after placing the mode switch to run.

QUESTION: 035 (1.00)

The following plant conditions exist:

Jet pumps 1-10 Differential Pressure [meter] = 3 psid
Jet pumps 11-20 Differential Pressure [meter] = 15 psid
RECIRC LOOP B ONLY OUT OF SERVICE [annunciator] = ON

The TOTAL CORE FLOW recorder would calculate core flow by which ONE of the following methods?

a. Loop A Jet Pump flow + loop B Jet Pump flow.

b. Loop A Jet Pump flow - loop B Jet Pump flow.

c. Loop A Jet Pump flow only.

d. Loop B Jet Pump flow only.

QUESTION: 036 (1.00)

Which ONE of the following correctly describes the operation of the RWM as power is reduced from 100% power to 25% of rated?

a. The "AUTO" indicator light will extinguish when power is decreased below the LPSP.

b. The system will enforce the loaded rod sequence when power is decreased below the LPSP.

c. Both steam flow and feed flow must decrease below a designated setpoint to place the system in service.

d. All system alarms and displays are operative while in the "Transition Zone".

QUESTION: 037 (1.00)

Unit 2 is at 20% power with the Turbine/Generator synchronized to the grid. The unit operator observes the following:

Condenser A, B, or C vacuum low Alarm.
Standby SJAE auto-started.
Off-Gas Flow to 6-Hour Holdup Volume is decreasing.

Which ONE of the following describes the action to be taken?

a. Trip the main turbine.

b. Reduce generator load using load-set.

c. Scram the reactor and trip the turbine.

d. Place mechanical vacuum pumps in service.

QUESTION: 038 (1.00)

Unit 2 is in the process of starting-up, and the Unit operator just completed pulling the first control rod to position 48. He then successfully selects a control rod from Group 4 rather than Group 1. Which ONE of the following describes the restrictions and/or limitations now imposed?

a. The withdrawn control rod shall be manually inserted to position 00.

b. The withdrawn control rod shall be individually scrammed.

c. Control rod withdrawal may continue as long as a second operator verifies proper movement.

d. Manual withdrawal and insertion of control rods shall be stopped.

QUESTION: 039 (1.00)

A leak in the steam tunnel at 100% power resulted in a Group 1 Isolation. The reactor failed to scram with an 80% hydraulic ATWS. Given these conditions, which ONE of the following describes the actions to restore the main condenser as a heat sink?

a. Reduce steam tunnel temperature, reset PCIS, and equalize around and open MSIVs

b. Start an auxiliary boiler, line up aux. steam to steam seals, start mechanical vacuum pumps, and open the MSIVs

c. The main condenser is unavailable as a heat sink due to the presence of a Group 1 Isolation signal.

d. Bypass Steam Tunnel High Temperature Isolation and reset PCIS to restore the main condenser as a heat sink.

QUESTION: 040 (1.00)

Following a reactor scram several rod positions on the full core display have no numbered position indications, and only a green background. Which ONE of the following describes their position?

a. They are fully inserted with failed position 00 reed switches.

b. They are somewhere between the 00 position and the 48 position and driving in as a result of the scram signal.

c. They over travelled beyond full in as a result of the scram signal and should settle back to "00" position after the scram is reset.

d. Their full in/full out bypass switches in the aux instrument room have been bypassed in the Full In position per EOI Appendix 9.

QUESTION: 041 (1.00)

Unit 2 is operating at 98% power, when the Unit Operator notices the red and green lights above the hand switch for MSIV 1-15 are on, and "C" MSL flow reads less than A, B, and D. Which ONE of the following describes the action to be taken in response to the above indications?

a. Insert control rods until below 80% reactor power.

b. Reduce reactor power to 70% of rated.

c. Reduce recirc flow to 45% of rated.

d. Reduce core flow to below 45% of rated.

QUESTION: 042 (1.00)

Which ONE of the following describes the correct method for rapidly reducing Rx power from 100% during an abnormal condition?

a. Reduce Recirc flow to 45%, then stop flow reduction, and insert control rods until below the 80% rod line.

b. Reduce power with recirc flow until APRM High Rod Block is reached; insert rods until blocks are clear; if necessary proceed with flow reduction.

c. Reduce recirc pump speed to <28%; then insert control rods until below the 80% rod line.

d. Reduce recirc pump speed to 28%, then Manually Scram the plant.

QUESTION: 043 (1.00)

Unit 2 is in cold shutdown with the reactor vessel head removed in preparation for refueling. RHR Pump 2A is in shutdown cooling. RHR Pump 2C is tagged for motor inspection. Both recirculation pumps are out of service. Smoke is observed coming from RHR Pump 2A, and the pump subsequently trips. While troubleshooting is in progress, reactor vessel temperature is noted to be >150 Deg F. Which operator action is appropriate given the above conditions?

a. Attempt to reestablish primary containment integrity.

b. Raise reactor water level to +60 inches to promote natural circulation.

c. Initiate Shutdown cooling using either Unit 1 or Unit 3 RHR via the cross-tie lines.

d. Evacuate the Refuel Floor and place RHR System II in Shutdown Cooling without flushing.

QUESTION: 044 (1.00)

Unit 2 is operating at 100% power when the Unit operator observes the following alarms:

H-2 Analyzer A and B Hi
Recombiner A Inlet Temp Low
Off-Gas Holdup Vol Press High
Off-Gas Holdup Vol Temp High

Which ONE of the following describes the action to be taken?

a. Manually scram the reactor.

b. Reduce reactor power to 50%.

c. Scram the reactor if >4% H-2 is confirmed by sample.

d. Scram the reactor if OG Post-Treatment High alarm is received.

QUESTION: 045 (1.00)

Which ONE of the following describes the action required to allow cycling of FCV-74-1 [RHR pump 2A Torus suction], while Unit 2 is operating at 100% power?

a. 2-XS-74-157 must be placed in the NORMAL position.

b. 2-XS-74-157 must be placed in the SHUTDOWN position.

c. 2-XS-74-158 must be placed in the NORMAL position.

d. 2-XS-74-158 must be placed in the SHUTDOWN position.

QUESTION: 046 (1.00)

Unit 2 is operating near rated power when control rod 38-23 begins to drift out [from notch 00]. Which ONE of the following actions is the immediate response in accordance with 2-AOI-85-6?

a. Manually insert control rods following the approved sequence.

b. Reduce Recirculation Pump speed by 10% to control possible power increase.

c. Manual scram the reactor.

d. Drive the rod in using the Emergency Rod In position of the CRD NOTCH OVERIDE SWITCH.

QUESTION: 047 (1.00)

Which ONE of the following conditions would result from failure of a Reactor Recirculation Pump #1 seal assembly at rated conditions?

a. A decrease in #1 seal cavity pressure from approximately 1000 psig to about 500 psig.

b. An increase in #1 seal cavity pressure from approximately 500 psig to approximately 1000 psig.

c. An increase in #2 seal cavity pressure from approximately 500 psig to approximately 1000 psig.

d. A decrease in #2 seal cavity pressure from approximately 500 psig to approximately 0 psig.

QUESTION: 048 (1.00)

Which ONE of the following signals from the Post-Treatment Radiation Monitoring System will initiate an Automatic isolation of the Off-Gas Discharge to the main stack [FCV-66-28]?

a. High trip in channel B.

b. High-High-High trip in Channel A

c. High trip in Channel A and Downscale trip in Channel B.

d. Downscale trip in Channel A and High-High-High trip in Channel B.

QUESTION: 049 (1.00)

Fuel loading is in progress on Unit 2. The following SRM readings were recorded before and after the loading of a bundle in Quadrant A.

	Before	After
SRM A	4 cps	5 cps
SRM B	1 cps	2 cps
SRM C	5 cps	5 cps
SRM D	3 cps	6 cps

Which ONE of the following actions describes the required action?

a. Must perform a response check of SRM C before loading more fuel.

b. Fuel loading may continue in quadrants A, C, and D.

c. Fuel loading shall be halted and a subcriticality check performed.

d. Fuel loading may continue in any quadrant.

QUESTION: 050 (1.00)

An AUO is performing an independent verification of a clearance. Which ONE of the following is the proper method for verifying a manual valve is open?

a. Check that stem position and local indicator shows valve open.

b. Turn valve hand wheel in the open direction and verify stem does not move.

c. Turn valve hand wheel in closed direction to verify movement, then fully re-open valve.

d. Count turns to close valve, then turn the same number of turns in the open direction and ensure valve opens.

QUESTION: 051 (1.00)

The boundaries of an existing CS pump clearance must be expanded to allow work on the pump's suction valve. Which ONE of the following correctly describes the method for accomplishing the above?

a. All work must stop, the existing clearance released, picked up, and a new clearance established.

b. Work could continue on the unaffected part of the clearance while the boundary change was being made, provided the SOS/ASOS and all persons holding the clearance agree that a safe clearance could be maintained during the change.

c. The person requesting the change would be responsible for notifying and obtaining authorization for the change from all persons holding the clearance and the notify the SOS/ASOS of their authorization.

d. A temporary change form [SDSP-217] must be completed and attached to the existing clearance prior to the boundary change.

QUESTION: 052 (1.00)

Maintenance is scheduled to perform work in a High Radiation area on your shift. How is access to this area obtained?

a. Checking out the key from the SOS.

b. Checking out the key from Radcon.

c. Radcon, with the SOS permission.

d. Radcon, with the ASOS permission.

QUESTION: 053 (1.00)

Which ONE of the following is the lowest event classification at which a Site Assembly MUST be performed?

a. Notification of Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

QUESTION: 054 (1.00)

During a Rx startup a single notch withdrawal reduced the reactor period to 55 seconds. Which ONE of of the following describes the action to be taken?

a. The Rx shall be shutdown until a thorough assesment has been performed.

b. Control rod[s] should be inserted to achieve a stable period > 60 seconds; the nuclear engineer and SOS should be contacted before pulling any more control rods.

c. The Rx shall be made subcritical and rod withdrawal stopped until permission is obtained from the nuclear engineer and SOS to continue.

d. No action is required unless a period of less than 30 seconds is observed.

QUESTION: 055 (1.00)

While operating at 17% rated thermal power, an EHC failure caused all of the Turbine Bypass Valves to open. When main steam line pressure drops to 785 psig, the MSIVs close causing a reactor scram. Level and pressure are controlled using HPCI.

Which ONE of the following safety limits has been exceeded for the above situation?

a. The Thermal Power Safety Limit has been exceeded.

b. The Power Transient Safety Limit has been exceeded.

c. The Reactor Vessel Water Level Safety Limit has been exceeded.

d. The Reactor Coolant System Integrity Safety Limit has been exceeded.

QUESTION: 056 (1.00)

EOI-1 was entered due to low water level, and five minutes later, while still in EOI-1, drywell pressure rises to 2.6 psig. Which statement below describes the requied SRO actions?

a. Enter EOI-2 and continue on in EOI-1.

b. Reenter EOI-1 at the beginning.

c. Reenter EOI-1 at the beginning and enter EOI-2.

d. Exit EOI-1 and enter EOI-2.

QUESTION: 057 (1.00)

Which ONE of the following describes the SLC system response to an initiation signal [Turning Key-Lock Switch SH-63-6A]?

a. Turning the key-lock switch in either direction will fire both explosive valves, start both A and B SLC pumps, and isolate RWCU.

b. Turning the key-lock switch in either direction will start only the selected SLC pump [A or B], fires the pump's associated explosive valves, and isolate RBCCW.

c. Turning the key-lock switch in either direction will fire both explosive valves, start the selected SLC pump [A or B], and isolate RWCU.

d. Turning the key-lock switch in either direction will start the selected SLC pump [A or B], and fire the associated explosive valve.

QUESTION: 058 (1.00)

Which ONE of the conditions required for ADS automatic initiation can be bypassed without any operator actions?

a. The 105 second timer.

b. High drywell pressure

c. A need to have an RHR pump or Corse Spray pumps operating.

d. A confirmatory low reactor water level [+18], and a low low Rx vessel level [-114"].

QUESTION: 059 (1.00)

Select the number of ADS valves that con be operated from the Remote Shutdown Panel.

a. 6

b. 3

c. 4

d. 2

QUESTION: 060 (1.00)

Select the ONE RCIC Turbine Trip signal, from those listed below, that does not result in closure of the RCIC Trip Throttle Valve [FCV 71-9] in < 0.3 seconds.

a. Electrical overspeed [125%]

b. High Turbine Exhaust pressure [25psig]

c. Manual Trip

d. High Reactor Vessel Water Level [+54"]

QUESTION: 061 (1.00)

Which ONE of the following describe the Class "A" Containment Isolation valve utilized at BFN?

a. Valves on lines that connect directly with the reactor vessel or coolant piping and penetrate the primary containment.

b. Valves with a closure time based on minimizing radioactive release in case of a LOCA

c. Valves on lines that penetrate the primary containment and connect directly with containment free air space.

d. Valves on lines that penetrate the primary containment and are located outside the containment in series.

QUESTION: 062 (1.00)

Which ONE of the following is not a characteristic of High Purity Liquid Radwaste?

a. Conductivity >100 umho/cm

b. Major sources are equipment drains.

c. Considered to be reclaimable for reuse in the plant

d. Must be sampled and analyzed by Chem Lab before transfer or release.

QUESTION: 063 (1.00)

Which ONE of the following statements describes the condition[s] for which operating with one Reactor Recirculation pump in service is acceptable?

a. Single Reactor Recirculation pump operation is acceptable when one Reactor Recirculation pump trips from 100% power operation, and the tripped pump will be returned to service within 30 to 36 hours.

b. Single Reactor Recirculation pump operation should never continue for longer than a 24 hour period during any mode of operation.

c. Single Reactor Recirculation pump operation is acceptable when NO Shutdown Cooling is available and the single recirculation pump is required to prevent stratification of Reactor Vessel water.

d. Single Reactor Recirculation pump operation is acceptable for periods greater than 24 hours only when approval in granted by the Plant Manager and Plant Engineering.

QUESTION: 064 (2.50)

MATCH each of the reactor scrams listed in Column A with the correct purpose for the scrams in Column B. [NOTE: The items in Column B may be used once, more than once, or not at all, and only a single answer may occupy one answer space.]

Column A

a. APRM Hi Flux Scram

b. MSIV Not Full Open Scram

c. Reactor Low Water Level Scram

d. Low Air Header Pressure Scram

e. Generator Load Reject Scram

Column B

1. Scrams the reactor when the core is in danger of inadequate cooling.

2. Scrams the reactor to limit the fission products released from the fuel.

3. Scrams the reactor due to the anticipation of the rapid pressure and neutron flux increase.

4. Scrams the reactor to protect the fuel cladding against high heat generation rates.

5. Scrams the reactor before rods begin drifting into the core and before the scram discharge volume is filled.

6. Scrams the reactor prior to exceeding the Reactor Coolant System Integrity Safety Limit.

QUESTION: 065 (1.00)

Unit 2 is operating at 100 % power when a small LOCA occurs. The ADS initiation logic is satisfied and the 120 second timer has started. Which ONE of the following failures would prevent the automatic initiation of ADS ?

a. Core Spray Pumps A and B fail to start.

b. Loss of Drywell Control Air.

c. RHR Pumps B and D trip after starting.

d. Loss of 250 VDC.

QUESTION: 066 (1.00)

A reactor startup is in progress on Unit 2 with the following plant conditions:

- Reactor power is below the RSCS LPSP
- Rod withdrawal sequence B is in effect
- No RWM errors or blocks exist
- The RWM was placed in BYPASS following withdrawal of all RSCS Group ~~6~~ 12 rods to the withdraw limit at position 36.

The operator attempts to continue withdrawing Group 6 rods to position 48. Which ONE of the following describes the plant response to this operator action ?

a. Rod withdrawal will occur with NO rod position restrictions.

b. Rod withdrawal will occur, provided all Group ~~6~~ 12 rods are maintained within 1 notch of each other.

c. Rod withdrawal will NOT occur beyond the RWM withdrawal limit for Group ~~6~~ 12.

d. Rod withdrawal will NOT occur beyond position 36, due to RSCS immediately imposing a rod block.

QUESTION: 067 (1.00)

Which ONE of the following would DEFEAT manual opening of the RHR Test Return Line Valves 74-57 and 74-59 ?

a. Reactor water level < -39 "

b. LPCI initiation

c. Minimum Flow Valves not fully shut

d. Containment Spray initiation

QUESTION: 068 (1.00)

Which ONE of the following will NOT generate a " RPS ANALOG TRIP UNIT TROUBLE " alarm ?

a. Master Trip Unit calibration in progress

b. Slave Trip Unit trip

c. RPS bus power lost

d. Slave Trip Unit card removed

QUESTION: 069 (1.00)

Which ONE of the following is correct concerning the Minimum Alternate Flooding Pressure [MAFP] used in EOI-1 C5, Level/Power Control ?

a. Once RPV pressure decreases below the MAFP, adequate core cooling is assured.

b. Once RPV pressure decreases below the MAFP, steam flow through the core does not provide adequate core cooling.

c. If there are no SRVs open and pressure remains above the MAFP, adequate core cooling is assured.

d. If pressure remains above the MAFP with at least 2 SRVs open, natural circulation assures adequate core cooling.

QUESTION: 070 (1.00)

Which ONE of the following is correct if ALL rod position indication is lost while operating at power ?

a. Obtain an OD-7 printout before and after moving any control rod.

b. Withdrawal of control rods is allowed using only single notch movement.

c. Control rod movement is allowed only by scram.

d. Control rod movement is allowed provided a Qualified Member of the Technical Staff independently verifies proper rod sequencing.

QUESTION: 071 (1.00)

Which ONE of the following describes the response of an SRM detector to a pinhole leak which causes a gradual decrease in Argon gas pressure ?

a. Gamma and neutron sensitivity would decrease.

b. Gamma sensitivity would decrease but neutron sensitivity would remain the same.

c. Gamma sensitivity would remain the same but neutron sensitivity would decrease.

d. Both gamma and neutron sensitivity would remain the same.

QUESTION: 072 (1.00)

Unit 2 is operating at 100 % power in 3-element control when one steam flow detector fails upscale. Which ONE of the following describes the plant response ?

a. Reactor water level will decrease and stabilize at a lower level.

b. Reactor water level will decrease and initiate a reactor scram.

c. Reactor water level will increase and stabilize at a higher level.

d. Reactor water level will increase and initiate a Turbine trip.

QUESTION: 073 (1.00)

Which ONE of the following statements is correct concerning the number of visitors that are allowed to be assigned to a single escort in accordance with SDSP-11.11 ?

a. A person may escort no more than three visitors in a Vital Area without authorization of the Plant Manager.

b. A person may, without exception, escort a maximum of five visitors in the Protected Area.

c. A person may escort seven visitors in the Protected Area with the authorization of the Plant Manager.

d. A person may escort six visitors in a Vital Area with the authorization of the Plant Manager.

QUESTION: 074 (1.00)

Which ONE of the following is procedurally correct for verifying a Locked Throttle Valve's position when performing a valve lineup in accordance with SDSP-3.15 ?

a. An Independent Verifier may observe the initial valve operator's action when the throttle valve is operated to determine its position.

b. An Independent Verifier should check the throttle valve's position by manipulating the valve to the closed position and then reopening the valve to its proper throttled position.

c. The Initial Verifier should verify the valve's position utilizing alternate verification techniques such as observation of the valve stem, etc..

d. The Initial Verifier should check the valves's position by manipulating the valve to the Full Open position and then, restoring the valve to its proper throttled position.

QUESTION: 075 (1.00)

Which ONE of the following is NOT a Purpose of the Reactor Protection System as described in Technical Specification 3.1 ?

The Reactor Protection automatically initiates a Reactor Scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be absorbed following a loss of coolant accident.

d. Minimize the production of N-16 gammas which could be released following a loss of coolant accident.

QUESTION: 076 (1.00)

Which ONE of the following is the most desirable method of fighting a high voltage electrical fire with water ?

a. Straight stream directly on fire from at least 50 ft.

b. Wide fog pattern from at least 10 ft.

c. Fog/stream combination from as close as possible.

d. Narrow fog pattern from at least 40 ft.

QUESTION: 077 (1.00)

A Reactor startup is in progress on Unit 2 when the operating CRD Pump trips due to motor overload [pump shaft sheared]. The backup pump is under repair and is expected to be operable in 2 hours. The following plant conditions exist:

Rx Power 5 %
Reactor Pressure 700 psig
Charging Water Pressure 1400 psig [decreasing slowly]
1 Accumulator light on Full Core Display illuminated
1 CRD High Temperature alarm
CRD Low Pressure alarm

Which ONE of the following would be the correct action to take in accordance with 2-AOI-85-3 ?

a. A Manual Scram is required because charging water pressure cannot be restored and maintained above 1410 psig.

b. A Manual Scram is required if another Control Rod High Temperature alarm is received in conjunction with the LOW CRD WATER PRESSURE alarm.

c. A Manual Scram is required if a second Accumulator alarm is received due to low pressure.

d. A Manual Scram is required if Reactor Pressure decreases to 650 psig.

QUESTION: 078 (1.00)

Which ONE of the following is an IMMEDIATE operation action in the event ONE Reactor Recirculation pump trips while operating at 100% Reactor power?

a. Immediately refer to the power to flow map to determine if the unit is operating in a safe region; withdraw control rods to exit instability region if necessary.

b. Immediately insert control rods in a sequence determined by the S.S.

c. Immediately take actions to assure that Reactor power is below the 90% rod line.

d. Immediately place the remaining "RFC RECIRC PUMP SPEED CONTROL" in manual.

QUESTION: 079 (1.00)

Unit 2 Refuel [fuel movement] is in progress when the following alarms are noted in the Control Room:

FUEL POOL SERV FLR AREA HIGH RADIATION
REFUELING ZONE EXHAUST HIGH RADIATION
AIR PARTICULATE MONITOR HIGH RADIATION

Which ONE of the actions below is the control room operator required to perform?

a. Evacuate all personnel from the reactor building.

b. Stop all fuel handling and evacuate all non-essential personnel from the refueling floor.

c. Continue to fuel movement operations and notify RADCON to evaluate radiation levels.

d. Ensure all personnel remaining on refuel floor are wearing proper respiratory gear and notify RADCON.

QUESTION: 080 (1.00)

Which ONE of the following is the Suppression Pool temperature that should be maintained under normal operating conditions to assure continued unit operation ?

a. Maintain a temp < or = to 85 Deg F.

b. Maintain a temp < or = to 95 Deg F.

c. Maintain a temp < or = to 105 Deg F.

d. Maintain a temp < or = to 110 Deg F.

QUESTION: 081 (2.00)

For each of the Main Condenser Vacuum reading listed in Column A, choose the automatic action that should occur from Column B. [NOTE: The items in Column A may be used once, more than once, or not at all, and only a single answer may occupy one answer space.]

Column A	Column B
a. 25" Hg vacuum	1. Condensate pumps trip
b. 21.8" Hg vacuum	2. Reactor Scram
c. 7" Hg vacuum	3. Standby SJAE automatically starts
d. .8" Hg vacuum	4. Recirc pumps trip
	5. Main turbine trip
	6. RFP turbine trip and main turbine bypass valves closure occurs

QUESTION: 082 (1.00)

Which ONE of the following would be classified as a Reactor Transient Event in accordance with PMI 15.8, "Unit Trip, Reactor Transient, and Plant Transient Analysis"?

a. A 10% power decrease as indicated by APRMs due to a recirculation flow controller failure and subsequent scoop tube lock.

b. An unplanned reactor water level oscillation of 3 inches before the operator can respond to a startup level controller failure.

c. The Off-Gas pretreatment radiation monitor increases by 5% over ONE 8 hour shift during steady state 100% power operation.

d. Generator megawatts[MWE] decreases by 17 MWe during scheduled turbine valve testing.

QUESTION: 083 (1.00)

You are executing EOI-1, "Reactor Control", in response to a reactor scram and subsequent low water level. Drywell cooling has been lost due to an electrical malfunction of the fans. After 10 minutes, drywell temperature is 163 Deg F, and slowly increasing.

Which ONE of the following is the correct action to take?

a. Re-enter EOI-1 at the beginning.

b. Continue in EOI-1 and enter EOI-2.

c. Exit EOI-1 and enter EOI-2.

d. Re-enter EOI-1 at the beginning and enter EOI-2.

QUESTION: 084 (1.00)

The results of a surveillance performed on the Standby Liquid Control System are as follows:

SLC Tank level :	3500 gal	[300 gal of water added to SLC Tank]
Sodium Pentaborate:	8.9 %	
Solution Temperature:	68 Deg F	
B-10 Enrichment:	63 %	
Pump Flow Rates:	A: 42 gpm	B: 39gpm
Relief Valve Setting:	A: 1330 psig	B: 1390 psig

Which ONE of the following statements describes the status of the SLC system operability ? [NOTE: Applicable Tech Spec Attached]

a. SLC Train "A" is inoperable and LCO 3.4.B.1 applies.

b. SLC Train "B" is inoperable and LCO 3.4.B.1 applies.

c. The SLC system is operable and no restrictions apply.

d. The SLC system is inoperable and LCO 3.4.E applies.

QUESTION: 085 (1.00)

Unit 2 is operating at 70 % power when Security reports that water from the Wheeler Reservoir has started to run across the CCW Pumping Station Deck. Which ONE of the following actions should be taken ?

a. Reduce Reactor Power in preparation for shutdown and be in at least Hot Standby within 24 hours.

b. An orderly shutdown should be initiated and the plant placed in the Cold Shutdown Condition.

c. Continue operating and monitor for any Inplant Flooding or loss of any plant Water Tight Integrity.

d. Manually Scram the Reactor and perform an Emergency Cooldown in preparation for loss of Condenser Vacuum.

QUESTION: 086 (1.00)

Which ONE of the following will cause the greatest biological damage to man from an external source ?

a. 0.1 Rad of Fast Neutron

b. 1.0 Rem of Gamma

c. 10 Rem of Beta

d. 0.05 Rad of Alpha

QUESTION: 087 (1.00)

Assuming there is a completed NRC Form-4 on file, and there has been no previous exposure, 10CFR20 states that the maximum allowable Whole-body dose a woman 21 years of age may receive in one year while performing non-emergency work is: [Select ONE]

a. 0.5 REM unless shown NOT to be pregnant.

b. 3 Rem per quarter not to exceed annual dose of 5 Rem.

c. 5 Rem

d. 12 Rem

QUESTION: 088 (1.00)

Which ONE of the following individuals has the authority, in accordance with EPIP-15 "Emergency Exposure", to determine the amount of emergency exposure that will be allowed in a Life Saving Action ?

a. Radcon Superintendent

b. Site Emergency Director

c. Man in charge at the scene

d. Assistant Shift Operations Supervisor

QUESTION: 089 (1.00)

Which ONE of the following individuals is the designated Medical Team Leader during a Medical Emergency Response ?

a. Shift Operations Supervisor

b. Incident Commander

c. Emergency Service Technician with the most advanced licensing level.

d. Shift Fire Captain

QUESTION: 090 (1.00)

Unit 2 is shutdown and the SLC system is under a clearance issued to three persons with extensive repairs in progress. An SLC Pump motor requires power to be restored to conduct a phase rotation check. One Job Supervisor has determined that the Hold Order Tag on the pump motor breaker needs to be lifted to allow closing of the breaker.

Which ONE of the following is the correct procedure to follow in accordance with SDSP-14.9 ?

a. The SOS/ASOS may authorize the removal of the hold tag and will assign a qualified operator to temporarily remove the tag and operate the breaker.

b. The SOS/ASOS should complete a Clearance Temporary Lift, to include the authorization of each person issued the clearance, and designate a qualified operator to remove the tag and operate the breaker.

c. The individual requesting the lift should authorize a Clearance Temporary Lift. The SOS/ASOS will have a thorough inspection of the equipment made to ascertain that all personnel are in the clear, then authorize the tag removal and the breakers operation.

d. The individual requesting the lift must obtain release authorization from all persons issued the clearance. The SOS/ASOS will then release the clearance and designate a qualified operator to remove the tags and operate the breaker.

QUESTION: 091 (1.00)

Which ONE of the following describes ALL of the requirements which must be met for a non-licensed person to operate the Recirculation Flow Control system during power operations.

a. The person must be engaged in a formal OJT program as part of the RO/SRO licensing process and have permission from the SOS.

b. The person must be directly supervised by an active licensed operator and the individual must be engaged in a formal training program.

c. The person must be directly supervised by any licensed operator and have permission from the SOS.

d. The person must be in a formal OJT program as part of the RO/SRO licensing process under the direct supervision of a licensed operator and have been signed off on certain plant systems.

QUESTION: 092 (1.00)

Which ONE of the following is the Ph band stated in Tech Specs that, if exceeded [while at 100% power] for more than 24 hours, requires the Unit to be shutdown ?

a. < 5.3 or > 8.6

b. < 5.6 or > 8.6

c. < 5.6 or > 8.3

d. < 5.3 or > 8.3

QUESTION: 093 (1.00)

Which ONE of the following is the bases for the precaution PC/P-1 of EOI-2, "Primary Containment Control", to vent the Primary Containment only when Drywell and Suppression Chamber temperature are both below 210 Deg F ?

a. Venting above 210 Deg F would severely damage the charcoal bed in the SBGT system due to the high temperature/moisture content of the vented atmosphere.

b. Venting above 210 Deg F would lead to chugging in the Downcomers from the Drywell to the Suppression Pool.

c. Venting above 210 Deg F could eventually result in the collapse of the Primary Containment.

d. Venting above 210 Deg F could result in a release in excess of 10CFR20 limits since Containment temperatures this high are indicative of a large LOCA.

QUESTION: 094 (1.00)

Unit 2 has just received a request to increase power to 100 % in order to accommodate peak Summertime loading. The following paramaters were noted by the Unit Operator prior to increasing power:

Rx Power 90 %
RBCCW temperature increasing
RBCCW Surge Tank level increasing [slowly]
No other alarms or abnormal indications present.

Which ONE of the following malfunctions would cause these indications ?

a. RBCCW to RCW Heat Exchanger[s] tube leak[s].

b. RBCCW Nonregenerative Heat Exchanger tube leak.

c. RBCCW Heat Exchanger Temperature Element Failed High.

d. Loss of control air to the Temperature Control Valves.

QUESTION: 095 (1.50)

Match the Reactor Coolant chemistry paramaters in Column A with the limits in Column B. [NOTE: The items in Column B may be used once, more than once, or not at all, and only a single answer may occupy one answer space.]

Column A	Column B
a. Maximum Conductivity during a startup.	1. .10 umho/cm
b. Conductivity at 100 % power which, if exceeded, requires placing the unit in Cold Shutdown.	2. .20 umho/cm
c. Maximum conductivity at 100% power.	3. 1.0 umho/cm
	4. 2.0 umho/cm
	5. 10 umho/cm
	6. 100 umho/cm

QUESTION: 096 (2.00)

The following plant conditions exist for UNIT 2:

Rx Power 90 %
SLC Pump A inoperable [estimated time to repair 4 days]
Thunderstorms in the area

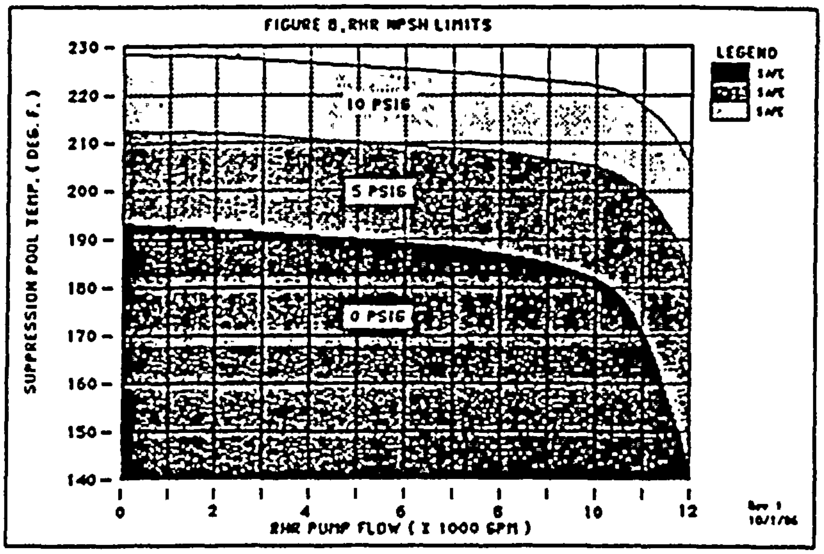
Match the event in Column A with the appropriate classification in Column B in accordance with EPIP-1 [attached]. Consider each event separately. [NOTE: The items in Column B may be used once, more than once, or not at all, and only a single answer may occupy one answer space.]

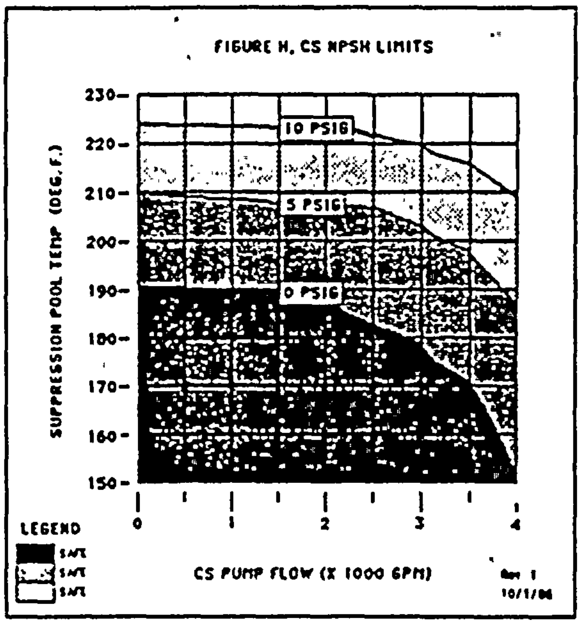
Column A	Column B
a. Loss of 120V 60 HZ Unit 2 preffered power to Panel 9-9, Cabinet 6 for two hours.	1. Not Applicable
b. Tornado sited over Decatur Alabama heading East.	2. NOUE
c. Security reports discovery of 3 personnel inside the Reactor Building with false identification badges.	3. ALERT
d. SLC Pump B is declared inoperable due to motor failure. Estimated time to repair - 2 days.	4. SITE AREA EMERGENCY
	5. GENERAL EMERGENCY

(********** END OF EXAMINATION **********)

CAUTION

MAINTAIN PUMP FLOW BELOW THE CS AND RHR NPSH LIMITS





NOTE: Curves indicate suppression chamber pressure. Below the curve for each suppression chamber pressure (0,5, and 10 psig) is the safe region for CS and RHR operation.

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LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the operating status of the Standby Liquid Control System.

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

A. Normal System Availability

1. Except as specified in 3.4.B.1, the Standby Liquid Control System shall be OPERABLE at all times when there is fuel in the reactor vessel and the reactor is not in a shutdown condition with Specification 3.3.A.1 satisfied.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. Verify pump OPERABILITY in accordance with Specification 1.0.MM.

2. At least once during each operating cycle:

   a. Check that the setting of the system relief valves is 1,425 ± 75 psig.

   b. Manually initiate the system, except explosive valves. Visually verify flow by pumping boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. After pumping boron solution, the system shall be flushed with demineralized water. Verify minimum

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.4.A Normal System Applicability

4.4.A.2.b. (Cont'd)

pump flow rate of 39 gpm against a system head of 1275 psig by pumping demineralized water from the Standby Liquid Control Test Tank.

c. Manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.

This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump operability. Replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date.

d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

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LIMITING CONDITIONS FOR OPERATION

3.4.B. Operation with Inoperable Components

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

3.4.C Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be OPERABLE, the following conditions shall be met:

1. At least 180 pounds Boron-10 must be stored in the Standby Liquid Control Solution tank and be available for injection.

2. The sodium pentaborate solution concentration must be equal to or less than 9.2% by weight.

SURVEILLANCE REQUIREMENTS

4.4.B. Surveillance with Inoperable Components

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

4.4.C Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Sodium Pentaborate Concentration check at least once per month. Also check concentration within 24 hours anytime water or boron is added to the solution.

3. Boron-10 Quantity:

   At least once per month, calculate and record the quantity of Boron-10 stored in the Standby Liquid Control Solution Tank.

4. Boron-10 Enrichment: At least once per 18 months and following each addition of boron to the Standby Liquid Control Solution Tank:

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	a. Calculate the enrichment within 24 hours. b. Verify by analysis within 30 days.
3.4.D The Standby Liquid Control System conditions must satisfy the following equation. $\frac{(C)(Q)(E)}{(13\ \text{wt.\%})(86\ \text{gpm})(19.8\ \text{atom\%})} \geq 1$ where, C = sodium pentaborate solution concentration (weight percent) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2.	
Q = pump flow rate (gpm) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b. E = Boron-10 enrichment (atom percent Boron-10) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.	4.4.D Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.
3.4.E If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem operable within 8 hours or the reactor shall be placed in a Shutdown Condition with all operable control rods fully inserted within the following 12 hours.	4.4.E No additional surveillance required.

ANSWER KEY

001	b
002	a
003	d
004	b
005	d
006	d
007	c
008	c
009	a
010	b
011	d
012	c
013	a
014	b
015	b
016	b
017	c
018	c
019	c
020	c
021	c
022	b
023	b
024	a
025	c

ANSWER KEY

026	c
027	a
028	c
029	b
030	c
031	c
032	b
033	c
034	c
035	b
036	b
037	a
038	d
039	c
040	c
041	b
042	a
043	d
044	a
045	b
046	d
047	c
048	d
049	b
050	c

ANSWER KEY

051	b
052	c
053	c
054	b
055	b
056	c
057	c
058	b
059	c
060	d
061	a
062	a
063	c
064	match with selected number in the blank
	a 4
	b 3
	c 1
	d 5
	e 3
065	d
066	b
067	b
068	b
069	b
070	c

ANSWER KEY

071 a

072 c

073 c

074 a

075 d

076 ~~d~~ b [illegible]

077 c

078 d

079 b

080 b

081 match with selected number in the blank

a 3

b 5

c 6

d 2

082 a

083 b

084 d

085 b

086 c

087 d

088 b

089 d

090 b

091 d

A N S W E R K E Y

092 b

093 c

094 d

095 match with selected number in the blank

a 4

b 5

c 3

096 match with selected number in the blank

a 2

b 1

c 3

d 2

(********** END OF EXAMINATION **********)

ANSWER: 001 (1.00)

b

REFERENCE:

LESSONS PLANS: OPL173.901/3C
BF EOI 5214
[3.5/3.9 K/A VALUE]

295031G012 ..(KA's)

ANSWER: 002 (1.00)

a

REFERENCE:

LESSON PLANS: OPL173.901.2
[3.7/4.4 K/A VALUES]

295015G012 ..(KA's)

ANSWER: 003 (1.00)

d

REFERENCE:

LESSON PLANS: OPL173.901/3A
[3.6/4.0 KA VALUES]

295024K301 ..(KA's)

ANSWER: 004 (1.00)

b

REFERENCE:

LESSON PLANS: OP174.822.6 TS 1.1.B
[4.2/4.6 KA VALUES]

295014A205 ..(KA's)

ANSWER: 005 (1.00)

d

REFERENCE:

LESSON PLANS: OPL174.822.6
[3.5/4.3 KA VALUES]

295025G003 ..(KA's)

ANSWER: 006 (1.00)

d

REFERENCE:

LESSON PLANS: OPL173.901/2
[3.9/4.6 KA VALUES]

295037G012 ..(KA's)

ANSWER: 007 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.901/2
[3.9/4.6 KA VALUES]

295037G012 ..(KA's)

ANSWER: 008 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.901.2 EOI 2
[3.9/4.5 KA VALUES]

295024G012 ..(KA's)

ANSWER: 009 (1.00)

a

REFERENCE:

LESSON PLANS : OPL173.901/3D
[3.8/4.1 KA VALUES]

295026K301 ..(KA's)

ANSWER: 010 (1.00)

b.

REFERENCE:

LESSON PLANS: OPL173.920
[4.0/4.1 KA VALUES]

295016K202 ..(KA's)

ANSWER: 011 (1.00)

d

REFERENCE:

LESSON PLANS: OPL176.007: OPL173.828;
[3.9/3.9 KA VALUES]
295009A101 ..(KA's)

ANSWER: 012 (1.00)

c

REFERENCE:

LESSON PLANS: OP;173.901.1
[3.8/4.5 KA VALUE]

295026G012 ..(KA's)

ANSWER: 013 (1.00)

a

REFERENCE:

LESSON PLANS OPL173.917 2-AOI-85-3
[3.7/3.5 KA VALUES]

295022G010 ..(KA's)

ANSWER: 014 (1.00)

b

REFERENCE:

LESSON PLANS OPL173.904 2-AOI-70-1
[3.4/3.3 KA VALUES]

295018G010 ..(KA's)

ANSWER: 015 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.920
[4.1/4.2 KA VALUE]

295016K301 ..(KA's)

ANSWER: 016 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.901/3C
[3.5/3.9 KA VALUES]

295029K301 ..(KA's)

ANSWER: 017 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.901.2

295026G012 ..(KA's)

ANSWER: 018 (1.00)

c

REFERENCE:

LESSON PLANS: OPL174.721.11 2010090101
[3.6/3.6 KA VALUES]

202002A105 ..(KA's)

ANSWER: 019 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.004 2170050101 OPL173.915.1 OPL173.923.1
[4.3/4.3 KA VALUES]

206000K407 ..(KA's)

ANSWER: 020 (1.00)

c

REFERENCE:

LESSON PLANS: OP;176.004 OPL173.917
[3.8/3.8 KA VALUES]

201001K405 ..(KA's)

ANSWER: 021 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.004 OPL174.845/2.8 OPL174.729/3
[3.2/3.1 KA VALUES]

201002A204 ..(KA's)

ANSWER: 022 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.038 LER86-26 OPL176.002
[3.5/3.7 KA VALUES]

241000K101 ..(KA's)

ANSWER: 023 (1.00)

b

REFERENCE:

LESSON PLANS: OPL176.002 OPL174.717/7 OPL172.014
KA21600A211

216000A211 ..(KA's)

ANSWER: 024 (1.00)

a

REFERENCE:

LESSON PLANS: OPL176.001 OPL173.733.1 OPL173.901/2 EOI-1
[3.9/4.6 KA VALUES]

295037G012 ..(KA's)

ANSWER: 025 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.001 OPL174.804.5 OPL173.827
KA239002K301

239002K301 ..(KA's)

ANSWER: 026 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.008
[3.5/3.5 KA VALUES]

259001K411 ..(KA's)

ANSWER: 027 (1.00)

a

REFERENCE:

LESSON PLANS: OPL174.705/3
[3.7/4.1 KA VALUES]

264000A209 ..(KA's)

ANSWER: 028 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.904
KA223001A1.02 [3.6/3.7 KA VALUES]

223001A102 ..(KA's)

ANSWER: 029 (1.00)

b

REFERENCE:

LESSON PLANS: OPL174.819/9
[3.6/3.7 KA VALUES]

212000K114 ..(KA's)

ANSWER: 030 (1.00)

c

REFERENCE:

LESSON PLAN: EDGS
[4.0/4.2 KA VALUES]

264000K402 ..(KA's)

ANSWER: 031 (1.00)

c

REFERENCE:

LESSON PLANS: OPLI73.904 2-AOI-47-2
[3.6/3.5 KA VALUES]

241000G014 ..(KA's)

ANSWER: 032 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.019
KA215004K406[3.2/3.2 KA VALUES]
215004G013 ..(KA's)

ANSWER: 033 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.035.E OPL171.035.F
[3.3/3.5 KA VALUES]

209001K409 ..(KA's)

ANSWER: 034 (1.00)

c

REFERENCE:

LESSON PLANS: OPL174.830 TS 3.7.A.5.6 TS 3.7.A.6.a.1
[3.2/3.6 KA VALUES]

223001G010 ..(KA's)

ANSWER: 035 (1.00)

b

REFERENCE:

LESSON PLAN: OPL171.007
[3.6/3.7 KA VALUES]

202001K101 ..(KA's)

ANSWER: 036 (1.00)

b

REFERENCE:

LESSON PLAN: OPL174.729 OPL176.006 171.024 Rev 2
[3.4/3.5 KA VALUES]

201006K404 ..(KA's)

ANSWER: 037 (1.00)

a

REFERENCE:

LESSON PLANS: OPL173.904 2-AOI-47-3
[3.5/3.6 KA VALUES]

245000A203 ..(KA's)

ANSWER: 038 (1.00)

d

REFERENCE:

LESSON PLANS: OPL174.830 TS 3.3.B.3.C 2-GOI-100-A
[4.0/3.6 KA VALUES]

201004G013 ..(KA's)

ANSWER: 039 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.004 OPL173.901.2
[3.4/3.5 KA VALUES]
295015G007 ..(KA's)

ANSWER: 040 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.004 OPL174.845/8
KA214000A2.02 [3.6/3.7 KA VALUES]

214000A202 ..(KA's)

ANSWER: 041 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.904
[3.8/3.9 KA VALUES]

239001A210 ..(KA's)

ANSWER: 042 (1.00)

a

REFERENCE:

LESSON PLANS: OPL173.904 2-GOI-100-1C
[3.7/3.8 KA VALUES]

201003A101 ..(KA's)

ANSWER: 043 (1.00)

d

REFERENCE:

LESSON PLANS: OPL173.904 2-AOI-74-1
[3.1/3.3 KA VALUES]

205000A201 ..(KA's)

ANSWER: 044 (1.00)

a

REFERENCE:

LESSON PLANS: OPL173.904 2-AOI-66-1
[3.5/3.9 KA VALUES]

271000A206 ..(KA's)

ANSWER: 045 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.903/4 171.044 LO 11
[3.6/3.5 KA VALUES]

205000A402 ..(KA's)

ANSWER: 046 (1.00)

d

REFERENCE:

LESSON PLANS: OPL173.904 2-AOI-85-6
[3.4/3.7 KA VALUES]

201003A203 ..(KA's)

ANSWER: 047 (1.00)

c

REFERENCE:

LESSON PLANS: 171.007 LO D
[3.6/3.6; 3.0/3.1; 3.3/3.3 KA VALUES]

202001K105 202001K404 202001A109 ..(KA's)

ANSWER: 048 (1.00)

d

REFERENCE:

LESSON PLANS: 171.033
[3.7/4.1 KA VALUES]

272000K402 ..(KA's)

ANSWER: 049 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.060 EO B.5
EO B.5
[3.1/3.3 KA VALUES]

234000G013 ..(KA's)

ANSWER: 050 (1.00)

c

REFERENCE:

LESSON PLAN:
BFSP NO. 6740
[3.7/3.7]

294001K101 ..(KA's)

ANSWER: 051 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.086 PMI 12.15
BFSP: NO.4769
[3.9/4.5]

294001K102 ..(KA's)

ANSWER: 052 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.065 3430220302
BFBRQ: NO. 5267
[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 053 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.913 OPL171.075
BFREP: NO.5597
[2.9/4.7]

294001A116 ..(KA's)

ANSWER: 054 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.830 2-GOI-100-1
BFGOI: NO. 6736
[3.9/3.9]
295014A202 ..(KA's)

ANSWER: 055 (1.00)

b

REFERENCE:

LESSON PLANS: OPL174.822.6 TS 1.1.B
BFTS: NO.4545
[3.8/4.4]

295006G003 ..(KA's)

ANSWER: 056 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.901.1
[3.9/4.5 KA VALUES]

295024G012 ..(KA's)

ANSWER: 057 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.039
KA211000K105 [3.4/3.6 KA VALUES]

211000K105 ..(KA's)

ANSWER: 058 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.043
KA218000K403 [3.8/3.8 KA VALUES]

218000K403 ..(KA's)

ANSWER: 059 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.043
KA218000K105 [3.9/3.9]

218000K105 ..(KA's)

ANSWER: 060 (1.00)

d

REFERENCE:

LESSON PLANS: OPL171.040
KA217000A201 [3.8/3.7 KA VALUES]

217000A201 ..(KA's)

ANSWER: 061 (1.00)

a

REFERENCE:

LESSON PLANS: OPL171.017
KA223002G004 [3.6/3.7 KA VALUES]

223002G004 ..(KA's)

ANSWER: 062 (1.00)

a

REFERENCE:

LESSON PLANS: OPL171.084 LO2
KA268000K502 [3.1/3.6 KA VALUES]

268000K502 ..(KA's)

ANSWER: 063 (1.00)

c

REFERENCE:

LESSON PLANS: 2AOI-74-1
KA295021A102 [3.3/3.4 KA VALUES]

295021A102 ..(KA's)

ANSWER: 064 (2.50)

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

REFERENCE:

OPL 171.028 EO B.5
4.0/4.2 3.7/3.9 3.4/3.6 .

212000K112 212000G007 212000K101 ..(KA's)

ANSWER: 065 (1.00)

d

REFERENCE:

OPL 171.043
3.8/4.0 3.8/3.8 3.9/4.1

218000K501 218000K403 218000K601 ..(KA's)

ANSWER: 066 (1.00)

b

REFERENCE:

OPL 171.025
3.4/3.6 3.4/3.5 3.2/3.3

201002K105 201002K106 201002K103 ..(KA's)

ANSWER: 067 (1.00)

b

REFERENCE:

OPL171.044 EO B.11
3.4/3.6 3.5/3.7 3.2/3.4

226001K102 226001K101 226001K409 ..(KA's)

ANSWER: 068 (1.00)

b

REFERENCE:

OPL 171.077
3.4/3.3

212000A111 ..(KA's)

ANSWER: 069 (1.00)

b

REFERENCE:

EOI-1 OPL171.082
4.4/4.5 4.0/4.2 4.3/4.5

295037K209 295037K204 295037K302 ..(KA's)

ANSWER: 070 (1.00)

c

REFERENCE:

AOI-85-4
3.3/3.1 3.5/3.3

214000G014 214000G008 ..(KA's)

ANSWER: 071 (1.00)

a

REFERENCE:

OPL 171.019 EO B.2
3.3/3.5 2.9/2.9 3.0/3.1

291002K122 215004A205 215004K604 ..(KA's)

ANSWER: 072 (1.00)

c

REFERENCE:

OPL 171.080, OPL 171.012
3.5/3.5

259002K605 ..(KA's)

ANSWER: 073 (1.00)

c

REFERENCE:

SDSP-11.11
3.2/3.7

294001K105 ..(KA's)

ANSWER: 074 (1.00)

a

REFERENCE:

SDSP-3.15
3.7/3.7

294001K101 ..(KA's)

ANSWER: 075 (1.00)

d

REFERENCE:

Technical Specification 3.1 , OPL171.028
4.2/4.3

212000G004 ..(KA's)

ANSWER: 076 (1.00)

~~d~~ b ~~AA~~

REFERENCE:
BFN FPP-3

3.5/3.8

294001K116 ..(KA's)

ANSWER: 077 (1.00)

c

REFERENCE:

2-AOI-85-3
3.4/3.3 3.3/3.4 3.7/4.0

295022G011 201001G015 201001A201 ..(KA's)

ANSWER: 078 (1.00)

d

REFERENCE:

2-AOI-68
3.5/3.6

295001A101 ..(KA's)

ANSWER: 079 (1.00)

b

REFERENCE:

2-AOI-79-1
3.4/4.1

295023A204 ..(KA's)

ANSWER: 080 (1.00)

b

REFERENCE:

TS
3.3/4.2

295013G003 ..(KA's)

ANSWER: 081 (2.00)

a. 3
b. 5
c. 6
d. 2

REFERENCE:

2-AOI-47-3
3.5/3.5 3.1/3.2

295002K202 295002K201 ..(KA's)

ANSWER: 082 (1.00)

a

REFERENCE:

PMI 15.8
[3.1/3.8 KA VALUES]

215005G002 ..(KA's)

ANSWER: 083 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.901.2
BFEOI: NO: 3714
[4.2/4.2]

294001A101 ..(KA's)

ANSWER: 084 (1.00)

d

REFERENCE:

Technical Specification 3.4
OPL171.039 V.B.9
3.4/4.1 3.6/4.4

211000G011 211000G005 ..(KA's)

ANSWER: 085 (1.00)

b

REFERENCE:

Technical Specification 3.2.H, 3.2 Bases
EPIP 01
3.2/4.1 2.2/3.6 3.7/3.8

295002G011 295002G008 295002G004 ..(KA's)

ANSWER: 086 (1.00)

c

REFERENCE:

10CFR 20.5
2.8/3.4

294001K103 ..(KA's)

ANSWER: 087 (1.00)

d

REFERENCE:

10CFR20
2.8/3.4

294001K103 ..(KA's)

ANSWER: 088 (1.00)

b

REFERENCE:

EPIP-15, RCI-1
3.3/3.8 2.9/4.7

294001A116 294001K103 ..(KA's)

ANSWER: 089 (1.00)

d

REFERENCE:

SDSP-34.2
3.6/4.2

294001A110 ..(KA's)

ANSWER: 090 (1.00)

b

REFERENCE:

SDSP-14.9
3.9/4.5
294001K102 ..(KA's)

ANSWER: 091 (1.00)

d

REFERENCE:

PMI-12.12
2.7/3.7

294001A103 ..(KA's)

ANSWER: 092 (1.00)

b

REFERENCE:

TECHNICAL SPECIFICATION 3.6B
2.9/3.4

294001A114 ..(KA's)

ANSWER: 093 (1.00)

c

REFERENCE:

EOI-2, OPL171.057
3.8/4.0 3.6/3.8

295010G007 295010K301 ..(KA's)

ANSWER: 094 (1.00)

d

REFERENCE:

OPL171.047
3.1/3.2

295018A202 ..(KA's)

ANSWER: 095 (1.50)

a. 4
b. 5
c. 3

REFERENCE:

TECHNICAL SPECIFICATION 3.6B
2.9/3.4

294001A114 ..(KA's)

ANSWER: 096 (2.00)

a. 2
b. 1
c. 3
d. 2

REFERENCE:

EPIP-1, PMI 15.4, TECH SPEC 3.4
2.9/4.7

294001A116 ..(KA's)

(********** END OF EXAMINATION **********)

QUESTION	VALUE	REFERENCE
001	1.00	9000488
002	1.00	9000489
003	1.00	9000490
004	1.00	9000491
005	1.00	9000492
006	1.00	9000493
007	1.00	9000494
008	1.00	9000495
009	1.00	9000496
010	1.00	9000497
011	1.00	9000498
012	1.00	9000499
013	1.00	9000500
014	1.00	9000501
015	1.00	9000502
016	1.00	9000503
017	1.00	9000504
018	1.00	9000505
019	1.00	9000506
020	1.00	9000507
021	1.00	9000508
022	1.00	9000509
023	1.00	9000510
024	1.00	9000511
025	1.00	9000512
026	1.00	9000513
027	1.00	9000514
028	1.00	9000515
029	1.00	9000516
030	1.00	9000517
031	1.00	9000519
032	1.00	9000520
033	1.00	9000521
034	1.00	9000523
035	1.00	9000524
036	1.00	9000525
037	1.00	9000526
038	1.00	9000527
039	1.00	9000528
040	1.00	9000529
041	1.00	9000530
042	1.00	9000531
043	1.00	9000532
044	1.00	9000533
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047	1.00	9000536
048	1.00	9000538
049	1.00	9000540
050	1.00	9000541
051	1.00	9000543
052	1.00	9000545
053	1.00	9000547
054	1.00	9000548

QUESTION	VALUE	REFERENCE
055	1.00	9000549
056	1.00	9000550
057	1.00	9000551
058	1.00	9000552
059	1.00	9000553
060	1.00	9000555
061	1.00	9000556
062	1.00	9000558
063	1.00	9000560
064	2.50	9000561
065	1.00	9000562
066	1.00	9000563
067	1.00	9000564
068	1.00	9000565
069	1.00	9000566
070	1.00	9000567
071	1.00	9000569
072	1.00	9000570
073	1.00	9000740
074	1.00	9000741
075	1.00	9000746
076	1.00	9000747
077	1.00	9000752
078	1.00	9000753
079	1.00	9000754
080	1.00	9000756
081	2.00	9000757
082	1.00	9000518
083	1.00	9000542
084	1.00	9000734
085	1.00	9000735
086	1.00	9000736
087	1.00	9000737
088	1.00	9000738
089	1.00	9000739
090	1.00	9000742
091	1.00	9000743
092	1.00	9000745
093	1.00	9000748
094	1.00	9000750
095	1.50	9000744
096	2.00	9000751

	100.00	

	100.00	

FU1

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

FISSION PRODUCT BARRIER DEGRADATION

UNUSUAL EVENT

Technical Specification 3.6.B.6 Exceeded For Iodine Spike (26 uCi/gm Dose Equivalent Iodine-131 in Coolant) by Radiochemical Laboratory Sample

OPERATING MODE APPLICABILITY: - All

BASIS:

The basis for this coolant activity is a computed dose to the thyroid of 36 REM at the exclusion area distance during the two-hour period following a steam line break accident. Technical Specification Basis 3.6.B.

NUREG-0654

High Coolant Activity Sample

FU2

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

FISSION PRODUCT BARRIER DEGRADATION

UNUSUAL EVENT

Main Steam Line Radiation Exceeding 1.5 X Normal Full Power Background (Alarm)

OPERATING MODE APPLICABILITY: - All
(Not required if steam lines were isolated before event)

BASIS:

Radiation levels at 1.5 times normal are indications of abnormal coolant activity spikes and possible abnormal core behavior. Technical Specification Basis 3.1, NUREG-0654.

FU3

Initiating Conditions and Emergency Action Levels For Notification of Unusual Event

FISSION PRODUCT BARRIEB DEGRADATION

UNUSUAL EVENT

Leakage Exceeding Tech Spec 3.6.C.1
1) Exceeding Five (5) gpm Unidentified Leakage, or 2) Exceeding Twenty Five (25) gpm Total Leakage, or 3) Exceeding Two (2) gpm Increase From Unidentified Sources in a 24-Hour Period (Except for the First 24-Hour Period After Going to RUN Mode) from either Control Room indication/alarm of Sump run times/fill rates, or SI-2.

OPERATING MODE APPLICABILITY: - Fuel in the vessel and coolant temperature is >212°F

BASIS:

For leakage on the order of 5 gpm, the probability is small that a crack or imperfection would propagate rapidly. The drywell sump pumps can each pump 40-45 gpm, so the 25 gpm limit is conservative. Technical Specification 3.6.C, NUREG-0654.

FU4

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

FISSION PRODUCT BARRIER DEGRADATION

UNUSUAL EVENT

Main Steam Relief or Safety-Related System Relief Valve Opening and Failure to Reclose as Expected (High Temp (TR-1-1) or Flow (FA-1-1) or Control Room Alarm (9-3), or Operator observation (Safety-Related Relief Valves)

OPERATING MODE APPLICABILITY: - Main Steam Relief Valves; whenever Reactor Coolant Temperature is greater than 212°F

ECCS operability is required

Safety-Related Relief Valves; any time ECCS operability is required

BASIS:

These are indicative of a degradation of the coolant boundary such that the cooldown rate of <100°F/hr. (Tech Spec 3.6.A.1) may be exceeding, or degradation of flow capability of an ECCS system if the system has a relief valve stuck open. NUREG-0654

FU5

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

FISSION PRODUCT BARRIER DEGRADATION

UNUSUAL EVENT

ECCS Initiation With Discharge to Vessel (HPCI, RCIC, LPCI, CSS, or ADS) Unless Part of a Planned Sequence During Testing or Maintenance

OPERATING MODE APPLICABILITY: - Fuel in the reactor vessel

BASIS:

The signals which cause auto initiation of these systems are indicative of a degraded plant condition far in excess of normal plant transients. If the auto initiation signals were false, the resultant plant transient would be minor. FSAR N6.5.4.1.4, NUREG-0654.

FU6

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

FISSION PRODUCT BARRIER DEGRADATION

UNUSUAL EVENT

Drywell leakage of greater than 542 SCFH determined by surveillance

OPERATING MODE APPLICABILITY: - Whenever Primary Containment is Required

BASIS:

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the volume assumed in the accident analysis. Technical Specification 3.7.A.2.C.

NUREG-0654

Exceeding either primary/secondary containment leak rate technical specification or primary system leak rate technical specification.

FU7

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

FISSION PRODUCT BARRIER DEGRADATION

UNUSUAL EVENT

1.0 Paragraph 0, Definition Not Met

OPERATING MODE APPLICABILITY: - Whenever Primary Containment is required

0. Primary Containment Integrity - Primary Containment integrity means that the Drywell and Pressure Suppression Chamber are intact and all of the following conditions are satisfied:

   1. All non-automatic containment isolation valves on lines connected to the reactor coolant systems or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

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

   3. All automatic containment isolation valves are operable or each line which contains an inoperable valve is isolated as required by Specification 3.7.D.2.

   4. All blind flanges and manways are closed.

BASIS:

The integrity of primary containment along with the operation of the ECCS systems ensure that the release of radioactive materials from the containment will be restricted to those release paths assumed in the accident analysis. Technical Specification 3.7.A basis, NUREG-0654.

FU8

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

FISSION PRODUCT BARRIER DEGRADATION

UNUSUAL EVENT

Isolation of Offgas System Due to Post Treatment Monitoring, RM-90-265/266 or Pretreatment Monitor RM-90-157 Alarms or 0.05 mrem/hr Above Background for One Hour at or Beyond Site Boundary

OR

Any Measurable Iodine at or Beyond Site Boundary

OPERATING MODE APPLICABILITY: - All

BASIS:

This condition, which could cause an isolation of the offgas system, is indicative of a fuel failure. NUREG-0654, Tech Spec 3.2.k.1.

FA1

Initiating Conditions and Emergency Action Levels
For Alert

FISSION PRODUCT BARRIER DEGRADATION

ALERT

Dose Equivalent Iodine Concentration >300 uCi/cc in Coolant by Radiochemical Laboratory Analysis.

OPERATING MODE APPLICABILITY: - All

BASIS:

Concentrations on this order of magnitude indicate damage to significantly more than one fuel rod, and extrapolates to less than the 10 CFR 100 limit (300 rem) for a steam line break. Technical Specification Basis 3.6.B/4.6.B, NUREG-0654.

NUREG-0654

Very High Coolant Activity Sample

FA2

Initiating Conditions and Emergency Action Levels
For Alert

FISSION PRODUCT BARRIER DEGRADATION

ALERT

Main Steam Line Radiation Exceeding 3 X Normal Full Power Background (Alarm and Group 1 Isolation)

OPERATING MODE APPLICABILITY: - All
(Not required if Main Steam Lines are isolated before the event.)

BASIS:

Radiation levels 3 times normal are indications of leaking fuel or activated contaminent leaving the vessel. Technical Specification Basis 3.1, NUREG-0654.

FA3

Initiating Conditions and Emergency Action Levels For Alert

FISSION PRODUCT BARRIER DEGRADATION

ALERT

Notification by Refuel Floor SRO of a Dropped or Damaged Fuel Bundle with a Release of Activity Indicated by One or More of the Following:

1. Refueling Zone Ventilation Isolation
2. Continuous Air Monitor Alarm RM-90-250
3. Area Radiation Monitors RM-90-1,2, or 3 alarm
4. Analysis of the Refuel Floor atmosphere indicates fission product release

OPERATING MODE APPLICABILITY: - All

BASIS:

Damage to a fuel element with a release of radioactive material has the potential for degradation of the plant safety level, release of radiation to the environment, and warrants increased monitoring. FSAR Section 14, NUREG-0654

FA4

Initiating Conditions and Emergency Action Levels For Alert

FISSION PRODUCT BARRIER DEGRADATION

ALERT

Total Primary System Leakage Exceeding 40 gpm (Identified and Unidentified Leakage)

Based on pumping rate as calculated in SI-2 or SOS judgment when isolated.

OPERATING MODE APPLICABILITY: - Fuel in the vessel and coolant temperature is >212°F

BASIS:

Leakage rates in excess of technical specifications indicates a potentially worsening problem that could challenge plant integrity. A leakage rate of 150 gpm has been calculated to be the minimum liquid leakage from a crack enough to propagate. The limit of 40 gpm allows time for corrective action. Also the NUREG-0654 suggested limit of 50 gpm can't be easily detected because a pump will only pump 40-45 gpm. If the pump(s) runs continuously leakage is greater than 40 gpm. FSAR 4.10.3.2, NUREG-0654.

FA5

Initiating Conditions and Emergency Action Levels
For Alert

FISSION PRODUCT BARRIER DEGRADATION

ALERT

Main Steam Line Not Isolated When Required
(Both Inboard and Outboard MSIVs fail to close on any line when an automatic or manual isolation required)

OPERATING MODE APPLICABILITY: - Reactor coolant temperature >212°F

BASIS:

This condition indicates a direct path from the reactor vessel to areas outside primary containment, and is indicative of a large failure of primary containment. Technical Specification 3.7.A, 3.7.D, 1.0.

NUREG-0654

Steam Line Break with MSIV malfunction causing leakage.

FS1

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

FISSION PRODUCT BARRIER DEGRADATION

SITE AREA EMERGENCY

Notification of Major Damage to Spent Fuel Bundles, or Loss of Spent Fuel Pool Water Level to the Extent that One or More Spent Fuel Bundles are Exposed.

OPERATING MODE APPLICABILITY: - All

BASIS:

NUREG/CR-4982, indicates that the long term health effects from this class of accident are potentially large, no prompt fatalities are predicted, and the risk of injury is low. Thus, a Site Area Emergency is appropriate for this event. FSAR Section 14.

NUREG-0654

Major damage to spent fuel in containment or fuel handling building.

FS2

Initiating Conditions and Emergency Action Levels For Site Area Emergency

FISSION PRODUCT BARRIER DEGRADATION

SITE AREA EMERGENCY

Main Steam Line Break Outside Containment Without the Ability to Isolate

OPERATING MODE APPLICABILITY: - Reactor coolant temperature >212°F.

BASIS:

This condition indicates a direct path from the reactor vessel to areas outside primary containment, under an accident condition. The PCIS system has not performed its function to limit the accident consequences. Because of this failure, progression to a General Emergency is possible.

NUREG-0654

BWR Steamline Break Outside Containment Without Isolation.

FS3

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

FISSION PRODUCT BARRIER DEGRADATION

SITE AREA EMERGENCY

Loss of coolant inventory greater than makeup pump capacity or without capability to restore. Reactor Water Level Decreases Below -150 inches. (Possible escalation to General Emergency dependent on length of time and fuel damage.)

OPERATING MODE APPLICABILITY: - Fuel in the vessel. (Refer to Note at 3.1)

BASIS:

Fuel may become uncovered and fuel damage can occur. The ECCS are designed to maintain reactor water level above -150 inches after some initial uncovery. BFNP-FSAR

NUREG-0654

Loss of coolant accident greater than makeup pump capacity.

FS4

Initiating Conditions and Emergency Action Levels For Site Area Emergency

FISSION PRODUCT BARRIER DEGRADATION

SITE AREA EMERGENCY

Break of Large Reactor Coolant System Line (RWCU - HPCI Injection - HPCI Steam - RCIC Injection - RCIC Steam) Outside Primary Containment Without Isolation

OPERATING MODE APPLICABILITY: - Reactor coolant temperature >212° F

BASIS:

This condition indicates a direct path from the reactor vessel to areas outside primary containment under an accident condition. The PCIS system has not performed its function to limit the accident consequences. Because of this failure progression to General Emergency is possible. (NUREG-0654)

FS5

Initiating Conditions and Emergency Action Levels For Site Area Emergency

FISSION PRODUCT BARRIER DEGRADATION

SITE AREA EMERGENCY

Degraded Core With Possible Loss of Coolable Geometry

OPERATING MODE APPLICABILITY: - All

BASIS:

This event is indicative of a situation that even with ECCS available the potential exists for severe releases to the environment. Could be indicated by problems with driving TIPS, IRMs, and SRMs.

NUREG-0654

Degraded core with possible loss of coolable geometry.

FG1

Initiating Conditions and Emergency Action Levels
For General Emergency

FISSION PRODUCT BARRIER DEGRADATION

GENERAL EMERGENCY

ECCS Failure (Pumps Not Running or Unable to Maintain Reactor Water Level Leading to Fuel Melt)

OPERATING MODE APPLICABILITY: - Fuel in the vessel

BASIS:

Probable dangerous exposure by plant personnel and the general public. This event is beyond those described in the FSAR, and is best captured by a General Emergency. NUREG-0654.

FG2

Initiating Conditions and Emergency Action Levels
For General Emergency

FISSION PRODUCT BARRIER DEGRADATION

GENERAL EMERGENCY

Drywell Pressure >50 psig or Drywell Temperature >280°F.

OPERATING MODE APPLICABILITY: - All

BASIS:

This condition indicates a pending breach of the reactor containment and reaching the containment pressure and temperature calculated for a design basis event. The drywell design pressure is 56 psig, and the design temperature is 281°F. FSAR 5.2.3.2, NUREG-0654

FG3

Initiating Conditions and Emergency Action Levels
For General Emergency

FISSION PRODUCT BARRIER DEGRADATION

GENERAL EMERGENCY

Loss of Any 2 of 3 Fission Product Barriers With a Potential Loss of Third Barrier.

Barriers are:

1. Fuel Cladding

2. Primary Coolant Boundary

3. Primary Containment

OPERATING MODE APPLICABILITY:
- Shutdown
- Hot Standby
- Startup
- Run

BASIS:

This indicates severe core damage, probable ECCS failure, and failures beyond the FSAR analysis. NUREG-0654

SU1

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

SYSTEM MALFUNCTION

UNUSUAL EVENT

Tech Spec LCO Reached Requiring Shutdown

OPERATING MODE APPLICABILITY: - Other Than Shutdown

BASIS:

The limiting conditions for operation specify the minimum acceptable levels of the system performance necessary to ensure the safe startup and operation of the facility. Failure to meet these conditions results in a reduced margin of safety for the plant operation. Tech Spec 1.C, NUREG-0654.

SU2

Initiating Conditions and Emergency Action Levels
For Unusual Event

SYSTEM MALFUNCTION

UNUSUAL EVENT

Reactor SCRAM Due to Failure of Main Turbine Rotating Component

OPERATING MODE APPLICABILITY: - Run

BASIS:

Loss of the turbine with the reactor at power can result in a significant addition of positive reactivity, and can cause the loss of the normal heat sink. FSAR page 7.2-8, #4 and 5. NUREG-0654

SU3

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

SYSTEM MALFUNCTION

UNUSUAL EVENT

Loss of All Offsite Power OR Onsite AC Power Supply to Any Unit

a. Loss of voltage to both 4kV shutdown buses (units 1 and 2), or loss of voltage from all offsite sources to all 4kV shutdown boards on unit 3.

OR

b. Two or more unit-related diesel generators inoperable simultaneously by unscheduled outage or failure (when not in cold shutdown).

OPERATING MODE APPLICABILITY: - All

BASIS:

This initiating condition is based on severe degradation of normal/emergency AC sources. FSAR Section 8.4.4.2 states "The normal and standby auxiliary (electric) sources shall be sufficient in number and of such electrical and physical independence that no single event, as a minimum requirement, can negate all auxiliary power at one time." This initiating condition would place the plant in a limiting condition for operation per Technical Specification 3.9.B. The objective of this specification is to ensure an adequate source of electrical power to operate facilities to cool the plant during shutdown and to operate engineered safeguards following an accident.

This initiating condition can affect more than one unit, and in that event, escalation of the emergency classification should be evaluated.

NUREG-0654

Loss of offsite power or loss of outside AC power capability.

SU5

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

SYSTEM MALFUNCTION

UNUSUAL EVENT

Loss of Computer Requiring Shutdown

OPERATING MODE APPLICABILITY: - $\geq$25% rated thermal power

BASIS:

Tech Spec 3.5.I, J, K, and L require core thermal limits to be determined daily whenever $\geq$25% rated thermal power. These LCOs cannot be met without the process computer. NUREG-0654.

SU4

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event.

SYSTEM MALFUNCTION

UNUSUAL EVENT

Loss of Control Room Surveillance Instruments/Alarms Listed in Technical Specification Table 3.2.F Requiring Shutdown

OPERATING MODE APPLICABILITY: - Fuel in the vessel and not in shutdown

BASIS:

There are two channels of instrumentation listed in table 3.2.F. By comparing readings between channels, a near continuous surveillance of plant parameters is possible and any deviation in readings will initiate early investigation thereby maintaining the quality of the systems monitored. Technical Specification Basis 3.2, NUREG-0654.

SU6

Initiating Conditions and Emergency Action Levels
For Unusual Event

SYSTEM MALFUNCTION

UNUSUAL EVENT

Loss of All Meteorological Instrumentation. (Both RRRMS and at Meteorological Tower) Listed in Tech Spec Table 3.2.I Except for Planned Outages Where Compensating Measures are in Place

OPERATING MODE APPLICABILITY: - All

BASIS:

The operability of the meteorological instrumentation ensures that sufficient data is available for estimating potential radiation doses to the public as a result of routine or accidental releases. A planned outage provides for planned compensatory measures to make this information available to the SOS. Technical Specification Basis 3.2, NUREG-0654.

SU7

Initiating Conditions and Emergency Action Levels
For Unusual Event

SYSTEM MALFUNCTION

UNUSUAL EVENT

Loss of RPIS Indication or Alarms

OPERATING MODE APPLICABILITY: - Refuel
- Startup
- Run

BASIS:

Indication of Control Rod position is required to adequately monitor reactor core condition and ensure shutdown margin. Technical Specification Basis 3.3.A.2, NUREG-0654.

SU8

Initiating Conditions and Emergency Action Levels
For Unusual Event

SYSTEM MALFUNCTION

UNUSUAL EVENT

Significant Loss of Offsite Telephone Communication Capability for more than 10 Minutes

OPERATING MODE APPLICABILITY: - All

BASIS:

NUREG-0654 requirement to ensure offsite communications are available to summon essential assistance and notify State and local authorities.

SU9

Initiating Conditions and Emergency Action Levels For Notification of Unusual Event

SYSTEM MALFUNCTION

UNUSUAL EVENT

Loss of Main Steam Line Radiation, Flow or Area Temperature Instrumentation Listed in Table 3.2.A, Requiring Shutdown

OPERATING MODE APPLICABILITY: - Startup
- Run

BASIS:

The temperature setting of 200°F is low enough to detect leaks on the order of 15 gpm and is therefore capable of covering the entire spectrum of breaks. For large breaks, the flow instruments serve as backup to the temperature indication. Technical Specification Basis 3.2, NUREG-0654.

SA1

Initiating Conditions and Emergency Action Levels For Alert

SYSTEM MALFUNCTION

ALERT

Failure to Initiate and Complete SCRAM to Bring the Reactor Subcritical from Any Expected Scram Signal

OPERATING MODE APPLICABILITY: - Reactor Critical

BASIS:

This condition indicates a failure of the reactor protection system to SCRAM the reactor and/or insert adequate control rods to make the reactor subcritical. NUREG-0654

SA2

Initiating Conditions and Emergency Action Levels
For Alert

SYSTEM MALFUNCTION

ALERT

Loss of Capability to Attain or Maintain Cold Shutdown Conditions Using All Available Systems

OPERATING MODE APPLICABILITY: - All

BASIS:

The decay heat is beyond the capabilities of available systems and a radiation release is possible.

NUREG-0654

Complete loss of any function needed for plant cold shutdown.

SA3

Initiating Conditions and Emergency Action Levels
For Alert

SYSTEM MALFUNCTION

ALERT

Main Turbine Mechanical Failure Resulting in Penetration of Turbine Casing

OPERATING MODE APPLICABILITY: - Main Turbine/Condenser being used for Heat Sink

BASIS:

Loss of the turbine with the reactor at power can result in a significant addition of positive reactivity, and can cause the loss of the normal heat sink. This condition could create an unanticipated path for the release of radioactive material and removes the normal heat sink from service. FSAR page 7.2-8.

NUREG-0654

Turbine failure causing casing penetration.

SA4

Initiating Conditions and Emergency Action Levels
For Alert

SYSTEM MALFUNCTION

ALERT

Loss of All Offsite (1) and All Onsite (3) AC Power Supply to Any Unit

(Potential Escalation to Site Area Emergency)

(1) Indicated by loss of voltage to both 4KV Shutdown Buses (Units 1 & 2), or loss of voltage from all offsite sources to all 4KV Shutdown Boards on Unit 3.

AND

(3) Loss of voltage to all 4KV Shutdown Boards on any unit.

OPERATING MODE APPLICABILITY: - All

BASIS:

A complete loss of offsite and onsite AC power would compromise all plant safety systems requiring AC electrical power including RHR, Core Spray, RHRSW, and EECW.

NUREG-0654

Loss of offsite power and loss of all onsite AC power.

SA5

Initiating Conditions and Emergency Action Levels
For Alert

SYSTEM MALFUNCTION

ALERT

Loss of Any 250V DC Unit Battery Board and Not in Cold Shutdown

OPERATING MODE APPLICABILITY: - All but Cold Shutdown

BASIS:

A loss of DC power compromises the ability to monitor and control plant safety functions. A prolonged loss of all DC could cause core uncovering and loss of containment integrity. NUREG-0654

SA6

Initiating Conditions and Emergency Action Levels For Alert

SYSTEM MALFUNCTION

ALERT

Loss of All Annunciators in Group I (Panel 9-3A thru F) or II (Panel 9-4A, 4B and 9-5A, 5B) for 15 Minutes

OPERATING MODE APPLICABILITY: - All

BASIS:

Loss of operator early warning system for significant safety and/or reactor monitoring systems.

NUREG-0654

Most or all alarms (annunicators) lost.

SS1

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

SYSTEM MALFUNCTION

SITE AREA EMERGENCY

Core Thermal Power Greater than 3% and Reactor Critical When Shutdown Required. This Power Level is After Initiating a Manual SCRAM, Mode Switch in Shutdown, and Recirculation Pumps Tripped.

OPERATING MODE APPLICABILITY:
- Hot Standby
- Startup
- Run

BASIS:

Under these conditions, the reactor is producing more decay heat load than the safety systems are designed to remove. Reactor shutdown may not be accomplished without threatening primary containment. NUREG-0654.

SS2

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

SYSTEM MALFUNCTION

SITE AREA EMERGENCY

Loss of All Offsite (a) Power AND Onsite (b) AC Power Supply to Any Unit for More Than 15 Minutes

Loss of all Offsite AND Onsite AC Power Capability as indicated by:

a. Loss of voltage to both 4kV Shutdown buses (Units 1&2), or loss of voltage from all offsite sources to all 4kV Shutdown boards on Unit 3.

AND

b. Loss of voltage to all 4kV Shutdown boards on any unit.

OPERATING MODE APPLICABILITY: - All

BASIS:

Loss of all offsite power compromises all plant safety systems requiring electric power including ECCS, Containment Cooling, and the Heat Sink provided by the Pressure Suppression Chamber Water. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity. Thus, this event can escalate to a General Emergency.

This initiating condition can affect more than one unit, and in that event, escalation of the emergency classification should be evaluated.

NUREG-0654

Loss of offsite power and loss of onsite AC power for more than 15 minutes.

SS3

Initiating Conditions and Emergency Action Levels For Site Area Emergency

SYSTEM MALFUNCTION

SITE AREA EMERGENCY

Loss of All Control Room Annunciators in Group I (Panel 9-3A thru F) or II (Panel 9-4A, 4B and 9-5A, 5B) and Plant Transient Initiated or in Progress

OPERATING MODE APPLICABILITY: - Fuel in Vessel

BASIS:

These are significant safety system annunciators which assist the operator in mitigating plant transients.

NUREG-0654

Most or all alarms (annunciators) lost and plant transient initiated or in progress.

SS4

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

SYSTEM MALFUNCTION

SITE AREA EMERGENCY

Loss of All 250V DC Unit Battery Boards for 15 Minutes

OPERATING MODE APPLICABILITY: - All

BASIS:

A loss of DC power compromises the ability to monitor and control plant safety functions. A prolonged loss of all DC could cause core uncovering and loss of containment integrity.

NUREG-0654

Loss of all vital onsite DC power for more than 15 minutes.

RU1

Initiating Conditions and Emergency Action Levels
For Unusual Event

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

UNUSUAL EVENT

Liquid Release Exceeding Tech Spec 3.8.A.1, 3, or 5

OPERATING MODE APPLICABILITY: - All

BASIS:

The concentration of radioactive material released in liquid effluents from the site will be less than the levels specified in 10 CFR 20. Technical Specification Basis 3.8.A.

NUREG-0654

Radiological Effluent Technical Specification Limits Exceeded

RU2

Initiating Conditions and Emergency Action Levels
For Unusual Event

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

UNUSUAL EVENT

Gas Release Exceeding Tech Spec 3.8.B.1, 3 or 5
Based on Calculations from 0-SI-4.8.B.1.a.1.2.1(2 or 3) SI-4.8.B.5.a OR 1 (2) SI-4.8.B.1.a.3 or 3-SI-4.8.B.1.a

OPERATING MODE APPLICABILITY: - All

BASIS:

This ensures that the dose at any time at the exclusion area boundary from gaseous effluents will be within the annual dose rate limits of 10 CFR 20 for unrestricted areas. Technical Specification Basis 3.8.B, NUREG-0654.

NUREG-0654

Radiological Effluent Technical Specification Limits Exceeded

RA1

Initiating Conditions and Emergency Action Levels For Alert

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

ALERT

Liquid Release Exceeding 10 Times Technical Specification 3.8.A.1, 3 or 5

OPERATING MODE APPLICABILITY: - All

BASIS:

NUREG-0654 and 10 CFR 20.105 allowable limits form the basis for this condition.

NUREG-0654

Radiological effluents greater than 10 times technical specification limits

RA2

Initiating Conditions and Emergency Action Levels
For Alert

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

ALERT

Liquid Release Exceeding 10 CFR 20 Limits by a Factor of 10

OPERATING MODE APPLICABILITY: - All

BASIS:

NUREG-0654.

RA3

Initiating Conditions and Emergency Action Levels For Alert

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

ALERT

Liquid Release That Cannot Be Terminated

OPERATING MODE APPLICABILITY: - All

BASIS:

State of Alabama commitment.

RA4

Initiating Conditions and Emergency Action Levels
For Alert

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

ALERT

Gas Release Exceeding 10 Times Technical Specification 3.8.B.1, 3 Based on Calculations from SI-4.8.B.1.a.1, SI-4.8.B.1.A.2, SI-4.8.B.1.A.3, SI-4.8.B.1.5.A or Environmental Measurements at or Beyond the Site Boundary of 0.5 mR/hr for One Hour Whole Body or 4.9 E-10 uCi/cc Iodine

OPERATING MODE APPLICABILITY: - All

BASIS:

NUREG-0654

Radiological effluents greater than 10 times technical specification instantaneous limits.

RA5

Initiating Conditions and Emergency Action Levels
For Alert

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

ALERT

Radiation Unexpectedly Abnormal, Increases by 1 R/hr (Alarm and RadCon Confirmation)

OPERATING MODE APPLICABILITY: - All

BASIS:

This condition addresses an event of safety significance to plant personnel and the general public with regard to control of radioactive materials.

NUREG-0654

Radiation levels or airborne contamination which indicate a severe degradation in the control of radioactive materials.

RA6

Initiating Conditions and Emergency Action Levels
For Alert

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

ALERT

Airborne Radiation Unexpectedly Increases by >100 MPC for a Controlled Area (CAM Alarm, RadCon Confirmation)

OPERATING MODE APPLICABILITY: - All

BASIS:

This condition addresses an event of safety significance to plant personnel and the general public with regard to control of radioactive materials.

NUREG-0654

Radiation levels or airborne contamination which indicate a severe degradation in the control of radioactive materials.

RS1

Initiating Conditions and Emergency Action Levels For Site Area Emergency

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

SITE AREA EMERGENCY

. Verified Total Plant Noble Gas Release for the Stack of 1000 Ci/sec for 30 Minutes, or 10,000 Ci/sec for ≥2 Minutes as Determined by Plant Computer, or EPIP-3, 4 or 5 Manual Calculations, or Environmental Measurements at or Beyond the Site Boundary Find One of the Following:

1. ≥500 MREM/HR Whole Body for ≥2 minutes
2. ≥7.31E-7 uce/cc Iodine for ≥2 minutes
3. ≥50 MREM/HR Whole Body for ≥30 minutes
4. ≥7.31 E-8 uci/cc Iodine for ≥30 minutes

OR

Projected Accumulated Dose at or Beyond the Site Boundary:

1. ≥1 REM Whole Body
2. ≥5 REM Thyroid

OPERATING MODE APPLICABILITY: - All

BASIS:

NUREG-0654

RG1

Initiating Conditions and Emergency Action Levels
For General Emergency

RADIATION LEVELS ABNORMAL/RADIOLOGICAL EFFLUENTS

GENERAL EMERGENCY

As projected from plant parameters and conditions, plus actual site meterological conditions, radiation monitors detect release sufficient to cause radiation dose rates at the site boundary of 1 REM/hr whole body or 5 REM/hr dose commitment rate to the thyroid. For stack releases this value is 20,000 Ci/sec.

OR

Environmental measurements at or beyond the site boundary find one of the following:

1 REM/hr Whole Body
1.46E-6 uCi/cc Iodine

OPERATING MODE APPLICABILITY: - ALL

BASIS:

- NUREG-0654

These dose rates were derived from the Manual of Protective Action Guides and Protective Actions for Nuclear Incidents EPA-520/1-75-001 and initiate consideration of evaluation of the two-mile radius of the site boundary.

- NUREG-0654.

HU1

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Security Threat - Attempted Entry (SED JUDGEMENT Based on Advice from Nuclear Security Supervisor)

OPERATING MODE APPLICABILITY: - All

BASIS:

Indicative of a potential degradation or loss of control of the plant facility. NUMARC, 1/89, pg 76.

NUREG-0654

Security threat or attempted entry or attempted sabotage.

HU2

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Attempted Sabotage (SED JUDGEMENT Based on Advice from Nuclear Security Supervisor)

OPERATING MODE APPLICABILITY: - All

BASIS:

Indicative of a potential degradation or loss of control of the plant facility.

NUREG-0654

Security threat or attempted entry or attempted sabotage.

HU3

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Crash Within Site Boundary or Unusual Activity Over Facility (SED JUDGEMENT)

OPERATING MODE APPLICABILITY: - All

BASIS:

Involves a situation requiring heightened awareness and could be a possible threat to the plant facility.

NUREG-0654

Aircraft crash onsite or unusual aircraft activity over facility.

HU4

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Injured and Contaminated Individual Transported to Offsite Hospital

OPERATING MODE APPLICABILITY: -- All

BASIS:

Involves a situation requiring heightened awareness due to possible spread of contaminated material.

NUREG-0654

Transportation of contaminated injured individual from site to offsite hospital.

HU5

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Toxic Gases Near or Onsite and May Impair Station Operability (SED Judgement)

OPERATING MODE APPLICABILITY: - All

BASIS:

Involves a situation that cannot be stopped or controlled by plant personnel and in concentrations within the site boundary that could affect the health of plant personnel or affect the safe operation of the facility.

NUREG-0654

. Near or onsite toxic or flammable gas release.

HU6

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Triaxial Operation >0.01 g (Seismic Accelerometers, Alarm "Start of Strong Motion Accelerometers")

Before Initiating EPIP-2 Confirm Building Movement and/or Call National Earthquake Information Center at (303)236-1500.

OPERATING MODE APPLICABILITY: - All

BASIS:

The maximum operating basis earthquake is 0.1 g maximum ground acceleration. FSAR 12.1.

NUREG-0654

Any earthquake felt in plant or detected on station seismic instrumentation.

HU7

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Reservoir Level ≥563.5' But ≤565.0'

OPERATING MODE APPLICABILITY: - All

BASIS:

Water levels of 563.5' would be unusual because of the control that TVA has over the Tennessee River. The plant is designed to operate without problem up to elevation 565'. Technical Specification Basis 3.2.

NUREG-0654

Fifty-Year Flood or Low Water, TSUNAMI, Hurricane Surge, SEICHE

HU8

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Tornado Onsite
(Onsite is considered to be owner-controlled area)

OPERATING MODE APPLICABILITY: - All

BASIS:

Browns Ferry Nuclear Plant (BFN) safety-related structures have been designed to withstand tornado wind loads of 300 mph. A tornado which has been sighted has the potential to do some amount of damage if it strikes BFN. FSAR 12.1.

NUREG-0654

Any tornado onsite.

HU9

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Sustained High Winds With Speed > 70 mph but < 85 mph as Measured at the Meteorological Tower as Indicated in the Main Control Room

OPERATING MODE APPLICABILITY: - All

BASIS:

This event signifies abnormal weather/meteorological conditions that have reduced the margin of safety concerning the plant design to withstand sustained wind speed (design 100 mph sustained wind speed). NUREG-0654.

HU10

Initiating Conditions and Emergency Action Levels
For Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Wheeler Reservoir Below Seasonal Level Due to Problem at Wheeler Dam (Observation/Notification of Unusually Low Level)

OPERATING MODE APPLICABILITY: - All

BASIS:

This event could precede subsequent problems regarding the effectiveness of the remaining reservoir to provide adequate cooling water for the plant. NUREG-0654.

HU11

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Fire Within Protected Area Longer Than 10 Minutes OR Involving Radioactive Material

OPERATING MODE APPLICABILITY: - All

BASIS:

This EAL is written to address fires that may (1) be potentially significant precursors to damage to safety systems or (2) have the potential to cause the release of or exposure to radioactive materials. This EAL applies for buildings or areas contiguous to plant vital areas or other significant buildings or areas (any fire within the protected area fence).

This initiating condition can affect more than one unit, and in that event, escalation of the emergency classification should be evaluated. NUREG-0654.

NUREG-0654

Fire within the plant lasting more than 10 minutes.

HU12

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Confirmed Report or Detection of Unplanned Explosion Near or Within Site Boundary

OPERATING MODE APPLICABILITY: - All

BASIS:

Possible damage to necessary safety systems or components or possible sabotage.

NUREG-0654

Near or Onsite Explosion

HU13

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Gas or Vapors, Near or Onsite, That May Impair Station Operability (SED Judgement) (Hazardous Gases, Vapors, etc. - Dependent on Present Location, Wind Direction, etc.)

OPERATING MODE APPLICABILITY: - All

BASIS:

Involves a situation that cannot be stopped or controlled by plant personnel and in concentrations within the site boundary that could affect the health of plant personnel or affect the safe operation of the facility.

NUREG-0654

Near or Onsite Toxic or Flammable Gas Release

HU14

Initiating Conditions and Emergency Action Levels
For Notification of Unusual Event

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

Conditions exist warranting increased awareness of plant operating staff, State or local government or involve other than normal controlled shutdown

These items are based on Site Emergency Director's professional judgement.

OPERATING MODE APPLICABILITY: - All

BASIS:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Site Emergency Director to fall under the Notification of Unusual Event emergency class description.

NUREG-0654

Plant conditions exist that warrant increased awareness on the part of a plant operation staff or State and/or local offsite authorities or require plant shutdown under technical specification requirements or involve other than normal controlled shutdown.

HA1

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Unauthorized Personnel Inside Plant Security Boundary (SED Judgement Based on Advice from Nuclear Security Supervisor)

OPERATING MODE APPLICABILITY: - All

BASIS:

Indicative of a potential degradation or loss of control of the plant facility.

NUREG-0654

Ongoing Security Compromised

HA2

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Actual Sabotage (SED Judgement)

OPERATING MODE APPLICABILITY: - All

BASIS:

This event warrants Alert classification based solely on actual sabotage due to the potential degradation of the level of safety of the plant. NUREG-0654.

HA3

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Visual Observation or Confirmation of Missile or Aircraft Crash Within the Protected Area

OPERATING MODE APPLICABILITY: - All

BASIS:

Involves a situation requiring heightened awareness and could be possible threat to the plant facility.

NUREG-0654

Aircraft crash on facility or missile impacts from whatever source on facility.

HA4

Initiating Conditions and Emergency Action Levels For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Toxic Gases Within the Protected Area Affecting Safe Operation

OPERATING MODE APPLICABILITY: - All

BASIS:

- Involves a situation that cannot be stopped or controlled by plant personnel and in concentrations within the site boundary that can affect the health of plant personnel or affect the safe operation of the facility.

NUREG-0654

Entry into facility of uncontrolled toxic or flammable gases.

HA5

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Evacuation or Anticipated Evacuation of the Control Room

OPERATING MODE APPLICABILITY: - All

BASIS:

This condition is intended to address situations where control of the plant may be transferred from its normal location to the backup control center.

NUREG-0654.

Evacuation of control room anticipated or required with control of shutdown systems established from local stations.

HA6

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Biaxial Operation >0.1 g (Seismic Triggers, Alarm Seismic Triggers A, B, or C)

Before Initiating Alert, Confirm Building Movement and Call National Earthquake Information Center at (303) 236-1500.

OPERATING MODE APPLICABILITY: - All

BASIS:

The maximum operating basis earthquake is 0.1 g maximum ground acceleration, and the maximum design basis earthquake is 0.2 g. FSAR 12.1.

NUREG-0654

Earthquake Greater Than OBE Levels

HA7

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Reservoir Level >565.0' (Water Running on CCW Pump Deck) and Plant Watertight

OPERATING MODE APPLICABILITY: - All

BASIS:

At elevation 565', the water would begin to cross the top of the pumping station and a shutdown is required. Technical Specification Basis 3.2.

NUREG-0654

Flood, Low Water, TSUNAMI, Hurricane Surge, SEICHE Near Design Levels

HA8

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Tornado Striking Plant

(The plant is considered to be Reactor, Turbine, Service, Diesel Generator Buildings, Intake Structure, or Switchyard.)

OPERATING MODE APPLICABILITY: - All

BASIS:

Browns Ferry Nuclear Plant (BFN) safety-related structures have been designed to withstand tornado wind loads of 300 mph. A tornado on the BFN complex could do some amount of damage to BFN. FSAR 12.1.

NUREG-0654

Any Tornado Striking Facility

HA9

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Sustained Winds >85 mph but <100 mph as Indicated in the Control Room

OPERATING MODE APPLICABILITY: - All

BASIS:

Browns Ferry Nuclear Plant has been designed to withstand sustained wind loads of 100 mph. FSAR 12.1.

NUREG-0654

Hurricane Winds Near Design Basis Level

HA10

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Catastrophic Failure of Wheeler Dam Causing Low Reservoir Level

OPERATING MODE APPLICABILITY: - All

BASIS:

A loss of Wheeler Dam should result in a river level of 529'. The RHRSW/EECW pumps should have sufficient NPSH to continue to operate at this level, a loss of RHRSW/EECW pump suction would mean a loss of the Ultimate Heat Sink for Browns Ferry Nuclear Plant. FSAR 2.4-3, NUREG-0654.

HA11

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Fire Threatening Vital Area or Safety System (SED Judgement)

OPERATING MODE APPLICABILITY: - All

BASIS:

Potentially significant precursor to damage to safety systems.

NUREG-0654

Fire potentially affecting safety systems.

HA12

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Explosion Within Protected Area Causing Damage to the Facility Affecting Plant Operation

OPERATING MODE APPLICABILITY: - All

BASIS:

Possible damage to plant systems or components, including safety systems. Possible sabotage.

NUREG-0654

Known Explosion Damage to Facility Affecting Plant Operation

HA13

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Gas or Vapors, Uncontrolled Within the Protected Area Affecting Safe Plant Operation (SED Judgement)

OPERATING MODE APPLICABILITY: - All

BASIS:

Involves a situation that cannot be stopped or controlled by plant personnel and result in immediate health threatening concentrations within the site boundary that will affect the health of plant personnel or affect the safe operation of the facility.

NUREG-0654

Entry Into Facility Environs of Uncontrolled Toxic or Flammable Gases

HA14

Initiating Conditions and Emergency Action Levels
For Alert

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Conditions Exist Warranting Activation of the TSC and CECC (Potential to Escalate to Site Area Emergency)

OPERATING MODE APPLICABILITY: - All

BASIS:

Based on SED judgement, increased immediate support is required to mitigate current abnormal occurrences or possible consequences of events not classified elsewhere.

NUREG-0654

Plant Conditions Exist That Warrant Precautionary Activation of Technical Support Center and Placing Near-Site Emergency Operations Facility and Other Key Emergency Personnel on Standby

HS1

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Unauthorized Personnel Inside Plant (Vital Areas) and Are Threatening Physical Control of Plant Facility (SEC Judgement Based on Advice from Nuclear Security Supervisor)

OPERATING MODE APPLICABILITY: - All

BASIS:

Indicative of a potential degradation or loss of control of the plant facility. NUREG-0654

HS2

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Missile or Aircraft Crash Involving Vital Areas, Structures, or Equipment and Not in Cold Shutdown (Includes Reactor Building, Diesel Building, Intake, and Control Building)

OPERATING MODE APPLICABILITY: - All except cold shutdown

BASIS:

Involves a situation with possible damage to safety-related systems affecting the potential to shutdown the plant safely.

NUREG-0654

Aircraft Crash Affecting Vital Structures By Impact or Fire

HS3

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Severe Damage to Vital Areas, Structures, or Components from Missiles (External or Internal) or Explosion While Not in Cold Shutdown

OPERATING MODE APPLICABILITY: - All except cold shutdown

BASIS:

Loss of safety systems while not in cold shutdown with a potential threat to redundant systems or redundant systems may be challenged.

NUREG-0654

Severe Damage to Safe Shutdown Equipment from Missiles or Explosion

HS4

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Toxic Gases Within Vital Areas That Impair Station Operability

OPERATING MODE APPLICABILITY: - All

BASIS:

Involves a situation that cannot be stopped or controlled by plant personnel and in concentrations within the site boundary that will affect the health of plant personnel or affect the safe operation of the facility.

NUREG-0654

Entry of Uncontrolled Toxic Gases Into Vital Areas Where Lack of Access to the Area Constitutes a Safety Problem

HS5

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Evacuation and Control from the Backup Control Panel Not Established Within 15 Minutes

OPERATING MODE APPLICABILITY: - All

BASIS:

Expeditious transfer of safety systems has not occurred because of equipment malfunction beyond the design of the backup control center or because of personnel error.

NUREG-0654

Evacuation of Control Room and Control of Shutdown Systems Not Established from Local Stations in 15 Minutes

HS6

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Reservoir Level >565' (Water Running on CCW Pump Deck) and Any Indication of Inplant Flooding or Loss of Any Plant Watertight Features and Units Not in Cold Shutdown

OPERATING MODE APPLICABILITY: - All but cold shutdown

BASIS:

The base elevation at Browns Ferry Nuclear Plant is 565'. A loss of water tight features with water level above this elevation could degrade safety system and components. FSAR 12.1.

NUREG-0654

Flood, Low Water, TSUNAMI, Hurricane Surge, SEICHE Greater Than Design Levels or Failure of Protection of Vital Equipment at Lower Levels

HS7

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Sustained Winds >100 mph as Indicated in the Control Room and Not in Cold Shutdown

OPERATING MODE APPLICABILITY: - All but cold shutdown

BASIS:

Browns Ferry Nuclear Plant has been designed to withstand sustained wind load of 100 mph. These winds exceed design limits.

NUREG-0654

Sustained Winds or Tornados in Excess of Design Levels

HS8

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Catastrophic Failure of Wheeler Dam and RHRSW/EECW Pumps Suction Lost and Not in Cold Shutdown

OPERATING MODE APPLICABILITY: - All except cold shutdown

BASIS:

A loss of Wheeler Dam should result in a river level of 529'. The RHRSW/EECW pumps should have sufficient NPSH to continue to operate at this level, a loss of RHRSW/EECW pump suction would mean a loss of the Ultimate Heat Sink for Browns Ferry Nuclear Plant and an event not envisioned in the design. FSAR 2.4-3.

NUREG-0654

Flood, Low Water, Tsunami, Hurricane Surge, Seiche Greater Than Design Levels or Failure of Protection of Vital Equipment at Lower Levels

HS9

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Fire Affecting Safety System or Vital Area (SED Judgement)

OPERATING MODE APPLICABILITY: - All

BASIS:

Potentially significant precursor to damage to safety systems.

NUREG-0654

Fire Compromising the Functions of Safety Systems

HS10

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Explosion Causing Major Damage Involving Vital Structures (Reactor Building, Control Building, Diesel Building, Intake) or Vital Equipment and Not in Cold Shutdown

OPERATING MODE APPLICABILITY: - All but cold shutdown

BASIS:

Degradation of safety systems or vital structures.

NUREG-0654

Severe Damage to Safe Shutdown Equipment from Missiles or Explosion

HS11

Initiating Conditions and Emergency Action Levels
For Site Area Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

Gas or Vapors, Uncontrolled Within Vital Areas, Affecting Safe Operation and Not in Cold Shutdown

OPERATING MODE APPLICABILITY: - All except cold shutdown

BASIS:

Involves a situation that cannot be stopped or controlled by plant personnel and results in immediate health threatening concentrations within the site boundary that will affect the health of plant personnel or affect the safe operation of the facility.

NUREG-0654

Entry of Uncontrolled Flammable Gases Into Vital Areas. Entry of Uncontrolled Toxic Gases Into Vital Areas Where Lack of Access to the Area Constitutes a Safety Problem.

HG1

Initiating Conditions and Emergency Action Levels
For General Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

GENERAL EMERGENCY

Loss of Physical Control of the Facility (SED Judgement)

OPERATING MODE APPLICABILITY: - All

BASIS:

Due to sabotage and/or intruders, control of the plant no longer is provided by licensed operators. NUREG-0654.

HG2

Initiating Conditions and Emergency Action Levels For General Emergency

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

GENERAL EMERGENCY

Major Internal or External Events Which Could Cause Massive Common Damage to Plant Systems

OPERATING MODE APPLICABILITY: - All

BASIS:

Unanticipated conditions or events causing or threatening to cause massive damage to multiple plant systems. NUREG-0654.

A.2 SITE EMERGENCY ORGANIZATION

BFN maintains an organization capable of responding to a radiological emergency. The TSC, OSC, and Control Room staffing for response to emergencies is shown in figure A-1. The minimum onshift emergency response staffing is found in figure A-2.

A.2.1 Site Director

The Site Director serves as a corporate interface for the SED, relieving him from duties which could distract from the SED's primary purpose of plant operations and accident mitigation activities. The Site Director shall provide assistance in the following areas:

1. Provides TVA policy direction to the Site Emergency Director.

2. Directs the site resources to support the Site Emergency Director in the accident mitigation activities.

3. Provides direct interface on overall site response activities with:

   a. NRC, FEMA, or other Federal organizations responding to the site.

   b. CECC Director.

   c. Onsite media.

4. At his discretion, may provide interface at the appropriate offsite location on the overall site response activities with:

   a. State and local agencies.

   b. NRC region/corporate.

   c. Joint Information Center.

5. Provides support to other emergency operation centers as *necessary.

A.2.2 Site Emergency Director

1. Directs onsite emergency accident mitigation activities.

2. Consults with CECC Director and Site Director on significant events and their related impacts.

3. Initiates onsite protective actions.

*Revision

4. Coordinates accident mitigation actions with NRC.

5. Initiates long-term 24-hour accident mitigation operations.

6. Prior to the CECC being staffed, makes recommendations for protective actions (if necessary) to State and local agencies through the Operations Duty Specialist. This responsibility cannot be delegated except to the CECC Director after the CECC is operational.

7. Responsible for determining the emergency classification. This responsibility cannot be delegated.

8. Makes final decision on personnel entrance to radiologically hazardous areas when RadCon recommends against the entry.

A.2.3 *Operations Manager

1. Directs operational activities.

2. Informs Site Emergency Director of plant status and operational problems.

3. Performs damage assessment as necessary.

4. Recommends solutions and mitigating action for operational problems.

A.2.4 *Technical Assessment Manager

1. Directs onsite effluent assessment.

2. Directs activities of technical assessment team.

3. Projects future plant status based on present plant conditions.

4. Keeps assessment team informed of plant status.

5. Provides information, evaluations, and projections to Site Emergency Director.

6. Coordinates assessment activities with the CECC plant assessment team.

A.2.5 *Maintenance Manager

1. Directs repairs and corrective actions.

2. Performs damage assessment.

3. *Directs activities of Operations Support Center.

*Revision

*

*4. Coordinates maintenance teams and ensures they have received proper briefings and are accompanied by a RadCon technician, as necessary.

A.2.6 TSC Clerks

1. Maintain log of events.

*

*2. Answer telephones.

*3. Distribute plant parameter data sheets.

*4. Maintain TSC organization board.

*5. Operate facsimile machine.

*6. Operate Emergency Data Information System.

*7. Other duties as assigned by Site Emergency Director.

A.2.7 TSC Communicator

1. Provides information from control room to Technical Assessment team.

2. Completes plant parameter data sheets.

*

A.2.8 Nuclear Security Manager

1. Directs activities of Nuclear Security Services personnel.

2. Controls access to site and control rooms.

3. Reports on site accountability/evacuation as defined in BFN-EPIPs.

A.2.9 *Radiological Control Manager

1. Directs and/or performs assessment of inplant and onsite radiological conditions.

2. Directs onsite RadCon activities.

*Revision

3. Coordinates additional RadCon support with CECC Radiological Communicator.

4. Makes recommendations for protective actions for onsite personnel.

5. Maintains status map of offsite radiological conditions.

6. Coordinates assessment of radiological conditions offsite with CECC Radiological Communicator.

7. Maintains inplant radiation status board.

8. Coordinates briefing of maintenance teams with maintenance manager and assigns a RadCon Technician to accompany them if necessary.

9. Makes final recommendations to the Site Emergency Director for personnel entry to radiologically hazardous environment.

A.2.10 *Chemistry and Environmental Manager

1. Coordinates assessment of radioactive effluents with CECC Plant Assessment Team.

2. Directs post-accident sampling activities.

3. Directs activities of the radiochemical laboratory.

4. Determines impact of incident on radwaste and various effluent treatment systems.

A.2.11 *Mechanical Maintenance Supervisor

1. Directs OSC (Mechanical).

2. Performs damage and repair assessment.

A.2.12 Technical Assessment Team Leader

1. Performs systems assessment as directed by Technical Support Superintendent.

2. Determines condition of reactor and nuclear fuel.

3. Acts as plant assessment team leader.

A.2.13 *Instrument Maintenance Supervisor

1. Directs OSC (Instrumentation).

2. Performs damage and repair assessment.

*Revision

A.2.14 *Electrical Maintenance Supervisor

1. Directs OSC (Electrical).

2. Performs damage and repair assessment.

A.2.15 Transmission and Customer Services Engineer

1. Performs damage and repair assessment.

A.2.16 NRC Coordinator

1. Acts as primary liaison with onsite NRC personnel.

2. Updates NRC personnel on plant status.

3. Provides information requests from NRC to TSC personnel.

A.2.17 Operations Specialist

1. Provides operational knowledge for status evaluation of plant systems.

2. Provides advice regarding technical specifications, system response, safety limits, etc.

3. Assists in development of recommended solutions to developing problems.

A.2.18 Radiological Emergency Coordinator

1. Advises Site Emergency Director regarding overall radiological emergency plan, use of implementing procedures, emergency equipment availability, and coordination with CECC.

2. Confirms TSC is operating properly.

A.2.19 *NE Manager

1. Serves as the primary interface with Nuclear Engineering.

*Revision

2. Provides for additional engineering support during and/or following a radiological emergency.

3. Coordinates the design and construction of emergency equipment and structures as necessary.

A.2.20 Technical Assessment Team

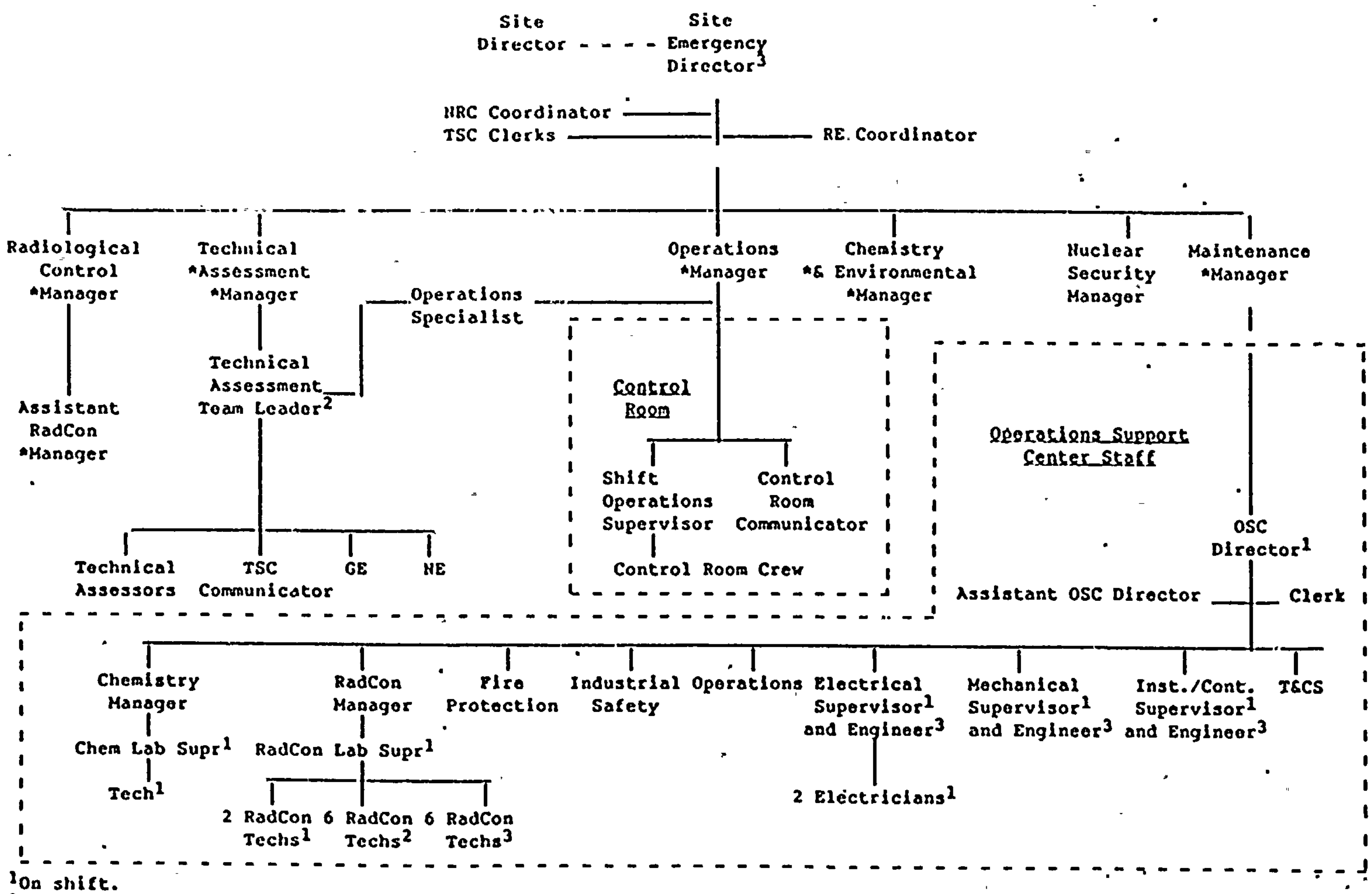
1. Prepares and provides periodic current assessments on plant conditions and provides this information to the CECC plant assessment team.

2. Projects future plant status based on present plant conditions.

3. Provides technical support to plant operations on mitigating actions.

*Revision.

FIGURE A-1
SITE EMERGENCY ORGANIZATION

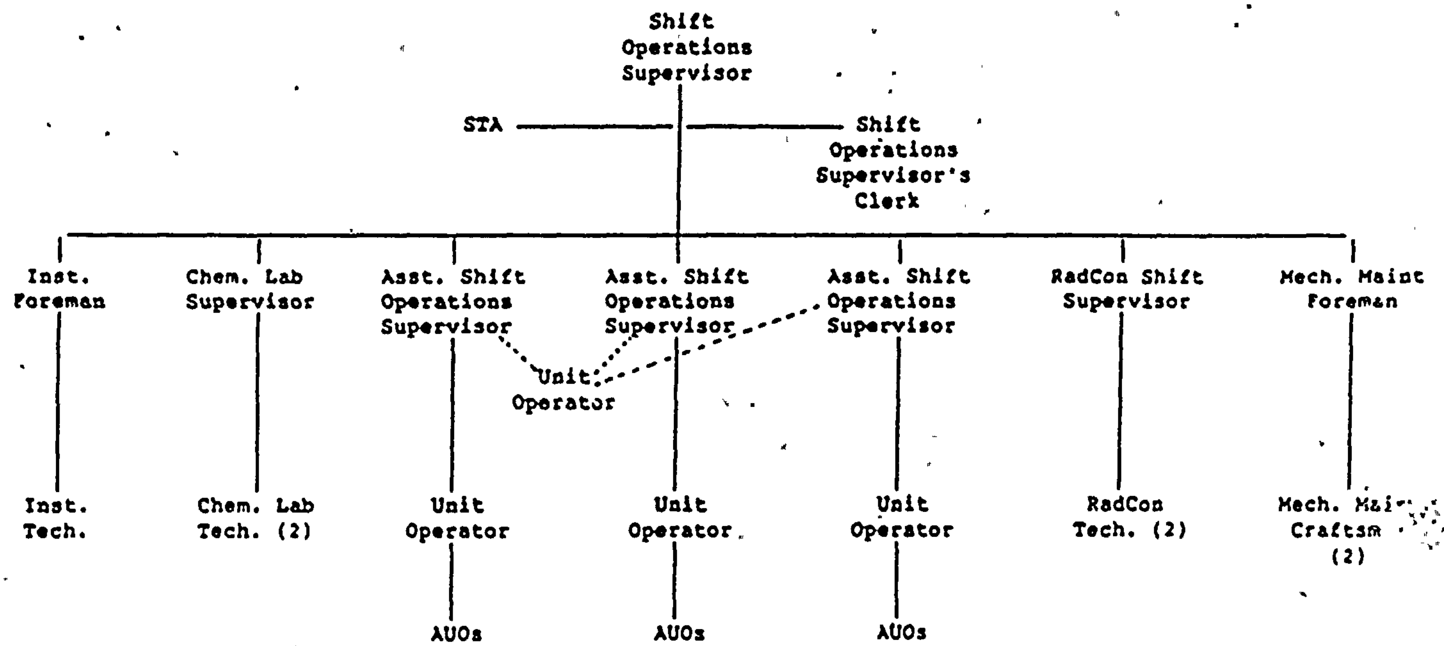


[1] On shift.
[2] Augment within approximately 30 minutes.
[3] Augment within approximately 60 minutes.

*Revision

FIGURE A-2

MINIMUM ONSHIFT EMERGENCY RESPONSE PERSONNEL



AUO - Assistant Unit Operator
5 AUOs per shift required by Tech. Specs.

A.3 EMERGENCY RESPONSE FACILITIES, EQUIPMENT, AND SUPPLIES

Specific plant areas, facilities, and equipment are selected and provided for use during a radiological emergency. The preselection, allocation, and inclusion of emergency facilities assure that needed services and equipment are available for use during emergency conditions.

A.3.1 Technical Support Center (TSC)

A specific area (between units 1 and 2 control room and unit 3 control room) in the control bay at elevation 617.0 is designated for use as the TSC. The room is provided with communication facilities for plant areas and areas external to the plant. The communication facilities include PAX telephone, Bell System telephones, a paging-intercom system, and two-way radio capabilities. This room is sufficiently shielded to ensure occupancy during an emergency and is designed to be continuously habitable during all radiological emergencies. All ventilating and air-conditioning facilities have redundant or backup systems. Toilet and shower facilities are available on the same elevation.

The diesel generators will provide emergency power when there is a loss of normal ac power, and cooling water for the air-conditioning equipment can be taken from the emergency equipment cooling water system. Emergency equipment is specifically designated and stored near the TSC for use during an emergency. Figure A-3 shows a detailed TSC layout.

Meteorological information is available both in the TSC and main control room and includes wind speed and direction at 10, 46, and 93 meters, and temperature at 10, 45, and 90 meters. Also available is information from the environmental monitors, both perimeter and local.

A.3.2 Operations Support Center (OSC)

The role of the OSC is to provide an assembly area for operations support personnel during an emergency situation and is under the supervision of the OSC Director. The restart operations area at elevation 580' in the service building (see figure A-4) is designated for use as the OSC. The OSC is provided with telephone communications. In the event that radiation conditions require evacuation of this area, OSC personnel will report to the office building, elevation 580.

TECHNICAL SUPPORT CENTER

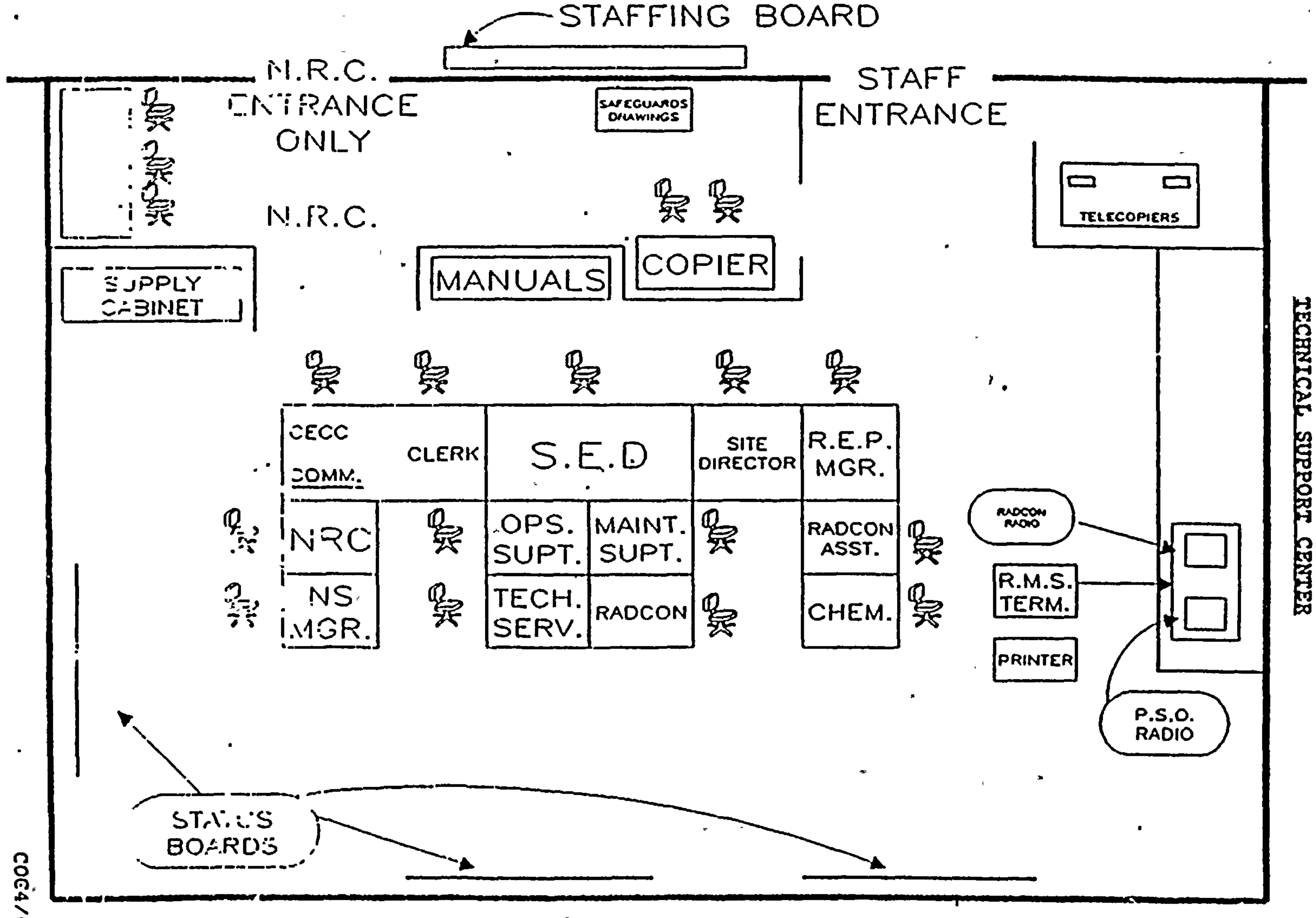


FIGURE A-3

TECHNICAL SUPPORT CENTER

OPERATIONS SUPPORT CENTER LAYOUT

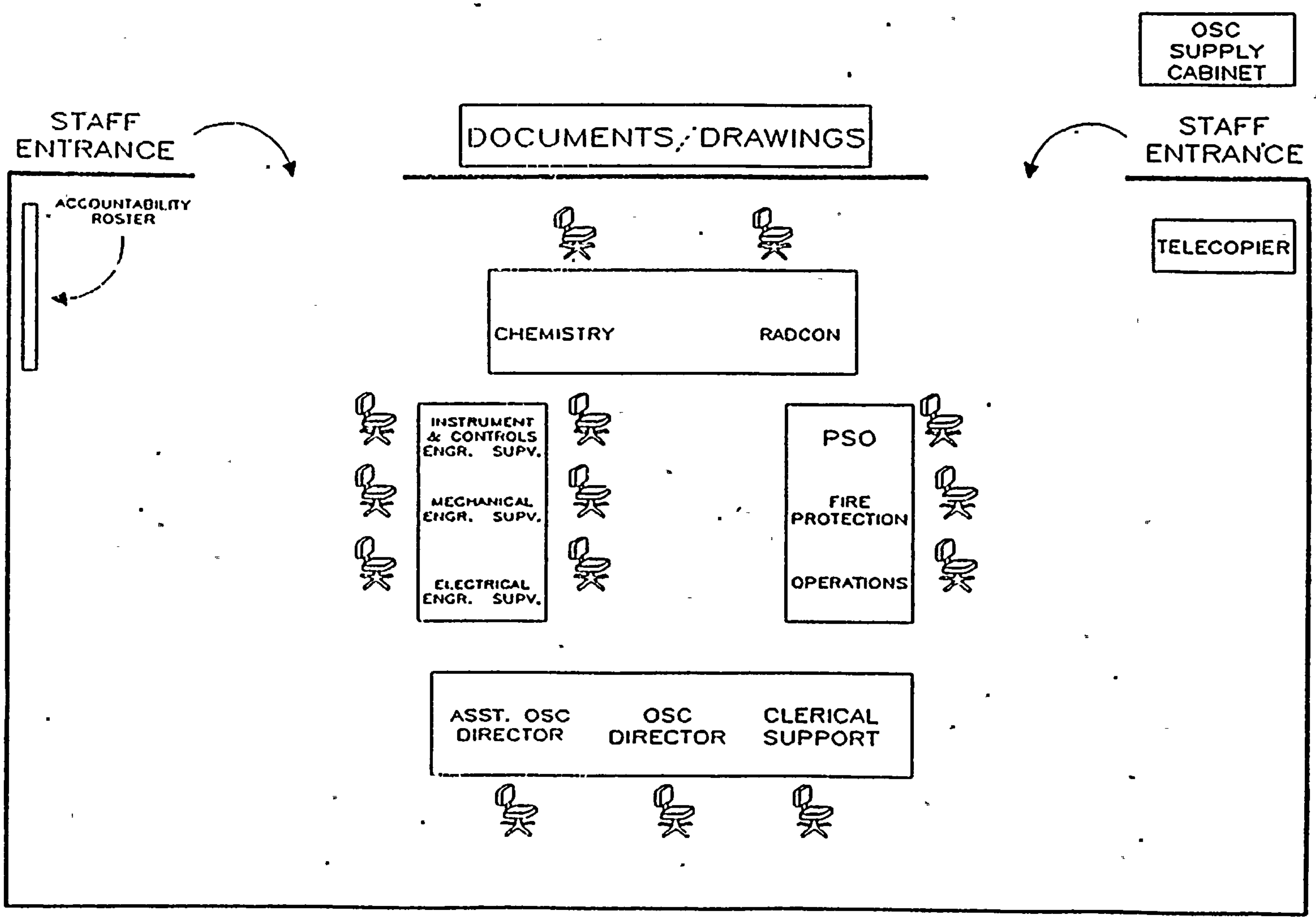


FIGURE A-4

OPERATIONS SUPPORT CENTERS

A.3.3 RadCon Laboratory and Equipment

The RadCon laboratory is located in the service building adjacent to the personnel corridor at elevation 565.0. The portable radiation monitoring and counting equipment normally used by the plant RadCon section is kept in this space and is available for use during an emergency. Sufficient reserves of instruments/equipment are available to replace those removed from service for calibration or repair. Calibration of equipment is carried out at intervals *specified by DIR-10.2.

A.3.4 Onsite Monitoring Systems and Equipment

A.3.4.1 Natural Phenomena

In the event an emergency is the result of a natural phenomena, there is instrumentation to monitor its severity. The Environmental Data Station is located onsite and contains instruments capable of measuring wind direction, wind speed, and temperatures. Seismic instrumentation is available in the plant to monitor acceleration levels of ground movement. Hydrological monitoring systems are installed to supply flow and level information for each site. Meteorological and seismic instrumentation have readily accessible readout in the main control room. More specific information on these systems can be found in the Browns Ferry FSAR.

A.3.4.2 Radiological Monitors

The installed Radiation Monitoring System consists of process monitors and area monitors which read out on local panels and in the control room.

A.3.4.2.1 Process Monitors (Radiological)

The process system continuously monitors selected lines containing or possibly containing radioactive effluents. The system's function is to warn personnel of increasing radiation levels, to give early warning of a system malfunction, and to record and control discharges of radioactive liquids and gases to the environment. The system consists of active and redundant channels. Examples of process monitors are:

1. Reactor Building Ventilation Monitoring System

2. Main Steam Line Radiation Monitoring System

3. Main Stack Radiation Monitoring System

4. Plant Ventilation Exhaust Radiation Monitoring System

5. Liquid Radwaste

*Revision

6. Raw Cooling Water

7. Reactor Building Closed Cooling Water

8. Residual Heat Removal Service Water Discharge

A.3.4.2.2 Area Monitors

Area monitors are placed at specific locations in the plant. Examples of area monitor locations are:

1. Reactor Building

2. New and Spent Fuel Storage Area

3. Turbine Building

4. Main Control Room

5. Radwaste Building

6. Off-Gas Stack

A.3.4.2.3 Portable Monitors

Portable radiation detection equipment consists of low- and high-range instruments to measure gamma radiation levels from 0.1 mR/hr to 1000 R/hr. Instruments for alpha, beta-gamma, and neutron radiation measurements are available. Sampling equipment is available to take low- or high-volume air samples. Air samplers can be used to collect low-volume samples either onsite or offsite. The counting room has a multichannel analyzer with shielded GeLi detectors, gas flow proportional counter, liquid scintillation counter, and gamma spectrometer with NaI detector.

A.3.4.2.4 Process Monitors (Nonradiological)

Installed in the main control room are the necessary instrumentation readouts to assess plant systems status including reactor coolant system pressure and temperature, containment pressure and temperature, liquid levels, flow rates, fire detection equipment, and meteorological instrumentation. More specific information on control room instrumentation can be found in the Browns Ferry FSAR.

A.3.4.2.5 Fire Protection

The plant's fire protection system is designed to furnish water and other extinguishing agents with the capability of extinguishing any single or probable combination of simultaneous fires that might occur. The use of combustible materials is minimized, and the greatest possible use of fire-retardant materials has been incorporated in plant design. The standards of the National Fire Protection Association and the recommendations of the nuclear insurers are considered in the system design to provide the following:

1. Supply of water for the fire protection system.

2. Automatic fire or smoke detection in the more critical areas.

3. Fire suppression by fixed equipment actuated automatically or manually.

4. Manually operated portable fire extinguishing equipment at strategic locations.

5. Compartmentation to limit the spread of fire.

In addition to the minimum standards prescribed in the technical specifications, Browns Ferry has one fully equipped Class A fire department pumper. Sufficient personnel are available to meet technical specification requirements for a fire brigade and provide the necessary personnel to operate the pumper.

A.3.4.2.6 Environment

Facilities available for assessing the impact of plant operations on the environment include atmospheric monitoring stations, direct gamma radiation detectors, and automatic water samplers. This equipment is used in the routine environmental radiological monitoring program and is available in the event of a radiological emergency condition.

The atmospheric monitoring network is divided into three subgroups. Local air monitors are located at or adjacent to the site boundary in the directions of predominant wind flow. Perimeter monitors are located three to 10 miles from the plant in areas of relatively high population densities and/or in the direction of predominant air flow. Remote monitors (controls) are located at sites greater than 10 miles from the plant.

At each monitor, air is continuously passed through a particulate filter at a regulated flow. In series with, but downstream of, the particulate filter is a charcoal filter used to collect iodine. Each monitor has a collection tray and storage container to collect rainwater on a continuous basis.

Thermoluminescent dosimeters (TLDs) are placed at approximately 40 sites around the plant. These TLDs are located typically in each of the 16 meteorological sectors at or near the site boundary and at a distance of approximately 4-5 miles. Three dosimeters are usually placed at each site.

Automatic water samplers are located above and below the plant discharge, at the first potable water supply downstream from the plant, and at a ground water well which is down gradient from the plant.

In addition to these facilities, established sampling points for milk, vegetation, soil, fish, and sediment are located in the vicinity of the plant. Samples may be collected from these stations on a nonroutine basis as needed.

All samples are returned to one of TVA's radiological laboratories for processing.

A.3.5 Emergency Equipment

Figure A-5 contains listings of emergency equipment and storage locations throughout the plant in effect at the time the plan was prepared. Updated listings and locations are included in the BFN-EPIPs. Required calibration of equipment is carried out at intervals recommended by the supplier of the equipment or as specified in the Browns Ferry FSAR.

A.3.6 First Aid and Medical Facilities

A.3.6.1 Decontamination Facilities

The site is responsible for maintaining supplies and equipment to establish a temporary decontamination area for the purpose of gross radiological decontamination of personnel who may also be injured.

The personnel decontamination room and emergency medical treatment area, complete with sink and shower facilities, is provided in the service building area at elevation 565.0. Equipment and cleaning solutions for the decontamination of personnel are available in this room. Stretcher-bound personnel can be decontaminated in the Radwaste Building at elevation 565.0.

FIGURE A-5

EMERGENCY EQUIPMENT

	Location	Description
1.	RadCon Laboratory Service Building	Emergency Supplies and Radiological survey meters
2.	Control Bay	Emergency SCBAs with additional cylinders
3.	Control Building Mechanical Equipment Room	Emergency supplies and radiological survey meters
5.	Emergency Van	General emergency supplies related to environs monitoring
6.	Huntsville Hospital & Decatur General Hospital Emergency Rooms	Supplies specific to radiological injuries

A.3.6.2 Health Stations and Supplies

Emergency medical equipment is strategically located throughout the plant, with trauma kits and other specialized equipment available for use by the MERT.

A first aid station, staffed by an EMT, is located in a separate building near the base of the off-gas stack within the security fence. Medical supplies and treatments for minor injuries are available. First aid treatment is available 24 hours a day.

A medical office, staffed by registered nurses and a physician, is located in the northwest corner of the Personal Services Building outside the east portal to the plant. Medical treatment and examinations (employment, routine, occupational) are available during the day shift, Monday - Friday.

Potassium iodide tablets for onsite personnel are controlled and stored by site RadCon. Specific information including authorization and dispersal of tablets is contained in the site EPIPS.

A.3.6.3 Receiving Hospitals and Supplies

Arrangements have been made with at least one hospital to receive patients from BFN. (See Sections 12.3 and 16.5.)

A.3.6.4 Ambulance Service

A TVA ambulance is available at the site and is maintained and staffed in conjunction with the MERT. Arrangements have been made for offsite ambulance assistance to BFN. (See Sections 12.2 and 16.5.)

A.3.7 Additional Local Support

A.3.7.1 Law Enforcement

Agreements (see section 16.5) are maintained with local law enforcement agencies to support TVA when necessary.

A.3.8 Vendor Support

If necessary, the NSSS vendor, General Electric, will be contacted by the CECC Director to provide assistance in the form of manpower, equipment, and technical backup. Other vendors will also be contacted if their assistance is needed.

A.3.9 Emergency Siren

Undulating sirens are provided in strategic plant areas for indicating the assembly of all personnel. Care is exercised in locating the sirens so that they are audible in all plant areas. A three-minute
*undulating blast on the siren is the signal for assembly. Site
*evacuation is a steady three-minute blast.

*Revision

The sirens are powered by redundant 120V ac supplies. The sirens can be activated from the electrical control desk in the units 1 and 2 control room or the unit 3 control room.

A.3.10 Local Recovery Center

The LRC for Browns Ferry will be a portion of the second floor of the Administration Building outside the protected area of the site.

A.3.10.1 Communications

The LRC has voice communication capabilities to enable personnel to communicate with the CECC and the Browns Ferry TSC. The following voice communication is available in the LRC area:

1. Bell Telephone

2. TVA Microwave Telephone System

3. Long Distance Service

Meteorological information and dose rate calculations are also available to LRC personnel.

Other equipment available for use by LRC personnel include:

1. Facsimile machine

2. Copy machines

3. Hand calculators

4. Tape recorders

5. Plant-specific drawings, manuals, procedures, etc.

A.3.11 REP Implementing Procedures

The following is a listing of the BFN-EPIPs.

A.3.11.1 BFN-EPIP-1--Emergency Plan Classification Logic

This procedure provides guidance to the Shift Operations Supervisor in determining the classification of an accident to ensure that appropriate predetermined actions are implemented. It details initiating conditions and directs shift personnel to appropriate notification and assessment procedures.

A.3.11.2 BFN-EPIP-2--Notification of Unusual Event

This procedure provides for the timoly notification of appropriate individuals when the Shift Operations Supervisor has determined by EPIP-1 that an incident has occurred which is classified as a Notification of Unusual Event. It details requirements for periodic reassessment and the implementation of appropriate actions.

A.3.11.3 BFN-EPIP-3--Alert

This procedure provides for the timely notification of appropriate individuals when the Shift Operations Supervisor has determined by EPIP-1 that incident has occurred which is classified as an Alert. It details requirements for periodic reassessment and the implementation of appropriate actions. It also contains information for performing offsite dose assessment.

A.3.11.4 BFN-EPIP-4--Site Area Emergency

This procedure provides for the timely notification of appropriate individuals when the Shift Operations Supervisor has determined by EPIP-1 that an incident has occurred which is classified as a Site Area Emergency. It details requirements for periodic reassessment and the implementation of appropriate actions. It also contains information for performing offsite dose assessment.

A.3.11.5 BFN-EPIP-5--General Emergency

This procedure provides for the timely notification of appropriate individuals when the Shift Operations Supervisor has determined by EPIP-1 that incident has occurred which is classified as a General Emergency.

It details requirements for periodic reassessment and the implementation of appropriate actions. It also contains information for performing offsite dose assessment. It also contains information for determining protective action recommendations for the public.

A.3.11.6 BFN-EPIP-6--Activation and Operation of the TSC

This procedure directs the activation and operation of the TSC during an Alert, Site Area Emergency, or General Emergency. It details notification requirements. Documents issued onsite contain the TSC call-out lists.

A.3.11.7 BFN-EPIP-7--Activation and Operation of the OSC

This procedure directs the activation and operation of the OSC during an Alert, Site Area Emergency, or General Emergency. It details notification requirements. Documents issued onsite contain the OSC call-out lists.

A.3.11.8 BFN-EPIP-8--Personnel Accountability and Evacuation

This procedure details the requirements for accountability of all personnel and visitors and the orderly evacuation of areas of the plant during a radiological emergency.

*

*A.3.11.9 BFN-EPIP-10--Medical Emergency Procedure

This procedure details actions to be followed during medical emergencies. It provides for the organization and activation of the onsite Medical Emergency Organization. It contains the duties and responsibilities of the onsite Medical Emergency Organization. The procedure provides guidance on the care and handling of patients who may have been exposed to or contaminated with radioactive material, including provisions for the transport of these individuals to offsite medical support facilities. Maps and appropriate instructions are also included.

*A.3.11.10 BFN-EPIP-11--Security and Access Control

This procedure details responsibilities and requirements for access control and accountability during a radiological emergency.

*

*A.3.11.11 BFN-EPIP-13--Radiochemical Laboratory Procedure

This procedure provides the instructions to be followed by the lab during an emergency.

*A.3.11.12 BFN-EPIP-14--Radiological Control Procedure

This procedure outlines the actions to be followed by RadCon personnel during a plant emergency. It details responsibilities and RadCon assessment actions and recordkeeping requirements. The procedure provides guidance regarding the administration of potassium iodide (KI) to inplant workers.

*A.3.11.13 BFN-EPIP-15--Emergency Exposure

This procedure provides guidance on acceptable personnel exposures for various conditions. It specifies absolute exposure and authorizes the Site Emergency Director to permit exposures in excess of 10 CFR 20 limits in order to perform the emergency mission.

*Revision

*A.3.11.14 BFN-EPIP-16--Deescalation and Recovery

This procedure outlines responsibilities and provides guidance on recovery after an emergency to ensure adequate planning for efficient utilization of resources and radiation exposure.

*A.3.11.15 BFN-EPIP-17--Emergency Equipment and Supplies

This procedure details requirements for periodic inspection and maintenance of emergency equipment and supplies. It assigns responsibility and specifies the inspection frequency and documentation requirements.

*

*A.3.11.16 BFN-EPIP-19--Communication System and Emergency Notification List

This procedure provides a ready reference of onsite communication capabilities and key telephone numbers onsite and offsite.

*A.3.11.17 BFN-EPIP-20--Plant Data

This procedure provides a system to collect and transfer plant data from the Control Room to the TSC.

*A.3.11.18 BFN-EPIP-21--Fire Emergency Procedure

This procedure provides the guidance for the management of the response to fire emergencies.

*Revision

A.4 PROMPT NOTIFICATION SYSTEM

The prompt notification system network consists of a combination of fixed sirens, mobile sirens, and tone-alert radios. The system is designed to provide warning within 15 minutes to the population within five miles of the plant and within a maximum of 45 minutes to the population located in the 5- to 10-mile area.

A.4.1 Fixed Sirens

The fixed siren component consists of 54 electromechanical sirens. The system is made up of fifty 125-dB rotating sirens and four 115-dB omnidirectional sirens. The sirens are radio activated by local emergency management agencies.

The siren system is activated on a monthly basis by the local civil defense agencies as a regularly scheduled test. A silent test is

conducted every two weeks to test the radio link to the sirens. A growl test is performed by TVA employees on a quarterly basis. This test consists of each siren being activated individually by a portable activation unit.

Preventive maintenance is performed by TVA on an annual basis commensurate with the manufacturers' recommendations. Unscheduled maintenance is performed on an as-needed basis.

A.4.2 Mobile Sirens

The mobile siren component consists of five siren-equipped vehicles driven by county employees which run predetermined routes. All mobile siren routes are in the 5- to 10-mile area.

The vehicles are regularly driven and maintained by local county employees. The sirens are tested on a quarterly basis by TVA employees. Maintenance is performed on the sirens on an as-needed basis by TVA employees.

A.4.3 Tone-Alert Radios

The tone-alert radio component consists of two types of radios. These are NOAA-activated radios and radios activated by county frequencies. The radios are placed in institutions, both public and private, where there are concentrations of people. A total of 48 NOAA radios and 17 county-activated radios have been placed. A number of NOAA radios are held in reserve in county emergency management offices.

The NOAA radios are activated by the National Weather Service, Huntsville, Alabama, at the request of the State of Alabama Emergency Management Agency. The radios operating on county frequencies are activated by the respective counties.

The NOAA radios are tested on a weekly basis by the National Weather Service, i.e., they broadcast the activation signal, but they do not check to see if the radios are activated. The county radios are tested by the counties on a regular basis.

Scheduled maintenance is performed by TVA on the NOAA radios on a six-month basis. Unscheduled maintenance is performed on an as-needed basis. Maintenance on the county frequency radios is performed by the respective counties.

ATTACHMENT
SRO MASTER
90-02

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Site Director Standard Practice

NOV 29 1989

Form Page 1 of 1
Form SDSP-1
SDSP-2.11

FORM SDSP-1
PROCEDURE REVIEW AND APPROVAL COVER SHEET

Procedure No.	Tracking No.	Title	(for SP use only) Rev. No.
EPIP-1	12	Emergency Plan Classification Logic	7

CHECK AS APPROPRIATE:
[ ] New Procedure
[✓] Permanent Change. Two Year Review per SDSP-7.4? ___ Yes ✓ No
[ ] Cancellation
[ ] Temporary change
Anticipated duration (date or condition): ______

SP LIST pages and forms affected after word processing. (For TCs list marked pages):
General Revision MB 5/11/90

Pages reissued for pagination only: ______

ORIGINATOR: Jesse D. Savage (PRINT your name) Date 2-2-90 Organization OPS Phone 3932

Radiological Emergency
Name of Responsible Organization

[X] Timothy W. Cornelius RPC Signature 2038 Phone 2/5/90 Date

AFFECTED MANAGERS CONCURRENCE SIGNATURES	DATE	AFFECTED ORGANIZATIONS CONCURRENCE SIGNATURES	DATE
[ ] Plant Manager		[X] Organization: Operations	2-5-
[ ] Project Controls & Financial Svcs Mgr		[X] A L Champion Organization: Training	2-6-
[ ] Site Support Manager		[X] Timothy W. Cornelius Organization: Rad. Emerg. (REP)	2/5
[ ] Site Licensing		[ ] Organization:	
[ ] Site Programs Manager		[ ] Organization:	
[ ] Nuclear Engineering (NE)		[ ] Organization:	
[ ] Modification (MODS)		[ ] Organization:	
[ ] Materials & Procurement Manager		[ ] Organization:	
[X] Billy M. King Site Quality Assurance	2/6/90	[X] ~~Beth Zoner 2-7-90~~ Site Procedures BT 2-6-90	2-6-90

ak 2-6-90

[X] Samuel H. Smith Responsible Organization Supervisor 2/6/90 Date (Sign AFTER all required signatures except PORC, Principal Manager, and Site Director, if applicable.)

For Instructions approved by a Supervisor, MARK the following approvals "NA."

[X] N/A Principal Manager ______ Date (Required signature for standard practices ONLY, except if Principal Manager is the Plant Manager, then mark "NA" here and check Plant Manager at bottom left.)

Title: N/A

PORC Review required? [✓] Yes [ ] No

[ ] N/A Site Director (SDSPs only) ______ Date

[ ] CB Swindell PORC Chairman or PORC Minutes No. 2/9/90 Date

[ ] CB Swindell Plant Manager or PORC Minutes No. 2/9/90 Date

Date released from Administrative Hold
5-11-90 (SP completes, if applicable)
Effective Date 5-14-90 (SP completes)

Retention Period: Lifetime
Responsibility: ...

HISTORY OF REVISION/REVIEW

REV. NO.	DATE	REVISED PAGES	REASON FOR CURRENT REVISION
0	09/25/87	ALL	Change IP-1 to EPIP-1.
ITCP	10/09/87	17	Correct a reference to EPIP-2 & 3 for clarification. (ITC-02)
1	05/20/88	All	General revision to bring the procedure in line with the emergency Action Levels of the Office of Nuclear Power Radiological Emergency Plan (supercedes the BFNP REP). Put in the format describes for EPIP's. Incorporate ITC-02.
ITC	06/17/88	13	Correct a typo on Emergency Class. (ITC-04)
2	08/26/88	12,13	Incorporate an existing ITC. Incorporates a correction to SI number on p. 12 (from Site Procedures).
TC-07	09/28/88	11	To allow the site meterological tower to be taken out of service for a planned outage. REASON FOR URGENCY: Outage is scheduled for 9-28-88.
3	12/03/88	11,12,19	Incorporate TC-EPIP-01-07. Add High winds classification for None and Alert. Correct Cross reference error per SI4.8.B.1.A. REASON FOR URGENCY: THe High winds information is due to NRC wihtin 45 days and must reach Nuclear Licensing by December 9, 1988.
4	06/19/89	18	To change the criteria for classifying an emergency event due to tornado. URGENCY: Nuclear Power management wanted the procedure changed as soon as possible.
5	09/29/89	5,6	To clarify the three fission product barriers. Add classification, Site Area Emergency, for large Reactor Coolant System line breaks. REASON FOR URGENCY: Must be in place before next REP drill ,scheduled 10/04/89.

HISTORY OF REVISION/REVIEW

REV. NO.	DATE	REVISED PAGES	REASON FOR CURRENT REVISION
6	01/09/90	1,6,18	This proposed change only clarifies the classification of three REP's. These 3 events have caused concern over the Ambiguity of the classification when a Parameter is exceeded . Then returns to normal, the change clarifies examples used in determining the classification. The clarifications concern: Anticipated transient without Scram (ATWS) there are 2 clarifications to this event. Modification to scope of procedure. REASON FOR URGENCY: to ensure clarifications are recieved prior to NRC visit. Date needed approved by 1/5/89. Effective as soon as possible after approval.
7	02/09/90	ALL	Change format to improve human factors. Delete requirement to declare notification of unusual event if flood protection instruments are inoperable. Corrects value of coolant activity spike required for notification of unusual event from 3.2 to 26 uci/gm. Clarifies other events which require action levels. Added emergency classification flowchart (NER/C NRC IN 89-072). URGENCY: Needed PORC approval by 02/09/90 for operations requal training.

EMERGENCY CLASSIFICATION LOGIC

REV 0007

1.0 PURPOSE

1.1 To provide guidance to the Shift Operations Supervisor (SOS) or Site Emergency Director (SED) on what constitutes an emergency classification.

1.2 To ensure that the emergency classification is consistent with that used by the local and state governments and the NRC.

1.3 To provide a cross reference between this procedure and the ONP-REP, Appendix A, for use by the SOS or SED for additional information in classifying events.

2.0 SCOPE

2.1 This procedure applies to those events, that in the professional judgment of the SOS <u>or</u> the SED constitutes an emergency. The SOS and the SED are the only individuals authorized to make the emergency class determination.

2.2 The events listed in the attachments to this procedure cannot possibly incorporate all events which can occur. Therefore, all classifications should be judged against the general guidance listed below:

2.2.1 Notification of Unusual Event

Unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

2.2.2 Alert

Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

2.0 (Continued)

2.2.3 Site Area Emergency

Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to exceed EPA Protective Action Guideline exposure levels except near site boundary.

2.2.4 General Emergency

Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

3.0 INSTRUCTION

3.1 Review Attachment 1 to determine if an event should be classified as an emergency.

Note: (1) If an emergency action level for a higher classification was exceeded, but the present situation indicates a lower classification, the fact that the higher classification occurred shall be reported to the NRC and the CECC (if staffed), but should not be declared. (2) If an emergency action level was met but the emergency has been totally resolved, the emergency class that was appropriate shall be declared and terminated at the same time.

3.1.1 Attachment 1 captures events in four broad categories:

Fission Product Barrier Degradation (F)
System Malfunction (S)
Radiation Levels Abnormal/Radiological Effluents (R)
Hazards and Other Conditions Affecting Plant Safety (H)

3.1.2 Each actual condition in a category is given an alphanumeric designator (FU1 = Fission Product Barrier Degradation (F) resulting in Notification of Unusual Event (U) #1).

3.1.3 The only significance of the alphanumeric designator is a cross-reference to ONP-REP, Appendix A, which provides additional information for the SOS/SED in classifying the event.

3.0 (Continued) REV 0007

3.2 If the event is determined to be one of the four emergency classification, the SOS assumes the responsibilities of SED.

3.2.1 Implement one of the following procedures as applicable:

EPIP-2 - Notification of Unusual Event
EPIP-3 - Alert
EPIP-4 - Site Area Emergency
EPIP-5 - General Emergency

3.2.2 Continue to review the emergency conditions in Attachment 1 to escalate, de-escalate or terminate the emergency as appropriate.

3.3 If the event is determined not to be one of the four emergency classifications, continue to monitor plant conditions.

4.0 ATTACHMENTS

Attachment 1, Emergency Classification Flowchart [NER/C NRC in 89-072]

END OF TEXT

Index to Emergency Classification Flow Chart

DESCRIPTION — PAGE #

I. Fission Product Barrier Degradation

A. Fuel Damage 1
B. Primary System Leakage 1
C. Primary Coolant Break or Loss of Inventory 2
D. Primary Containment Integrity 2
E. Loss of Fission Product Barriers 2

II. System Malfunction

A. Tech Spec LCO 3
B. RPS/Core 3
C. Thermal Power 3
D. Shutdown 2
E. Turbine and Condenser 3
F. AC Power 4
G. DC Power 4
H. Instrumentation, Controls, and Communications 5

III. Radiation Levels Abnormal/ /Effluents

A. Radiological Effluents (Liquid) 6
B. Radiological Effluents (Gaseous) 7
C. Area Radiation 8

IV. Hazards and Other Conditions Affecting Plant Safety

A. Security Threat 9
B. Missiles or Aircraft 9
C. Injuries 10
D. Uncontrolled Toxic Gases 10
E. Control Room 10
F. Earthquake 11
G. Flood 11
H. Tornado 11
I. High Winds 11
J. Low Reservoir 12
K. Fire 12
L. Explosion 12
M. Flammable Gas or Vapors 12
N. Other 12

REV 0007

EMERGENCY CLASSIFICATION FLOWCHART

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

FISSION PRODUCT BARRIER DEGRADATION

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
FUEL DAMAGE	FU1 TECH SPEC 3.6.B.6 EXCEEDED FOR IODINE SPIKE (26 uCi|gm DOSE EQUIVALENT I-131 IN COOLANT) FU2 MAIN STEAM LINE RADIATION 1.5 X NORMAL FULL POWER BACKGROUND (ALARM)	FA1 COOLANT ACTIVITY EQUIVALENT TO IODINE CONCENTRATION ≥300 uCi|cc IN COOLANT BY ANALYSIS FA2 MAIN STEAM LINE RADIATION 3 X NORMAL FULL POWER BACKGROUND (ALARM AND ISOLATION) FA3 DAMAGE OR DROPPED FUEL BUNDLE AND RELEASE OF RADIOACTIVE MATERIALS	FS1 MAJOR DAMAGE TO SPENT FUEL OR FUEL POOL WATER LEVEL BELOW TOP OF SPENT FUEL	
PRIMARY SYSTEM LEAKAGE	FU3 LEAKAGE EXCEEDING TECH SPEC 3.6.C.1 > 5 GPM UNIDENTIFIED LEAKAGE OR > 25 GPM TOTAL LEAKAGE OR > 2 GPM INCREASE IN UNIDENTIFIED LEAKAGE IN A 24 HOUR PERIOD (EXCEPT FOR FIRST 24 HOURS AFTER GOING TO RUN MODE) FU4 MAIN STEAM RELIEF OR SAFETY RELATED SYSTEM RELIEF VALVE OPENING AND FAILURE TO RECLOSE AS EXPECTED. (HIGH TEMP OR FLOW OR CONTROL ROOM ALARM (TR-1-1, FA-1-1)	FA4 TOTAL LEAKAGE > 40 GPM (IDENTIFIED AND UNIDENTIFIED LEAKAGE) (BASED ON PUMPING RATE DETERMINED BY SI-2 OR SED JUDGMENT)		

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EMERGENCY CLASSIFICATION FLOWCHART

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

FISSION PRODUCT BARRIER DEGRADATION

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
PRIMARY COOLANT BREAK OR LOSS OF INVENTORY	FU5 ECCS INITIATION WITH DISCHARGE TO VESSEL (HPCI, RCIC, LPCI, CSS, OR ADS) UNLESS PART OF A PLANNED SEQUENCE DURING TESTING OR MAINTENANCE		FS2 MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT WITHOUT ISOLATION FS3 LOSS OF COOLANT INVENTORY GREATER THAN MAKEUP PUMP CAPACITY OR REACTOR WATER LEVEL DECREASES BELOW -150 INCHES, WITHOUT CAPABILITY TO RESTORE FS4 BREACH OF LARGE REACTOR COOLANT SYSTEM LINE (RWCU - HPCI|RCIC INJECTION OR STEAM) OUTSIDE CONTAINMENT WITHOUT ISOLATION	FG1 ECCS FAILURE (PUMPS NOT RUNNING OR UNABLE TO MAINTAIN REACTOR WATER LEVEL LEADING TO FUEL MELT)
PRIMARY CONTAINMENT INTEGRITY	FU6 DRYWELL LEAKAGE OF > 542 SCFH FU7 TECH SPEC 1.0, PARAGRAPH O DEFINITION NOT MET (PRIMARY CONTAINMENT)	FA5 MAIN STEAM LINE NOT ISOLATED WHEN REQUIRED. (BOTH INBOARD AND OUTBOARD MSIVS FAIL TO CLOSE)		FG2 DRYWELL PRESSURE > 50 PSIG OR DRYWELL TEMPERATURE > 280°F
LOSS OF FISSION PRODUCT BARRIERS	FU8 ISOLATION OF OFF GAS SYSTEM DUE TO POST TREATMENT MONITOR RM-90-265|266 OR PRETREATMENT MONITOR RM-90-157 ALARMS OR 0.05 MREM|HR ABOVE BACKGROUND FOR ONE HOUR AT OR BEYOND SITE BOUNDARY OR ANY MEASURABLE IODINE AT OR BEYOND SITE BOUNDARY		FS5 DEGRADED CORE WITH POSSIBLE LOSS OF COOLABLE GEOMETRY	FG3 LOSS OF 2 OF 3 FISSION PRODUCT BARRIERS WITH A POTENTIAL LOSS OF 3RD BARRIER BARRIERS ARE: FUEL CLADDING, PRIMARY COOLANT BOUNDARY, AND PRIMARY CONTAINMENT

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EMERGENCY CLASSIFICATION FLOWCHART

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

SYSTEM MALFUNCTION

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
TECH SPEC LCO	SU1 TECH SPEC LCO REACHED REQUIRING SHUTDOWN			
RPS|CORE THERMAL POWER [NRC|C 84-42-01]		SA1 FAILURE TO INITIATE AND COMPLETE A SCRAM TO BRING RX SUBCRITICAL FROM ANY EXPECTED SCRAM SIGNAL	SS1 CORE THERMAL POWER > 3% AND REACTOR CRITICAL WHEN SHUTDOWN REQUIRED (POWER > 3% AFTER INITIATING MANUAL SCRAM, MODE SWITCH IN SHUTDOWN, AND RECIRCULATION PUMPS TRIPPED)	
SHUTDOWN		SA2 LOSS OF CAPABILITY TO ATTAIN OR MAINTAIN COLD SHUTDOWN CONDITIONS USING ALL AVAILABLE SYSTEMS		
TURBINE AND CONDENSER	SU2 REACTOR SCRAM DUE TO FAILURE OF MAIN TURBINE ROTATING COMPONENT	SA3 MAIN TURBINE MECHANICAL FAILURE RESULTING IN PENETRATION OF TURBINE		

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EMERGENCY CLASSIFICATION FLOWCHART

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

SYSTEM MALFUNCTION

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
AC POWER	SU3 LOSS OF ALL OFFSITE (1) OR ONSITE AC POWER SUPPLY (2) TO ANY UNIT	SA4 LOSS OF ALL OFFSITE (1) AND ALL ONSITE AC POWER SUPPLY (3) TO ANY UNIT (POTENTIAL ESCALATION TO SITE AREA EMERGENCY)	SS2 LOSS OF ALL OFFSITE (1) AND ALL ONSITE AC POWER SUPPLY (3) TO ANY UNIT FOR MORE THAN 15 MINUTES	
DC POWER		SA5 LOSS OF ANY 250V DC UNIT BATTERY BOARD AND NOT IN COLD SHUTDOWN SA6 LOSS OF ALL ANNUNCIATORS IN GROUP I (PANEL 9-3A THRU F) OR II (PANEL 9-4A, 4B AND 9-5A, B) FOR > 15 MINUTES	SS3 LOSS OF ALL CONTROL ROOM ANNUNCIATORS IN GROUP I (PANEL 9-3A THRU F) OR II (PANEL 9-4A, B, AND 9-5A, B) AND PLANT TRANSIENT INITIATED OR IN PROGRESS SS4 LOSS OF ALL 250V DC UNIT BATTERY BOARDS FOR > 15 MINUTES	

(1) Indicated by loss of voltage to both 4kv shutdown buses (Units 1 & 2), or loss of voltage from all offsite sources to all 4KV shutdown boards on Unit 3.
(2) Two or more unit related diesel generators inoperable simultaneously by unscheduled outage or failure (when not in cold shutdown).
(3) Loss of voltage to all 4KV shutdown boards on any unit.

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EMERGENCY CLASSIFICATION FLOWCHART

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

SYSTEM MALFUNCTION

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
INSTRUMENTATION, CONTROLS, AND COMMUNICATIONS	SU4 LOSS OF CONTROL ROOM SURVEILLANCE INSTRUMENTS OR ALARMS LISTED IN TECH SPEC TABLE 3.2.F REQUIRING SHUTDOWN SU5 LOSS OF COMPUTER REQUIRING SHUTDOWN SU6 LOSS OF ALL METEOROLOGICAL INSTRUMENTATION (BOTH RRRMS AND AT METEOROLOGICAL TOWER) LISTED IN T.S. TABLE 3.2.I EXCEPT FOR PLANNED OUTAGES WHERE COMPENSATING MEASURES ARE IN PLACE SU7 LOSS OF RPIS INDICATION OR ALARMS SU8 SIGNIFICANT LOSS OF OFFSITE COMMUNICATIONS CAPABILITY FOR > 10 MINUTES SU9 LOSS OF MSL RADIATION, FLOW, OR AREA TEMPERATURE INSTRUMENTATION LISTED IN T.S. TABLE 3.2.A OR ALARMS REQUIRING SHUTDOWN			

EMERGENCY CLASSIFICATION FLOWCHART

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NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

RADIATION LEVELS ABNORMAL|RADIOLOGICAL EFFLUENTS

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
RADIOLOGICAL EFFLUENTS LIQUID	RU1 LIQUID RELEASE EXCEEDING TECH SPECS 3.8.A.1, 3 OR 5	RA1 LIQUID RELEASE EXCEEDING 10 TIMES TECH SPECS 3.8.A.1., 3, OR 5 RA2 LIQUID RELEASE EXCEEDING 10 CFR 20 LIMITS BY A FACTOR OF 10 RA3 LIQUID RELEASE THAT CANNOT BE TERMINATED		

EMERGENCY CLASSIFICATION FLOWCHART

REV 0007

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

RADIATION LEVELS ABNORMAL|RADIOLOGICAL EFFLUENTS

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
RADIOLOGICAL EFFLUENTS GASEOUS (COMMITMENT TO STATE OF ALABAMA)	RU2 GAS RELEASE RATE EXCEEDING TECH SPECS 3.8.B.1, 3 OR 5 BASED ON CALCULATIONS FROM 0-SI-4.8.B.1.a.1, 2 1 (2 OR 3)-SI-4.8.B.5.a, OR 1 (2)-SI-4.8.B.1.a.3 3-SI-4.8.B.1.a	RA4 GAS RELEASE RATE EXCEEDING 10 TIMES TECH SPECS 3.8.B.1., 3, OR 5 BASED ON CALCULATIONS FROM SI-4.8.B.1.a.1, SI-4.8.B.1.A.2, SI-4.8.B.1.A.3, SI-4.8.B.1.5.A OR ENVIRONMENTAL MEASUREMENTS AT OR BEYOND THE SITE BOUNDARY OF 0.5 MREM|HR FOR ONE HOUR WHOLE BODY OR 4.9E-10 uCi|CC IODINE NRC|C	RS1 VERIFIED TOTAL PLANT NOBLE GAS RELEASE FOR THE STACK OF 1,000 Ci|SEC FOR > 30 MINUTES OR 10,000 Ci|SEC FOR > 2 MINUTES AS DETERMINED BY PLANT COMPUTER OR SITE EPIP-3, 4, OR 5 MANUAL CALCULATIONS OR ENVIRONMENTAL MEASUREMENTS AT OR BEYOND THE SITE BOUNDARY FIND ONE OF THE FOLLOWING: 1. ≥ 500 MREM|HR WHOLE BODY FOR > 2 MINUTES 2. ≥ 7.31E-7 uCi|CC IODINE FOR > 2 MINUTES 3. ≥ 50 MREM|HR WHOLE BODY FOR > 30 MINUTES 4. ≥ 7.31E-8 uCi|CC IODINE FOR > 30 MINUTES OR PROJECTED ACCUMULATED DOSE AT OR BEYOND THE SITE BOUNDARY: 1. > 1 REM WHOLE BODY 2. > 5 REM THYROID	RG1 AS PROJECTED FROM PLANT PARAMETERS AND ACTUAL METEOROLOGICAL CONDITIONS, MONITORS DETECT RELEASE SUFFICIENT TO CAUSE RADIATION DOSE RATES AT THE SITE BOUNDARY OF 1 REM|HR WHOLE BODY OR 5 REM|HR DOSE COMMITMENT RATE TO THE THYROID. FOR BFN STACK RELEASE THIS VALUE IS 20,000 Ci|SEC. OR ENVIRONMENTAL MEASUREMENTS AT OR BEYOND THE SITE BOUNDARY FIND ONE OF THE FOLLOWING: 1. > 1 REM|HR WHOLE BODY 2. > 1.46E-6 uCi|CC IODINE

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EMERGENCY CLASSIFICATION FLOWCHART

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

RADIATION LEVELS ABNORMAL|RADIOLOGICAL EFFLUENTS

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
AREA RADIATION		RA5 RADIATION UNEXPECTEDLY ABNORMAL INCREASE BY 1 R|HR (ALARM|RADCON CONFIRMATION RA6 AIRBORNE RADIATION UNEXPECTEDLY INCREASES BY 100 MPC FOR A CONTROLLED AREA. (CAM ALARM|RADCON CONFIRMATION)		

(1) MPC = MPC FOR A CONTROLLED AREA.

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EMERGENCY CLASSIFICATION FLOWCHART

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
SECURITY THREAT	HU1 SECURITY THREAT - ATTEMPTED ENTRY (SOS|SED JUDGMENT BASED ON ADVICE FROM NUCLEAR SECURITY SUPERVISOR) HU2 ATTEMPTED SABOTAGE (SOS|SED JUDGMENT BASED ON ADVICE FROM NUCLEAR SECURITY SUPERVISOR)	HA1 UNAUTHORIZED PERSONNEL INSIDE PLANT SECURITY BOUNDARIES (SED JUDGMENT BASED ON ADVICE FROM NUCLEAR SECURITY SUPERVISOR) HA2 ACTUAL SABOTAGE (SED JUDGMENT)	HS1 UNAUTHORIZED PERSONNEL INSIDE PLANT (VITAL AREAS) THREATENING PHYSICAL CONTROL OF FACILITY (SED JUDGMENT BASED ON ADVICE FROM NUCLEAR SECURITY SUPERVISOR)	HG1 LOSS OF PHYSICAL CONTROL OF FACILITY (SED JUDGMENT)
MISSILES OR AIRCRAFT	HU3 CRASH WITHIN SITE BOUNDARY OR UNUSUAL ACTIVITY OVER FACILITY, (SOS|SED JUDGMENT)	HA3 CRASH WITHIN THE PROTECTED AREA	HS2 CRASH INVOLVING VITAL AREAS, STRUCTURES OR EQUIPMENT AND NOT IN COLD SHUTDOWN HS3 SEVERE DAMAGE TO SAFE SHUTDOWN EQUIPMENT FROM MISSILE OR EXPLOSION AND NOT IN COLD SHUTDOWN	

EMERGENCY CLASSIFICATION FLOWCHART

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NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

NOTE

REFER TO EPIP - 10 FOR ANY MEDICAL EMERGENCY

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
INJURIES	HU4 INJURED AND CONTAMINATED INDIVIDUAL TRANSPORTED TO OFFSITE HOSPITAL			
UNCONTROLLED TOXIC GASES	HU5 TOXIC GASES NEAR OR ONSITE AND MAY IMPAIR STATION OPERABILITY (SOS JUDGMENT)	HA4 TOXIC GASES WITHIN PROTECTED AREA AFFECTING SAFE OPERATION	HS4 TOXIC GASES WITHIN VITAL AREAS AFFECTING OPERATIONS AND NOT IN COLD SHUTDOWN	
CONTROL ROOM		HA5 EVACUATION OR ANTICIPATED EVACUATION OF THE CONTROL ROOM	HS5 EVACUATION AND CONTROL FROM BACKUP CONTROL PANEL NOT ESTABLISHED WITHIN 15 MINUTES	

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EMERGENCY CLASSIFICATION FLOWCHART

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
EARTHQUAKE	HU6 TRIAXIAL OPERATION > 0.01g (SEISMIC ACCELEROMETERS, ALARM START OF STRONG MOTION ACCELEROMETERS) (REFER TO NOTE 1)	HA6 BIAXIAL OPERATION > 0.1g (SEISMIC TRIGGERS, ALARM SEISMIC TRIGGER A, B, OR C) (REFER TO NOTE 2)		
FLOOD	HU7 RESERVOIR LEVEL > 563.5' BUT < 565.0'	HA7 RESERVOIR LEVEL > 565.0' (WATER RUNNING ON CCW PUMP DECK) AND PLANT WATERTIGHT	HS6 RESERVOIR LEVEL > 565.0' (WATER RUNNING ON CCW PUMP DECK) AND ANY INDICATION OF INPLANT FLOODING OR LOSS OF ANY PLANT WATERTIGHT FEATURES AND UNITS NOT IN COLD SHUTDOWN	
TORNADO (REFER TO 0-AOI-100-7 ALSO FOR WATCH OR WARNING)	HU8 TORNADO ONSITE (OWNER CONTROLLED PROPERTY)	HA8 TORNADO STRIKING PLANT (REACTOR, TURBINE, SERVICE, OR DIESEL GENERATOR BLDGS, INTAKE OR SWITCHYARD)		
HIGH WINDS	HU9 SUSTAINED WINDS > 70 MPH BUT < 85 MPH AS INDICATED IN THE CONTROL ROOM	HA9 SUSTAINED WINDS > 85 MPH BUT < 100 MPH AS INDICATED IN THE CONTROL ROOM	HS7 SUSTAINED WINDS > 100 MPH AS INDICATED IN THE CONTROL ROOM AND NOT IN COLD SHUTDOWN	

Note 1. BEFORE INITIATING EPIP-2, CONFIRM BUILDING MOVEMENT AND|OR CALL NATIONAL EARTHQUAKE INFORMATION CENTER AT (303) 236-1500.

Note 2. BEFORE INITIATING EPIP-3, CONFIRM BUILDING MOVEMENT AND CALL NATIONAL EARTHQUAKE INFORMATION CENTER AT (303) 236-1500.

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EMERGENCY CLASSIFICATION FLOWCHART

NOTE: The NP-REP, Appendix A, contains information or detail, related to emergency classifications or emergency action levels.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

DESCRIPTION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION	INITIATING CONDITION
LOW RESERVOIR	HU10 WHEELER RESERVOIR BELOW SEASONAL LEVEL DUE TO PROBLEM AT WHEELER DAM	HA10 CATASTROPHIC FAILURE OF WHEELER DAM CAUSING LOW RESERVOIR LEVEL	HS8 CATASTROPHIC FAILURE OF WHEELER DAM AND RHRSW| EECW PUMPS SUCTION LOST AND NOT IN COLD SHUTDOWN	
FIRE	HU11 FIRE WITHIN THE PROTECTED AREA LONGER THAN 10 MINUTES OR INVOLVING RADIOACTIVE MATERIAL	HA11 FIRE THREATENING VITAL AREA OR SAFETY SYSTEM (SOS|SED JUDGMENT)	HS9 FIRE AFFECTING SAFETY SYSTEM OR VITAL AREA (SED JUDGMENT)	
EXPLOSION	HU12 UNPLANNED EXPLOSION NEAR OR WITHIN SITE BOUNDARY	HA12 EXPLOSION WITHIN PROTECTED AREA CAUSING DAMAGE TO THE FACILITY AFFECTING PLANT OPERATION	HS10 EXPLOSION CAUSING MAJOR DAMAGE INVOLVING VITAL STRUCTURES OR EQUIPMENT AND NOT IN COLD SHUTDOWN	
FLAMMABLE GAS OR VAPORS	HU13 GAS OR VAPORS, NEAR OR ONSITE AND MAY IMPAIR STATION OPERABILITY (SOS|SED JUDGMENT)	HA13 GAS OR VAPORS, UNCONTROLLED WITHIN PROTECTED AREA AFFECTING SAFE OPERATION (SOS|SED JUDGMENT)	HS11 GAS OR VAPORS, UNCONTROLLED WITHIN VITAL AREAS AFFECTING SAFE OPERATION AND NOT IN COLD SHUTDOWN	
OTHER (1)	HU14 CONDITIONS EXIST WARRANTING INCREASED AWARENESS OF PLANT OPERATING STAFF, STATE OR LOCAL GOVERNMENT, OR INVOLVE OTHER THAN NORMAL CONTROLLED SHUTDOWN	HA14 CONDITIONS EXIST WHICH WARRANT ACTIVATION OF THE TSC AND CECC (POTENTIAL TO ESCALATE TO SITE AREA EMERGENCY)		HG2 MAJOR INTERNAL OR EXTERNAL EVENTS WHICH COULD CAUSE MASSIVE COMMON DAMAGE TO PLANT SYSTEMS

1) These items are based on Site Emergency Director's professional judgment.

LAST PAGE

APPENDIX A

BROWNS FERRY NUCLEAR PLANT

MASTER

ENCLOSURE 2

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

FACILITY: Browns Ferry 1, 2, & 3

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 90/06/25

CANDIDATE: ______________________

INSTRUCTIONS TO CANDIDATE:

Points for each question are indicated in parentheses after the question. T. pass this examination, you must achieve an overall grade of at least 80%. Examination papers will be picked up four and one half (4 1/2) hours after the examination starts.

NUMBER QUESTIONS	TOTAL POINTS	CANDIDATE'S POINTS	CANDIDATE'S OVERALL GRADE (%)
94	100.00		

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

7. You may write your answers on the examination question page or on a separate sheet of paper. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

8. If you write your answers on the examination question page and you need more space to answer a specific question, use a separate sheet of the paper provided and insert it directly after the specific question. DO NOT WRITE ON THE BACK SIDE OF THE EXAMINATION QUESTION PAGE.

9. Print your name in the upper right-hand corner of the first page of answer sheets whether you use the examination question pages or separate sheets of paper. Initial each of the following answer pages.

10. 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.

11. If you are using separate sheets, number each answer and skip at least 3 lines between answers to allow space for grading.

12. Write "Last Page" on the last answer sheet.

13. 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.

14. 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.

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

16. 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.

17. Proportional grading will be applied. Any additional wrong information that is provided may 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, and 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 got the four correct answers.

18. If the intent of a question is unclear, ask questions of the examiner only.

19. When turning in your examination, assemble the completed examination with examination questions, examination aids and answer sheets. In addition, turn in all scrap paper.

20. To pass the examination, you must achieve an overall grade of 80% or greater.

21. There is a time limit of (4 1/2) hours for completion of the examination. (or some other time if less than the full examination is taken.)

22. 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: 001 (1.00)

A reactor start-up and heat-up is in progress on Unit 2. Reactor pressure is 520 psig and the 1B CRD pump is out of service for bearing replacement. The following alarms/indicators are received on Unit 2:

PA-85-1, CRD pump A suct press low
2A CRD pump breaker trips
CRD drive water HDR diff press. is 175 psid
TA-85-1, Control Rod Drive Temp High

Which ONE of the following describes the action to be taken:

a. Insert a manual scram.

b. When the second accumulator light comes in, manually scram.

c. If charging water pressure is < 1410 psig, manually scram.

d. If CRD system not restored in 1 hour, manually scram.

QUESTION: 002 (1.00)

A plant heatup is in progess from cold shutdown to rated pressure. During the heatup actual reactor water level remains constant. The Emergency System Range Level Indicators (LI-3-58A and LI-3-58B) will decrease and come on scale due to which ONE of the following?

a. Due to reference leg heat up.

b. Due to variable leg heat up.

c. Due to increased downcomer subcooling.

d. Due to increased core flow.

QUESTION: 003 (1.00)

During an ATWS SLC was injected per the EOI's. When would SLC injection be terminated?

a. All control rods are inserted to 00.

b. Suppression pool temperature is less than 110 deg. F.

c. SLC tank level has decreased to 18%.

d. Indicated Reactor Power < 3%.

QUESTION: 004 (1.00)

Which ONE of the following is the purpose of the recirculation pump 75% limiter?

a. Prevent caviatation of the Recirculation Pumps.

b. Prevent overspeed trip of the remaining Feedpump.

c. Prevent a low reactor water level scram.

d. Prevent Jetpump vibration.

QUESTION: 005 (1.00)

During a Rx startup operating temperature and pressure was reached at 0100 hrs and the Mode Switch was placed in Run at 0345 hrs. Which ONE of the following describes the requirements for satisfying primary containment limitations?

a. Nitrogen inerting must be completed by 0100 hrs the next day and delta P control must be established by 0345 hrs the next day.

b. Both inerting and delta P control must be completed by 0100 the next day.

c. Nitrogen inerting must be completed by 0345 hrs the next day and delta P control must be established by 0100 hrs the next day.

d. Both inerting and delta P control must be completed within 24 hours after placing the mode switch to run.

QUESTION: 006 (1.00)

The following conditions exist on Unit 2:

Reactor scrammed from 100% power due to low level [loss of all reactor feed pumps].

All control rods are fully inserted.
RCIC and HPCI failed to operate.
Reactor water level is -150 inches.
No injection system or alternate injection is available.

Which ONE of the following describes the operator actions required?

a. Enter C2, Emergency Rx Depressurization, and attempt to restore a low pressure injection system to operation in accordance with C1, Alternate Level Control.

b. Enter C3, Steam Cooling and attempt to restore an injection system to operation in accordance with C1, Alternate Level Control.

c. Enter C4, Reactor Flooding Pressure, and when pressure is below the Minimum Alternate Flooding Pressure, return to RC/L Level Control.

d. No action is required until reactor water level reaches the top of the active fuel. Then enter C7, Core Cooling Without Level Restoration.

QUESTION: 007 (1.00)

The following plant conditions exist:
Unit 2 has scrammed from rated conditions.
Twelve control rods failed to insert.
A condition exists requiring Emergency Depressurization.
Injection has been terminated and prevented.
All six ADS valves have been manually opened.

Which ONE of the following describes the appropriate point at which injection to the vessel can be re-established?

a. Reactor pressure is <190 psig.

b. Reactor pressure is <450 psig.

c. Reactor water level decreases to TAF.

d. Reactor water level decreased to -150 inches.

QUESTION: 008 (1.00)

Which ONE of the following describes the purpose of the Drywell Spray Initiation Limit Curve?

a. Assures the prevention of equipment failure due to unstable steam condensation during an ADS blowdown.

b. Assures that actuation of ADS will not result in damage to the pool or any submerged structure within the suppression pool.

c. Assures that any steam released in the drywell will be directed under water in the suppression pool.

d. Assures that the drywell will not collapse or otherwise fail due to negative pressure.

QUESTION: 009 (1.00)

With the Reactor at 94% rated thermal power and Recirc Flow at 90%, a malfunction of the EHC system results in a slow increase in reactor pressure and power until the reactor scrams on High Neutron Flux at 118%.

Which ONE of the following statements is correct concerning the above situation?

a. The Thermal Power Safety Limit has been exceeded.

b. The Power Transient Safety Limit has been exceeded.

c. The Reactor Vessel Water Level Safety Limit has been exceeded.

d. The Reactor Coolant System Integrity Safety Limit has been exceeded.

QUESTION: 010 (1.00)

Unit 2 is operating at 80% power when a steam leak in the tunnel results in a Group 1 Isolation. The reactor scrams due to MSIV closure. ONE SRV with a normal setpoint of 1125 psig fails to open, and peak reactor pressure reaches 1150 psig.

Which ONE of the following statements is correct concerning the above situation?

a. The Thermal Power Safety Limit has been exceeded.

b. The Power Transient Safety Limit has been exceeded.

c. The Reactor Coolant System Integrity Safety Limit has been exceeded.

d. No safety limit has been exceeded.

QUESTION: 011 (1.00)

Unit 2 has experienced a failure to scram from 100% power. The following plant parameters and conditions exist:

Reactor power : 7%
Suppression pool temperature : 118 F
Rx. pressure : 920 psig
SLC tank level : 40%
MSIV's : open
Rx. water level : -75 inches.

Which ONE of the following describes the conditions necessary to start a cooldown:

a. Suppression pool temperatures < 110 F.

b.. Reactor power < 3%.

c. Reactor subcritical.

d. SLC injected to a level of 6%.

QUESTION: 012 (1.00)

While operating at power, a transient occurs. The MSIVs close due to low water level. The reactor automatically scrams, however, many rods fail to insert. HPCI fails, but RCIC slowly recovers level. Suppression pool cooling is placed in service. Pressure control is established on the SRVs. The following plant conditions exist:

Reactor power = 5%
Reactor pressure = 1000 psig
Reactor level = -45
Suppression pool temperature = 100 Deg F [Slowly increasing]
Suppression pool level = +1"

Which ONE of the following actions should be performed?

a. Reactor water level should be deliberately lowered to control reactor power.

b. SLC injection should be initiated.

c. MSIVs should be opened to re-establish the main condenser as a heat sink.

d. Emergency depressurization should be conducted.

QUESTION: 013 (1.00)

A loss of coolant accident occurs from normal operating conditions. All emergency systems [PRS, PCIS, ECCS] respond normally. Use the following parameters to answer the question below.

Reactor pressure = 100 psig
Reactor level = -20"
Drywell pressure = 26 psig
Drywell Temperature = 180 Deg. F
Suppression pool level = 18.5 ft.
Suppression pool temperature = 165 Deg. F
Suppression chamber temperature 160 Deg. F
Suppression chamber pressure = 26 psig

Which ONE of the following actions is appropriate?

a. Initiate Drywell and Suppression pool sprays.

b. Initiate Drywell spray, but not Suppression Pool spray.

c. Initiate Suppression pool sprays, but not Drywell sprays.

d. Do not initiate Drywell or Suppression pool sprays.

QUESTION: 014 (1.00)

Which ONE of the following describes the purpose of the Heat Capacity Temperature Limit?

a. Assures the prevention of equipment failure due to unstable steam condensation during an ADS blowdown.

b. Assures that actuation of ADS will not result in damage to the pool or any submerged structure within the suppression pool.

c. Assures that any steam released in the drywell will be directed under water in the suppression pool.

d. Assures that the drywell will not collapse or otherwise fail due to negative pressure.

QUESTION: 015 (1.00)

The following plant conditions exist on Unit 2:

Reactor scrammed
Control Room evacuated
Plant cooldown required using SRV's

Suppression Pool Cooling is required during the cooldown. Which ONE of the following is the location from which RHR pump 2C would be started.

a. The Backup Control Panel.

b. 4KV Shutdown Board B .

c. 4KV Shutdown Board C.

d. Locally at the pump.

QUESTION: 016 (1.00)

A Unit 2 start-up is in progress, at 20% power, with the turbine rolling at 1800 rpm. The RFP "A" control signal is lost, and the MGU is locked. For the given conditions, which ONE of the following describes the response of the "A" RFP if the MGU lock-out were reset [Hydraulic Jack switch is on]:

a. Speed would increase to the MGU High Speed Stop.

b. Speed would decrease to the MSC LSS.

c. Speed would be controlled by the master FWLCS controller.

d. Speed would remain the same, controlled by the MSC.

QUESTION: 017 (1.00)

EOI-1 Reactor Control, is being executed following a scram due to a turbine trip at high power. During the initial phase of the transient, pressure increased to the SRV lift setpoint, and ONE of the SRV, sticks open. Suppression pool temperature has reached 95 Deg. F. Which ONE of the following actions should take place?

a. Re-enter EOI-1 at the beginning.

b. Renter EOI-1 at the beginning and enter EOI-2.

c. Continue in EOI-1 and enter EOI-2.

d. Continue in EOI-1.

QUESTION: 018 (1.00)

Unit 2 is operating at 55% power when the RBCCW essential loop isolation valve [70-47] closes and will not respond to an open signal. Which ONE of the following describes the required action?

a. If temperature limits are exceeded on A or B recirc pump, Manually scram the Rx and trip the associated recirc pump.

b. Manually scram the Rx and initiate a cooldown at 90 deg. F/hour.

c. Reduce Rx power in an attempt to reduce drywell and recirc pump temperatures.

d. If drywell temperature exceeds 160 deg. F, enter EOI-1 and EOI-2.

QUESTION: 019 (1.00)

With Unit 2 operating at power and Unit 1 in cold shutdown, an event occurs causing the SOS to decide that the control room must be evacuated. All required immediate actions are taken. Which ONE of the following describes the condition of Unit 2?

a. Hot standby

b. Hot shutdown

c. Cold shutdown

d. Operating at reduced power

QUESTION: 020 (1.00)

Which ONE of the following describes the purpose of the Suppression Pool Load Limit Curve?

a. Assures the prevention of equipment failure due to unstable steam condensation during an ADS blowdown.

b. Assures that actuation of ADS will not result in damage to the pool or any submerged structure within the suppression pool.

c. Assures that any steam released in the drywell will be directed under water in the suppression pool.

d. Assures that the drywell will not collapse or otherwise fail due to negative pressure.

QUESTION: 021 (1.00)

The following plant condition exist:

Unit 2 scrammed
MSIV's closed
Suppression chamber pressure 3 psig.
Suppression Pool Temp 200 Deg. F

Which ONE of the following states the maximum allowable RHR pump flow?
[Refer to EOI Charts]

a. 3500 gpm

b. 5000 gpm

c. 8500 gpm

d. 11000 gpm

QUESTION: 022 (1.00)

The reactor is operating at 70% rated thermal power and 70% rated core flow. Which ONE of the following describes how you would decrease reactor power to 60% of rated, while maintaining core flow constant [70%]?

a. Insert control rods.

b. Withdraw control rods and decrease recirculation pump speed.

c. Insert control rods and decrease recirculation pump speed.

d. Insert control rods and increase recirculation pump speed.

QUESTION: 023 (1.00)

Following a reactor scram, reactor water level is restored with the use of HPCI and RCIC. If Rx level was allowed to increase to +54", which ONE of the following describes how HPCI and RCIC would respond?

a. Both would trip, if Rx level then decreased to -51.5", HPCI would auto initiate but RCIC would not.

b. Both would trip and neither would auto initiate if RX level later decreased to -51.5".

c. Both would trip, if Rx level then decreased to -51.5", both would auto initiate.

d. Both would trip, if Rx level then decreased to -51.5", RCIC would auto initiate but HPCI would not.

QUESTION: 024 (1.00)

Which ONE of the following explains why the CRD system flow controller [FIC 85-11] could indicate flow off-scale high [>100 gpm] with the flow control valves [FCV's 85-11A and 11B] both indicating closed.

a. Due to a rupture of the differential pressure detector bellows of the flow element [FE 85-11].

b. Due to drive water header flow directed to the CRD under piston area sensed by 85-11, as a result of a scram, but supplied through FCV's 85-11A and 11B.

c. Due to charging water header flow to the CRD under-piston area sensed by 85-11, as a result of a scram, but not supplied through FCV's 85-11A and 11B.

d. Due to a stem/disc seperation on Charging Water Throttle Valve resulting in CRD pump runout.

QUESTION: 025 (1.00)

If manual rod insertion Per RC/Q is required, choose the correct statement describing which interlocks, if any, are bypassed and how each bypass is accomplished.

a. RWM by use of the Full In/Full Out bypass switches; RSCS by use of the Emergency In position of HS 85-47.

b. RSCS by use of the Full In/Full Out bypass switches; and RWM by use of Manual Bypass switch.

c. RSCS by Mode Switch in Refuel and the Emergency In position of HS85-47; RWM by use of Manual Bypass switch.

d. Place Mode Switch in Refuel to allow selection of any rod; use the Emergency In position of HS 85-47 to bypass RSCS and RWM.

QUESTION: 026 (1.00)

Unit 2 is operating at 100% power with the following initial conditions:

Turbine load set	100%
Pressure set	920 psig
Max. combine flow set	100 [125% steam flow]
Load limit set	100%
Recirc flow control	Master manual

An electrical fault causes the load reject relay to pick-up. Which ONE of the following correctly describes the plant's response to this transient? Assume no operator action.

a. Reactor pressure will increase due to rapid closure of the Turbine Control Valves resulting in a high pressure reactor scram.

b. Control oil pressure will decrease due to rapid closure of the Turbine Control Valves resulting in a low control oil pressure scram.

c. Reactor power will increase due to rapid closure of the Turbine Control Valves resulting in a high flux reactor scram.

d. The Turbine Control Valves will rapidly close resulting in the Bypass Valves opening to control Pressure with the Plant eventually stabilizing at 60% power.

QUESTION: 027 (1.00)

Assume the plant is operating with the Bypass Valves open. If an SRV suddenly fails 100% open, how would the Turbine Bypass Valves respond? [Select ONE]

a. All Bypass Valves would close, and throttle pressure would decrease.

b. No change in Bypass Valve position of throttle pressure.

c. Bypass Valves would close down in response to the decrease in throttle pressure.

d. All Bypass Valves would close and throttle pressure would remain constant.

QUESTION: 028 (1.00)

With Unit 2 at 100% power, Unit 1 and 3 defueled, a loss of 500KV system occurs, followed a minute later by a loss of the 161KV system. All Diesel Generators start and are operating properly. After 30 minutes, Unit 2 is shutdown, with pressure and level under control. Which ONE of the following describes the requirements for paralleling D/G's to establish the main condenser as a heat sink.

a. Conditions for paralleling D/G's are not met.

b. D/G's B and 3EB should be paralleled.

c. D/G's A and 3EA should be paralleled.

d. D/G's A and B should be paralleled.

QUESTION: 029 (1.00)

Due to an inadvertent Group 6 Isolation, the SOS has determined either the plant control air system or the CAD system must be aligned to the drywell control air system. Which ONE of the following describes the concerns that must be addressed when either system is lined up?

a. Plant control air could cause a drywell pressure increase. CAD could cause an increase of drywell pressure and oxygen concentration.

b. Both plant control air and CAD could cause drywell pressure and oxygen concentration to increase.

c. Plant control air could cause an increase of drywell pressure and oxygen concentration. CAD could cause an increase of drywell pressu

d. The only concerns for either plant control air or CAD would be an increase of drywell pressure.

QUESTION: 030 (1.00)

Unit 2 is operating at 100% power, when RPS M/G set 2B trips. Five minutes later, the RPS bus is re-energized, from the alternate power supply. If the Reactor Mode Switch is placed in Shutdown before the Half Scram is reset, which ONE of the following RPS scram functions would cause the reactor to scram?

a. Mode switch in S/D

b. APRM Hi-Hi neutron flux

c. APRM Hi-Hi thermal

d. IRM Hi-Hi

QUESTION: 031 (1.00)

Which ONE of the following conditions will result in a trip of the Diesel Generator following an automatic start due to Hi Drywell Pressure?

a. Low oil pressure of 20 psig

b. Reverse power

c. High differential current

d. Low cooling water pressure of 20 psig

QUESTION: 032 (1.00)

Unit 2 is at approximately 10% power, with preparations in progress to roll the main turbine. The unit operator observes the following:

All bypass valves open
Rx pressure is decreasing
Rx power is decreasing
Rx level is increasing

Which ONE of the following describes the action to be taken.

a. Place EHC pumps in pull to lock.

b. Scram the reactor and break condenser vacuum.

c. Scram the reactor and close the MSIV's

d. Use max. combined flow to close the bypass valves.

QUESTION: 033 (1.00)

Which ONE of the following condition[s] is required to automatically bypass SRM rod blocks during a startup ?

a. SRM rod blocks are automatically bypassed ONLY when the Mode Switch in the RUN position.

b. SRM rod blocks are automatically bypassed when all IRM range switche are on range 8 or above, or the Mode Switch is in the RUN position.

c. SRM rod blocks will be automatically bypassed when all IRM switchs are on range 2 and the SRMs are reading less than 100 cps.

d. SRM rod blocks can only be bypassed when the Unit Operator has fully withdrawn all SRMs after verifying correct IRM over lap.

QUESTION: 034 (1.00)

Unit 2 is operating at 80 % rated thermal power when an accident signal occurs. Offsite power is available. Which ONE of the following describes the Unit 2 Core Spray system response?

a. Only the 2B and 2D Core Spray pumps start.

b. All Unit 2 Core Spray pumps start after 7 seconds.

c. 2B Core Spray pump starts, in 7 seconds, 2C Core Spray pump starts in 14 seconds.

d. 2B and 2D Core Spray pumps start in 7 seconds: 2A and 2C Core Spray pumps start in 14 seconsds.

QUESTION: 035 (1.00)

Which ONE of the following describes the purpose of the Diesel Generator Air Dryers?

a. Filter the corrosion products from the Diesel Generators starting air system.

b. Remove the moisture from the starting air system to reduce moisture related corrosion.

c. Heat the air in order to reduce the moisture in the D/G air system.

d. Remove the oil and debris from the system before going to the air start motors.

QUESTION: 036 (1.00)

The following plant conditions exist:

Jet pumps 1-10 Differential Pressure [meter] = 3 psid
Jet pumps 11-20 Differential Pressure [meter] = 15 psid
RECIRC LOOP B ONLY OUT OF SERVICE [annunciator] = ON

The TOTAL CORE FLOW recorder would calculate core flow by which ONE of the following methods?

a. Loop A Jet Pump flow + loop B Jet Pump flow.

b. Loop A Jet Pump flow - loop B Jet Pump flow.

c. Loop A Jet Pump flow only.

d. Loop B Jet Pump flow only.

QUESTION: 037 (1.00)

Which ONE of the following correctly describes the operation of the RWM as power is reduced from 100% power to 25% of rated?

a. The "AUTO" indicator light will extinguish when power is decreased below the LPSP.

b. The system will enforce the loaded rod sequence when power is decreased below the LPSP.

c. Both steam flow and feed flow must decrease below a designated setpoint to place the system in service.

d. All system alarms and displays are operative while in the "Transition Zone".

QUESTION: 038 (1.00)

Unit 2 is at 20% power with the Turbine/Generator synchronized to the grid. The unit operator observes the following:

Condenser A, B, or C vacuum low Alarm.
Standby SJAE auto-started.
Off-Gas Flow to 6-Hour Holdup Volume is decreasing.

Which ONE of the following describes the action to be taken?

a. Trip the main turbine.

b. Reduce generator load using load-set.

c. Scram the reactor and trip the turbine.

d. Place mechanical vacuum pumps in service.

QUESTION: 039 (1.00)

Unit 2 is in the process of starting-up, and the Unit operator just completed pulling the first control rod to position 48. He then successfully selects a control rod from Group 4 rather than Group 1. Which ONE of the following describes the restrictions and/or limitations now imposed?

a. The withdrawn control rod shall be manually inserted to position 00.

b. The withdrawn control rod shall be individually scrammed.

c. Control rod withdrawal may continue as long as a second operator verifies proper movement.

d. Manual withdrawal and insertion of control rods shall be stopped.

QUESTION: 040 (1.00)

A leak in the steam tunnel at 100% power resulted in a Group 1 Isolation. The reactor failed to scram with an 80% hydraulic ATWS. Given these conditions, which ONE of the following describes the actions to restore the main condenser as a heat sink?

a. Reduce steam tunnel temperature, reset PCIS, and equalize around and open MSIVs

b. Start an auxiliary boiler, line up aux. steam to steam seals, start mechanical vacuum pumps, and open the MSIVs

c. The main condenser is unavailable as a heat sink due to the presence of a Group 1 Isolation signal.

d. Bypass Steam Tunnel High Temperature Isolation and reset PCIS to restore the main condenser as a heat sink.

QUESTION: 041 (1.00)

Following a reactor scram several rod positions on the full core display have no numbered position indications, and only a green background. Which ONE of the following describes their position?

a. They are fully inserted with failed position 00 reed switches.

b. They are somewhere between the 00 position and the 48 position and driving in as a result of the scram signal.

c. They over travelled beyond full in as a result of the scram signal and should settle back to "00" position after the scram is reset.

d. Their full in/full out bypass switches in the aux instrument room have been bypassed in the Full In position per EOI Appendix 9.

QUESTION: 042 (1.00)

Unit 2 is operating at 98% power, when the Unit Operator notices the red and green lights above the hand switch for MSIV 1-15 are on, and "C" MSL flow reads less than A, B, and D. Which ONE of the following describes the action to be taken in response to the above indications?

a. Insert control rods until below 80% reactor power.

b. Reduce reactor power to 70% of rated.

c. Reduce recirc flow to 45% of rated.

d. Reduce core flow to below 45% of rated.

QUESTION: 043 (1.00)

Which ONE of the following describes the correct method for rapidly reducing Rx power from 100% during an abnormal condition?

a. Reduce Recirc flow to 45%, then stop flow reduction, and insert control rods until below the 80% rod line.

b. Reduce power with recirc flow until APRM High Rod Block is reached; insert rods until blocks are clear; if necessary proceed with flow reduction.

c. Reduce recirc pump speed to <28%; then insert control rods until below the 80% rod line.

d. Reduce recirc pump speed to 28%, then Manually Scram the plant.

QUESTION: 041 (1.00)

Unit 2 is in cold shutdown with the reactor vessel head removed in preparation for refueling. RHR Pump 2A is in shutdown cooling. RHR Pump 2C is tagged for motor inspection. Both recirculation pumps are out of service. Smoke is observed coming from RHR Pump 2A, and the pump subsequently trips. While troubleshooting is in progress, reactor vessel temperature is noted to be >150 Deg F. Which operator action is appropriate given the above conditions?

a. Attempt to reestablish primary containment integrity.

b. Raise reactor water level to +60 inches to promote natural circulation.

c. Initiate Shutdown cooling using either Unit 1 or Unit 3 RHR via the cross-tie lines.

d. Evacuate the Refuel Floor and place RHR System II in Shutdown Cooling without flushing.

QUESTION: 045 (1.00)

Unit 2 is operating at 100% power when the Unit operator observes the following alarms:

H-2 Analyzer A and B Hi
Recombiner A Inlet Temp Low
Off-Gas Holdup Vol Press High
Off-Gas Holdup Vol Temp High

Which ONE of the following describes the action to be taken?

a. Manually scram the reactor.

b. Reduce reactor power to 50%.

c. Scram the reactor if >4% H-2 is confirmed by sample.

d. Scram the reactor if OG Post-Treatment High alarm is received.

QUESTION: 046 (1.00)

Which ONE of the following describes the action required to allow cycling of FCV-74-1 [RHR pump 2A Torus suction], while Unit 2 is operating at 100% power?

a. 2-XS-74-157 must be placed in the NORMAL position.

b. 2-XS-74-157 must be placed in the SHUTDOWN position.

c. 2-XS-74-158 must be placed in the NORMAL position.

d. 2-XS-74-158 must be placed in the SHUTDOWN position.

QUESTION: 047 (1.00)

Unit 2 is operating near rated power when control rod 38-23 begins to drift out [from notch 00]. Which ONE of the following actions is the immediate response in accordance with 2-AOI-85-6?

a. Manually insert control rods following the approved sequence.

b. Reduce Recirculation Pump speed by 10% to control possible power increase.

c. Manual scram the reactor.

d. Drive the rod in using the Emergency Rod In position of the CRD NOTCH OVERIDE SWITCH.

QUESTION: 048 (1.00)

Which ONE of the following conditions would result from failure of a Reactor Recirculation Pump #1 seal assembly at rated conditions?

a. A decrease in #1 seal cavity pressure from approximately 1000 psig to about 500 psig.

b. An increase in #1 seal cavity pressure from approximately 500 psig to approximately 1000 psig.

c. An increase in #2 seal cavity pressure from approximately 500 psig to approximately 1000 psig.

d. A decrease in #2 seal cavity pressure from approximately 500 psig to approximately 0 psig.

QUESTION: 049 (1.00)

The main turbine is at 1800 rpm with the generator output breakers SHUT.

Which ONE of the following will occur if the "All Valves Closed" pushbutton is depressed?

a. All the Control Vales [TCVs and ICVs] and Main Stop Valves [MSVs] will remain OPEN.

b. The Turbine Control Valves [TCVs] and Main Stop Valves will close, but the Intercept Valves [ICVs] will remain OPEN.

c. All of the Control Valves [TCVs and ICVs] and Main Stop Valves [MSVs] will CLOSE.

d. The Control Valves [TCVs and CIVs] will CLOSE, but the Main Stop Valves [MSVs] will remain OPEN.

QUESTION: 050 (1.00)

Which ONE of the following signals from the Post-Treatment Radiation Monitoring System will initiate an Automatic isolation of the Off-Gas Discharge to the main stack [FCV-66-28]?

a. High trip in channel B.

b. High-High-High trip in Channel A

c. High trip in Channel A and Downscale trip in Channel B.

d. Downscale trip in Channel A and High-High-High trip in Channel B.

QUESTION: 051 (1.00)

Which ONE of the following describes the function of the"NORMAL/SHUTDOWN" switch on operation of the RHR pump Torus Suction Valves?

a. The Torus Suction Valves can only be operated from the Rx. MOV BD's with the switch in "SHUTDOWN".

b. The Torus Suction Valves can only be operated from the Unit 2 CR with the switch in "NORMAL".

c. The Torus Suction Valves cannot be operated from any location with the switch in "NORMAL"

d. The Torus Suction Valves cannot be operated from any location with the switch in "SHUTDOWN".

QUESTION: 052 (1.00)

Fuel loading is in progress on Unit 2. The following SRM readings were recorded before and after the loading of a bundle in Quadrant A.

	Before	After
SRM A	4 cps	5 cps
SRM B	1 cps	2 cps
SRM C	5 cps	5 cps
SRM D	3 cps	6 cps

Which ONE of the following actions describes the required action?

a. Must perform a response check of SRM C before loading more fuel.

b. Fuel loading may continue in quadrants A, C, and D.

c. Fuel loading shall be halted and a subcriticality check performed.

d. Fuel loading may continue in any quadrant.

QUESTION: 053 (1.00)

An AUO is performing an independent verification of a clearance. Which ONE of the following is the proper method for verifying a manual valve is open?

a. Check that stem position and local indicator shows valve open.

b. Turn valve hand wheel in the open direction and verify stem does not move.

c. Turn valve hand wheel in closed direction to verify movement, then fully re-open valve.

d. Count turns to close valve, then turn the same number of turns in the open direction and ensure valve opens.

QUESTION: 054 (1.00)

The boundaries of an existing CS pump clearance must be expanded to allow work on the pump's suction valve. Which ONE of the following correctly describes the method for accomplishing the above?

a. All work must stop, the existing clearance released, picked up, and a new clearance established.

b. Work could continue on the unaffected part of the clearance while the boundary change was being made, provided the SOS/ASOS and all persons holding the clearance agree that a safe clearance could be maintained during the change.

c. The person requesting the change would be responsible for notifying and obtaining authorization for the change from all persons holding the clearance and the notify the SOS/ASOS of their authorization.

d. A temporary change form [SDSP-217] must be completed and attached to the existing clearance prior to the boundary change.

QUESTION: 055 (1.00)

Which ONE of the following individuals would be allowed to make changes to Reactor Recirculation Pump speed, during power operations, without direct supervision?

a. An inactive SRO license who is currently on a 40 hour break-in to get his license active.

b. An un-licensed individual currently in training to obtain an RO liscense.

c. A unit operator assigned to the unit, and holds an active license for that unit, but is not the lead operator.

d. SRO certified instructor assigned to shift for training.

QUESTION: 056 (1.00)

Maintenance is scheduled to perform work in a High Radiation area on your shift. How is access to this area obtained?

a. Checking out the key from the SOS.

b. Checking out the key from Radcon.

c. Radcon, with the SOS permission.

d. Radcon, with the ASOS permission.

QUESTION: 057 (1.00)

During an emergency, the Site Emergency Director can authorize an increase exposure limits. What maximum exposure limit can the Site Emergency Director authorize for life saving activities?

a. 1.25 rem

b. 25 rem

c. 3 rem

d. 75 rem

QUESTION: 058 (1.00)

Which ONE of the following is the lowest event classification at which a Site Assembly MUST be performed?

a. Notification of Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

QUESTION: 059 (1.00)

During a Rx startup a single notch withdrawal reduced the reactor period to 55 seconds. Which ONE of of the following describes the action to be taken?

a. The Rx shall be shutdown until a thorough assesment has been performed.

b. Control rod[s] should be inserted to achieve a stable period > 60 seconds; the nuclear engineer and SOS should be contacted before pulling any more control rods.

c. The Rx shall be made subcritical and rod withdrawal stopped until permission is obtained from the nuclear engineer and SOS to continue.

d. No action is required unless a period of less than 30 seconds is observed.

QUESTION: 060 (1.00)

While operating at 17% rated thermal power, an EHC failure caused all of the Turbine Bypass Valves to open. When main steam line pressure drops to 785 psig, the MSIVs close causing a reactor scram. Level and pressure are controlled using HPCI.

Which ONE of the following safety limits has been exceeded for the above situation?

a. The Thermal Power Safety Limit has been exceeded.

b. The Power Transient Safety Limit has been exceeded.

c. The Reactor Vessel Water Level Safety Limit has been exceeded.

d. The Reactor Coolant System Integrity Safety Limit has been exceeded.

QUESTION: 061 (1.00)

EOI-1 was entered due to low water level, and five minutes later, while still in EOI-1, drywell pressure rises to 2.6 psig. Which statement below describes the requied SRO actions?

a. Enter EOI-2 and continue on in EOI-1.

b. Reenter EOI-1 at the beginning.

c. Reenter EOI-1 at the beginning and enter EOI-2.

d. Exit EOI-1 and enter EOI-2.

QUESTION: 062 (1.00)

Which ONE of the following describes the SLC system response to an initiation signal [Turning Key-Lock Switch SH-63-6A]?

a. Turning the key-lock switch in either direction will fire both explosive valves, start both A and B SLC pumps, and isolate RWCU.

b. Turning the key-lock switch in either direction will start only the selected SLC pump [A or B], fires the pump's associated explosive valves, and isolate RBCCW.

c. Turning the key-lock switch in either direction will fire both explosive valves, start the selected SLC pump [A or B], and isolate RWCU.

d. Turning the key-lock switch in either direction will start the selected SLC pump [A or B], and fire the associated explosive valve.

QUESTION: 063 (1.00)

Which ONE of the conditions required for ADS automatic initiation can be bypassed without any operator actions?

a. The 105 second timer.

b. High drywell pressure

c. A need to have an RHR pump or Corse Spray pumps operating.

d. A confirmatory low reactor water level [+18], and a low low Rx vessel level [-114"].

QUESTION: 064 (1.00)

Select the number of ADS valves that con be operated from the Remote Shutdown Panel.

a. 6

b. 3

c. 4

d. 2

QUESTION: 065 (1.00)

Following an auto start of SBGT, due to High Drywell pressure, how long will SBGT run?

a. SBGT will continue to run until the high drywell condition no longer exist.

b. SBGT will continue to run until manually stopped by the operator.

c. SBGT will continue to run until the high drywell condition, which is cleared and reset.

d. SBGT will continue to operate until the relative humidity heaters trip on high flow of 4000 scfm.

QUESTION: 066 (1.00)

Select the ONE RCIC Turbine Trip signal, from those listed below, that does not result in closure of the RCIC Trip Throttle Valve [FCV 71-9] in < 0.3 seconds.

a. Electrical overspeed [125%]

b. High Turbine Exhaust pressure [25psig]

c. Manual Trip

d. High Reactor Vessel Water Level [+54"]

QUESTION: 067 (1.00)

Which ONE of the following describe the Class "A" Containment Isolation valve utilized at BFN?

a. Valves on lines that connect directly with the reactor vessel or coolant piping and penetrate the primary containment.

b. Valves with a closure time based on minimizing radioactive release in case of a LOCA

c. Valves on lines that penetrate the primary containment and connect directly with containment free air space.

d. Valves on lines that penetrate the primary containment and are located outside the containment in series.

QUESTION: 068 (1.00)

Which ONE of the following methods is used to stop TIP Dry Tube [in-core] leakage when the TIP detector can not be retracted?

a. The TIP Ball Valve should be repositioned [closed].

b. The Shear Valve should be repositioned, by repositioning its key-lock switch on the TIP Panel.

c. The operator should initiate a Group 2 or Group 8 Isolation.

d. Select an alternate indexing mechanism to withdraw TIP detector and close the TIP Ball Balve.

QUESTION: 069 (1.00)

Which ONE of the following is not a characteristic of High Purity Liquid Radwaste?

a. Conductivity >100 umho/cm

b. Major sources are equipment drains.

c. Considered to be reclaimable for reuse in the plant

d. Must be sampled and analyzed by Chem Lab before transfer or release.

QUESTION: 070 (1.00)

Which ONE of the following Fuel Pool temperatures would be an indication to the Control Room Operator that a possible Fuel Pool Cooling and Clean-up System malfunction exists ?

a. A Fuel Pool temperature of 75 Deg F.

b. A Fuel Pool temperature of 90 Deg F.

c. A Fuel Pool temperature of 110 Deg F.

d. A Fuel Pool temperature of 126 Deg F.

QUESTION: 071 (1.00)

Which ONE of the following statements describes the condition[s] for which operating with one Reactor Recirculation pump in service is acceptable?

a. Single Reactor Recirculation pump operation is acceptable when one Reactor Recirculation pump trips from 100% power operation, and the tripped pump will be returned to service within 30 to 36 hours.

b. Single Reactor Recirculation pump operation should never continue for longer than a 24 hour period during any mode of operation.

c. Single Reactor Recirculation pump operation is acceptable when NO Shutdown Cooling is available and the single recirculation pump is required to prevent stratification of Reactor Vessel water.

d. Single Reactor Recirculation pump operation is acceptable for periods greater than 24 hours only when approval in granted by the Plant Manager and Plant Engineering.

QUESTION: 072 (1.00)

Unit 2 is operating at 100 % power when a small LOCA occurs. The ADS initiation logic is satisfied and the 120 second timer has started. Which ONE of the following failures would prevent the automatic initiation of ADS ?

a. Core Spray Pumps A and B fail to start.

b. Loss of Drywell Control Air.

c. RHR Pumps B and D trip after starting.

d. Loss of 250 VDC.

QUESTION: 073 (1.00)

A reactor startup is in progress on Unit 2 with the following plant conditions:

- Reactor power is below the RSCS LPSP
- Rod withdrawal sequence B is in effect
- No RWM errors or blocks exist
- The RWM was placed in BYPASS following withdrawal of all RSCS Group ~~6~~ 12 AH rods to the withdraw limit at position 36.

The operator attempts to continue withdrawing Group 6 rods to position 48. Which ONE of the following describes the plant response to this operator action ?

a. Rod withdrawal will occur with NO rod position restrictions.

b. Rod withdrawal will occur, provided all Group ~~6~~ 12 AH rods are maintained within 1 notch of each other.

c. Rod withdrawal will NOT occur beyond the RWM withdrawal limit for Group ~~6~~ 12 AH

d. Rod withdrawal will NOT occur beyond position 36, due to RSCS immediately imposing a rod block.

QUESTION: 074 (1.00)

Which ONE of the following would DEFEAT manual opening of the RHR Test Return Line Valves 74-57 and 74-59 ?

a. Reactor water level < -39 "

b. LPCI initiation

c. Minimum Flow Valves not fully shut

d. Containment Spray initiation

QUESTION: 075 (1.00)

Which ONE of the following will NOT generate a " RPS ANALOG TRIP UNIT TROUBLE " alarm ?

a. Master Trip Unit calibration in progress

b. Slave Trip Unit trip

c. RPS bus power lost

d. Slave Trip Unit card removed

QUESTION: 076 (1.00)

Which ONE of the following is correct concerning the Minimum Alternate Flooding Pressure [MAFP] used in EOI-1 C5, Level/Power Control ?

a. Once RPV pressure decreases below the MAFP, adequate core cooling is assured.

b. Once RPV pressure decreases below the MAFP, steam flow through the core does not provide adequate core cooling.

c. If there are no SRVs open and pressure remains above the MAFP, adequate core cooling is assured.

d. If pressure remains above the MAFP with at least 2 SRVs open, natural circulation assures adequate core cooling.

QUESTION: 077 (1.00)

Which ONE of the following is correct if ALL rod position indication is lost while operating at power ?

a. Obtain an OD-7 printout before and after moving any control rod.

b. Withdrawal of control rods is allowed using only single notch movement.

c. Control rod movement is allowed only by scram.

d. Control rod movement is allowed provided a Qualified Member of the Technical Staff independently verifies proper rod sequencing.

QUESTION: 078 (1.00)

Which ONE of the following is NOT correct concerning the number 4 and 5 Low Pressure Feedwater Heaters ?

a. Both the "4" and "5" heaters are horizontal U-tube heat exchangers with intergral drain cooler sections.

b. Both the "4" and "5" heaters are located inside the neck of the main condenser to minimize piping runs.

c. Both the "4" and "5" heaters are continuously in service whenever the turbine is operating.

d. Heater drains form the "3" heater cascade into the "4" heater, but no heater drains cascade into the "5" heater.

QUESTION: 079 (1.00)

Which ONE of the following describes the response of an SRM detector to a pinhole leak which causes a gradual decrease in Argon gas pressure ?

a. Gamma and neutron sensitivity would decrease.

b. Gamma sensitivity would decrease but neutron sensitivity would remain the same.

c. Gamma sensitivity would remain the same but neutron sensitivity would decrease.

d. Both gamma and neutron sensitivity would remain the same.

QUESTION: 080 (1.00)

Unit 2 is operating at 100 % power in 3-element control when one steam flow detector fails upscale. Which ONE of the following describes the plant response ?

a. Reactor water level will decrease and stabilize at a lower level.

b. Reactor water level will decrease and initiate a reactor scram.

c. Reactor water level will increase and stabilize at a higher level.

d. Reactor water level will increase and initiate a Turbine trip.

QUESTION: 081 (1.00)

Which ONE of the following will cause the greatest biological damage to man from an external source ?

a. 0.1 Rad of Fast Neutron

b. 1.0 Rem of Gamma

c. 10 Rem of Beta

d. 0.05 Rad of Alpha

QUESTION: 082 (1.00)

Which ONE of the following statements is correct concerning the number of visitors that are allowed to be assigned to a single escort in accordance with SDSP-11.11 ?

a. A person may escort no more than three visitors in a Vital Area without authorization of the Plant Manager.

b. A person may, without exception, escort a maximum of five visitors in the Protected Area.

c. A person may escort seven visitors in the Protected Area with the authorization of the Plant Manager.

d. A person may escort six visitors in a Vital Area with the authorization of the Plant Manager.

QUESTION: 083 (1.00)

Which ONE of the following is procedurally correct for verifying a Locked Throttle Valve's position when performing a valve lineup in accordance with SDSP-3.15 ?

a. An Independent Verifier may observe the initial valve operator's action when the throttle valve is operated to determine its position.

b. An Independent Verifier should check the throttle valve's position by manipulating the valve to the closed position and then reopening the valve to its proper throttled position.

c. The Initial Verifier should verify the valve's position utilizing alternate verification techniques such as observation of the valve stem, etc..

d. The Initial Verifier should check the valves's position by manipulating the valve to the Full Open position and then, restoring the valve to its proper throttled position.

QUESTION: 084 (1.00)

Which ONE of the following is NOT a Purpose of the Reactor Protection System as described in Technical Specification 3.1 ?

The Reactor Protection automatically initiates a Reactor Scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be absorbed following a loss of coolant accident.

d. Minimize the production of N-16 gammas which could be released following a loss of coolant accident.

QUESTION: 085 (1.00)

Which ONE of the following is the most desirable method of fighting a high voltage electrical fire with water ?

a. Straight stream directly on fire from at least 50 ft.

b. Wide fog pattern from at least 10 ft.

c. Fog/stream combination from as close as possible.

d. Narrow fog pattern from at least 40 ft.

QUESTION: 086 (1.00)

Which ONE of the following statements is correct concerning the RHRSW Process Radiation Monitors ?

a. RHRSW Process Radiation Monitors measure the cumulative activity release to the environment from the RHRSW system.

b. RHRSW Process Radiation Monitors are only required to be energized when in the Shutdown Cooling Mode of operation.

c. RHRSW Process Radiation Monitors measure the activity of the fluid which passes through the tubes of the RHR Heat Exchanger.

d. RHRSW Process Radiation Monitors are not sensitive enough to detect a heat exchanger tube leak which could be in excess of 10CFR20 limits. Grab samples should be taken every 4 hours when on-line.

QUESTION: 087 (1.00)

A Reactor startup is in progress on Unit 2 when the operating CRD Pump trips due to motor overload [pump shaft sheared]. The backup pump is under repair and is expected to be operable in 2 hours. The following plant conditions exist:

Rx Power 5 %
Reactor Pressure 700 psig
Charging Water Pressure 1400 psig [decreasing slowly]
1 Accumulator light on Full Core Display illuminated
1 CRD High Temperature alarm
CRD Low Pressure alarm

Which ONE of the following would be the correct action to take in accordance with 2-AOI-85-3 ?

a. A Manual Scram is required because charging water pressure cannot be restored and maintained above 1410 psig.

b. A Manual Scram is required if another Control Rod High Temperature alarm is received in conjunction with the LOW CRD WATER PRESSURE alarm.

c. A Manual Scram is required if a second Accumulator alarm is received due to low pressure.

d. A Manual Scram is required if Reactor Pressure decreases to 650 psig.

QUESTION: 088 (1.00)

Which ONE of the following is an IMMEDIATE operation action in the event ONE Reactor Recirculation pump trips while operating at 100% Reactor power?

a. Immediately refer to the power to flow map to determine if the unit is operating in a safe region; withdraw control rods to exit instability region if necessary.

b. Immediately insert control rods in a sequence determined by the S.S.

c. Immediately take actions to assure that Reactor power is below the 90% rod line.

d. Immediately place the remaining "RFC RECIRC PUMP SPEED CONTROL" in manual.

QUESTION: 089 (1.00)

Unit 2 Refuel [fuel movement] is in progress when the following alarms are noted in the Control Room:

FUEL POOL SERV FLR AREA HIGH RADIATION
REFUELING ZONE EXHAUST HIGH RADIATION
AIR PARTICULATE MONITOR HIGH RADIATION

Which ONE of the actions below is the control room operator required to perform?

a. Evacuate all personnel from the reactor building.

b. Stop all fuel handling and evacuate all non-essential personnel from the refueling floor.

c. Continue to fuel movement operations and notify RADCON to evaluate radiation levels.

d. Ensure all personnel remaining on refuel floor are wearing proper respiratory gear and notify RADCON.

QUESTION: 090 (1.00)

Which ONE of the following is the Suppression Pool temperature that should be maintained under normal operating conditions to assure continued unit operation ?

a. Maintain a temp < or = to 85 Deg F.

b. Maintain a temp < or = to 95 Deg F.

c. Maintain a temp < or = to 105 Deg F.

d. Maintain a temp < or = to 110 Deg F.

QUESTION: 091 (2.50)

MATCH each of the reactor scrams listed in Column A with the correct purpose for the scrams in Column B. [NOTE: The items in Column B may be used once, more than once, or not at all, and only a single answer may occupy one answer space.]

Column A

a. APRM Hi Flux Scram

b. MSIV Not Full Open Scram

c. Reactor Low Water Level Scram

d. Low Air Header Pressure Scram

e. Generator Load Reject Scram

Column B

1. Scrams the reactor when the core is in danger of inadequate cooling.

2. Scrams the reactor to limit the fission products released from the fuel.

3. Scrams the reactor due to the anticipation of the rapid pressure and neutron flux increase.

4. Scrams the reactor to protect the fuel cladding against high heat generation rates.

5. Scrams the reactor before rods begin drifting into the core and before the scram discharge volume is filled.

6. Scrams the reactor prior to exceeding the Reactor Coolant System Integrity Safety Limit.

QUESTION: 092 (3.00)

Match each of the systems listed in Column A with its correct automatic initiation or trip setpoint listed in Column B. [Note: The items in Column B may be used once, more than once, or not at all, and only a single answer may occupy one answer space.]

Column A	Column B
a. Core Spray automatic initiation	1. +60.0" Level
b. LPCI Mode of RHR automatic initiation	2. +54.0" Level
c. HPCI automatic initiation	3. +11.0" Level
d. RCIC Turbine trip	4. -45.0" Level
e. Recirc Pump trip	5. -51.5" Level
f. RCIC automatic initiation	6. -114.5" Level
	7. -155.5" Level

QUESTION: 093 (2.00)

For each of the Main Condenser Vacuum reading listed in Column A, choose the automatic action that should occur from Column B. [NOTE: The items in Column A may be used once, more than once, or not at all, and only a single answer may occupy one answer space.]

Column A	Column B
a. 25" Hg vacuum	1. Condensate pumps trip
b. 21.8" Hg vacuum	2. Reactor Scram
c. 7" Hg vacuum	3. Standby SJAE automatically starts
d. .8" Hg vacuum	4. Recirc pumps trip
	5. Main turbine trip
	6. RFP turbine trip and main turbine bypass valves closure occurs

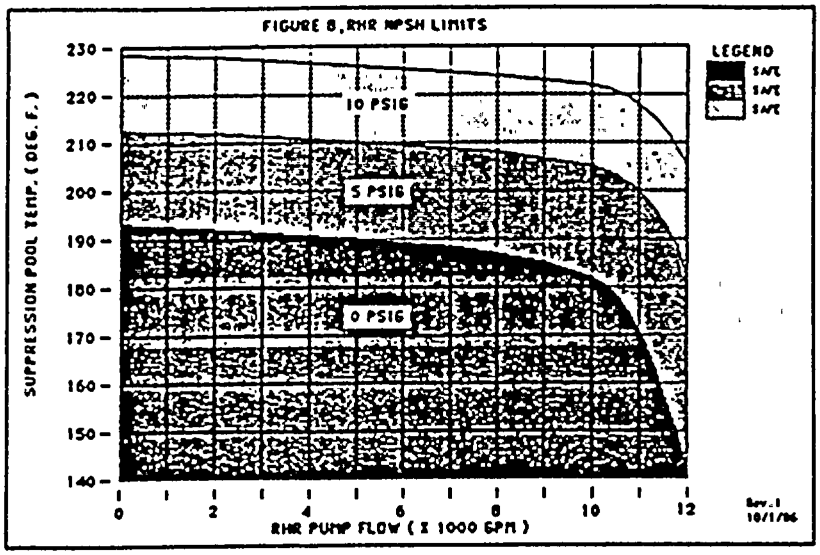
QUESTION: 094 (2.50)

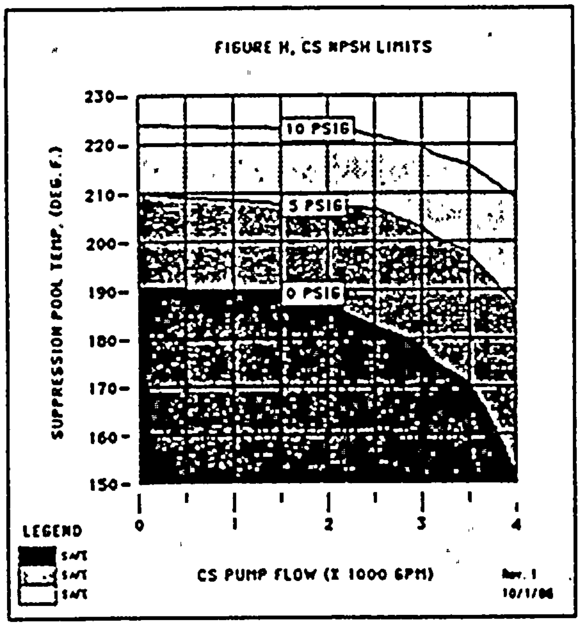
Match each of the systems in Column A with the isolations that affect it Column B.

Column A		Column B
a.	RCIC [steam supply]	1. Group 1 Isolation
b.	TIP	2. Group 2 Isolation
c.	RBCCW	3. Group 3 Isolation
d.	MSIV's	4. Group 4 Isolation
e.	RWCU	5. Group 5 Isolation
		6. Group 6 Isolation
		7. Group 7 Isolation
		8. Group 8 Isolation
		9. None

(********** END OF EXAMINATION **********)

CAUTION
MAINTAIN PUMP FLOW BELOW THE CS AND RHR NPSH LIMITS





NOTE: Curves indicate suppression chamber pressure. Below the curve for each suppression chamber pressure (0,5, and 10 psig) is the safe region for CS and RHR operation.

ANSWER KEY

001	a
002	b
003	a
004	c
005	c
006	b
007	a
008	d
009	b
010	d
011	d
012	c
013	c
014	a
015	b
016	d
017	c
018	b
019	b
020	b
021	c
022	c
023	c
024	c
025	c

ANSWER KEY

026	b
027	c
028	a
029	c
030	b
031	c
032	c
033	b
034	c
035	b
036	b
037	b
038	a
039	d
040	c
041	c
042	b
043	a
044	d
045	a
046	b
047	d
048	c
049	a
050	d

ANSWER KEY

051	c
052	b
053	c
054	b
055	c
056	c
057	d
058	c
059	b
060	b
061	c
062	c
063	b
064	c
065	b
066	d
067	a
068	b
069	a
070	d
071	c
072	d
073	b
074	b
075	b

ANSWER KEY

076 b

077 c

078 a

079 a

080 c

081 c

082 c

083 a

084 d

085 ~~d~~ b APA

086 c

087 c

088 d

089 b

090 b

091 match with selected number in the blank

a 4

b 3

c 1

d 5

e 3

092 match with selected number in the blank

a 6

b 6

c 5

ANSWER KEY

d 3

e 4

f 5

093 match with selected number in the blank

a 3

b 5

c 6

d 2

094 match with selected number in the blank

a 5

b 8

c 9

d 1

e 3

(********** END OF EXAMINATION **********)

ANSWER: 001 (1.00)

a

REFERENCE:

LESSON PLANS OPL173.917 2-AOI-85-3
[3.7/3.5 KA VALUES]

295022G010 ..(KA's)

ANSWER: 002 (1.00)

b

REFERENCE:

LESSON PLANS: OPL176.002 OPL174.717/7 OPL172.014
KA21600A211

216000A211 ..(KA's)

ANSWER: 003 (1.00)

a

REFERENCE:

LESSON PLANS: OPL176.001 OPL173.733.1 OPL173.901/2 EOI-1
[3.9/4.6 KA VALUES]

295037G012 ..(KA's)

ANSWER: 004 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.008
[3.5/3.5 KA VALUES]

259001K411 ..(KA's)

ANSWER: 005 (1.00)

c

REFERENCE:

LESSON PLANS: OPL174.830 TS 3.7.A.5.6 TS 3.7.A.6.a.1
[3.2/3.6 KA VALUES]

223001G010 ..(KA's)

ANSWER: 006 (1.00)

b

REFERENCE:

LESSONS PLANS: OPL173.901/3C
BF EOI 5214
[3.5/3.9 K/A VALUE]

295031G012 ..(KA's)

ANSWER: 007 (1.00)

a

REFERENCE:

LESSON PLANS: OPL173.901.2
[3.7/4.4 K/A VALUES]

295015G012 ..(KA's)

ANSWER: 008 (1.00)

d

REFERENCE:

LESSON PLANS: OPL173.901/3A
[3.6/4.0 KA VALUES]

295024K301 ..(KA's)

ANSWER: 009 (1.00)

b

REFERENCE:

LESSON PLANS: OP174.822.6 TS 1.1.B
[4.2/4.6 KA VALUES]

295014A205 ..(KA's)

ANSWER: 010 (1.00)

d

REFERENCE:

LESSON PLANS: OPL174.822.6
[3.5/4.3 KA VALUES]

295025G003 ..(KA's)

ANSWER: 011 (1.00)

d

REFERENCE:

LESSON PLANS: OPL173.901/2
[3.9/4.6 KA VALUES]

295037G012 ..(KA's)

ANSWER: 012 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.901/2
[3.9/4.6 KA VALUES]

295037G012 ..(KA's)

ANSWER: 013 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.901.2 EOI 2
[3.9/4.5 KA VALUES]

295024G012 ..(KA's)

ANSWER: 014 (1.00)

a

REFERENCE:

LESSON PLANS : OPL173.901/3D
[3.8/4.1 KA VALUES]

295026K301 ..(KA's)

ANSWER: 015 (1.00)

b.

REFERENCE:

LESSON PLANS: OPL173.920
[4.0/4.1 KA VALUES]

295016K202 ..(KA's)

ANSWER: 016 (1.00)

d

REFERENCE:

LESSON PLANS: OPL176.007; OPL173.828;
[3.9/3.9 KA VALUES]
295009A101 ..(KA's)

ANSWER: 017 (1.00)

c

REFERENCE:

LESSON PLANS: OP;173.901.1
[3.8/4.5 KA VALUE]

295026G012 ..(KA's)

ANSWER: 018 (1.00)

b

REFERENCE:

LESSON PLANS OPL173.904 2-AOI-70-1
[3.4/3.3 KA VALUES]

295018G010 ..(KA's)

ANSWER: 019 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.920
[4.1/4.2 KA VALUE]

295016K301 ..(KA's)

ANSWER: 020 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.901/3C
[3.5/3.9 KA VALUES]

295029K301 ..(KA's)

ANSWER: 021 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.901.2

295026G012 ..(KA's)

ANSWER: 022 (1.00)

c

REFERENCE:

LESSON PLANS: OPL174.721.11 2010090101
[3.6/3.6 KA VALUES]

202002A105 ..(KA's)

ANSWER: 023 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.004 2170050101 OPL173.915.1 OPL173.923.1
[4.3/4.3 KA VALUES]

206000K407 ..(KA's)

ANSWER: 024 (1.00)

c

REFERENCE:

LESSON PLANS: OP;176.004 OPL173.917
[3.8/3.8 KA VALUES]

201001K405 ..(KA's)

ANSWER: 025 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.004 OPL174.845/2,8 OPL174.729/3
[3.2/3.1 KA VALUES]

201002A204 ..(KA's)

ANSWER: 026 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.038 LER86-26 OPL176.002
[3.5/3.7 KA VALUES]

241000K101 ..(KA's)

ANSWER: 027 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.001 OPL174.804.5 OPL173.827
KA239002K301

239002K301 ..(KA's)

ANSWER: 028 (1.00)

a

REFERENCE:

LESSON PLANS: OPL174.705/3
[3.7/4.1 KA VALUES]

264000A209 ..(KA's)

ANSWER: 029 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.904
KA223001A1.02 [3.6/3.7 KA VALUES]

223001A102 ..(KA's)

ANSWER: 030 (1.00)

b

REFERENCE:

LESSON PLANS: OPL174.819/9.
[3.6/3.7 KA VALUES]

212000K114 ..(KA's)

ANSWER: 031 (1.00)

c

REFERENCE:

LESSON PLAN: EDGS
[4.0/4.2 KA VALUES]

264000K402 ..(KA's)

ANSWER: 032 (1.00)

c

REFERENCE:

LESSON PLANS: OPLI73.904 2-AOI-47-2
[3.6/3.5 KA VALUES]

241000G014 ..(KA's)

ANSWER: 033 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.019
KA215004K406[3.2/3.2 KA VALUES]
215004G013 ..(KA's)

ANSWER: 034 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.035.E OPL171.035.F
[3.3/3.5 KA VALUES]

209001K409 ..(KA's)

ANSWER: 035 (1.00)

b

REFERENCE:

LESSON PLANS: OPL122.916/1 OPL173.912/1
[3.2/3.2 KA VALUES]

264000K106 ..(KA's)

ANSWER: 036 (1.00)

b

REFERENCE:

LESSON PLAN: OPL171.007
[3.6/3.7 KA VALUES]

202001K101 ..(KA's)

ANSWER: 037 (1.00)

b

REFERENCE:

LESSON PLAN: OPL174.729 OPL176.006 171.024 Rev 2
[3.4/3.5 KA VALUES]

201006K404 ..(KA's)

ANSWER: 038 (1.00)

a

REFERENCE:

LESSON PLANS: OPL173.904 2-AOI-47-3
[3.5/3.6 KA VALUES]

245000A203 ..(KA's)

ANSWER: 039 (1.00)

d

REFERENCE:

LESSON PLANS: OPL174.830 TS 3.3.B.3.C 2-GOI-100-A
[4.0/3.6 KA VALUES]

201004G013 ..(KA's)

ANSWER: 040 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.004 OPL173.901.2
[3.4/3.5 KA VALUES]
295015G007 ..(KA's)

ANSWER: 041 (1.00)

c

REFERENCE:

LESSON PLANS: OPL176.004 OPL174.845/8
KA214000A2.02 [3.6/3.7 KA VALUES]

214000A202 ..(KA's)

ANSWER: 042 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.904
[3.8/3.9 KA VALUES]

239001A210 ..(KA's)

ANSWER: 043 (1.00)

a

REFERENCE:

LESSON PLANS: OPL173.904 2-GOI-100-1C
[3.7/3.8 KA VALUES]

201003A101 ..(KA's)

ANSWER: 044 (1.00)

d

REFERENCE:

LESSON PLANS: OPL173.904 2-AOI-74-1
[3.1/3.3 KA VALUES]

205000A201 ..(KA's)

ANSWER: 045 (1.00)

a

REFERENCE:

LESSON PLANS: OPL173.904 2-AOI-66-1
[3.5/3.9 KA VALUES]

271000A206 ..(KA's)

ANSWER: 046 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.903/4 171.044 LO 11
[3.6/3.5 KA VALUES]

205000A402 ..(KA's)

ANSWER: 047 (1.00)

d

REFERENCE:

LESSON PLANS: OPL173.904 2-AOI-85-6
[3.4/3.7 KA VALUES]

201003A203 ..(KA's)

ANSWER: 048 (1.00)

c

REFERENCE:

LESSON PLANS: 171.007 LO D
[3.6/3.6; 3.0/3.1; 3.3/3.3 KA VALUES]

202001K105 202001K404 202001A109 ..(KA's)

ANSWER: 049 (1.00)

a

REFERENCE:

LESSON PLANS: 171.014
[3.1/3.2; 3.0/3.2 KA VALUES]
245000K409 262001K105 ..(KA's)

ANSWER: 050 (1.00)

d

REFERENCE:

LESSON PLANS: 171.033
[3.7/4.1 KA VALUES]

272000K402 ..(KA's)

ANSWER: 051 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.928 OPL173.903
[3.6/3.5 KA VALUES]

205000A402 ..(KA's)

ANSWER: 052 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.060 EO B.5
EO B.5
[3.1/3.3 KA VALUES]

234000G013 ..(KA's)

ANSWER: 053 (1.00)

c

REFERENCE:

LESSON PLAN:
BFSP NO. 6740
[3.7/3.7]

294001K101 ..(KA's)

ANSWER: 054 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.086 PMI 12.15
BFSP: NO.4769
[3.9/4.5]

294001K102 ..(KA's)

ANSWER: 055 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.908
BFPMI: NO.5233
[3.3/4.2]

294001A109 ..(KA's)

ANSWER: 056 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.065 3430220302
BFBRQ: NO. 5267
[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 057 (1.00)

d

REFERENCE:

LESSON PLANS: OPL174.821.10 BFN-EPIP-15
BFREP: NO. 4587
[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 058 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.913 OPL171.075
BFREP: NO.5597
[2.9/4.7]

294001A116 ..(KA's)

ANSWER: 059 (1.00)

b

REFERENCE:

LESSON PLANS: OPL173.830 2-GOI-100-1
BFGOI: NO. 6736
[3.9/3.9]
295014A202 ..(KA's)

ANSWER: 060 (1.00)

b

REFERENCE:

LESSON PLANS: OPL174.822.6 TS 1.1.B
BFTS: NO.4545
[3.8/4.4]

295006G003 ..(KA's)

ANSWER: 061 (1.00)

c

REFERENCE:

LESSON PLANS: OPL173.901.1
[3.9/4.5 KA VALUES]

295024G012 ..(KA's)

ANSWER: 062 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.039
KA211000K105 [3.4/3.6 KA VALUES]

211000K105 ..(KA's)

ANSWER: 063 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.043
KA218000K403 [3.8/3.8 KA VALUES]

218000K403 ..(KA's)

ANSWER: 064 (1.00)

c

REFERENCE:

LESSON PLANS: OPL171.043
KA218000K105 [3.9/3.9]

218000K105 ..(KA's)

ANSWER: 065 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.018
KA261000K401 [3.7/3.8 KA VALUES]

261000K401 ..(KA's)

ANSWER: 066 (1.00)

d

REFERENCE:

LESSON PLANS: OPL171.040
KA217000A201 [3.8/3.7 KA VALUES]

217000A201 ..(KA's)

ANSWER: 067 (1.00)

a

REFERENCE:

LESSON PLANS: OPL171.017
KA223002G004 [3.6/3.7 KA VALUES]

223002G004 ..(KA's)

ANSWER: 068 (1.00)

b

REFERENCE:

LESSON PLANS: OPL171.02 LO2
KA215001A207 [3.4/3.7KA VALUES]

215001A207 ..(KA's)

ANSWER: 069 (1.00)

a

REFERENCE:

LESSON PLANS: OPL171.084 LO2
KA268000K502 [3.1/3.6 KA VALUES]

268000K502 ..(KA's)

ANSWER: 070 (1.00)

d

REFERENCE:

LESSON PLANS: OPL 171.052
KA233000K301 [3.2/3.4 KA VALUES]

233000K301 ..(KA's)

ANSWER: 071 (1.00)

c

REFERENCE:

LESSON PLANS: 2AOI-74-1
KA295021A102 [3.3/3.4 KA VALUES]

295021A102 ..(KA's)

ANSWER: 072 (1.00)

d

REFERENCE:

OPL 171.043
3.8/4.0 3.8/3.8 3.9/4.1

218000K501 218000K403 218000K601 ..(KA's)

ANSWER: 073 (1.00)

b

REFERENCE:

OPL 171.025
3.4/3.6 3.4/3.5 3.2/3.3

201002K105 201002K106 201002K103 ..(KA's)

ANSWER: 074 (1.00)

b

REFERENCE:

OPL171.044 EO B.11
3.4/3.6 3.5/3.7 3.2/3.4

226001K102 226001K101 226001K409 ..(KA's)

ANSWER: 075 (1.00)

b

REFERENCE:

OPL 171.077
3.4/3.3

212000A111 ..(KA's)

ANSWER: 076 (1.00)

b

REFERENCE:

EDI-1 OPL171.082
4.4/4.5 4.0/4.2 4.3/4.5

295037K209 295037K204 295037K302 ..(KA's)

ANSWER: 077 (1.00)

c

REFERENCE:

ADI-85-4
3.3/3.1 3.5/3.3

214000G014 214000G008 ..(KA's)

ANSWER: 078 (1.00)

a

REFERENCE:

OPL 171.011 EO B.17
2.8/2.9 3.4/3.4

259001K402 259001G004 ..(KA's)

ANSWER: 079 (1.00)

a

REFERENCE:

OPL 171.019 EO B.2
3.3/3.5 2.9/2.9 3.0/3.1

291002K122 215004A205 215004K604 ..(KA's)

ANSWER: 080 (1.00)

c

REFERENCE:

OPL 171.080, OPL 171.012
3.5/3.5

259002K605 ..(KA's)

ANSWER: 081 (1.00)

c

REFERENCE:

10CFR 20.5
2.8/3.4

294001K103 ..(KA's)

ANSWER: 082 (1.00)

c

REFERENCE:

SDSP-11.11
3.2/3.7

294001K105 ..(KA's)

ANSWER: 083 (1.00)

a

REFERENCE:

SDSP-3.15
3.7/3.7

294001K101 ..(KA's)

ANSWER: 084 (1.00)

d

REFERENCE:

Technical Specification 3.1 , OPL171.028
4.2/4.3

212000G004 ..(KA's)

ANSWER: 085 (1.00)

~~d~~ b [illegible]

REFERENCE:
BFN - FPP-3

3.5/3.8

294001K116 ..(KA's)

ANSWER: 086 (1.00)

c

REFERENCE:

OPL171.046 EO 11
3.3/3.4 3.4/3.7

272000A211 272000G004 ..(KA's)

ANSWER: 087 (1.00)

c

REFERENCE:

2-ADI-85-3
3.4/3.3 3.3/3.4 3.7/4.0

295022G011 201001G015 201001A201 ..(KA's)

ANSWER: 088 (1.00)

d

REFERENCE:

2-ADI-68
3.5/3.6

295001A101 ..(KA's)

ANSWER: 089 (1.00)

b

REFERENCE:

2-ADI-79-1
3.4/4.1

295023A204 ..(KA's)

ANSWER: 090 (1.00)

b

REFERENCE:

TS
3.3/4.2

295013G003 ..(KA's)

ANSWER: 091 (2.50)

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

REFERENCE:

OPL 171.028 EO B.5
4.0/4.2 3.7/3.9 3.4/3.6

212000K112 212000G007 212000K101 ..(KA's)

ANSWER: 092 (3.00)

a. 6
b. 6
c. 5
d. 3
e. 4
f. 5

REFERENCE:

*REFERENCE
2-AOI-3-1
3.9/4.0
295009K201 ..(KA's)

ANSWER: 093 (2.00)

a. 3
b. 5
c. 6
d. 2

REFERENCE:

2-AOI-47-3
3.5/3.5 3.1/3.2

295002K202 295002K201 ..(KA's)

ANSWER: 094 (2.50)

a. 5
b. 8
c. 9
d. 1
e. 3

REFERENCE:

2-AOI-64-2A 2-AIOI-64-2B TS Table 3.7A
3.8/3.9 3.9/3.8 3.7/3.5

295010G005 295031G005 295008K303 ..(KA's)

(********** END OF EXAMINATION **********)

TEST CROSS REFERENCE

QUESTION	VALUE	REFERENCE
001	1.00	9000500
002	1.00	9000510
003	1.00	9000511
004	1.00	9000513
005	1.00	9000523
006	1.00	9000488
007	1.00	9000489
008	1.00	9000490
009	1.00	9000491
010	1.00	9000492
011	1.00	9000493
012	1.00	9000494
013	1.00	9000495
014	1.00	9000496
015	1.00	9000497
016	1.00	9000498
017	1.00	9000499
018	1.00	9000501
019	1.00	9000502
020	1.00	9000503
021	1.00	9000504
022	1.00	9000505
023	1.00	9000506
024	1.00	9000507
025	1.00	9000508
026	1.00	9000509
027	1.00	9000512
028	1.00	9000514
029	1.00	9000515
030	1.00	9000516
031	1.00	9000517
032	1.00	9000519
033	1.00	9000520
034	1.00	9000521
035	1.00	9000522
036	1.00	9000524
037	1.00	9000525
038	1.00	9000526
039	1.00	9000527
040	1.00	9000528
041	1.00	9000529
042	1.00	9000530
043	1.00	9000531
044	1.00	9000532
045	1.00	9000533
046	1.00	9000534
047	1.00	9000535
048	1.00	9000536
049	1.00	9000537
050	1.00	9000538
051	1.00	9000539
052	1.00	9000540
053	1.00	9000541
054	1.00	9000543

QUESTION	VALUE	REFERENCE
055	1.00	9000544
056	1.00	9000545
057	1.00	9000546
058	1.00	9000547
059	1.00	9000548
060	1.00	9000549
061	1.00	9000550
062	1.00	9000551
063	1.00	9000552
064	1.00	9000553
065	1.00	9000554
066	1.00	9000555
067	1.00	9000556
068	1.00	9000557
069	1.00	9000558
070	1.00	9000559
071	1.00	9000560
072	1.00	9000562
073	1.00	9000563
074	1.00	9000564
075	1.00	9000565
076	1.00	9000566
077	1.00	9000567
078	1.00	9000568
079	1.00	9000569
080	1.00	9000570
081	1.00	9000736
082	1.00	9000740
083	1.00	9000741
084	1.00	9000746
085	1.00	9000747
086	1.00	9000749
087	1.00	9000752
088	1.00	9000753
089	1.00	9000754
090	1.00	9000756
091	2.50	9000561
092	3.00	9000755
093	2.00	9000757
094	2.50	9000758
	100.00	
	100.00	

ENCLOSURE 3

Facility Comment

Question: RO number 85, SRO number 76

The training received by BFN operators and reinforced by BFN procedure FPP-3 page 8 step 5.8.4 (attached), stressed a (narrow) 30 Degrees F fog (cone) pattern with a minimum distance of 10 feet for 161 kv and 15 feet for 500 kv. Additionally, conversations with site fire training instructors indicate the following limitations for a narrow fog pattern (30 Degrees F cone):

> a narrow fog (30 Deg) with a 1 1/2 hose will reach 15-20 ft.
> a narrow fog (30 Deg) with a 2 1/2 hose will reach 20-25 ft.

Based upon this information none of the selections are correct.

Resolution: Delete the question from the RO and SRO exam.

NRC Resolution of Facility Comment

Comment partially accepted: The reference material supplied by the facility does not mention the width of the fog as being narrow or 30 Degrees F. However, it does refer to a minimum distance of 10 feet as being required to utilize a fog spray on a high voltage (166 KV) bus. Based upon this information, the answer was changed to choice (b), which is correct in accordance with the information supplied from FPP-3.

ENCLOSURE 4

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee: Tennessee Valley Authority

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

Operating Tests Administered on: June 26-29, 1990

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 test, the following items were observed:

ITEM	DESCRIPTION
ICs	The initial conditions available for this examination were void of decay heat. Several ICs should be available with decay heat models that provide realistic post trip conditions which challenge the operators abilities to control containment parameters utilizing EOI-2. The lack of decay heat can result in negative training since operators may get accustomed to the uneventful conditions which follow a reactor scram.
SLC Inj. of Boron	With the reactor at 100 percent power, SLC injection with 1 pump resulted in power decreasing to < 1 percent within 15 minutes. This response is much faster than would be expected. OPL 171.039 states, "Depending on the pump injection rate and the number of control rods that failed to insert, it should take from 15 to 45 minutes to shut down the reactor". With a 100 percent ATWS (no rod movement) a more conservative time to shutdown, vice the minimum, would be required to enhance the transient fidelity.
Full Core Display	The full core display exhibits numerous ghost images making it extremely unreliable for determining a rods' position. Operators are experiencing negative training and were observed to ignore the display during ATWS events for checking rod positions.