

Nuclear Regulatory Commission  
Operator Licensing  
Examination

PERRY NUCLEAR POWER PLANT  
JUNE 21, 2010

## U.S. Nuclear Regulatory Commission

### Site-Specific RO Written Examination

#### Applicant Information

Name:

Date:

Facility/Unit:

Region: I ☐ II ☐ III ☒ IV ☐

Reactor Type: W ☐ CE ☐ BW ☐ GE ☒

Start Time:

Finish Time:

#### Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

#### Applicant Certification

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

\_\_\_\_\_

Applicant's Signature

#### Results

Examination Value 75 Points

Applicant's Score \_\_\_\_\_ Points

Applicant's Grade \_\_\_\_\_ Percent

**U.S. Nuclear Regulatory Commission**  
**Site-Specific SRO Written Examination**

**Applicant Information**

Name:

Date:

Facility/Unit: Quad Cities U1/U2

Region:      I ☐ II ☐ III ☒ IV ☐

Reactor Type: W ☐ CE ☐ BW ☐ GE ☒

Start Time:

Finish Time:

**Instructions**

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

**Applicant Certification**

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

\_\_\_\_\_

Applicant's Signature

**Results**

RO/SRO-Only/Total Examination Values   75   /   25   /  100  Points

Applicant's Scores        /        /        Points

Applicant's Grade        /        /        Percent

## Examination Rules

### General Guidelines

1. Cheating on any part of the examination will result in a denial of your application and/or action against your license.
2. If you have any questions concerning the administration of any part of the examination, do not hesitate to ask them before starting that part of the test.
3. SRO applicants will be tested at the level of responsibility of the senior licensed shift position (i.e., shift manager).
4. You must pass every part of the examination to receive a license or to continue performing license duties. Applicants for an SRO-upgrade license may require remedial training in order to continue their RO duties if the examination reveals deficiencies in the required knowledge and abilities.
5. The NRC examiner is not allowed to reveal the results of any part of the examination until they have been reviewed and approved by NRC management. Grades provided by the facility licensee are preliminary until approved by the NRC. You will be informed of the official examination results about 30 days after all the examinations are complete.

### Part B: Written Examination Guidelines

1. After you complete the examination, 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.
2. To pass the examination, you must achieve an overall grade of 80.00 percent or greater, with 70.00 percent or greater on the SRO-only items, if applicable. SRO-upgrade applicants who take the RO portion of the exam and score below 80.00 percent on that part of the exam can still pass overall, but may require remediation. Grades will not be rounded up to achieve a passing score. Every question is worth one point.
3. For an initial examination, the nominal time limit for completing the examination is 8 hours for the combined RO/SRO exam. Notify the proctor if you need more time.
4. You may bring pens, pencils, and calculators into the examination room; however, programmable memories must be erased. Use dark pencil to facilitate machine grading.
5. Print your name in the blank provided on the examination cover sheet and on the answer sheet.
6. Mark your answers on the answer sheet provided. If you are recording your answers on a machine-gradable form that offers more than four answer choices (e.g., "a" through "e"), be careful to mark the correct column. Use your examination pages for scrap paper. The examination will be retained by the facility licensee.

## Examination Rules

7. If you have any questions concerning the intent or the initial conditions of a question, do not hesitate to ask them before answering the question. Note that questions asked during the examination are taken into consideration during the grading process and when reviewing applicant appeals. Ask questions of the NRC examiner or the designated facility instructor only. A dictionary is available if you need it.

When answering a question, do not make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question. For example, you should not assume that any alarm has activated unless the question so states or the alarm is expected to activate as a result of the conditions that are stated in the question. Similarly, you should assume that no operator actions have been taken, unless the stem of the question or the answer choices specifically state otherwise. Finally, answer all questions based on actual plant operation, procedures, and references. If you believe that the answer would be different based on simulator operation or training references, you should answer the question based on the actual plant.

8. Restroom trips are permitted, but only one applicant at a time will be allowed to leave. Avoid all contact with anyone outside the examination room to eliminate even the appearance or possibility of cheating.
9. When you complete the examination, sign the statement on the examination cover sheet indicating that the work is your own and that you have neither given nor received assistance in completing the examination, and bring it together with your answer sheet to the exam proctor.
10. After turning in your examination, leave the examination area as defined by the proctor or NRC examiner (this classroom & the hall in front of this classroom). If you are found in this area while the examination is still in progress, your license may be denied or revoked.
11. Do you have any questions

**QUESTION 001**

The Plant is operating at 95% rated thermal power. Feedwater Heater 6B level switch, 1N25-N0263B, failed causing 6B Feedwater Heater to isolate.

The appropriate ONI's have been entered. However, no operator actions have been taken.

If no operator action is taken, Reactor Power will initially \_\_\_\_.

- a. rise then return to the initial value
- b. rise then stabilize at a higher value
- c. lower then return to the initial value
- d. lower then stabilize at a lower value

**QUESTION 002**

The following conditions exist:

- The plant has been shutdown for a refuel outage
- Fuel shuffle is in progress
- RHR Shutdown Cooling has been removed from service per the outage schedule for maintenance

Which of the following is a responsibility of the Reactor Operator ATC during core alterations?

- a. Monitor reactor coolant temperature
- b. Authorize commencement of fuel movements
- c. Verify required refueling surveillances are current
- d. Ensure the Control Room fuel tag board is maintained current

**QUESTION 003**

The following sequence of events occurred in rapid succession:

1. The plant was operating at 100% rated thermal power
2. An MSIV isolation occurred
3. Reactor pressure peaked at 1115 psig
4. The reactor scrammed on high drywell pressure
5. Safety Relief Valves (SRV's) cycled to relieve pressure during this event
6. No other scram signals have been actuated

Current conditions are:

- Reactor pressure is 960 psig and lowering
- 5 SRV's remain open

Based on these conditions, RPS \_\_(1)\_\_ operate as designed and the SRV's \_\_(2)\_\_ operate as designed.

	__(1)__	__(2)__
a.	did	did <u>not</u>
b.	did	did
c.	did <u>not</u>	did <u>not</u>
d.	did <u>not</u>	did

## QUESTION 004

The following conditions exist:

- A plant startup is in progress
- Reactor power is above the Low Power Setpoint
- SVI-C11-T1022, Rod Pattern Control System – Rod Withdrawal Limiter is in progress
- The Rod Withdrawal Limiter failed to inhibit withdrawal at the 4-notch limit

Which one of the following describes the required actions to be taken?

- a. Immediately suspend control rod withdrawal.
- b. Within one (1) hour take action to insert all control rods.
- c. Immediately suspend control rod movement except by scram only.
- d. Within one (1) hour take action to be in mode 2 within 7 hours.



**QUESTION 005**

The plant was operating at rated power when a scram was inserted.

An Anticipated Transient Without a Scram (ATWS) occurred resulting in the following conditions:

- Injection Systems: Terminated and Prevented IAW EOP 1A Level Power Control
- SRVs: 8 ADS valves are open IAW EOP 04-2 Emergency Depressurization
- Reactor Pressure: 200 psig and lowering
- Reactor Level: -20" and lowering
- Reactor Power: 2% and lowering

Which of the following identifies if a Technical Specification Safety Limit value has been exceeded and if adequate core cooling currently exists?

A Technical Specification Safety Limit value \_\_(1)\_\_ been exceeded and IAW EOP Bases, adequate core cooling \_\_(2)\_\_ exist.

- |    | __(1)__ | __(2)__  |
|----|---------|----------|
| a. | has     | does     |
| b. | has     | does not |
| c. | has not | does     |
| d. | has not | does not |

**QUESTION 006**

As the Field Supervisor, you must assign non-licensed operators to perform a surveillance test on the Fuel pool Cooling Cleanup system (FPCC). Per PAP-0114, Radiation Protection Program, you must maintain dose ALARA.

The following conditions exist for a job to be performed.

- The general area radiation levels are 10 mrem/hr in the room
- The hot spot in the heat exchanger room is a pipe elbow that has a radiation level of 100 mrem/hr
- The job will be performed near the hot spot area

Additional information:

- All 4 cases below have the same transition time to and from destinations
- Dose rate for all shielding placement and removal is at 100 mrem/hr
- Dose rate for the hot spot with shielding in place is 10 mrem/hr

Which method complies with ALARA procedural requirements for performance of the surveillance?

- a. The job is performed using 2 operators for 3 hrs each on the job at the hot spot.
- b. The job is performed using 3 operators for 1 hr each on the job at the hot spot and a fourth operator reading instructions in the general room area for 1 hr.
- c. The job is performed using 2 operators for 2 hrs each on the job at the hot spot and a third operator reading instructions in the general room area for 2 hrs.
- d. The placement and removal of lead shielding on the hot spot is performed by 2 Radiation Protection personnel in 1.5 hours. The job is performed after the lead shielding is in place using 2 operators for 3 hrs each on the job.

**QUESTION 007**

The plant is operating at 100% power.

Offgas Post Treatment Radiation monitors 1D17-K601A and K601B have alarmed on a High-High Radiation signal (OG ISOL OG POST-TREAT PRCS RAD A/B 3XHI – H13-P604-0001-A5).

Which one of the following describes the effect on Offgas and the Main Condenser?

- a. Only the Offgas charcoal adsorber's inlet valves will isolate causing a loss of main condenser vacuum.
- b. Offgas will shift into the Bypass Mode of operation causing a loss of main condenser vacuum.
- c. Offgas will shift to Treat Mode allowing main condenser vacuum to remain constant.
- d. Offgas System will isolate causing a loss of main condenser vacuum.

**QUESTION 008**

The following conditions exist:

- The plant scrambled on high reactor pressure
- The reactor is shutdown
- All feedwater pumps tripped
- Level control is on Reactor Core Isolation Cooling
- Pressure control is on the Bypass Valves

The assigned level band should be \_\_(1)\_\_, but can be expanded if other \_\_(2)\_\_ actions have a higher priority.

- |    | __(1)__      | __(2)__ |
|----|--------------|---------|
| a. | 150" to 219" | EOP     |
| b. | 150" to 219" | ONI     |
| c. | 178" to 219" | EOP     |
| d. | 178" to 219" | ONI     |

**QUESTION 009**

The reactor is at 96% rated thermal power with a plant down power in progress for monthly surveillance testing.

The following occurs:

- ANN PWR SUPPLY FAIL is illuminated
- Alarms that were locked in on H13-P601 have deactivated

The Control Room actions would be to dispatch an operator to \_\_ (1) \_\_ to investigate and \_\_ (2) \_\_.

	1	2
a.	D-1-A	suspended power maneuvering
b.	D-1-A	continue lowering reactor power
c.	ED-1-A	suspended power maneuvering
d.	ED-1-A	continue lowering reactor power

**QUESTION 010**

The following time-line is provided:

- At 0817 a small break LOCA was identified by control room operators.
- The Shift Manager classified and declared the event at 0829.
- The emergency communicator notified the state and local officials of the event classification and declaration at 0839.

Based on the above time-line, the NRC must be notified no later than \_\_ (1) \_\_ using the \_\_ (2) \_\_.

- |    | __ (1) __ | __ (2) __                     |
|----|-----------|-------------------------------|
| a. | 0844      | Red Phone                     |
| b. | 0854      | Dialogics System              |
| c. | 0917      | Commercial telephone system   |
| d. | 0929      | Emergency Notification System |

**QUESTION 011**

The plant was operating at 100% rated thermal power when both Reactor Recirculation Pumps down shifted.

The plant is now operating in the Immediate Exit Required region of the OPRM Operable - Two Loop Power - Flow Map.

Which of the following will raise the likelihood of power oscillations?

- a. Xenon concentration change caused by the pump down shift.
- b. Control rod insertion to exit the Immediate Exit Region.
- c. Feedwater temperature reduction as amount of extraction steam lowers
- d. Lower pressure regulator setpoint as the pressure regulator attempts to keep the plant stable.

**QUESTION 012**

A Total Loss of AC power (TLAC) is in progress.

ONI-SPI D1, Maintaining System Availability, directs that the Division 3 Battery Room door be opened within two (2) hours.

Which of the following describes the location and the specific reason given for performing this action?

- a. Control Complex 620': Dissipate Heat
- b. Control Complex 620': Prevent Hydrogen build up
- c. Control Complex 638': Dissipate Heat
- d. Control Complex 638': Prevent Hydrogen build up

**QUESTION 013**

The plant was operating at 80% rated thermal power.

Then, due to a system fault, power is lost to 125 VDC Bus ED-1-A.

Which of the following loses power and what is the immediate operator action?

- a. High Pressure Core Spray (HPCS) Logic – Verify RCIC Operable
- b. Low Pressure Core Spray (LPCS) Logic – Rack-out breaker EH1111 for LPCS pump
- c. Residual Heat Removal (RHR) C Logic – Place RHR C loop on Alternate Keep-fill
- d. Reactor Recirculation Breaker 3 Control Logic – Insert a manual reactor scram

**QUESTION 014**

With the reactor at 100% power, which of the following conditions will result in a reactor scram and a direct automatic transfer of the Recirculation Pumps from fast speed to slow speed?

- a. Main turbine trip
- b. Reactor feedwater pump trip
- c. Drywell pressure high - 1.68 psig
- d. Reactor water level high - Level 8

**QUESTION 015**

The following plant conditions exist:

- RCIC was running for testing
- RHR loop A was running in Suppression Pool Cooling mode to support RCIC testing

The following then occurred:

- The Control Room was evacuated due to toxic gas
- All Immediate Actions of ONI-C61, Evacuation of the Control Room, have been completed
- No other operator actions have been performed.

Which of the following describes the Suppression Pool Temperature instrumentation available on the Remote Shutdown Panel(s) that is(are) indicating accurate Suppression Pool Temperature, if any, at this time?

- a. No Remote Shutdown Panel Suppression Pool temperature instrumentation.
- b. Only the Division 1 Remote Shutdown Panel Suppression Pool temperature instrumentation.
- c. Only the Division 2 Remote Shutdown Panel Suppression Pool temperature instrumentation.
- d. Both the Division 1 and Division 2 Remote Shutdown Panel Suppression Pool temperature instrumentation.

**QUESTION 016**

The following plant conditions existed:

- The plant was operating at 100% rated thermal power
- NCC pumps A & B were running
- NCC pump C was tagged out for maintenance
- NCC C heat exchanger was in dry layup
- NCC B heat exchanger was removed from service due to indications of a tube leak

Then, NCC temperature control valve (NCC HX A TCV) P43-F006A began to slowly fail closed.

The Unit Supervisor entered ONI-P43, Loss Of Nuclear Closed Cooling.

The following conditions currently exist:

- Validated SPDS Drywell temperature is rising at 0.5°F/minute.
- Average Drywell temperature is 138°F
- Highest Drywell temperature is 148°F
- Suppression Pool Average temperature is 96°F rising slowly
- Suppression Pool Level is 18.1' rising slowly
- Containment pressure is 0.3 psig stable
- Drywell pressure is 0.6 psig rising slowly
- Containment Average temperature is 87°F rising slowly

Based on these conditions, the following action(s) is/are required IMMEDIATELY.

- a. Insert a manual reactor scram.
- b. Continue with ONI-P43 actions only.
- c. Suspend ONI-P43 actions and enter EOP-2 Primary Containment Control.
- d. Continue with ONI-P43 actions and enter EOP-2 Primary Containment Control.



**QUESTION 017**

The following plant conditions exist:

- The reactor is operating at 50% power
- The Service Air and Instrument Air Systems are in their normal lineup

An air leak occurs resulting in the following:

- Instrument Air receiver pressure is 85 psig and lowering
- Service Air receiver pressure is 95 psig and lowering

Which of the following describes how the Service Air/Instrument Air Cross-Connect Valves, 1P52-F050 & 2P52-F050, respond to these conditions, including the bases for this response?

The Service Air/Instrument Air Cross-Connect Valves \_\_\_\_.

- a. close to completely isolate the Service Air and Instrument Air headers
- b. close to prevent a leak in the Service Air header from impacting the Instrument Air header
- c. remain open; however they will close if Service Air receiver pressure lowers to 90 psig in order to completely isolate the Service Air and Instrument Air headers
- d. remain open; however they will close if Instrument Air receiver pressure lowers to 80 psig in order to prevent a leak in the Service Air header from impacting the Instrument Air header

**QUESTION 018**

The following conditions exist:

- It is day 20 of a refuel outage
- Fuel Pool Cooling Cleanup (FPCC) is providing decay heat removal
- Fuel shuffle is complete
- Core verification is in progress
- Div 2 electrical outage in progress
- Reactor coolant temperature is 85°F

FPCC A pump then trips on a ground fault.

Based on the above conditions, determine time to boil in the core?

**Reference Provided: PBD-A016 & PDB-A017**

- a. 5 hours
- b. 8.5 hrs
- c. 20 hrs
- d. 35 hrs

**QUESTION 019**

The following plant conditions exist:

- A full core offload is in progress

The following then occurred:

- An irradiated fuel bundle is being unloaded from the Fuel Handling Building IFTS Upender
- Bubbles are observed coming from the irradiated fuel bundle
- The Fuel Handling Building Evacuation Alarm Sounded

Based on these conditions, in order to minimize unnecessary exposure, immediate evacuation of \_\_\_\_ from the Fuel Handling Building is required.

1. Platform Operator
2. FME Coordinator
3. Fuel Handling Building Crane Operator
4. Fuel Handling Supervisor
5. Radiation protection Technician
6. Site Protection Officer
7. Spotter

- a. 1, 3, & 7
- b. 1, 4, & 6
- c. 2, 5, & 6
- d. 3, 5, & 7

**QUESTION 020**

The following plant conditions exist:

- The plant is operating at 100% power
- No equipment is out of service

A large-break LOCA occurs in the Drywell and the following conditions exist:

- Drywell pressure is at 12 psig and lowering
- Reactor pressure is at 140 psig and lowering
- Reactor water level is at 5 inches and rising

Which of the following describes the loading sequence to limit starting transients on the 4160 V Emergency Buses?

- a. RHR C starts 5 seconds from initiation signal  
HPCS starts 10 seconds from initiation signal  
LPCS starts 15 seconds from initiation signal
- b. RHR C starts 5 seconds from initiation signal  
HPCS starts 14 seconds from initiation signal  
LPCS starts 19 seconds from initiation signal
- c. RHR C starts immediately after initiation signal  
HPCS starts 10 seconds from initiation signal  
LPCS starts 15 seconds from initiation signal
- d. RHR C starts immediately after initiation signal  
HPCS starts 14 seconds from initiation signal  
LPCS starts 19 seconds from initiation signal

**QUESTION 021**

The plant was operating at rated power when a transient occurred that resulted in the following plant parameters:

- Reactor pressure: 1040 psig
- RPV level: 50"
- Drywell temperature: 185°F
- Drywell pressure: 4 psig
- Suppression Pool level: 18.5 feet
- Suppression Pool temperature: 114°F
- Containment pressure: 2.5 psig
- Containment temperature: 125°F

Which one of the following conditions will cause the margin to the Heat Capacity Limit (HCL) to improve?

- a. RPV level rises
- b. RPV pressure lowers
- c. Suppression Pool level lowers
- d. Suppression Pool temperature rises

**QUESTION 022**

The following plant conditions exist:

- Reactor power: 85%
- Suppression Pool Average temperature: 76°F

The following then occurs:

- Safety Relief valve 1B21-F047G inadvertently opens
- ONI-B21, SRV Inadvertent Opening/Stuck Open, actions were not successful in closing the SRV
- Suppression Pool Average Temperature is rising 3°F/minute
- No other operator actions are being performed

A Technical Specification requirement to immediately scram the reactor will first be met in \_\_\_\_.

- a. 7 minutes
- b. 10 minutes
- c. 12 minutes
- d. 15 minutes

**QUESTION 023**

EOP-2, Containment Control, was entered due to a scram discharge volume rupture.

Containment Spray was initiated per EOP-SPI 3.1, Containment Spray Operation.

In accordance with EOP-2, Containment Spray operation is required to be terminated \_\_\_\_.

- a. before Containment Pressure is reduced below 0.0 psig
- b. before Containment Pressure is reduced below 0.5 psig
- c. after Containment Temperature is reduced below 95°F
- d. after Containment Temperature is reduced below 185°F

**QUESTION 024**

The following conditions exist:

- EOP-2, Containment Control, has been entered
- Suppression Pool level is 17.7 feet and lowering slowly
- Suppression Pool Average temperature is 83°F
- Containment Average temperature is 96°F and stable
- Drywell Average temperature is 146°F and rising
- Drywell pressure is 0.6 psig and stable

Based on these conditions, the required EOP-2 action(s) is(are) to \_\_\_\_.

- a. Maximize Suppression Pool Cooling only
- b. Operate all available Drywell Cooling only
- c. Operate all available Drywell Cooling and  
Operate all available Containment Cooling
- d. Maximize Suppression Pool Cooling and  
Operate all available Drywell Cooling and  
Operate all available Containment Cooling

**QUESTION 025**

The following conditions exist:

- The reactor was scrammed from 25% rated power
- A low power ATWS is in progress
- Emergency Depressurization was required
- All SRVs failed to open
- RCIC suction is on the suppression pool
- RCIC is providing RPV level and pressure control
- Suppression Pool level is lowering 1 inch per minute
- Current suppression pool level is 15 feet 2 inches

Per EOP Bases, the earliest that continued operation of RCIC will be threatened due to possible RCIC equipment damage is in \_\_\_\_ minutes.

- a. 11
- b. 35
- c. 95
- d. 113



**QUESTION 026**

The following conditions exist:

- A LOCA in the drywell occurred
- Bus EH11 experienced a lockout
- High Pressure Core Spray failed to initiate
- Validated RPV water level is 25 inches and lowering
- Validated RPV pressure is 900 psig and lowering
- Drywell pressure is 1.8 psig and rising
- 1B21-N695B, RPV Level – Lvl 3, failed high
- All other systems are functioning as designed.

Which of the following describes the operation of the Automatic Depressurization System (ADS) under the current conditions?

- a. ADS SRVs can be opened by their individual key-lock switches only.
- b. ADS logic automatically initiates to open ADS SRVs after 105 second time delay relay times out
- c. ADS SRVs can be opened by arming and depressing the ADS LOGIC MANUAL INITIATION switches.
- d. ADS logic automatically initiates to open ADS SRVs 105 seconds after the RPV level set point is reached.

**QUESTION 027**

The plant was operating at 85% rated power with the following equipment out of service:

- CRD B pump tagged out due to an oil leak

A manual reactor scram was inserted.

The following conditions now exist:

- Reactor Scram Hardcard actions were completed
- Reactor power is at 5%
- Reactor pressure is at 940 psig
- The GP1A light above the 'A' manual scram switch is illuminated
- Bus EH11 locked out
- Full Core Display indicates numerous control rods withdrawn at various positions
- SCRAM VLV AIR HEADER PRESS LO annunciator (H13-P680-5A-D6) has alarmed

Which of the following methods would successfully insert control rods for the above listed conditions?

- a. EOP-SPI 1.1, PULLING SCRAM FUSES
- b. EOP-SPI 1.3, MANUAL ROD INSERTION
- c. EOP-SPI 1.5, VENTING CRD OVERPISTON VOLUMES
- d. EOP-SPI 1.6, INCREASED COOLING WATER  $\Delta P$

**QUESTION 028**

The following conditions exist:

- The plant is operating at full power
- Fuel inspections are in progress in the Fuel Handling Building
- FHB HVAC Supply Fan A (M40-C001A) is in operation
- FHB HVAC Exhaust Fans A and B (M40-C002A(B)) are in operation

An event occurs that damages numerous Spent Fuel Bundles with the following results:

- All FHB Ventilation Exhaust Airborne Radiation Monitor (D17-K710) module indications are offscale high.
- Spent Fuel Pool (D21-K332) and Fuel Prep Pool (D21-K322) Area Radiation Monitor indications are reading 12,000 mR/hr.
- The radiation release rate is at the General Emergency level
- The Unit Supervisor entered EOP-3, Secondary Containment Control and EOP-5, Radioactivity Release Control

The off-site release path is through the \_\_ (1) \_\_. The Control Room Operator would \_\_ (2) \_\_.

\_\_ (1) \_\_

\_\_ (2) \_\_

- |    |                   |  |
|----|-------------------|--|
| a. | Unit 1 Plant Vent | verify FHB HVAC Supply Fan A is tripped      |
| b. | Unit 1 Plant Vent | scram the Reactor and Emergency Depressurize |
| c. | Unit 2 Plant Vent | verify FHB HVAC Supply Fan A is tripped      |
| d. | Unit 2 Plant Vent | scram the Reactor and Emergency Depressurize |

**QUESTION 029**

The following conditions exist:

- The plant is operating at 100% power
- SAS reports that Zone 1 and Zone 2 heat detectors for Reactor Recirculation Pump A are in alarm
- The Unit Supervisor enters ONI-P54, Fire

Which one of the following actions is required for a fire in Reactor Recirculation Pump A per ONI-P54, Fire?

- a. Operator confirms the CO2 System has automatically dumped CO2
- b. Operator opens 1P54-F395, DW CO2 SUPPLY OTBD ISOL to commence CO2 dump
- c. Operator opens 1P54-F340, CTMT CO2 SUPPLY OTBD ISOL to commence CO2 dump
- d. Fire Brigade member opens 1P54-F3590, RX RCIRC PMP A SELECTOR VLV to commence CO2 dump

**QUESTION 030**

SCC informs the Control Room of a degraded grid condition and the following occurs:

1. ONI-S11, Hi/Low Voltage, is entered
2. Grid voltage and generator voltage swings are observed

What is the nominal bus voltage at which the undervoltage relays will drop out AND what is the basis for this protection?

- a. 3000 Vac (~75% nominal), protection is provided to prevent damage to running safety related equipment.
- b. 3800 Vac (~95% nominal), protection is provided to prevent damage to safety related power transformers.
- c. 3800 Vac (~95% nominal), protection is provided to prevent jeopardizing the reliability of starting safety related motors.
- d. 3000 Vac (~75% nominal), protection is provided to prevent jeopardizing the reliability of starting safety related motors

**QUESTION 031**

The plant is operating at 70% power when Inboard MSIV, B21-F022A, inadvertently closes.

- Reactor pressure rises 20 psig and stabilizes
- No RPS setpoints are exceeded
- No operator actions are taken

Which one of the following describes the response of the reactor to this event?

Reactor power initially \_\_\_\_.

- a. lowers and then stabilizes at a lower value.
- b. lowers and then returns to its original value.
- c. rises and then stabilizes at a higher value.
- d. rises and then returns to its original value.

**QUESTION 032**

The following conditions exist:

- The plant is operating at 50% power
- RFPT A is out of service for an oil leak

RFPT B tripped.

The following then occurred:

- The reactor operator inserted a reactor scram
- Reactor pressure peaked at 1045 psig
- The Motor Feed pump tripped on an electrical fault
- All other systems functioned as designed
- EOP-1, RPV Control, was entered

Based on the above conditions which of the following valves will be closed?

- |               |                                     |
|---------------|-------------------------------------|
| 1. 1B21-F019  | MSL DRN & MSIV BYP OTBD ISOL        |
| 2. 1D17-F081B | CNTMT RAD MON INBD SUCT ISOL        |
| 3. 1E12-F073B | RHR B HX'S SECOND VENT TO SUPR POOL |
| 4. 1G42-F010  | SPCU PUMP FIRST SUCT ISOL           |
| 5. 1P43-F055  | NCC CNTMT SUPPLY OTBD ISOL          |
| 6. 1P51-F150  | SA SUPPLY HDR CNTMT ISOL.           |

- a. 3, 4, & 5
- b. 2, 3, & 6
- c. 1, 5, & 6
- d. 1, 2, & 4

**QUESTION 033**

The plant is operating at 100% power when a trip of Containment Vessel Chilled Water Chiller A occurred.

- Containment temperature and pressure are slowly rising
- Drywell temperature and pressure are stable
- Alarm CONTAINMENT TEMP A HIGH H13-P601-020-F4 has been received
- Alarm CONTAINMENT TEMP B HIGH H13 P601-017-D2 has been received
- No EOP Entry Conditions exist

Which one of the following conditions will occur if Containment temperature and pressure continue to rise with no operator action taken?

- a. Indicated Containment Upper Pool level will lower.
- b. Indicated Suppression Pool level will rise.
- c. Containment Vacuum Breakers will open.
- d. Drywell Vacuum Breakers will open.

**QUESTION 034**

A plant startup is in progress after completion of a refuel outage. Plant conditions are as follows:

- Mode Switch is in STARTUP
- APRM Power is 3%
- IRMs are on Range 8

To protect the reactor from an inadvertent reactivity addition due to a control rod withdrawal accident the primary scram signal is \_\_\_\_.

- a. SRM High-High Flux
- b. IRM Neutron Flux-High
- c. APRM Neutron Flux-High
- d. APRM Neutron Flux-High Setdown

**QUESTION 035**

The following conditions exist:

- The plant is operating at rated power
- Surveillance SVI-B21-T0061-A, RPV LOW LEVEL 1 AND 2 CHANNEL A FUNCTIONAL FOR 1B21-N681A, is in progress

The I&C tech just dialed down 1B21-N681A, RPV LEVEL 1, until it tripped.

If 1B21-N681B, RPV LEVEL 1, now fails downscale, the effect would be a(n) \_\_\_\_.

- a. direct reactor scram to preserve the integrity of the fuel cladding
- b. Main Steam Line drain valve isolation only to prevent off-site dose limits from being exceeded
- c. RWCU valve isolation only to ensure peak fuel cladding temperatures remain below the limits of 10 CFR 50.46
- d. MSIV closure and subsequent reactor scram in anticipation of the complete loss of the normal heat sink and subsequent over-pressurization transient



**QUESTION 036**

The High Pressure Core Spray (HPCS) Pump, 1E22-C001, is being operated in the CST to CST Mode for testing.

During the test, actual Suppression Pool level rose to 18.5'.

Which one of the following statements describes the effect on HPCS operation?

- a. HPCS operation will be unaffected
- b. HPCS pumps the Suppression Pool to the CST
- c. HPCS Pump operates on minimum flow CST to Suppression Pool
- d. HPCS Pump operates on minimum flow with suction on the Suppression Pool

**QUESTION 037**

The following conditions exist:

- The plant is shutdown for a refuel outage
- Fuel shuffle is in progress
- Containment Vessel and Drywell Purge is operating in Refuel Mode

A fuel bundle was dropped and the following annunciators alarmed on 1H13-P680:

- CNTMT VENT EXH RAD HI
- CNTMT VENT EXH RAD A/D HI HI/INOP

Observation of radiation monitors in the Control Room reveal various monitors with elevated readings but only the Containment Ventilation Exhaust Radiation Monitors D17-K609A and D17-K609D have reached their alarm setpoints.

Based on these conditions, \_\_ (1) \_\_ Containment Vessel and Drywell Purge fans will be running and \_\_ (2) \_\_ of the Containment Vessel and Drywell Purge containment isolation valves will be closed.

	__ (1) __	__ (2) __
a.	no	all
b.	no	half
c.	all	none
d.	some	half

**QUESTION 038**

The following conditions exist:

- The plant is in MODE 4.
- RHR Loop A is running in the Shutdown Cooling Mode.
- RHR Loop B is running in the Suppression Pool Cooling Mode for surveillance testing

Which one of the following describes the automatic response of the RHR system if a valid RPV Level 1 Reactor water level condition occurs?

- a. RHR Pump A trips.  
RHR Loop B realigns to the LPCI mode.
- b. RHR Pump A continues to operate in the Shutdown Cooling Mode.  
RHR Loop B realigns to the LPCI mode.
- c. RHR Pumps A and B trip.  
RHR Loop B then shifts to the LPCI mode and RHR Pump B restarts.
- d. RHR Pump A trips.  
RHR Loop B continues to operate in the Suppression Pool Cooling mode.

**QUESTION 039**

RHR Loop A has just been placed into the Shutdown Cooling Mode of operation using the Normal Return Path.

The cooldown rate is excessive.

In accordance with SOI-E12, Residual Heat Removal System, which of the following actions will lower the cooldown rate?

- a. Throttle close the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle open the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 7000-7100 gpm.
- b. Throttle open the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle close the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 7000-7100 gpm.
- c. Throttle open the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle close the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 2575-7100 gpm.
- d. Throttle ESW flow through the RHR Heat Exchanger using RHR A HX'S ESW OUTLET VALVE, P45-F068A.

**QUESTION 040**

The following conditions exist:

- A plant startup is in progress
- Reactor power 25%
- Reactor Recirculation Flow Control Valves are full open
- APRM A is in Bypass

OPRM A receives a Core Flow signal from APRM \_\_ (1) \_\_ and if power oscillations occur, OPRM A \_\_ (2) \_\_ trip RPS.

	__ (1) __	__ (2) __
a.	A	will not
b.	E	will not
c.	A	will
d.	E	will

**QUESTION 041**

The Plant is operating at 75% rated power.

One hour ago the blue pressure permissive light for the LPCS Injection Valve, 1E21-F005, extinguished.

Control Room Operators confirmed the blue light bulb was good.

Which one of the following describes the operation of the LPCS Injection Valve control logic if a Loss-of Coolant Accident now occurs?

The LPCS Injection Valve \_\_\_\_\_.

- a. remains closed and cannot be opened with its control switch until RPV pressure lowers to 600 psig
- b. automatically opens, irrespective of RPV pressure, due to the LPCS LOCA initiation signal.
- c. remains closed and cannot automatically open until RPV pressure lowers to 530 psig
- d. remains closed and cannot automatically open until RPV pressure lowers to 600 psig

**QUESTION 042**

The following conditions exist:

- A LOCA occurred
- EOP-SPI 6.5 LPCS Injection has been performed
- Low Pressure Core Spray is injecting
- EOP-SPI 2.9 HPCS Suction Override to CST has been performed
- High Pressure Core Spray Injection Valve, 1E22F004 was overridden closed for level control
- No other injection systems or subsystems are available
- The Suppression Pool water level is approaching the SRV Tail Pipe Level Limit

To prevent the Suppression Pool water level from reaching the SRV Tail Pipe Level Limit of \_\_ (1) \_\_ feet, the required operator action is to shutdown the \_\_ (2) \_\_ pump.

- |    | __ (1) __ | __ (2) __ |
|----|-----------|-----------|
| a. | 23.0      | LPCS      |
| b. | 24.5      | HPCS      |
| c. | 24.5      | LPCS      |
| d. | 45.5      | HPCS      |

**QUESTION 043**

The following conditions exist:

- An ATWS occurred
- Standby Liquid Control (SLC) was initiated
- The SLC A Squib valve, 1C41-F004A, failed to fire
- The SLC B Pump, 1C41-C001B, failed to start
- Reactor Pressure is 1000 psig

Considering these failures, which one of the following describes the indications for SLC Pump A and B discharge pressure that would be observed on 1H13-P601.

SLC Pump A discharge pressure will be approximately \_\_(1)\_\_ and SLC Pump B discharge pressure will be approximately \_\_(2)\_\_.

- |    | __(1)__   | __(2)__   |
|----|-----------|-----------|
| a. | 1100 psig | 1100 psig |
| b. | 1100 psig | 0 psig    |
| c. | 1300 psig | 1300 psig |
| d. | 1300 psig | 0 psig    |



**QUESTION 044**

At 10:00 the following conditions exist:

- The plant is operating at 100% rated power

At 10:01 the following conditions exist:

- A load rejection with a loss of turbine EHC occurred
- Safety/Relief valves responded as designed

At 10:02 the following conditions exist:

- Motor Feedwater pump is running on minimum flow
- Reactor power is 5%
- Scram Hardcard actions have been completed

At 10:10 the following conditions exist:

- All rods are in
- Reactor pressure is being controlled on Bypass Valves
- Setpoint Setdown has been reset
- Motor Feedwater pump is running in manual, feeding the RPV
- The Unit Supervisor is operating in ONI-C71-1, Reactor Scram

At this time, the Redundant Reactivity Control System logic \_\_ (1) \_\_ be reset and RPS logic \_\_ (2) \_\_ be reset.

- |    | __ (1) __ | __ (2) __ |
|----|-----------|-----------|
| a. | can       | can       |
| b. | can not   | can       |
| c. | can       | can not   |
| d. | can not   | can not   |

**QUESTION 045**

The plant is operating at 100% power with the Reactor Protection System MG SET TRANSFER switch in ALT A.

The following occurs:

- Numerous Control Room Alarms are received
- Half scram is indicated
- Outboard BOP isolation has occurred

This is an indication of a loss of power from Bus \_\_\_\_.

- a. F1B08
- b. F1C08
- c. F1C12
- d. F1D12

**QUESTION 046**

The plant is operating at 50% rated thermal power.

RPS Bus B EPA's tripped open.

When the RPS bus is reenergized, which of the following is required?

- a. Reopen DW VAC RLF MOV ISOL VALVE, 1M16F010B
- b. Reset ½ scram caused by APRM D failing down scale
- c. Reboot ERIS computer
- d. Place IRM H on range 3

**QUESTION 047**

The following plant conditions exist:

- A reactor startup is in progress following replacement of all fuel bundles
- Reactor Protection System shorting links are removed
- Reactor power is rising with a stable positive period of 150 seconds
- SRM Channel 'A' detector is stuck and will not withdraw
- SRM Channel 'A' indication rises to  $2.5 \times 10^5$  cps
- No operator actions are performed

Which one of the following subsequently describes SRM Channel 'A' indicated reactor power and reactor period?

Indicated reactor power will \_\_(1)\_\_ and reactor period will \_\_(2)\_\_

- |    | __(1)__                           | __(2)__  |
|----|-----------------------------------|--|
| a. | lower                             | become negative  |
| b. | continue to rise, then stabilizes | stabilize at infinity ( $\infty$ ) as the reactor reaches the point of adding heat                               |
| c. | continue to rise                  | become longer as neutrons leakage increases  |
| d. | continue to rise, then drop       | remain stable and positive until reactor power reaches the Neutron Flux-High Setdown setpoint and a scram occurs |

**QUESTION 048**

The following conditions exist:

- The plant is operating at 48% rated power
- Total Core Flow is 50 MLbs/Hr
- Both Recirculation are operating in fast speed

What is the current Upscale Thermal Power Trip setpoint rounded to the nearest %?

**Reference Provided: PDB-A0012, Recirc Drive Flow vs. Total Core Flow**

- a. 107%
- b. 108%
- c. 111%
- d. 113%

**QUESTION 049**

The following conditions exist:

- The reactor scrammed due to a loss of the feedwater system
- High Pressure Core Spray pump tripped due to a seized shaft
- Reactor Core Isolation Cooling is maintaining RPV level at 30 inches
- Reactor pressure is 870 psig and lowering slowly

The receipt of which annunciator would indicate an immediate threat to RCIC's ability to maintain RPV level?

- a. RCIC ISOL DIAPHRAGM RUPTURED, (H13-P601-0021-B1)
- b. STEAM TUNNEL LD AMB TEMP P632, (H13-P601-0019-G4)
- c. RCIC TURBINE OIL COOLER OUT TEMP HIGH, (H13-P601-0021-C4)
- d. RCIC SUPR POOL SUCT VLV OPEN SUPR PL LVL HI, (H13-P601-0021-G5)

**QUESTION 050**

The following plant conditions exist:

- The reactor is operating at 100% power
- A loss of ED-1-A occurs

Which of the following describes the ability of the SRVs to function in the Pressure-Relief Mode and in the ADS Mode?

In the Pressure-Relief Mode, \_\_(1)\_\_ SRVs will function and in the ADS Mode, actuation of the Div. II ADS logic \_\_(2)\_\_ open the ADS valves.

- |    | __(1)__ | __(2)__         |
|----|---------|-----------------|
| a. | no      | will            |
| b. | no      | will <u>not</u> |
| c. | all     | will            |
| d. | all     | will <u>not</u> |

**QUESTION 051 THIS QUESTION HAD NO CORRECT ANSWER - DELETED**

An isolation signal was generated that resulted in the following light indications on panel H13-P622, Div 2 Aux Relay Panel Inboard Valves:

- NS4 MSL DRN ISOL INBD LOGIC TEST B21H-DS1B light on
- NS4 BOP ISOL INBD LOGIC TEST B21H-DS3B light on
- RX WTR SMPL VLV B33-F019 LOGIC TEST B21H-DS7B light off
- RHR ISOLATION INBD LOGIC TEST B21H-DS5B light off

Based on these indications, select the isolation that occurred.

**Reference Provided: Picture of panel H13-P622**

- a. BOP isolation
- b. RWCU isolation
- c. RHR LOCA isolation
- d. RHR Radwaste Valve isolation

**QUESTION 052**

The plant is operating at full power.

An indication that an SRV is open is \_\_ (1) \_\_.

In accordance with ONI-B21-1, SRV Inadvertent Opening/Stuck Open, do not attempt to close the SRV until reactor power is lowered to \_\_ (2) \_\_ percent.

- |    | __ (1) __   | __ (2) __ |
|----|---|-----------|
| a. | Main generator electrical output lowers               | 90        |
| b. | Main generator electrical output lowers               | 95        |
| c. | Indicated steam flow on the affected steam line rises | 90        |
| d. | Indicated steam flow on the affected steam line rises | 95        |

**QUESTION 053**

The following conditions exist:

- The plant is at 100% power
- Digital Feedwater Operator Rx Level Setpoint was set at 200 inches

A manual scram is now inserted.

Which of the following describes the response of the DFWLCS following the scram?

- Upon receipt of the scram signal, the level demand signal will be 196 inches for 10 seconds and then lower to 178 inches.
- Upon receipt of the scram signal, the level demand signal will be 200 inches for 10 seconds and then lower to 178 inches.
- When level reaches 178 inches, the level demand signal will be 196 inches for 10 seconds and then lower to 178 inches.
- When level reaches 178 inches, the level demand signal will be 200 inches for 10 seconds and then lower to 178 inches.

**QUESTION 054**

The following plant conditions exist:

- The reactor is operating at 90% power
- Annulus Exhaust Gas Treatment System (AEGTS) Train 'B' is in service
- An unplanned gaseous radioactive release occurs in the Annulus

Which one of the following Airborne Radiation Monitors (ABRM) would detect this radioactive release in the Annulus?

- Unit 1 (U1) Plant Vent, 1D17-K0780
- Unit 2 (U2) Plant Vent, 2D17-K0780
- Intermediate Bldg Vent, D17-K730
- Aux Bldg Vent, 1D17-K700

**QUESTION 055**

The plant is operating at full power with Division 1 Diesel Generator operating in parallel with the grid when the following valid alarms were received:

- BUS EH11 VOLTAGE DEGRADATION (H13-P877-0001-B1)
- BUS EH12 VOLTAGE DEGRADATION (H13-P877-0002-B1)
- BUS EH12 BUS STRIPPED UNDERVOLTAGE (H13-P877-0002-C1)

What additional condition is required to initiate the LOOP logic?

- a. 3 seconds elapses.
- b. 12 seconds elapses.
- c. Bus EH11 frequency at 58 hertz.
- d. Bus EH12 frequency at 57 hertz.

**QUESTION 056**

A fire in the Control Room forced all personnel to abandon the Control Room.

A reactor scram could not be initiated prior to evacuating the Control Room.

Which one of the following describes the preferred method for initiating a reactor scram, including the bases for this method?

Cycle the specified \_\_\_\_.

- a. ATWS UPS distribution panel breakers since this will not cause a MSIV closure
- b. ATWS UPS distribution panel breakers since this will not cause a loss of LPRMs/APRMs
- c. RPS power distribution panel breakers since this will not cause a MSIV closure
- d. RPS power distribution panel breakers since this will not cause a loss of LPRMs/APRMs



**QUESTION 057**

Which of the following is the power supply for the listed load?

	1N43-C007, Turb Emg Bearing Oil Pump	RHR B/C Logic	Main Generator Trip Logic
a.	D1A	ED1A	D1B
b.	D1B	ED1B	D1B
c.	D1A	ED1A	D1A
d.	D1B	ED1B	D1A

**QUESTION 058**

The following conditions exist:

- A Loss of Off-Site Power (LOOP) occurred
- Divisional diesel generators are supplying their respective buses
- DIESEL GENERATOR CONTROL TRANSFER switch is in CONT RM on 1H51-P055A
- DIESEL GENERATOR CONTROL TRANSFER switch is in LOCAL on 1H51-P055B

The following alarms are then received on H13-P877:

Division 1

- DIV 1 DIESEL GENERATOR TROUBLE, (H13-P877-0001-D5)
- DG TRIP\* LUBE OIL TEMP HIGH, (H13-P877-0001-C3)

Division 2

- DIV 2 DIESEL GENERATOR TROUBLE, (H13-P877-0002-D5)
- DG TRIP DIFF RELAY LOCKOUT, (H13-P877-0002-E4)

What will be the status of Buses EH11 and EH12?

	<u>EH11</u>	<u>EH12</u>
a.	Energized	Energized
b.	Energized	De-energized
c.	De-energized	Energized
d.	De-energized	De-energized

**QUESTION 059**

Which of the following will automatically trip on high temperature on a complete loss of Nuclear Closed Cooling?

- a. Control Complex Chiller B, P47-B001B
- b. Control Rod Drive Pump A, 1C11-C001A
- c. Reactor Recirculation Pump A, 1B33-C001A
- d. Unit 1 Instrument Air Compressor, 1P52-C001

**QUESTION 060**

The following conditions exist:

- The plant is operating at 75% reactor power.
- SVI-C71-T0039, MSIV Closure Channel Functional, is in progress
- MSL B INBD MSIV B21-F022B control switch is in the TEST position.

The Control Room Operator depresses the MSL B INBD MSIV TEST pushbutton 1B21H-S3B.

Which one of the following describes the response of MSL B INBD MSIV B21-F022B?

- a. Safety-Related Instrument Air bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.
- b. Instrument Air bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.
- c. Safety-Related Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.
- d. Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

**QUESTION 061**

The plant was operating at 75% rated thermal power with the following conditions:

- Bus EH12 was powered from the Alternate Preferred source
- All other electrical buses were powered from their Preferred or Normal sources

The following then occurred:

- Transformer LH-2-A experienced a lockout
- The Diesel Generators responded as designed

What is the consequence of this electrical transient?

- a. Service Water Pump C, P41C001C, will trip if running
- b. Control Rod Hydraulic Pump, 1C11C001B, can not be started
- c. Unit 2 Service Air Compressor, 2P51C001, will trip if running
- d. Nuclear Closed Cooling Pump C, P43C001C, can not be started

**QUESTION 062**

Breaker EH1104, 4.16KV TO 480V XFMR EHF-1-A TO BUS EF-1-A, tripped.

Which one of the following has lost power?

- a. Suppression Pool Cleanup Pump, 1G42-C001
- b. Standby Liquid Control Pump A, 1C41-C001A
- c. Reactor Water Cleanup Pump A, 1G33 C001A
- d. Control Complex Chill Water Pump C, P47-C001C

**QUESTION 063**

The following conditions exist:

- The plant is operating at 100% power
- Lake water temperature is 55°F
- TBCC Temperature Control Valve, 1P44-F300 is at midposition

The lake rolls over causing lake water temperature to rise to 63°F.

The rise in lake water temperature causes the temperature control valve to \_\_\_\_.

- raise the TBCC flow through the heat exchanger
- lower the TBCC flow through the heat exchanger
- raise the Service Water flow through the heat exchanger
- lower the Service Water flow through the heat exchanger

**QUESTION 064**

The plant is operating at full power.

A scram signal then occurs.

The Scram Inlet Valve, EP-126, for control rod 30-31 fails to open

Which of the following describes how control rod 30-31 is inserted?

The Control Rod Drive ball check valve repositions directing \_\_\_\_ the drive piston to insert the control rod.

- reactor water above
- reactor water below
- accumulator water above
- accumulator water below

**QUESTION 065**

Which of the following describes the operational implications of maintaining control rods within designated withdrawal distances during control rod movements above the high power setpoint?

- a. Establish a 2 notch limit to mitigate the consequences of a control rod drop accident by limiting the amount and rate of reactivity increase.
- b. Establish a 4 notch limit to mitigate the consequences of a control rod drop accident by limiting the amount and rate of reactivity increase.
- c. Establish a 2 notch limit to provide protection for a control rod withdrawal error event to preclude a MCPR safety limit violation.
- d. Establish a 4 notch limit to provide protection for a control rod withdrawal error event to preclude a MCPR safety limit violation.

**QUESTION 066**

The following conditions exist:

- Reactor is operating at rated power
- Reactor Recirculation FCV's are at 70% open
- APRM A is in Bypass
- APRM E ramped to 125%
- No operator actions were taken

The FCV's are now \_\_(1)\_\_ open .

If APRM E reading returned to normal, the FCV's will \_\_(2)\_\_ position.

	__(1)__	__(2)__
a.	30%	remain at the runback
b.	30%	return to the pre-runback
c.	60%	remain at the runback
d.	60%	return to the pre-runback

**QUESTION 067**

Which of the following parameters are monitored via installed instrumentation on the Division 1 Remote Shutdown Panel?

1. Drywell Temperature
2. Containment Pressure
3. Reactor Vessel Pressure
4. RPV Metal Temperature
5. Reactor Vessel Water Level

- a. 1, 2, & 4
- b. 1, 3, & 5
- c. 2, 3, & 4
- d. 3, 4, & 5

**QUESTION 068**

A high Drywell pressure scram occurred.

Eleven (11) minutes later, the following plant conditions exist:

- Reactor pressure is 400 psig and lowering
- Reactor water level is +12 inches and stable
- Drywell pressure is 4 psig and slowly rising
- Containment pressure is 6 psig and slowly rising
- No operator actions have been performed

Which one of the following describes the operating condition of RHR Loop 'A'?

RHR Loop 'A' is \_\_\_\_.

- a. spraying Containment
- b. operating on minimum flow; Containment Spray mode can be manually initiated
- c. injecting into the reactor vessel; Containment Spray mode can be manually initiated
- d. operating on minimum flow; Containment Sprays mode cannot be manually initiated



**QUESTION 069**

Refueling operations are in progress and the Inclined Fuel Transfer System (IFTS) is in operation.

The IFTS Carriage Assembly has just been raised to the RAISE FILL/DRAIN STOP position in the Fuel Handling Building.

The Bottom Valve and Drain Valve have closed.

Which one of the following describes what the Control Room operator will observe on H13-P601 meter, G43-R022A P601, POOL LEVEL A-SEPARATION?

The Upper Containment Pool water level will initially \_\_\_\_.

- a. lower when the IFTS Transfer Tube is filled with water; water level must be manually restored with makeup water from the Condensate Transfer and Storage System
- b. lower when the IFTS Transfer Tube is filled with water; water level is restored when water from the FPCC Surge Tanks are subsequently pumped back to the Upper Containment Pool
- c. rise due to the displacement of water by the IFTS Carriage Assembly; water level is restored when the IFTS Carriage Assembly is subsequently lowered to the Fuel Handling Building
- d. rise due to the displacement of water by the IFTS Carriage Assembly; water level is automatically restored via an automatic drain valve to the Fuel Storage Pool in the Fuel Handling Building

**QUESTION 070**

Select the statement that describes how the Low-Low Set (LLS) function of the Safety Relief Valves (SRVs) automatically operates following a main steam isolation from high power, to maintain RPV Pressure?

- a. 1B21-F051C opens at 1103 psig arming LLS which opens five LLS SRVs at 1113 psig.  
1B21F051C cycles between 1073 psig and 936 psig.
- b. 1B21-F051C and 1B21-F051D open at 1103 psig arming LLS.  
Four LLS SRVs cycle between 1113 psig and 936 psig.  
1B21F051C and 1B21-F051D cycle between 1073 psig and 926 psig.
- c. 1B21-F051D opens at 1103 psig arming LLS which opens 1B21-F051C.  
Four LLS SRVs cycle between 1113 psig and 946 psig.  
1B21-F051C and 1B21-F051D cycle between 1033 psig and 976 psig.
- d. 1B21-F051D opens at 1103 psig arming LLS which opens 1B21-F051C.  
Four LLS SRVs cycle between 1113 psig and 946 psig.  
1B21-F051C closes at 936 psig.  
1B21-F051D cycles between 1033 psig and 926 psig.

**QUESTION 071**

The following conditions exist:

- A plant startup is in progress
- The main turbine is reset

If Main Turbine speed rises from 100% to 107% of rated, which of the following describes the response of the Main Turbine Control Valves (CVs) and Intercept Valves (IVs)?

- a. The CVs and IVs start to throttle closed together
- b. The IVs throttle closed and the CVs remain open
- c. The IVs throttle closed first followed by the CVs
- d. The CVs throttle closed first followed by the IVs

**QUESTION 072**

The following conditions exist:

- The plant is operating at approximately 63% power
- RFPT B on 3 element (3E) control in AUTO
- RFPT B is operating at the suction flow limit of 23.1 Kgpm per SOI-C34, Feedwater Control System
- The Motor Feed pump is shutdown to Casing Warmup
- RFPT A is shutdown on the turning gear

The plant then entered ONI-P52, Loss Of Service And/Or Instrument Air, due to an air leak which resulted in the following:

- The source of the air leak was isolated from the Instrument Air header by closing 1P52F591, Instrument Air To Heater Bay Bldg Isol
- Instrument Air pressure in the Heater Bay is 50 psig and dropping steadily

Which one of the following describes the required operator response to the above conditions?

- a. Take RECIRC FLOW CONTROLLER MINIMUM OUTPUT, 1N27-K162B\_OP1 controller to 0% to close the RFP B RECIRC CONTROL VALVE and maintain vessel level 192 to 200 inches.
- b. Start the Motor Feed pump per SOI-C34, Feedwater Control System, and maintain vessel level 192 to 200 inches.
- c. Scram the Reactor due to lowering vessel level and enter ONI-C71-1, Reactor Scram.
- d. No action required RFPT B will recover vessel level in automatic.

**QUESTION 073**

Which of the following will cause Offgas After Filter flow to rise?

- a. Moisture in adsorber trains
- b. Excessive steam flow rate to SJAES
- c. Low level in condenser seal troughs
- d. Prefilter Line Drain Loop Seal Level Low

**QUESTION 074**

The following conditions exist:

- The plant is shutdown for a refueling outage
- M25/26 Control Room Ventilation train A is running in NORMAL mode
- M25/26 Control Room Ventilation train B is in STANDBY

A loss of input from Control Room Atmosphere Gas Module D17-K776 to the M25/26 logic has been experienced due to an electrical switching error on buses K1A and K1N.

What is the current status of the Control Room HVAC system?

	<u>Train A</u>	<u>Train B</u>
a.	running in NORMAL mode	remains in STANDBY
b.	running in NORMAL mode	running in EMERGENCY RECIRC mode
c.	running in EMERGENCY RECIRC mode	remains in STANDBY
d.	running in EMERGENCY RECIRC mode	running in EMERGENCY RECIRC mode

**QUESTION 075**

The following conditions exist:

- The reactor was shutdown due to electrical problems
- The Plant Computer lost power
- Cooldown is in progress
- SVI-B21-T1176, RCS HEATUP AND COOLDOWN SURVEILLANCE is in progress
- Preparation in progress to place Shutdown Cooling in service

While lowering reactor pressure to place Shutdown Cooling in service the following reactor pressure readings were obtained from recorder 1B21-R622, REACTOR LEVEL & PRESS, at the indicated times:

<u>Time</u>	<u>Rx pressure</u>
1200	1000 psig
1300	425 psig
1400	100 psig
1500	25 psig

Which one of the following choices completes the following statement?

The reactor cooldown rate specified in SVI-B21-T1176 and Technical Specification 3.4.11, RCS Pressure and Temperature (P/T) Limits \_\_\_\_.

**Reference provided – Steam Tables**

- has not been exceeded
- was exceeded between 1200 and 1300
- was exceeded between 1300 and 1400
- was exceeded between 1400 and 1500

**QUESTION 076**

A staff Senior Reactor Operator had an Active and Valid license on December 31, 2009.

He then stood the following 12-hour shifts:

- Jan 12, 2010 Shift Engineer
- Jan 31, 2010 ATC
- Feb 21, 2010 ATC
- Feb 22, 2010 BOP
- Feb 23, 2010 ATC
- Mar 23, 2010 Field Supervisor
- Apr 3, 2010 Unit Supervisor

No additional shifts were stood as of April 30, 2010.

As of May 1, 2010 the staff SRO \_\_\_\_\_.

- a. maintained proficiency as SRO
- b. maintained proficiency as RO only
- c. maintained proficiency as Shift Engineer
- d. did not maintain proficiency as a licensed operator

**QUESTION 077**

Refuel outage is in progress with the following conditions:

- Reactor cavity water level is at 22 feet above the RPV flange
- Only the thrice burnt fuel bundles have been off-loaded
- Control rod blade replacement is in progress
- Control Rod 30-31 position indication channel 2 is inoperable

Based on the above conditions, Technical Specifications require the Refueling SRO to \_\_ (1) \_\_.  
The basis for this action is \_\_ (2) \_\_.

	__ (1) __	__ (2) __
a.	suspend movement of <u>irradiated</u> fuel assemblies within the RPV	Insufficient water necessary to retain halogen (iodine) fission product activity in the water in the event of a fuel handling accident.
b.	suspend movement of <u>irradiated</u> fuel assemblies within the RPV	A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment.
c.	suspend movement of <u>new</u> fuel assemblies and handling of control rod blades within the RPV	A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment.
d.	suspend movement of <u>new</u> fuel assemblies and handling of control rod blades within the RPV	Insufficient water necessary to retain halogen (iodine) fission product activity in the water in the event of a fuel handling accident.

**QUESTION 078**

The plant is operating at full power.

- A problem develops with the Residual Heat Removal 'A' pump and the Unit Supervisor declares RHR A INOPERABLE at 0500 on June 1
- On June 3, a fire occurs in breaker EH1208 for RHR 'B' pump and RHR 'B' is declared INOPERABLE at 1700
- RHR 'A' is returned to OPERABLE status three hours later at 2000

If RHR 'B' cannot be returned to OPERABLE status, what is the latest time that the plant must be in MODE 3?

**Reference provided: Tech Spec 3.5.1**

- a. June 8, 0500
- b. June 9, 0500
- c. June 9, 1700
- d. June 10, 1700



**QUESTION 079**

The following plant conditions exist:

- Reactor Mode Switch in SHUTDOWN
- Reactor Coolant Temperature is 250°F
- Unit 2 Div. 1 battery (2R42-S002) is INOPERABLE
- Unit 1 Div.1 battery (1R42-S002) is on float
- RCIC System is in Secured Status

While performing weekly battery voltage surveillance, Electrical Maintenance discovered that the Unit 1 Division 1 battery (1R42-S002) terminal voltage to be 105 VDC.

As the Unit Supervisor, select the statement that describes the actions required.

**Reference provided: Technical Specifications 3.8.4 & 3.8.5**

- a. Restore the Unit 1, Div. 1 battery to OPERABLE within 2 hours or be in COLD SHUTDOWN within the following 24 hours
- b. Restore either Div. 1 battery to OPERABLE within 2 hours or be in COLD SHUTDOWN within the following 36 hours
- c. Restore both Div.1 batteries to OPERABLE within 2 hours or be in Cold Shutdown within 36 hours
- d. Declare affected required feature(s) INOPERABLE, immediately

**QUESTION 080**

The following plant conditions exist:

- A plant transient results in fuel damage
- The Shift Manager acting as the Emergency Coordinator declared a Site Area Emergency 18 minutes ago
- It is necessary for an operator to enter the Containment to terminate the offsite radioactive release
- Radiation Protection estimates the expected exposure for this task is 3 Rem
- The operator volunteering for the task has 500 mrem TEDE for the year

Describe the occupational dose impact and the required approval(s) for this task?

- a. Only a Site Administrative Control Level (ACL) limit will be exceeded;  
Only Shift Manager approval is required.
- b. A Site Administrative Control Level (ACL) limit and the Federal Dose Limit will be exceeded;  
Only Shift Manager approval is required.
- c. Only a Site Administrative Control Level (ACL) limit will be exceeded;  
Radiation Protection Manager and the Radiation Protection Technician's approvals are required.
- d. A Site Administrative Control Level (ACL) limit and the Federal Dose Limit will be exceeded;  
Plant Manager and the Radiation Protection Manager's approvals are required.

**QUESTION 081**

The plant is operating at rated power.

At 0700 on 3/1/2010, while reviewing the Tech Spec Rounds, the Unit Supervisor discovers the following:

- The last time that a Channel Check was performed on Control Room Airborne radiation monitor D17-K776 was 24 hours ago

Per Technical Specifications, D17-K776 is \_\_ (1) \_\_. The surveillance is required to be performed no later than \_\_ (2) \_\_.

**Reference provided: Tech Spec 3.3.7.1**

	__ (1) __	__ (2) __
a.	INOPERABLE	1300 on 3/2/2010
b.	OPERABLE	1300 on 3/2/2010
c.	INOPERABLE	0700 on 3/2/2010
d.	OPERABLE	0700 on 3/2/2010

**QUESTION 082**

The following conditions exist:

- An Alert has been declared
- The TSC is not yet activated

Which of the below listed duties are allowed to be delegated by the Shift Manager?

1. Provide information and assistance to the Public Information Organization.
  2. Determine the emergency classification including reclassification or termination.
  3. Recommend protective actions for the general public to State and local County Officials.
  4. Coordinate and direct the actions necessary to terminate or mitigate the effects of the emergency.
  5. Request mobilization of the Corporate Planning Assistance Center (CPAC) in accordance with NOBP-LP-5001
  6. Provide an interface with FirstEnergy Corporation organizational management and senior levels of outside organizations.
- 
- a. 1,2,3,
  - b. 2,3,4
  - c. 3,4,5
  - d. 4,5,6

**QUESTION 083**

The following conditions exist:

- The plant is shutdown for a refuel outage
- Core Alterations are in progress
- High Pressure Core Spray Pump is out of service for a motor change out
- Division 1 ECCS pumps are out of service for LLRT testing
- Buses EH11, EH12, & EH13 are aligned to LH-1-A
- The Perry Transmission Yard East Bus (Bus # 1) is de-energized for maintenance

A lockout occurs on Unit 1 Startup transformer

Ten (10) minutes after the Unit 1 transformer lockout the following conditions exist:

- Division 1 Diesel Generator is operating supplying Bus EH11
- Division 2 Diesel Generator failed to start
- Division 3 Diesel Generator is operating supplying Bus EH13

As the Unit Supervisor, which abnormal procedure will you enter to mitigate the consequences of the conditions above?

- a. ONI-R10, LOSS OF AC POWER - SBO
- b. ONI-R10, LOSS OF AC POWER - TLAC
- c. ONI-R10, LOSS OF AC POWER - LOOP
- d. ONI-R22, Loss OF AN ESSENTIAL AND/OR A STUB 4.16KV BUS

**QUESTION 084**

Given the following conditions:

- D-1-A Voltage 125 VDC
- D-1-B Voltage 100 VDC
- ED-1-A Voltage 100 VDC
- ED-1-B Voltage 0 VDC
- ED-1-C Voltage 90 VDC
- Reactor Coolant Temperature 180°F
- All control rods are inserted

Entry into Off-normal instruction(s) \_\_ (1) \_\_ is(are) required and, the Emergency Plan classification will be \_\_ (2) \_\_.

**Reference Provided: EPI-A1 Attachments 1 and 2**

	(1)	(2)
a.	ONI-R42-2, LOSS OF DC BUS ED-1-B and ONI-R42-3, LOSS OF DC BUS ED-1-C	ES-1
b.	ONI-R42-2, LOSS OF DC BUS ED-1-B and ONI-R42-3, LOSS OF DC BUS ED-1-C	EU-1
c.	ONI-R42-2, LOSS OF DC BUS ED-1-B	ES-1
d.	ONI-R42-2, LOSS OF DC BUS ED-1-B	EU-1

**QUESTION 085**

Surveillance SVI-C71-T0046, TURBINE STOP VALVE CLOSURE AND TURBINE CONTROL VALVE FAST CLOSURE CHANNEL FUNCTIONAL FOR 1C71-N006A, B, C, D, E, F, G, H AND 1C71-N005A, B, C, D commenced at 12:00 on June 2.

The Unit Supervisor signed the SVI for channel INOPERABILITY at 14:00 on June 2

At 16:00 on June 2, Turbine Stop Valve #3 failed to initiate a half scram signal to RPS when tested.

The Technical Specification Required Actions are to place channel in trip by \_\_(1)\_\_ on June 3. The bases of the Turbine Stop Valve closure scram is to ensure the \_\_(2)\_\_.

**Reference Provided: Technical Specification 3.3.1.1 – partial – modified**

	__(1)__	__(2)__
a.	04:00	fuel peak cladding temperature remains below the limits of 10CRF50.46
b.	08:00	fuel peak cladding temperature remains below the limits of 10CRF50.46
c.	04:00	Minimum Critical Power Ratio (MCPR) Safety Limit is not exceeded
d.	08:00	Minimum Critical Power Ratio (MCPR) Safety Limit is not exceeded

**QUESTION 086**

Which one of the following identifies a condition that would require declaration of an ALERT only?

**Reference Provided: EPI-A1 Attachments 1 and 2**

- a.     RPV level is lowering slowly  
Fifty Control Rods are withdrawn at various positions  
Reactor Power is 10%  
Mode Switch is in SHUTDOWN
- b.     RPV level at 175 inches and lowering  
Control Rods 30-31, 18-55, 14-15, 46-15, 46-31, & 46-47 are at position 48  
Nuclear Instruments are fully inserted  
IRM's are indicating on range 4
- c.     Mode Switch is placed in SHUTDOWN  
Control Rod 46-15 is at position 48  
Nuclear Instruments are fully inserted  
Power is indicating middle of the source range
- d.     RPV level at 179 inches and lowering  
Mode Switch is placed in SHUTDOWN  
All Control Rods indicate 00  
Nuclear Instruments are fully inserted  
Power is indicating middle of the source range



**QUESTION 087**

Refueling activities are in progress.

The following conditions exist on the Refuel Floor:

- A fuel bundle just arrived from the Fuel Handling Building with the IFTS Upender vertical
- A fuel bundle is in transit from the RPV to the fuel storage racks via F15 Bridge
- An unexplained drop in upper pool level occurs

Which of the following actions is required concerning the status of the two bundles?

- a. Incline the IFTS Upender and return the fuel bundle on the F15 Bridge back to any open vessel location.
- b. Incline the IFTS Upender and continue the fuel movement with the F15 Bridge to the fuel storage racks.
- c. Continue fuel movement with the F15 Bridge to the fuel storage racks, then, transfer the fuel bundle in IFTS to the fuel storage racks.
- d. Transfer the fuel bundle in IFTS down to the Fuel Handling Building and return the fuel bundle on the F15 Bridge back to the vessel location from which it was removed.

**QUESTION 088**

The following conditions exist:

- SVI-E51-T2001, RCIC Pump and Valve Operability test is in progress following maintenance
- SVI-D23-T1213, Suppression Pool Average Temperature is being performed by an I&C tech

Based on the information from the attached SPDS screen print, select the required action.

**Reference Provided: SPDS screen print and Technical Specification 3.6.2.1**

- a. Trip the RCIC turbine  
Restore Suppression Pool Average Temperature to  $\leq 95^{\circ}\text{F}$  within 24 hours  
Verify Suppression Pool Average Temperature  $\leq 110^{\circ}\text{F}$  once per 30 minutes
- b. Trip the RCIC turbine  
Verify Suppression Pool Average Temperature  $\leq 110^{\circ}\text{F}$  once per hour  
Restore Suppression Pool Average Temperature to  $\leq 95^{\circ}\text{F}$  within 24 hours
- c. Initiate Suppression Pool Cooling  
Restore Suppression Pool Average Temperature to  $\leq 95^{\circ}\text{F}$  within 24 hours  
Verify Suppression Pool Average Temperature  $\leq 120^{\circ}\text{F}$  once per 30 minutes
- d. Initiate Suppression Pool Cooling  
Reduce reactor power to  $\leq 1\%$  RTP in 12 hours  
Verify Suppression Pool Average Temperature  $\leq 110^{\circ}\text{F}$  once per hour

**QUESTION 089**

While the Unit Supervisor was reviewing Technical Specification Rounds he discovered the following:

- Containment Average Air Temp 78°F
- Suppression Pool validated level 17.2 feet
- Half of the suppression pool temperature detectors on H13P868 and H13-P869 read 79°F
- Half of the suppression pool temperature detectors on H13P868 and H13-P869 read 85°F

Based on the readings from Tech Spec Rounds, select the correct statement below.

- a. All Suppression Pool temperature instruments are OPERABLE
- b. The Suppression Pool temperature instruments reading 79°F are OPERABLE
- c. The Suppression Pool temperature instruments reading 85°F are OPERABLE
- d. None of the Suppression Pool temperature instruments are OPERABLE

**QUESTION 090**

The Reactor is operating at 75% power when the following alarms occur:

- DRYWELL PRESS A HIGH
- DRYWELL PRESS B HIGH
- DRYWELL AIR COOLERS DRAIN FLOW HI
- DW UNIDENTIFIED RATE OF CHANGE HIGH

The operator reports:

- DW Floor Drain Sump Level/Rate Recorder 1E31R618 is pegged high at 10 GPM.
- Drywell pressure is 0.6 psig

A / An \_\_(1)\_\_ would be the cause of all of these alarms and require the Emergency Coordinator to declare an \_\_(2)\_\_.

**Reference Provided: EPI-A1, Emergency Action Levels, Attachments 1 & 2**

__(1)__	__(2)__
a. RPV Condensing Pot leak	Unusual Event
b. RPV Condensing Pot leak	Alert
c. Upper Drywell Cooler tube leak	Unusual Event
d. Upper Drywell Cooler tube leak	Alert

**QUESTION 091**

What is the Basis for the Drywell Air Temperature Limiting condition for Operation?

In the event of a Design Basis Accident (DBA), initial average drywell air temperature is assumed to be less than or equal to \_\_\_\_.

- a. 135°F to ensure that peak LOCA drywell temperature does not exceed the maximum design temperature of 300°F during a feedwater line break accident
- b. 135°F to ensure that peak LOCA drywell temperature does not exceed the maximum allowable temperature of 330°F during a small steam line break accident
- c. 145°F to ensure that peak LOCA drywell temperature does not exceed the maximum design temperature of 300°F during a feedwater line break accident
- d. 145°F to ensure that peak LOCA drywell temperature does not exceed the maximum allowable temperature of 330°F during a small steam line break accident

**QUESTION 092**

The following plant conditions exist:

- Reactor water level is -10 inches
- Reactor pressure is 900 psig
- Drywell pressure is + 1.1 psig
- Drywell temperature is 130°F
- Containment pressure is +1.5 psig.
- Containment temperature is 100°F
- Suppression Pool temperature is 116°F
- Suppression Pool level is 24.8 feet
- All appropriate EOP's have been entered

Which one of the following actions is the Unit Supervisor required to direct, including the bases for this action?

- a. Anticipate emergency depressurization IAW EOP-1; the RPV cannot be permitted to remain at pressure if operation of SRVs may cause the SRV discharge line or associated components to fail.
- b. Anticipate emergency depressurization IAW EOP-1; if primary containment water level rises above the elevation of the SRV solenoids, the SRVs may no longer be operable.
- c. Emergency depressurize IAW EOP-4-2; the RPV cannot be permitted to remain at pressure if operation of SRVs may cause the SRV discharge line or associated components to fail.
- d. Emergency depressurize IAW EOP-4-2; if primary containment water level rises above the elevation of the SRV solenoids, the SRVs may no longer be operable.

**QUESTION 093**

The following conditions exist:

- First startup after refueling outage in progress
- The IRM Linearity Check was performed

The table below shows the IRM linearity data after final adjustments by I&C.

IRM LINEARITY TABLE				
IRM	Range 6 (40 to 75)	Range 6 Divided By 3.1623	Range 7	Sat / Unsat $\pm 3/125$
A	65	20.5547	21	S / U
E	58	18.3411	19	S / U
C	49	15.4951	19	S / U
G	55	17.3924	14	S / U
B	45	14.2301	13	S / U
F	65	20.5547	22	S / U
D	50	15.8113	16	S / U
H	51	16.1275	20	S / U

Based on the above data, which of the following actions is required?

**Reference provided: Technical Specification 3.3.1.1 – Partial (Modified)**

- Within 1 hour, insert a half scram.
- Within 6 hours, either place one IRM channel in trip or insert a half scram.
- Within 12 hours, place the plant in MODE 3.
- Within 12 hours, either place one IRM channel in trip or insert a half scram.

**QUESTION 094**

The following conditions exist:

- The plant is operating at full power
- Work was performed on the 1E51-F031, RCIC PUMP SUPR PL SUCT ISOL
- 1E51-F031 is now down powered
- RCIC is in standby with suction aligned to the Condensate Storage Tank

A Technical Specification function of RCIC is to \_\_ (1) \_\_.

RCIC is \_\_ (2) \_\_.

**Reference Provided: PMT data**

__ (1) __		__ (2) __
a.	maintain coolant inventory, as well as vessel level, if a small break occurs in the RPV while the RCS is still pressurized	OPERABLE
b.	maintain coolant inventory, as well as vessel level, if a small break occurs in the RPV while the RCS is still pressurized	INOPERABLE
c.	operate either automatically or manually following a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level	OPERABLE
d.	operate either automatically or manually following a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level	INOPERABLE



**QUESTION 095**

The following conditions exist:

- The plant is at 100% power
- I&C is performing surveillance SVI-B21-T0072-C, MAIN STEAM LINE LOW PRESSURE CHANNEL C CALIBRATION FOR 1B21-N076C
- The Unit Supervisor signed for Channel INOPERABILITY at 10:17 May 2<sup>nd</sup>
- All other MSL Low Pressure instruments are OPERABLE
- At 12:17 on May 2<sup>nd</sup>, the technician reports to you that he is unable to adjust the 1B21-N076C instrument within the allowable value.

Based on this information, the isolation function for Main Steam Line Pressure Low \_\_ (1) \_\_ maintained. The Required Action(s) is to \_\_ (2) \_\_.

**Reference Provided: Technical Specification 3.3.6.1 (partial – modified) and Plant Data Book Tab I - partial**

- |    | __(1)__ | __(2)__   |
|----|---------|---|
| a. | is      | place channel in trip by 10:17 on May 3 <sup>rd</sup> |
| b. | is      | place channel in trip by 12:17 on May 3 <sup>rd</sup> |
| c. | is not  | be in Mode 2 by 17:17 on May 2 <sup>nd</sup>          |
| d. | is not  | be in Mode 2 by 19:17 on May 2 <sup>nd</sup>          |

**QUESTION 096**

The following conditions exist:

- The reactor scrammed due to a small-break LOCA
- The only available injection source is from the Condensate Transfer system
- To maximize injection, Emergency Depressurization was initiated approximately 20 minutes ago and all ADS SRVs were verified open
- RPV level is -10 inches and rising slowly
- The SRV OPEN annunciator just reset

You have directed the Reactor Operator to verify the status of the ADS SRVs.

The ADS SRVs are reported to be \_\_(1)\_\_. You would direct the panel operators to \_\_(2)\_\_.

\_\_(1)\_\_

\_\_(2)\_\_

- |    |  |   |
|----|--|---|
| a. | closed based on stable SRV tailpipe temperatures   | open additional SRVs IAW EOP 4-2 Emergency Depressurization                                       |
| b. | closed based on SRV tailpipe temperatures slowly lowering  | perform EOP-SPI 8.1, RPV VENTING USING MAIN STEAM LINES   |
| c. | open based on SRV tailpipe temperatures of approximately 250°F and stable. Injection is occurring  | restore and maintain RPV level greater than 16.5" IAW EOP 4-1 ALTERNATE LEVEL CONTROL             |
| d. | open based on SRV tailpipe temperatures of approximately 330°F and slowly rising due to decay heat | open additional SRVs IAW EOP 4-2 Emergency Depressurization and continue to monitor for injection |

**QUESTION 097**

Given the following:

- The plant is operating at 75% rated power
- On June 4<sup>th</sup> at 12:15, AEGT A fan failed and was declared INOPERABLE
- On June 10<sup>th</sup> at 08:15, Bus EH12 experienced a bus lockout
- On June 10<sup>th</sup> at 08:15, Annulus  $\Delta P$  is reported to be 0 inches H<sub>2</sub>O

Per Technical Specifications the plant is required to be in Mode 3 no later than \_\_\_\_.

**Reference Provided: Technical Specification 3.6.4.1 & 3.6.4.3**

- a. June 10 at 15:15 and in Mode 4 no later than June 10 at 21:15
- b. June 10 at 21:15 and in Mode 4 no later than June 11 at 21:15
- c. June 11 at 00:15 and in Mode 4 no later than June 12 at 00:15
- d. June 12 at 00:15 and in Mode 4 no later than June 13 at 00:15

**QUESTION 098**

The following conditions exist:

- The plant is operating at 100% power with a high load line
- FMEOD (Fraction of Maximum Extended Operating Domain) at 0.985.

During a planned power reduction to 80% using the Recirc Flow Control Valves, the SCC Dispatcher requests Perry to hold power at 90%.

A short time later, the Reactor Operator reports that FMEOD is 0.992.

What is(are) the Unit Supervisor's required action(s)?

- a. No actions required since FMEOD is  $< 1.000$
- b. Per Reactor Engineering direction insert control rods IAW SOI-C11(RCIS), Rod Control And Information System
- c. Enter ONI-C51, Unplanned Change in Reactor Power or Reactivity and insert Cram Rods IAW FTI-B002, Control Rod Movements
- d. Enter ONI-C51, Unplanned Change in Reactor Power or Reactivity and inserting control rods per ONI direction

**QUESTION 099**

The plant was operating in the Nominal Feedwater Temperatures Back-up Stability Protection Regions – Two Loop Power - Flow Map with the following conditions:

- Reactor power is 98% with coast-down in progress
- Load line is 95.8%

The following events occurred in sequence:

1. An APRM circuit malfunction caused both Reactor Recirculation Flow Control Valves to run back.
2. Both FCV's ran back at 8% per second
3. The reactor operator locked up both FCV's.
4. After both FCV's were locked up, the A Reactor Recirculation Pump tripped.

The following stable conditions currently exist:

- Reactor Power is 45%
- Core flow is 38 Mlbm/hr

Based on the above conditions, per Technical specification 3.4.2, Flow Control Valves Bases, both FCV's are \_\_(1)\_\_. The Unit Supervisor would direct \_\_(2)\_\_.

	<u>__(1)__</u>	<u>__(2)__</u>
a.	OPERABLE	entry into ONI-C71-1, Reactor Scram
b.	OPERABLE	insertion of CRAM Rods; per FTI-B0002, Control Rod Movements
c.	INOPERABLE	entry into ONI-C71-1, Reactor Scram
d.	INOPERABLE	insertion of CRAM Rods; per FTI-B0002, Control Rod Movements

**QUESTION 100**

The following conditions exist:

- The reactor is shutdown
- Control Rod Drive Hydraulic system is shutdown

Which of the following conditions require system manipulations to be directly supervised by the Shift Manager or Unit Supervisor per OAI-0201, Operations General Instructions and Operating Practices?

- a. Adjusting FPCC Upper Pool flow while in MODE 3
- b. Placing a RWCU Filter/Demin in service while in MODE 4
- c. Shifting NCC heat exchangers from 'A' to 'C' while in MODE 4
- d. Placing RHR B loop in Suppression Pool Cooling with RHR A loop is in Shutdown Cooling while in MODE 3

## ANSWER KEY

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001	b.	021	b.	041	d.	061	d.	081	d.
002	a.	022	c.	042	b.	062	b.	082	d.
003	c.	023	a.	043	a.	063	a.	083	b.
004	a.	024	c.	044	b.	064	b.	084	d.
005	a.	025	c.	045	b.	065	c.	085	c.
006	b.	026	c.	046	d.	066	b.	086	b.
007	d.	027	c.	047	a.	067	b.	087	b.
008	c.	028	a.	048	c.	068	b.	088	b.
009	a.	029	c.	049	a.	069	b.	089	c.
010	d.	030	d.	050	c.	070	d.	090	a.
011	c.	031	c.	051	deleted	071	d.	091	d.
012	a.	032	b.	052	a.	072	c.	092	c.
013	d.	033	d.	053	d.	073	c.	093	d.
014	a.	034	b.	054	b.	074	d.	094	d.
015	c.	035	d.	055	c.	075	c.	095	b.
016	d.	036	d.	056	a.	076	d.	096	c.
017	b.	037	b.	057	d.	077	a.	097	b.
018	d.	038	a.	058	b.	078	c.	098	b.
019	c.	039	b.	059	d.	079	b.	099	a.
020	c.	040	d.	060	d.	080	a.	100	d.

**QUESTION 001**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.1.43	
	Importance Rating	4.1	
<p>K&amp;A: Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.</p>			
<b>Generic</b>			
<p>Explanation: <b>Answer B</b> – FW Heater 6B isolation will cause a rise in Rx power due to the loss of FW heating. Rx power will stabilize at a higher value. For this reason, the immediate action of ONI-N36 is to lower Rx power.</p> <p>A – incorrect – Rx power will not return to the initial value without operator intervention</p> <p>C – incorrect – Rx power will rise due to a loss of FW heating not lower</p> <p>D – incorrect – Rx power will rise due to a loss of FW heating not lower</p>			
Technical Reference(s): ONI-N36 rev 4 & GFE chapter 5 rev p		Reference Attached: ONI-N36 p 4 & GFE Chapter 5 p 15	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-N36_25_26-J.4			
Question Source:	Bank # Modified Bank # New x		
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41 x 55.43		
Comments: Level of Difficulty = x			



**QUESTION 002**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.1.44	
	Importance Rating	3.9	
<p>K&amp;A: Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.</p>			
<b>Generic</b>			
<p>Explanation: <b>Answer A</b> – IOI-9, Refueling, requires moderator temperature to be maintained greater than 68 F whether fuel is irradiated or new. Tech Spec Rounds (TS 3.9.8 C.2) monitors for this condition when no RHR loop is in operation.</p> <p>B and D – incorrect – this is the responsibility of the Unit Supervisor.</p> <p>C – incorrect – this is the responsibility of the Refueling Supervisor and the Fuel Handling Supervisor</p>			
Technical Reference(s): TS 3.9.8 C.2 & IOI-9 rev 26		Reference Attached: TS 3.9.8 C.2 p 3.9.11 & IOI-9 p 7	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-12(LP)-E			
Question Source:	Bank # Modified Bank # New	Perry 2009	
Question History:	Previous NRC Exam: Perry 2009		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 003**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.2.37	
	Importance Rating	3.6	
K&A: Ability to determine operability and/or availability of safety related equipment.			
<b>Generic</b>			
<p>Explanation: <b>Answer C</b> – RPS did not scram on High Rx Pressure and did not scram on MSIV closure. On an MSIV isolation from 100% power, more than 2 of the SRV's will initially open as the Relief pressure is reached. Low-Low Set will lower the Relief setpoints of 2 SRV's, but 2 SRV's are not enough to relieve the Rx pressure. At current RPV pressure, 6 SRV's should be open, not 5. SRV's did not operate as designed.</p> <p>A &amp; B – incorrect – RPS did not operate as designed (two functions)</p> <p>B &amp; D – incorrect – at this pressure, 6 SRV's should be open</p>			
Technical Reference(s): ONI-B21-1 rev 9 and ARI-H13-P680-05-A8 rev 12, & ARI-H13-P680-05-A6 rev 12		Reference Attached: ONI-B21-1 p. 11 and ARI-H13-P680-05-A8 p12, & ARI-H13-P680-05-A6 p 19	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C71-P.E & OT-COMBINED-B21_N11-H			
Question Source:	Bank # Modified Bank # New	Peach Bottom 2009	
Question History:	Previous NRC Exam: Peach Bottom 2009		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41            x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 004**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.2.39	
	Importance Rating	3.9	
K&A: Knowledge of less than or equal to one hour Technical Specification action statements for systems.			
<b>Generic</b>			
<p>Explanation: <b>Answer A</b> – per TS 3.3.2.1 Condition A, this is the correct action.</p> <p>B – incorrect – this is the required action for more than 2 SRM's INOP in modes 3 or 4 - this is not a required action for an INOP RWL</p> <p>C – incorrect – this is the required action for an INOP rod pattern controller</p> <p>D – incorrect – this is the required action when the 1-hour action has not been met for TS 3.3.4.2 &amp; 3.3.6.1 - not a required action for INOP RWL</p>			
Technical Reference(s): TS 3.3.2.1 & svo-C11-T1022 rev 11		Reference Attached: TS 3.3.2.1 p 3.3-15 & SVI-C11-T1022 p 2	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C11_RCIS-J.5			
Question Source:	Bank # Modified Bank # New	INL-0971	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 005**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.2.42	
	Importance Rating	3.9	
K&A: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.			
<b>Generic</b>			
<p>Explanation: <b>Answer A</b> – The Reactor Vessel Water Level SL (2.1.1.3) is applicable in all Modes. RPV water level must be maintained greater than or equal to zero inches. Greater than 0” is also considered Adequate Core Cooling. EOP’s direct lowering RPV water level, but the SL is still violated. Per EOP Bases, Adequate Core Cooling is also achieved by maintaining RPV pressure &gt; Minimum Steam Cooling pressure. With 8 SRVs open, this would be any pressure higher than 140 psig. The Safety Limit is violated but Adequate Core Cooling currently exists.</p> <p>B &amp; D – incorrect – Adequate Core Cooling currently exists</p> <p>C &amp; D – incorrect – the Reactor Vessel Water Level SL has been exceeded</p>			
Technical Reference(s): Tech Spec 2.1, TS 2.1.1.3 Bases rev 4 & EOP-1A Bases rev 1		Reference Attached: Tech Spec 2.1 p 2.0-1, TS 2.1.1.3 Bases p B 2.0-4 & EOP-1A Bases pp. 55-56	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3037-08 B			
Question Source:	Bank # Modified Bank # New	Hatch 2009	
Question History:	Previous NRC Exam: Hatch 2009		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 006**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.3.13	
	Importance Rating	3.4	
<p>K&amp;A: Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.</p>			
<b>Generic</b>			
<p>Explanation: <b>Answer B</b> – &gt; The job is performed by 3 operators for 1 hr each on the job at the hot spot and a fourth operator reading instructions in the general room area for 1hr. (3 operators x 100 mrem/hr x 1 hr) + (1 operator x mrem/hr x 1 hr) = 310 mrem</p> <p>A – incorrect - The job is performed using 2 operators for 3 hrs each on the job at the hot spot - (2 operators x 100 mrem/hr x 3 hrs) = 600 mrem</p> <p>C – incorrect - The job is performed by 2 operators for 2 hrs each on the job at the hot spot and a third Operator reading instructions in the general room area for 2 hrs - (2 operators x 100 mrem/hr x 2 hr) + (1 operators X 10 mrem/hr) = 420 mrem</p> <p>D – incorrect - Two RP personnel hang and remove lead shielding on the hot spot in 1.5 hours - The job is performed after the lead shielding is in place using 2 operators for 3 hrs each on the job. (2 RP techs x 100 mrem/hr x 1.5hrs) + (2 operators x 10 mrem/hr x 3 hr) = 360 mrem</p>			
Technical Reference(s): PAP-0114 rev 15		Reference Attached: PAP-0114 p 15	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): None			
Question Source:	Bank # Modified Bank # New	Hope Creek 2009	
Question History:	Previous NRC Exam: Hope Creek 2009		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 007**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.3.15	
	Importance Rating	2.9	
K&A: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.			
<b>Generic</b>			
<p>Explanation: <b>Answer D</b> – a Hi-Hi signal from both rad monitors causes the Offgas system to isolate. When OG isolates, main condenser vacuum will degrade.</p> <p>A – incorrect – OG Discharge Isolation, Cooler Condenser, Prefilter Inlet, and Holdup Line drain valves all isolate</p> <p>B – incorrect – OG will not auto shift to Bypass Mode</p> <p>C – incorrect – OG should be shifted from Auto Mode to Treat Mode at 5% power</p>			
Technical Reference(s): ARI-H13-P604-001-A5 rev 5		Reference Attached: ARI-H13-P604-001-A5 p 13	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-N64-M.6			
Question Source:	Bank # Modified Bank # New	River Bend 2008	
Question History:	Previous NRC Exam: River Bend 2008		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 008**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.4.14	
	Importance Rating	3.8	
K&A: Knowledge of general guidelines for EOP usage.			
<b>Generic</b>			
<p>Explanation: <b>Answer C</b> – both EOP-1 and ONI-C71-1 direct a level band of 178-219". Only the EOP allows expanding the level band (General Guidelines). Scram on high reactor pressure is an EOP entry.</p> <p>A – incorrect – this is the expected level band when pressure control is on SRV's.</p> <p>B – incorrect – ONI's do not allow expansion of the level band, they direct entry into the EOP's.</p> <p>D – incorrect – correct level band, but the ONI's do not allow expansion of the level band.</p>			
Technical Reference(s): EOP-1 Bases rev 1 & EOP-1 Chart rev B.		Reference Attached: EOP-1 Bases p 27 & EOP-1 Chart (partial)	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): x			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 009**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.4.32	
	Importance Rating	3.6	
K&A: Knowledge of operator response to loss of all annunciators.			
<b>Generic</b>			
<p>Explanation: <b>Answer A</b> – a loss of D-1-A-06 will cause ANN PWR SUPPLY FAIL window to illuminate and other annunciator windows to deactivate. Operator actions are to maintain the plant stable – suspend power maneuvering.</p> <p>B &amp; D – incorrect –the action to continuing to lower power is wrong</p> <p>C &amp; D – incorrect – ED-1-A does supply optical isolators which could cause the annunciator windows on H13-P601 to deactivate, but it will not cause ANN PWR SUPPLY FAIL window to illuminate</p>			
Technical Reference(s): ARI-H13-P680-007-E5 rev 17 & ONI-R61 rev 3		Reference Attached: ARI-H13-P680-007-E5 p 137 & ONI-R61 p 4	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-05(LP)-A.5			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			



**QUESTION        010**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.4.43	
	Importance Rating	3.2	
K&A: Knowledge of emergency communications systems and techniques.			
<b>Generic</b>			
<p>Explanation: <b>Answer D</b> – per EPI-B1, Emergency Notification System, Section 5.4.5, the NRC is contacted within 1 hour of classification via the ENS.10 CFR 50.72(a)(3)</p> <p>A – incorrect – The 15 minute notification is after classification &amp; declaration</p> <p>B – incorrect – The 15 minute notification after classification is for the State and Local Counties</p> <p>C – incorrect – The time is 1 hour after the event, not the notification, the notification system is not the ENS</p>			
Technical Reference(s): EPI-B1 rev 20, 10 CFR 50.72(a)(3)		Reference Attached: EPI-B1 p 15	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3039-01.B			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis        x		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 011**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295001	AK1.04
	Importance Rating	2.5	
<p>K&amp;A: Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Limiting cycle oscillation: Plant-Specific</p>			
<b>Partial or Complete Loss of Forced Core Flow Circulation / 1 &amp; 4</b>			
<p>Explanation: <b>Answer C</b> – lowering feedwater temperature can cause increase in reactor power and increase the likelihood of power oscillations. For this reason, RPV level is lowered during an ATWS.</p> <p>A – incorrect – xenon build-up will cause Rx power to lower which lowers the possibility of power oscillations</p> <p>B – incorrect – inserting control rods to lower power lowers the possibility of power oscillations</p> <p>D – incorrect – reducing the pressure regulator setpoint will not have any effect on power oscillations</p>			
Technical Reference(s): EOP Bases rev 0		Reference Attached: EOP Bases p 6	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): x			
Question Source:	Bank # Modified Bank # New	INL-1312	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 012**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295003	AK2.01
	Importance Rating	3.2	
K&A: Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: Station batteries			
<b>Partial or Complete Loss of AC / 6</b>			
<p>Explanation: <b>Answer A</b> – per ONI-SPI- D-1, battery room doors are opened to dissipate heat.</p> <p>B &amp; D – incorrect – hydrogen buildup is a concern when designing battery room ventilation, but it is not the reason given in ONI-SPI D1 for opening the door within two hours</p> <p>C &amp; D – incorrect – Control Complex 638' has Div 1 &amp; 2 battery rooms, but not the Div 3 battery room</p>			
Technical Reference(s): ONI-SPI-D-1 rev 0 & ONI-R10 flowchart rev B		Reference Attached: ONI-SPI-D-1 p 3 & ONI-R10 flowchart	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-18(LP) A.4			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION 013**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295004	AK3.03
	Importance Rating	3.1	
K&A: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Reactor SCRAM:			
<b>Partial or Total Loss of DC Pwr / 6</b>			
<p>Explanation: <b>Answer D</b> – ED-1-A supplies control logic for Recirc Breakers 3A &amp; 3B. When power is lost, Recirc breakers 5A &amp; 5B trip open. Per ONI-C51, with no Recirc pumps running, place Mode Switch in Shutdown.</p> <p>A – incorrect - HPCS logic is powered from ED1C</p> <p>B – incorrect - LPCS logic is powered from ED1A, however, there are no required operator actions to rack-out the LPCS pump breaker</p> <p>C – incorrect - RHR C logic is powered from ED1B</p>			
Technical Reference(s): ONI-R42-1 rev 7 & ONI-C51 rev 24		Reference Attached: ONI-R42-1 p 4 & ONI-C51 p 6	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-R42-F & OT-3035-05(LP)-A.1			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION 014**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295005	AA1.01
	Importance Rating	3.1	
K&A: Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Recirculation system:			
<b>Main Turbine Generator Trip / 3</b>			
<p>Explanation: <b>Answer A</b> – A main turbine trip from &gt;38% power will initiate a reactor scram and EOC-RPT logic will initiate a downshift of Recirc pumps. This is based on the MT Stop valve position (direct)</p> <p>B – incorrect – RFPT trip will cause a FCV runback when RPV hits L4</p> <p>C – incorrect – DW pressure high will cause a Rx scram, but not a direct down shift of Recirc pumps – the subsequent lowering of feedwater flow will cause a RR Pump downshift after a time delay</p> <p>D – incorrect – RPV water level high will cause a Rx scram, but not a direct down shift of Recirc pumps – the subsequent lowering of feedwater flow will cause a RR Pump downshift after a time delay</p>			
Technical Reference(s): ONI-N32 rev 9 & ARI-H13-P680-004-A3 rev 14		Reference Attached: ONI-N32 p 4 & ARI-H13-P680-004-A3 p 9	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-B33-E.3			
Question Source:	Bank # Modified Bank # New	INL-1294	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 015**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295016	AA2.04
	Importance Rating		
K&A: Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Suppression pool temperature			
<b>Control Room Abandonment / 7</b>			
<p>Explanation: <b>Answer C</b> – Division 2 Remote Shutdown Panel instruments do not have transfer switches and are normally energized.</p> <p>A – incorrect – Div 2 panel instrumentation is providing accurate information</p> <p>B – incorrect – Div 1 panel instruments must be transferred to energize the instruments, this action has not been done yet</p> <p>D – incorrect – same as B</p>			
Technical Reference(s): SDM-C61 rev 6		Reference Attached: SDM-C61 p 10	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-13(LP)-A.5			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 016**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295018	2.4.8
	Importance Rating	3.8	
K&A: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.			
<b>Partial or Total Loss of CCW / 8</b>			
<p>Explanation: <b>Answer D</b> – Entry into EOP-2 is required based on Suppression Pool temperature. EOPs are the higher tier documents, but ONI actions should be taken if they don not interfere with EOP's.</p> <p>A – incorrect – this action is only appropriate if <u>all</u> NCC flow was lost after lowering flow to 58 mlbm/hr</p> <p>B – incorrect –entry condition for EOP-2 is met</p> <p>C – incorrect – it is inappropriate to suspend ONI-P43 actions since entry conditions for ONI-P43 still exist</p>			
Technical Reference(s): EOP Bases rev 0 & EOP-2 Bases rev 0		Reference Attached: EOP Bases p 27 & EOP-2 Bases pp. 6 & 15	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-16(LP)-A.3			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 017**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295019	AA1.04
	Importance Rating	3.3	
K&A: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Service air isolations valves:			
<b>Partial or Total Loss of Inst. Air / 8</b>			
<p>Explanation: <b>Answer B</b> – the cross-connect valves will close when Instrument Air receiver pressure is &lt;90 psig to protect the Instrument Air system from a leak in the Service Air system.</p> <p>A – incorrect – check valves around the P52-F050 valves allow Service Air to continue to supply Instrument Air when the F050 valves are closed</p> <p>C – incorrect – P52-F050 valves are closed - Service air can still supply instrument air header. Therefore, they are not completely isolated from each other</p> <p>D – P52-F050 valves are closed. There are no automatic actions at 80 psig in the IA receiver.</p>			
Technical Reference(s): SOI-P51/52 rev 25 & Lesson Plan OT-COMBINED-P51_P52 rev 1		Reference Attached: SOI-P51/52 p 4 & Lesson Plan OT-COMBINED-P51_P52 p 14	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-P51_P52-E.7			
Question Source:	Bank # Modified Bank # New	Perry 2002	
Question History:	Previous NRC Exam: Perry 2002		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			



**QUESTION 018**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295021	AK2.05
	Importance Rating	2.7	
K&A: Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: Fuel pool cooling and cleanup system			
<b>Loss of Shutdown Cooling / 4</b>			
<p>Explanation: <b>Answer D</b> – FPCC is the alternate decay heat removal (SDC) system. The decay heat load is 20 MBTU/hr based on curves for decay after 248 bundles discharged and water level at 23' above the RPV flange, the time to boil is approximately 35 hours. With Div 2 outage in progress, B FPCC pump is not available.</p> <p>A – incorrect – plausible if use time to boil curve at vessel flange and decay heat curve prior to bundle discharge</p> <p>B – incorrect – plausible if use time to boil curve at vessel flange and decay heat curve after bundle discharge</p> <p>C – incorrect – plausible if use time to boil curve at 23' above flange and decay heat curve prior to bundle discharge</p>			
Technical Reference(s): PBD-A016 rev 9 & PDB-A017 rev 10		Reference Attached: PBD-A016 pp. 3 & 5 & PDB-A017 pp 9 & 11	
Proposed references to be provided to applicants during examination: PBD-A016 & PDB-A017			
Learning Objective (As available): OT-3035-11(LP)-A.1			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION 019**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295023	AK3.01
	Importance Rating	3.6	
K&A: Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS: Refueling floor evacuation			
<b>Refueling Acc / 8</b>			
<p>Explanation: <b>Answer C</b> – Per ONI-J11-2 Necessary Personnel are defined as those personnel necessary to place the equipment or fuel in a ‘safe condition’. SOI-F11, fuel handling Platform, identifies personnel required for fuel handling in the FHB. At Perry, ‘necessary personnel’ are the FH Supervisor, the Platform Operator, and the Spotter – all other personnel are to be evacuated.</p> <p>A – incorrect – bridge driver and spotter are necessary personnel</p> <p>B – incorrect – bridge driver and FH Supervisor are necessary personnel</p> <p>D – incorrect – the spotter is considered necessary personnel</p>			
Technical Reference(s): ONI-J11-2 rev 13 and SOI-F11 rev 10		Reference Attached: ONI-J11-2 pp. 5-7 and SOI-F11 p 5	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-14(LP)-A.5			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 020**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295024	EA1.10
	Importance Rating	3.4	
K&A: Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: A.C. distribution			
<b>High Drywell Pressure / 5</b>			
<p>Explanation: <b>Answer C</b> – a high DW pressure signal will initiate HPCS, LPCS, &amp; RHR. Per Tech Spec Bases, pump starting is sequenced due to the high starting currents to control emergency bus and transformer loading.</p> <p>A &amp; B – incorrect – RHR C starts immediately.</p> <p>B &amp; D – incorrect – HPCS starts 10 seconds from initiation. Nineteen seconds for LPCS and 14 seconds for HPCS is plausible because on a loss of bus power the respective pump, start relays are energized for 19 &amp; 14 seconds upon restoration of power.</p>			
Technical Reference(s): TS 3.3.5.1 Bases rev 7, SDM-E12 rev 9, SDM-E21 rev 10, & SDM-E22 rev 7		Reference Attached: TS 3.3.5.1 Bases p B3.3-97, SDM-E12 p 37, SDM-E21 p 21, & SDM-E22 p 22	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): x			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION 021**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295025	EA2.03
	Importance Rating	3.9	
K&A: Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression pool temperature			
<b>High Reactor Pressure / 3</b>			
<p>Explanation: <b>Answer B</b> – lowering reactor pressure will move farther away from HCL.</p> <p>A – incorrect – plausible due to misconception of effect of RPV level on HCL. Raising RPV level will have no effect on HCL margin</p> <p>C – incorrect – lowering suppression pool level will reduce margin to HCL</p> <p>D – incorrect – raising suppression pool temperature lowers margin to HCL</p>			
Technical Reference(s): EOP Bases rev 0		Reference Attached: EOP bases p 63	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3402-06-C.1			
Question Source:	Bank # Modified Bank # New	Peach Bottom 2008	
Question History:	Previous NRC Exam: Peach Bottom 2008		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 022**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295026	2.2.22
	Importance Rating	4	
K&A: Knowledge of limiting conditions for operations and safety limits.			
<b>Suppression Pool High Water Temp. / 5</b>			
<p>Explanation: <b>Answer C</b> – with a heat-up rate of 3 deg/min, suppression pool temperature will exceed 110°F in just over 11 minutes. Per tech specs, a scram is required if suppression pool temp exceeds 110°F.</p> <p>A – incorrect – this is the time to exceed 95°F, which is the requirement to enter TS 3.6.2.1 with no testing in progress adding heat to suppression pool</p> <p>B – incorrect – this is the time to exceed 105°F, which is the requirement to enter TS 3.6.2.1 with testing in progress adding heat to suppression pool</p> <p>D – incorrect – this is the time to exceed 120°F, which is requirement to depressurize the reactor</p>			
Technical Reference(s): Tech Spec 3.6.2.1		Reference Attached: Tech Spec 3.6.2.1 pp. 3.6-36 - 38	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3037-10.B			
Question Source:	Bank # Modified Bank # New	Oyster Creek 2008	
Question History:	Previous NRC Exam: Oyster Creek 2008		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 023**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295027	EK1.03
	Importance Rating	3.8	
<p>K&amp;A: Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): Containment integrity</p>			
<b>High Containment Temperature / 5</b>			
<p>Explanation: <b>Answer A</b> – Containment Spray must be terminated prior to reaching 0 psig in containment in order to maintain a margin to the negative design pressure of containment.</p> <p>B – incorrect – 0.5 psig is a 'rule of thumb' when securing from containment spray operation</p> <p>C – incorrect – 95°F is the entry condition for containment temperature leg, where containment sprays can be started</p> <p>D – incorrect – 185°F is the containment design limit for containment temperature for ED</p>			
Technical Reference(s): EOP-2 Bases rev 0		Reference Attached: EOP-2 Bases p 10	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3402-07-C.2			
Question Source:	Bank # Modified Bank # New	Perry 2007-1	
Question History:	Previous NRC Exam: Perry 2007-1		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION 024**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295028	EK2.04
	Importance Rating	3.6	
K&A: Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell ventilation			
<b>High Drywell Temperature / 5</b>			
<p>Explanation: <b>Answer C</b> – Containment temperature and DW temperature exceeds the value for operating all containment and DW cooling. Per the EOP bases, the determination that DW temp can not be maintained can be made prior to exceeding the value.</p> <p>A – incorrect – SP temperature is 13 deg F from the limit</p> <p>B – incorrect – this is only partially correct, containment temperature exceeds the value to operate cooling</p> <p>D – incorrect – SP temperature is 13 deg F from the limit and is rising slowly</p>			
Technical Reference(s): EOP Bases rev 0 & EOP-2 Bases rev 0		Reference Attached: EOP Bases p 36 & EOP-2 Bases pp 17 and 50	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3402-8.C			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 025**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295030	EK3.03
	Importance Rating	3.6	
K&A: Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: RCIC operation			
<b>Low Suppression Pool Wtr Lvl / 5</b>			
<p>Explanation: <b>Answer C</b> – with suppression pool level lowering at 1"/minute, it will take 95 minutes to reach 7.25'. At this level, pump damage from operation below the vortex limit becomes a concern.</p> <p>A – incorrect – This is the time to uncover the SRV tailpipes</p> <p>B – incorrect – this is the time to uncover the horizontal vents</p> <p>D – incorrect –this is the time when damage to RHR and LPCS may occur</p>			
Technical Reference(s): EOP Bases rev 0		Reference Attached: EOP Bases p 48	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3402-1-b.2			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			



**QUESTION 026**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295031	EA1.06
	Importance Rating	4.4	
K&A: Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Automatic depressurization system			
<b>Reactor Low Water Level / 2</b>			
<p>Explanation: <b>Answer C</b> – the Arm &amp; Depress switches will work anytime the respective LPCI/LPCS pump is providing sufficient discharge pressure.</p> <p>A – incorrect – the manual SRV switches are not the <u>only</u> way to open the valves</p> <p>B &amp; D – incorrect – the ADS B system will not automatically initiate with the Level 3 instrument failed high and ADS A will not automatically initiate with no LPCI/LPCS pump running</p>			
Technical Reference(s): PDB-I-005 rev 9		Reference Attached: PDB-I005 pp.13-14	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-B21C E.1			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 027**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295037	EA2.05
	Importance Rating	4.2	
<p>K&amp;A: Ability to determine and/or interpret the following as they apply to SCRAM          CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR          UNKNOWN: Control rod position</p>			
<b>SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1</b>			
<p>Explanation: <b>Answer C</b> – CRD B pump is tagged and CRD A pump is not running due to EH11 bus lockout. Therefore, venting the over-piston volumes is the only method that would work.</p> <p>A – incorrect – RPS did not fail to de-energize, therefore, this method would not be successful</p> <p>B &amp; D – incorrect – a CRD pump must be running to use these sections</p>			
Technical Reference(s): EOP-SPI 1.1 rev 0, EOP-SPI 1.3 rev 1, EOP-SPI 1.5 rev 0, EOP-SPI 1.6 rev 1		Reference Attached: EOP-SPI 1.1 p 2, EOP-SPI 1.3 p 2, EOP-SPI 1.5 p 2, EOP-SPI 1.6 p 2	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): 3402-03-D			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 028**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295038	2.4.31
	Importance Rating	4.2	
K&A: Knowledge of annunciator alarms, indications, or response procedures.			
<b>High Off-site Release Rate / 9</b>			
<p>Explanation: <b>Answer A</b> – Release path from FHB is through the UNIT 1 Plant Vent, EOP-3 requires verification of supply fans that should have tripped trip.</p> <p>B &amp; D – incorrect – this would be true if the leak was from a primary system per the EOP's</p> <p>C &amp; D – incorrect – unit 2 plant vent is not release pathway</p>			
Technical Reference(s): ARI-H13-P902-001-B1 rev 4 & EOP-3 Bases rev 0		Reference Attached: ARI-H13-P902-001-B1 p 7 & EOP-3 Bases p 12	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3402-17-D.2, OT-COMBINED-M40-B.3, & OT-3402-15-C.2			
Question Source:	Bank # Modified Bank # New	Perry 2007-1	
Question History:	Previous NRC Exam: Perry 2007-1		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 029**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	600000	AK1.02
	Importance Rating	2.9	
K&A: Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire Fighting			
<b>Plant Fire On Site / 8</b>			
<p>Explanation: <b>Answer C</b> – Per ONI-P54 Immediate operator action for Recirculation pump fire is to open 1P54F340.</p> <p>A – incorrect - 2 heat detectors automatically initiate CO2, but system will not dump with 1P54F340 closed</p> <p>B – incorrect – wrong valve, valve is already open</p> <p>D – incorrect – Fire Brigade is not required to open the selector valve for a Recirculation Pump fire</p>			
Technical Reference(s): ONI-P54 rev 15		Reference Attached: ONI-P54 p 4	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED P54-CO2-D.1.B, OT-3035-05(LP)-A.9			
Question Source:	Bank # Modified Bank # New	Perry 2007-1	
Question History:	Previous NRC Exam: Perry 2007-1		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 030**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	700000	AK2.01
	Importance Rating	3.1	
K&A: Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Motors			
<b>Generator Voltage and Electric Grid Disturbances / 6</b>			
<p>Explanation: <b>Answer D</b> – 75% voltage is to protect starting safety related motors</p> <p>A – incorrect – 3000 vac is protection for starting motors, not running equipment</p> <p>B – incorrect – Bus voltage of 3800 V does not effect the reliability of the Safety Related Transformers</p> <p>B &amp; C – incorrect – 3800 volts is to protect running equipment</p>			
Technical Reference(s): OT-COMBINED-R10 (LP) rev 0		Reference Attached: OT-COMBINED-R10 (LP) pp. 49-50	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-R10-D.3			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 031**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295007	AA2.02
	Importance Rating	4.1	
K&A: Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor power			
<b>High Reactor Pressure / 3</b>			
<p>Explanation: <b>Answer C</b> – Rx pressure rising causes voids to collapse causing a positive reactivity addition causing Rx power rises. Power will stabilize at a higher value due to the higher pressure.</p> <p>A &amp; B – incorrect – Rx power will initially rise due to void collapse, not lower</p> <p>D – incorrect – Rx power will not return to its original value</p>			
Technical Reference(s): OT-3301-04 (LP) rev 4 (GFE-Rx Theory Chapter 4) and IOI-3 rev 44		Reference Attached: OT-3301-04 (LP) p49 and IOI-3 p 7	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3301-04-7 & 9			
Question Source:	Bank # Modified Bank # New	Perry 2003	
Question History:	Previous NRC Exam: Perry 2003		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 032**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295009	2.4.2
	Importance Rating	4.5	
K&A: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.			
<b>Low Reactor Water Level / 2</b>			
<p>Explanation: <b>Answer B</b> – based on the given conditions, EOP-1 should be entered when vessel hits Level 2. The Level 2 and Level 3 isolations will occur. Additionally, HPCS and RCIC will initiate on L2 to restore level and not allow level to drop to Level 1.</p> <p>A &amp; C – incorrect - 1P43-F055 isolates on L1 not L2</p> <p>C &amp; D – incorrect - 1B21-F019 isolates on L1 not L2</p>			
Technical Reference(s): EOP-1 Bases rev 1 & OAI-1703 rev 3		Reference Attached: EOP-1 Bases p22 & OAI-1703 pp. 16 & 17	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3402-02-B & OT-COMBINED-B21-NS4-D			
Question Source:	Bank #	Modified Bank #	
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 033**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295011	AK1.01
	Importance Rating	4	
<p>K&amp;A: Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): Containment pressure: Mark-III</p>			
<b>High Containment Temp / 5</b>			
<p>Explanation: <b>Answer D</b> – with stable DW pressure, as containment pressure exceeds the DW vacuum breaker opening setpoint, the DW vacuum breakers will open</p> <p>A – These conditions will have no effect on indicated Containment Upper Pool level.</p> <p>B – These conditions will cause indicated Suppression Pool level to lower, not raise</p> <p>C – These conditions will cause Containment Vacuum breakers to remain closed, not open.</p>			
Technical Reference(s): ARI H13-P601-20(E4) & (F4) & SDM-M16 rev 4		Reference Attached: ARI H13-P601-20(E4) (F4) pp. 71 & 83 and & SDM-M16 p 5	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-T23-K10.D & OT-COMBINED-D23-F.4			
Question Source:	Bank # Modified Bank # New	Perry 2003	
Question History:	Previous NRC Exam: Perry 2003		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			



**QUESTION 034**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295014	AK2.01
	Importance Rating	3.9	
K&A: Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: RPS			
<b>Inadvertent Reactivity Addition / 1</b>			
<p>Explanation: <b>Answer B</b> – per Tech Spec Bases, IRM Neutron Flux-High trip is the primary scram signal.</p> <p>A – incorrect – SRM scram is bypassed for RFO12 startup</p> <p>C – incorrect – this APRM scram is active in Mode 1</p> <p>D – incorrect – this is a back-up or secondary scram to the IRM scram for the given condition.</p>			
Technical Reference(s): Tech Spec Bases 3.3.1.1 rev 0		Reference Attached: Tech Spec Bases 3.3.1.1 p B 3.3-4	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C71-F.2			
Question Source:	Bank # Modified Bank # New	Perry 2007-2	
Question History:	Previous NRC Exam: Perry 2007-2		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 035**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295020	AK3.01
	Importance Rating	3.8	
K&A: Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: Reactor SCRAM			
<b>Inadvertent Cont. Isolation / 5 &amp; 7</b>			
<p>Explanation: <b>Answer D</b> – the reactor scrams on a MSIV isolation in anticipation of the complete loss of the normal heat sink and subsequent over-pressurization transient. Channels A &amp; B give full MSIV isolation.</p> <p>A – incorrect – simultaneous trips on B21-N0681 A &amp; B channels cause all MSIVs to isolate which causes a reactor scram on MSIV position</p> <p>B &amp; C – incorrect – more than the stated actions occur</p>			
Technical Reference(s): PDB-I005 rev 9, SDM-B21(NS4) rev 6, & Tech Spec Bases 3.3.1.1 rev 0		Reference Attached: PDB-I005 pp. 1, 17-20, & 32, SDM-B21(NS4) 46, & TS Bases p B 3.3-14	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-B21(NS4)-E, I & J			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 036**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295029	EA1.02
	Importance Rating	3.1	
K&A: Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: HPCS			
<b>High Suppression Pool Wtr Lvl / 5</b>			
<p>Explanation: <b>Answer D</b> – Suppression Pool high level causes suction shift to the Suppression Pool. Both Test Valves Close. This loss of flow path causes the Minimum Flow Valve to open.</p> <p>A – incorrect - misconception when operating in TEST, if in suppression pool test mode (vice CST to CST test mode) no affect</p> <p>B – incorrect – misconception of CST Test Return Valve closure logic</p> <p>C – incorrect – misconception of Suction Valve transfer logic in TEST</p>			
Technical Reference(s): ARI-H13-P601-016-G5 Rev 12 and SDM-E22A rev 7		Reference Attached: ARI-H13-P601-016-G5 p 83 and SDM-E22A pp.67 & 69	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-E22A-E.3			
Question Source:	Bank # Modified Bank # New	Perry 2007-1	
Question History:	Previous NRC Exam: Perry 2007-1		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 037**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295033	EA1.02
	Importance Rating	3.7	
K&A: Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Process radiation monitoring system			
<b>High Secondary Containment Area Radiation Levels / 9</b>			
<p>Explanation: <b>Answer B</b> – the CVDWP radiation monitors are located in the Intermediate Building (secondary containment area) and will initiate an isolation signal to prevent a spread of radiation/contamination into secondary containment. A HI/HI alarm on D17-K609A and D radiation monitor causes an M14 Outboard Isolation while a HI/HI on the B and C radiation monitors causes an iM14 inboard isolation. The isolation causes loss of suction path trip for the fans. All fans trip but only 1/2 (the outboard) valves go closed.</p> <p>A – incorrect – the inboard isolation valves remain open</p> <p>C – incorrect – all of the fans trip and half of the isolation valves close</p> <p>D – incorrect – all of the fans trip</p>			
Technical Reference(s): ARI-H13-P680-007 rev 18 and EOP-3 Bases rev 0		Reference Attached: ARI-H13-P680-007 pp 27, 29-30 and EOP-3 Bases p 7	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3407-17-C			
Question Source:	Bank # Modified Bank # New		
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 038**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	203000	K4.09
	Importance Rating	3.1	
<p>K&amp;A: Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: Surveillance for all operable components</p>			
<b>RHR/LPCI: Injection Mode</b>			
<p>Explanation: <b>Answer A</b> – RHR pump A trips due to RHR-SDC valves F008 and F009 going close on the L3 signal. RHR B loop will realign for LPCI injection at L1. There is no automatic realignment of RHR from SDC to LPCI.</p> <p>B – incorrect – RHR pump A trips due to RHR-SDC valves F008 and F009 going close on the L3 signal</p> <p>C – incorrect – RHR Pump B does not trip and then restart</p> <p>D – incorrect – RHR B loop will realign for LPCI injection at L1</p>			
Technical Reference(s): ARI-H13-P601-017-C4 rev 12, ARI-H13-P601-020-F6 rev 15, PDB-I-005 rev 9, & SOI-E12 rev. 46		Reference Attached: ARI-H13-P601-017-C4 p 37, ARI-H13-P601-020-F6 p 89, PDB-I005 p 35, & SOI-E12 pp.9-10	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-E12-H.1.A			
Question Source:	Bank #	INL-36851	
	Modified Bank #		
	New		
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 039**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	205000	K5.03
	Importance Rating	2.8	
K&A: Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Heat removal mechanisms			
<b>Shutdown Cooling</b>			
<p>Explanation: <b>Answer B</b> – this is the approved method for adjusting the cooldown rate while in Normal Mode SDC.</p> <p>A – incorrect – this action will raise the cooldown rate</p> <p>C – incorrect – this is the required flow rate band when using the alternate return path via E12-F042A</p> <p>D – incorrect –not an approved method to control the cooldown rate per SOI-E12</p>			
Technical Reference(s): SOI-E12 rev 26		Reference Attached: SOI-E12 pp 26, 31-32	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-E12-B.4.C			
Question Source:	Bank # Modified Bank # New	Perry 2001	
Question History:	Previous NRC Exam: Perry 2001		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 040**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215005	K6.04
	Importance Rating	3.2	
K&A: Knowledge of the effect that a loss or malfunction of the following will have on the OSCILLATING POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: Trip units			
<b>OPRM</b>			
<p>Explanation: <b>Answer D</b> – when the APRM is bypassed, the associated OPRM automatically receives a core flow signal from the other APRM in the same panel. The OPRM trip is enabled at &gt; 23.3 % power and drive flow &lt; 63.5%.</p> <p>A &amp; C – incorrect – OPRM A receives core flow signal from APRM E</p> <p>A &amp; B – incorrect – the OPRM trip is enabled and will input a trip signal into RPS</p>			
Technical Reference(s): SOI-C51(APRM) rev 9 and ARI-H13-P601-006-A2 rev 7		Reference Attached: SOI-C51(APRM) p 14 and ARI-H13-P601-006-A2 p 7	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C51-AP_OPRM-E.2 & 3			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		
		x	
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 041**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209001	A1.03
	Importance Rating	3.8	
K&A: Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURECORE SPRAY SYSTEM controls including: Reactor water level			
<b>LPCS</b>			
<p>Explanation: <b>Answer D</b> – the Blue permissive light indicates that pressure downstream of the injection valve is &lt;600 psig and will allow the injection valve to automatically open.</p> <p>A – incorrect – the injection valve control switch bypasses the pressure permissive</p> <p>B – incorrect – the blue light off indicates pressure permissive is not met → no automatic opening</p> <p>C – incorrect – 600 psig is LPCS injection valve open permissive, 530 psig for RHR</p>			
Technical Reference(s): ARI-H13-P601-021-A6 rev 13 and SDM-E21 rev 10		Reference Attached: ARI-H13-P601-021-A6 p 17 and SDM-E21 pp.23-25	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-E21-E.1 & 3			
Question Source:	Bank # Modified Bank # New	Perry 2001-2	
Question History:	Previous NRC Exam: Perry 2001-2		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			



**QUESTION 042**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209002	A2.10
	Importance Rating	2.7	
<p>K&amp;A: Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve openings: BWR-5,6</p>			
<b>HPCS</b>			
<p>Explanation: <b>Answer B</b> – The SRV Tail Pipe Level limit is 24.5 feet. HPCS min flow valve will auto open when the Injection Valve is closed as flow lowers to 725 gpm to prevent the pump from running at shutoff head. With HPCS lined up to the CST, min flow is returned to the Suppression Pool causing pool level to go up.</p> <p>A &amp; C – incorrect – LPCS takes its suction from the suppression pool. Using LPCS will not cause suppression pool level to rise. 23.0 feet is the minimum water level over fuel.</p> <p>C &amp; D – incorrect – the SRVTPLL is 24.5 feet, the top of the PCL curve is 45.5 feet</p>			
Technical Reference(s): EOP Bases rev 0		Reference Attached: EOP Bases p 65	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-E22A-E.3 & OT-3039-01-B			
Question Source:	Bank # Modified Bank # New	Columbia 2006	
Question History:	Previous NRC Exam: Columbia 2006		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
<p>Comments: Level of Difficulty = x</p> <p>Level Of Knowledge: Candidate has to understand that HPCS will continue to add water to the suppression pool since the min flow valve opened when the inj valve was over ridden closed</p>			

**QUESTION 043**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000	A3.01
	Importance Rating	3.5	
K&A: Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: Pump discharge pressure			
<b>SLC</b>			
<p>Explanation: <b>Answer A</b> – both instruments show discharge pressure even though only one pump starts. Since one squib valve is open, a flow path exists, thus pressure will be slightly above reactor pressure.</p> <p>B – incorrect - due to the cross-connect line between the pumps, both meters will show discharge pressure</p> <p>C &amp; D – incorrect – since the pumps are rated at 1250 psig, a 1300 psig is an indication that the relief valve (1C41-F029A/B) is lifting, which is not the case since one squib valve fired</p>			
Technical Reference(s): SOI-C41 rev 18, Dwg. 302-691 rev V (Partial), & SDM-C41 rev 8		Reference Attached: SOI-C41 pp 6 & 9, Dwg. 302-691 rev V (Partial), & SDM-C41 p 8	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C41-C.3 & OT-3402-D.2			
Question Source:	Bank # Modified Bank # New	RQL-0047	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION        044**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	212000	2.4.11
	Importance Rating	4	
K&A: Knowledge of abnormal condition procedures.			
<b>RPS</b>			
<p>Explanation: <b>Answer B</b> – at time 10:10 there are no RPS scram signals nor RRCS initiation signals present. Therefore, RPS can be reset. However, ONI-C71-1, Reactor Scram contains a NOTE that informs the operator that RRCS logic contains a 12 minute timer, which does not allow reset of the RRCS logic until the timer times out.</p> <p>A &amp; C – incorrect – RRCS logic can not be reset due to the 12 minute timer</p> <p>C &amp; D – incorrect – RPS logic can be reset as there are no scram signals present</p>			
Technical Reference(s): ONI-C71-2 rev 8		Reference Attached: ONI-C71-2 p 7	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT—3035-11-(LP)-A.2			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		
			x
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 045**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	212000	K1.01
	Importance Rating	3.4	
K&A: Knowledge of the physical connections and/or cause/effect relationships between REACTOR PROTECTION SYSTEM and the following: A.C. electrical distribution			
<b>RPS</b>			
<p>Explanation: <b>Answer B</b> – with RPS MG Set Transfer switch in ALT A position, RPS bus A is powered from F1C08. Loosing RPS bus A causes an outboard isolation and ½ scram.</p> <p>A – incorrect – this is the normal power supply for RPS A</p> <p>C – incorrect – this is the normal power supply for RPS B</p> <p>D – incorrect – this is the alternate power supply for RPS B</p>			
Technical Reference(s): PDB-H014 rev 1, PDB-H15 rev 2, PDB-H16 rev 1, & SDM-C71 rev 10		Reference Attached: PDB-H15 p 5 & SDM-C71 p 78	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C71 D.1			
Question Source:	Bank # Modified Bank # New	Perry 2007-2	
Question History:	Previous NRC Exam: Perry 2007-2		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 046**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215003	K2.01
	Importance Rating	2.5	
K&A: Knowledge of electrical power supplies to the following: IRM channels/detectors			
<b>IRM</b>			
<p>Explanation: <b>Answer D</b> – RPS B supplies IRM H. When power is lost to the IRM power supply, the IRM fails down scale. It will be re-ranged to range 3 (in Mode 1) per SOI-C71</p> <p>A – incorrect – M16F010B receives an isolation signal on loss of RPS B, however, it is normally closed and will open on vacuum condition</p> <p>B – incorrect – APRMs are supplied from ATWS UPS's</p> <p>C – incorrect – the ERIS computer is supplied from the ATWS UPS</p>			
Technical Reference(s): SOI-C71 rev 16 & SDM-C51(IRM) rev 7		Reference Attached: SOI-C71 pp 50 & 111 and SDM-C51(IRM) p 53	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C51(IRM)-B.3.E			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 047**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO		
	Tier #	2			
	Group #	1			
	K/A#	215004	K3.04		
	Importance Rating	3.7			
K&A: Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: Reactor power and indication					
<b>Source Range Monitor</b>					
<p>Explanation: <b>Answer A</b> – with shorting links removed, SRM trips are active and non-coincident. As power rises above 2E+5, the reactor will scram on high flux and reactor period will be negative.</p> <p>B – incorrect – following the scram, reactor power lowers and Rx period becomes negative for several minutes before stabilizing at infinity</p> <p>C &amp; D – incorrect – Rx power will lower due to the scram</p> <p>D – incorrect – The Neutron Flux-High Setdown trip is a function of the APRM and higher than 2.5E5</p>					
Technical Reference(s): SOI-C51(SRM) rev 6		Reference Attached: SOI-C51(SRM) p 3			
Proposed references to be provided to applicants during examination: None					
Learning Objective (As available): OT-COMBINED-C51(SRM)-D.4 & OT-COMBINED –C71-O.5					
Question Source:	Bank # Modified Bank # New	Perry 2002			
Question History:	Previous NRC Exam: Perry 2002				
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x				
10 CFR Part 55 Content:	55.41      x 55.43				
Comments: Level of Difficulty = x					

**QUESTION 048**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215005	K4.07
	Importance Rating	3.7	
K&A: Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Flow biased trip setpoints			
<b>APRM / LPRM</b>			
<p>Explanation: <b>Answer C</b> – the APRM Upscale Thermal Power Trip setpoint is calculated using the formula <math>\text{Setpoint} = 0.628W_r + 61\%</math> (where <math>W_r</math> is % drive flow) and is clamped at 111%.</p> <p>A – incorrect – this is the current APRM Upscale Thermal Power Alarm setpoint</p> <p>B – incorrect – this is the clamped value for the APRM Upscale Thermal Power Alarm setpoint</p> <p>D – incorrect – this is the calculated setpoint, however, it is clamped at 111%</p>			
Technical Reference(s): SDM-C51 (PRM & OPRM) rev 10 and PDB A12 rev 15		Reference Attached: SDM-C51 (PRM & OPRM) pp 23-25 and PDB-A12 p 6	
Proposed references to be provided to applicants during examination: PDB-A0012, Recirc Drive Flow vs. Total Core Flow			
Learning Objective (As available): OT-COMBINED-C51(AP_OPRM)-D.8			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 049**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	217000	K5.07
	Importance Rating	3.1	
K&A: Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Assist core cooling			
<b>RCIC</b>			
<p>Explanation: <b>Answer A</b> – an exhaust diaphragm rupture will cause an immediate RCIC turbine trip.</p> <p>B – incorrect – the steam tunnel high temperature has a 29 minute time delay to isolate</p> <p>C – incorrect – high RCIC lube oil temp does not cause an immediate loss of the turbine</p> <p>D – incorrect – a high suppression level will cause a suction shift, but the suppression pool valve opens fully before the CST valve closes – in this case it doesn't matter which source is lined up.</p>			
Technical Reference(s): ARI-H13-P601-021 rev 13, ARI-H13-P601 rev 11, & SOI-E31 rev 8		Reference Attached: ARI-H13-P601-021 pp. 19, 39, & 89, ARI-H13-P601 p111, & SOI-E31 p 19	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-E51-D.3			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 x 55.43		
Comments: Level of Difficulty = x			



**QUESTION 050**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	218000	K6.06
	Importance Rating	3.4	
K&A: Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: D.C. power:			
<b>ADS</b>			
<p>Explanation: <b>Answer C</b> – the Pressure Relief mode will function using the B solenoids and the ADS function will function using the Div 2 logic</p> <p>A &amp; B – incorrect – the pressure relief mode will still function utilizing the B solenoids.</p> <p>D – incorrect – Div 2 ADS logic is independent of Div 1 power supply and can still open the ADS valves</p>			
Technical Reference(s): PDB-I005 rev 9 and SDM-B21C rev 6		Reference Attached: PDB-I005 p 14 and SDM-B21 p 22	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-B21C-E.1 & OT-COMBINED-R42-H			
Question Source:	Bank # Modified Bank # New	Perry 2003	
Question History:	Previous NRC Exam: Perry 2003		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 051**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	223002	A1.01
	Importance Rating	3.5	
<p>K&amp;A: Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: System indicating lights and alarms</p>			
<b>PCIS/Nuclear Steam Supply Shutoff</b>			
<p>Explanation: <b>Answer D</b> – the RHR Radwaste Valve isolation is Level 3 and 1.68 psig. The light goes out if isolation signal present. The candidate needs to know if the lights are on or off for the isolation.</p> <p>A – incorrect – the BOP isolation is actuated on Level 2 and 1.68 psig in DW – the light would be off</p> <p>B – incorrect – the RWCU isolation is actuated on Level 2 and Leak Detection signals – the BOP light would be off if hit L2</p> <p>C – incorrect – the RHR LOCA isolation is actuated on Level 1 and 1.68 psig in DW – the light would be off</p>			
Technical Reference(s): IOI-18 rev 6 and DWGS 208-013-008 rev P & 208-013-012 rev FF		Reference Attached: IOI-18 p 82 and DWGS 208-013-008 & 208-013-012	
Proposed references to be provided to applicants during examination: Picture of panel H13-P622			
Learning Objective (As available): OT-COMBINED-B21(NS4)-M			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 052**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	239002	A2.03
	Importance Rating	4.1	
<p>K&amp;A: Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV</p>			
<b>SRVs</b>			
<p>Explanation: <b>Answer A</b> – an indication that an SRV is open is lowering of main generator MWe – per ONI-B21-1, Rx power is lowered to 90%</p> <p>B &amp; D – incorrect – reactor power is to be lowered to less than 90% prior to attempting to close an SRV</p> <p>C &amp; D – incorrect – the flow orifices are down stream of the SRV's and indicated steam flow will lower not rise</p>			
Technical Reference(s): ONI-B21-1 rev 9		Reference Attached: ONI-B21-1 pp.3-4	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): x			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 053**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	259002	A3.06
	Importance Rating	3.0	
K&A: Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: Reactor water level setpoint setdown following a reactor scram: Plant-Specific			
<b>Reactor Water Level Control</b>			
<p>Explanation: <b>Answer D</b> – with the Operator Rx Level Setpoint set at 200", when RPV level drops below L3, Setpoint Setdown logic demands the Operator Rx Level Setpoint setting for 10 seconds then lowers to 178"</p> <p>A &amp; B – incorrect – the scram signal does not initiate Setpoint Setdown logic</p> <p>A &amp; C – incorrect – the Operator Rx Level Setpoint was set at 200", therefore the Setpoint Setdown logic demands 200" not 196"</p>			
Technical Reference(s): OT-COMBINED-C32 LP rev 4		Reference Attached: OT-COMBINED-C32 LP p 32	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C34-F.5.c			
Question Source:	Bank # Modified Bank # New	INL-3034	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION 054**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	261000	A4.01
	Importance Rating	3.2	
K&A: Ability to manually operate and/or monitor in the control room: Off-site release levels			
<b>SGTS</b>			
<p>Explanation: <b>Answer B</b> – The discharge of each AEGTS fan branches into two headers. One header recirculates air flow back into the annulus. The other header exhausts air to the unit's vent, AEGTS subsystem A to the Unit 1 Vent, AEGTS subsystem B to the Unit 2 Vent.</p> <p>A – incorrect – AEGTS A exhausts to U1 plant vent</p> <p>C &amp; D – plausible due to misconception that AEGTS exhausts to these ventilation systems</p>			
Technical Reference(s): Dwg 912-605 rev W		Reference Attached: Dwg 912-605	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-M15-B.1			
Question Source:	Bank # Modified Bank # New	Perry 2002	
Question History:	Previous NRC Exam: Perry 2002		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 055**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262001	2.4.45
	Importance Rating	4.1	
K&A: Ability to prioritize and interpret the significance of each annunciator or alarm.			
<b>AC Electrical Distribution</b>			
<p>Explanation: <b>Answer C</b> – the LOOP logic requires both bus EH11 &amp; bus EH12 to have low voltage (3KV) or low frequency (59 Hz). The low voltage is made up for the EH12 bus, so when frequency on bus EH11 lowers below 59 Hz, the logic will be made up.</p> <p>A – incorrect – this conditions is already fulfilled when the Bus Stripped alarm is received</p> <p>B – incorrect – this conditions is true if a LOCA signal were present</p> <p>D – incorrect – the LOOP logic is already made up for bus EH12, therefore this signal will have no effect</p>			
Technical Reference(s): SOI-S11 rev 5		Reference Attached: SOI-S11 pp. 3-4	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-R10-D.3			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		
			x
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 056**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262002	K1.19
	Importance Rating	2.9	
K&A: Knowledge of the physical connections and/or cause/effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Power range neutron monitoring system: Plant-Specific			
<b>UPS (AC/DC)</b>			
<p>Explanation: <b>Answer A</b> – cycling the ATWS breakers is preferred because this method deenergizes the APRM to cause a scram and allows the MSIV's to remain open.</p> <p>B – incorrect – this will cause the APRM's to lose power</p> <p>C – incorrect this will cause the MSIV's to close</p> <p>D – incorrect – this is not the preferred method</p>			
Technical Reference(s): ONI-C61 rev 6		Reference Attached: ONI-C61 p 5	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): x			
Question Source:	Bank # Modified Bank # New	Perry 2002	
Question History:	Previous NRC Exam: Perry 2002		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 057**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	263000	K2.01
	Importance Rating	3.1	
K&A: Knowledge of electrical power supplies to the following: Major D.C. loads			
<b>DC Electrical Distribution</b>			
<p>Explanation: <b>Answer D</b> – DC bus D1B supplies Emergency Bearing oil pump, D1A supplies the main generator trip logic, and ED1B supplies RHR B/C logic.</p> <p>A – incorrect – all three selections are incorrect</p> <p>B – incorrect – D1B does not supply main generator trip logic</p> <p>C – incorrect – wrong selections for emergency bearing oil pump and RHR logic</p>			
Technical Reference(s): PDB-H002 rev 2, PDB-H004 rev 3, & PDB-H005 rev 2		Reference Attached: PDB-H002 p 12, PDB-H004 pp. 13-14, & PDB-H005 p 2	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-R42-B.2.C			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			



**QUESTION 058**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	264000	K3.02
	Importance Rating	3.9	
K&A: Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: A.C. electrical distribution			
<b>EDGs</b>			
<p>Explanation: <b>Answer B</b> – the Lube Oil high temperature trip is bypassed with a LOOP signal present when the DG CONTROL TRANSFER switch is in CR position and the DIFF RELAY LOCKOUT trip (Div 2) is always active</p> <p>A – incorrect – EH12 will be deenergized</p> <p>C &amp; D – incorrect – EH11 will be energized</p>			
Technical Reference(s): SOI-R43 rev 36, ARI-H13-P877-001 rev 10, ARI-H13-P877-002 rev 11, & ARI-H51-P054B rev 7		Reference Attached: SOI-R43 p 6, ARI-H13-P877-001 pp 31& 44, ARI-H13-P877-002 pp. 44 & 53, & ARI-H51-P054B p 97	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-R43_48-D.9			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 059**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	300000	K4.03
	Importance Rating	2.8	
K&A: Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Securing of IAS upon loss of cooling water			
<b>Instrument Air</b>			
<p>Explanation: <b>Answer D</b> – NCC cools the IAC lube oil. On a loss of NCC, lube oil will heat up to the trip setpoint.</p> <p>A – incorrect –CC chiller B is supplied from ECC, CC chiller C is supplied from NCC</p> <p>B &amp; C – incorrect – CRD and Recirc pumps do not have automatic trips on high temp</p>			
Technical Reference(s): ARI-H13-P904 rev 9, ARI-H13-P680-004 rev 14, & ONI-P43 rev 11		Reference Attached: ARI-H13-P904 p 53, ARI-H13-P680-004 pp. 107 & 129 & ONI-P43 p 17	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-P43-H & OT-3035-16(LP)-A.3			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION 060**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	300000	K1.05
	Importance Rating	3.1	
K&A: Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: Main Steam Isolation Valve air			
<b>Instrument Air</b>			
<p>Explanation: <b>Answer D</b> – air is bled off the MSIV air cylinder allowing it to slow close.</p> <p>A &amp; B – incorrect – bleeding air off the MSIV allows it to close slowly – not 3-5 seconds which is the normal stroke time</p> <p>A &amp; C – incorrect – Safety Related Instrument Air (P57) is not the source of operating air to the MSIV's</p>			
Technical Reference(s): SDM-B21/N11 rev 10		Reference Attached: SDM-B21/N11 pp.25 & 58	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-B21_N11-E.2			
Question Source:	Bank # Modified Bank # New	Perry 2001	
Question History:	Previous NRC Exam: Perry 2001		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 061**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	400000	K2.01
	Importance Rating	2.9	
K&A: Knowledge of electrical power supplies to the following: CCW pumps			
<b>Component Cooling Water</b>			
<p>Explanation: <b>Answer D</b> – Transformer LH-2-A supplies Bus EH21 which supplies Bus XH21 which is the power supply for NCC pump C</p> <p>A – incorrect – SW Pump C is powered from bus H22 which is powered from LH-2-C</p> <p>B – incorrect – Containment Vessel chiller is powered from Bus H21 which is powered from LH-2-B</p> <p>C – incorrect – U2 SAC is powered from Bus H22 which is powered from LH-2-C</p>			
Technical Reference(s): ELI-R22 rev 8, ARI-2H13-P877-01 rev 3, & ARI-2H13-P870-001 rev 5		Reference Attached: ELI-R22 p 25, ARI-2H13-P877-01 p 3, & ARI-2H13-P870-001 p 39	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-P43-C.2			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION        062**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000	K2.01
	Importance Rating	2.9	
K&A: Knowledge of electrical power supplies to the following: SBLC pumps			
<b>SLC</b>			
<p>Explanation: <b>Answer B</b> – Standby Liquid Control Pump A, 1C41-C001A is powered from Bus EF-1-A, Breaker EH1104 on Bus EH11 supplies power to Bus EF-1-A</p> <p>A – incorrect – is plausible; powered from F-1-E bus</p> <p>C – incorrect – is plausible; Powered from F-1-C bus</p> <p>D – incorrect – is plausible; powered from EF-2-A Bus</p>			
Technical Reference(s): Dwg 206-021 rev KKKK		Reference Attached: Dwg 206-021	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C41-E.1, OT-COMBINED-R10-C.17			
Question Source:	Bank # Modified Bank # New	Perry 2007-1	
Question History:	Previous NRC Exam: Perry 2007-1		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION 063**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	400000	A1.02
	Importance Rating	2.8	
K&A: Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: CCW temperature			
<b>Component Cooling Water</b>			
<p>Explanation: <b>Answer A</b> – the TBCC temperature control valve splits the TBCC flow through the HX and bypassing the HX. As the lake temperature rises, SW temperature rises. This causes the TCV to bypass less TBCC flow around the HX to maintain TBCC outlet temperature constant. This is a recent modification to the plant to throttle TBCC vs. SW flows.</p> <p>B – incorrect – this is true if lake water temperature lowers</p> <p>C &amp; D – incorrect – SW is full flow through the TBCC HX's – plausible is candidate does not understand which medium is throttled</p>			
Technical Reference(s): SDM-P44 rev 6		Reference Attached: SDM-P44 pp 4 & 23	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-P44-B.2			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 064**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	201003	K1.02
	Importance Rating	2.9	
K&A: Knowledge of the physical connections and/or cause/effect relationships between CONTROL ROD AND DRIVE MECHANISM and the following: Reactor water			
<b>Control Rod and Drive Mechanism</b>			
<p>Explanation: <b>Answer B</b> – when the scram outlet valve opens, the ball check valve will reposition since it senses accumulator pressure &lt; Rx pressure and directs Rx water under the drive piston to insert the control rod.</p> <p>A – incorrect – Rx water is directed below the piston not above</p> <p>C &amp; D – incorrect – the ball check valve does not reposition to allow accumulator water to drive the rod – the accumulator is normally lined up to drive the rod</p>			
Technical Reference(s): SDM-C22(CRDH) rev 5		Reference Attached: SDM-C22(CRDH) pp. 6 & 37	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C11(CRDM)-C.1			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 065**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	201005	K5.09
	Importance Rating	3.5	
K&A: Knowledge of the operational implications of the following concepts as they apply to ROD CONTROL AND INFORMATION SYSTEM (RCIS): High power setpoints			
<b>RCIS</b>			
<p>Explanation: <b>Answer C</b> – RC&amp;IS enforces the limits of the rod withdrawal limiter above the HPSP and limits rod withdrawal to 2 notches to mitigate the consequences of a rod withdrawal error.</p> <p>A &amp; B – incorrect – the rod pattern controller of RC&amp;IS is designed to mitigate the consequences of a control rod drop accident</p> <p>B &amp; D – incorrect – the 4-notch limit is enforced between the LPSP and the HPSP</p>			
Technical Reference(s): Tech Spec Bases 3.3.2.1 rev 0 & SOI-C11(RCIS) rev 26		Reference Attached: Tech Spec Bases 3.3.2.1 pp. B 3.3-42 & 43 & SOI-C11(RCIS) p 6	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-C11(RCIS)-G & J.5			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			



**QUESTION 066**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	202002	A3.01
	Importance Rating	3.6	
K&A: Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: Flow control valve operation: BWR-5,6			
<b>Recirculation Flow Control</b>			
<p>Explanation: <b>Answer B</b> – the FCV's will runback 40% from the original starting point. The FCV's will return to the original position if no operator action is taken.</p> <p>A &amp; C – incorrect – if the runback was not caused by AFDL, then the FCV's would remain in the current position</p> <p>C &amp; D – incorrect – this is correct if the FCV's would have started at 100% open – plausible if the candidate thinks FCV's will close 40% from 100%</p>			
Technical Reference(s): SDM-B33 rev 9		Reference Attached: SDM-b33 p 43	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-B33-E.16			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 067**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	216000	K4.01
	Importance Rating	3.6	
K&A: Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following: Reading of nuclear boiler parameters outside the control room			
<b>Nuclear Boiler Inst.</b>			
<p>Explanation: <b>Answer B</b> – RPV level, pressure, and DW temperature are monitored via instrumentation from the RSD.</p> <p>A &amp; C – incorrect – containment pressure and RPV metal temperature are not monitored parameters</p> <p>D – incorrect – RPV metal temperature is not a monitored parameter</p>			
Technical Reference(s): PDB-R001 rev 25		Reference Attached: PDB-R001 p 91	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-05(LP)-A.10 & OT-3037-08-B			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			

**QUESTION 068**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	226001	A1.07
	Importance Rating	3.1	
K&A: Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: System pressure			
<b>RHR/LPCI: CTMT Spray Mode</b>			
<p>Explanation: <b>Answer B</b> – the high DW pressure signal will initiate RHR A. However, Rx pressure is too high for LPCI injection and Containment Spray mode will not auto initiate until containment pressure exceeds 8 psig.</p> <p>A – incorrect – Containment Spray auto initiate logic is not complete</p> <p>C – incorrect – LPCI injection valve opens at 530 psig but system injection doesn't start until ~280 psig due to the discharge pressure of RHR pumps</p> <p>D – incorrect – RHR Containment Spray mode can be manually initiated when drywell pressure is above 1.68 psig.</p>			
Technical Reference(s): ARI-H13-P601-020 rev 15, SOI-E12 rev 46, & SDM-E12 rev 9		Reference Attached: ARI-H13-P601-020 p 11, SOI-E12 p 9, & SDM-E12 p 97	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-E12-G.1 & G.2			
Question Source:	Bank # Modified Bank # New	Perry 2002	
Question History:	Previous NRC Exam: Perry 2002		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 069**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	233000	A4.04
	Importance Rating	2.9	
K&A: Ability to manually operate and/or monitor in the control room: Pool level			
<b>Fuel Pool Cooling/Cleanup</b>			
<p>Explanation: <b>Answer B</b> – First, upper pool level lowers due to IFTS tube filling, then level is restored when the FTT drain pumps pump water into the FPCC surge tanks and water is returned to the upper pools.</p> <p>A – incorrect – The Upper Containment Pool level is restored via the Fuel Transfer Tube Drain Tank Pump to the FPCC Surge Tanks and then pumped back to the Upper Containment Pool</p> <p>C &amp; D – incorrect - Upper containment pool level will initially lower as the transfer tube is filled (until the FPCC Upper Pool return can restore pool level)</p>			
Technical Reference(s): SDM-G41 rev 5		Reference Attached: SDM-G41 pp.5 & 50	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-G41-B.1, B.2.c, & B.2.d			
Question Source:	Bank # Modified Bank # New	Perry 2002	
Question History:	Previous NRC Exam: Perry 2002		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 070**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	239001	K5.09
	Importance Rating	3.4	
K&A: Knowledge of the operational implications of the following concepts as they apply to MAIN AND REHEAT STEAM SYSTEM: Decay heat removal			
<b>Main and Reheat Steam</b>			
<p>Explanation: <b>Answer D</b> – 1B21F051D opens at 1103 psig arming LLS, which opens 1B21F051C and F047F, F051A, F051B, F051G. SRVs 47F, 51A, 51B, and 51G cycle between 946 and 1113 psig. F051C cycles between 936 and 1073 psig and F051D cycles between 926 and 1033 psig.</p> <p>A – incorrect – F051C does not arm LLS logic</p> <p>B – incorrect – cycle pressures for all LLS SRVs is not correct</p> <p>C – incorrect – cycle pressure for F051C and D is not correct</p>			
Technical Reference(s): ONI-B21-1 rev 9, SOI-B21 rev 16, SDM-B21 rev 10		Reference Attached: ONI-B21-1 p 11, SOI-B21 p 4, SDM-B21 pp. 21 & 29	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-B21/N11-E.1 and E.2			
Question Source:	Bank # Modified Bank # New	Perry 2007-1	
Question History:	Previous NRC Exam: Perry 2007-1		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41    x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 071**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	241000	A1.13
	Importance Rating	2.7	
K&A: Ability to predict and/or monitor changes in parameters associated with operating the REACTOR/TURBINE PRESSURE REGULATING SYSTEM controls including: Main turbine speed			
<b>Reactor/Turbine Pressure Regulator</b>			
<p>Explanation: <b>Answer D</b> – The CVs will throttle closed first and will be fully closed at 105.5% of rated speed. Then the master IVs will throttle closed and all CVs will be fully closed at 107.5% of rated speed.</p> <p>A – incorrect – the CVs AND IVs will not throttle together</p> <p>B – incorrect – the IVs will NOT throttle closed first - the CVs will throttle closed first and will be fully closed at 105.5% of rated speed, then the master IVs will throttle closed AND all IVs will be fully closed at 107.5% of rated speed</p> <p>C – incorrect – the CVs will NOT remain open - the CVs will throttle closed first and will be fully closed at 105.5% of rated speed, then the master IVs will throttle closed and all IVs will be fully closed at 107.5% of rated speed</p>			
Technical Reference(s): SDM-N32/C85 rev 6		Reference Attached: SDM-N32/C85 pp. 34-35 & 115	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-N32/C85-G			
Question Source:	Bank # Modified Bank # New	Nine Mile Point 2002	
Question History:	Previous NRC Exam: Nine Mile Point 2002		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41 55.43	x	
Comments: Level of Difficulty = x			

**QUESTION 072**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	259001	A2.05
	Importance Rating	3	
<p>K&amp;A: Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of applicable plant air systems</p>			
<b>Reactor Feedwater</b>			
<p>Explanation: <b>Answer C</b> – the RFPT recirc valve fails open on a loss of air. Since the RFPT is operating at the suction limit to maintain RPV level, the &gt; 7000 gpm going to recirc will cause RPV level to lower. Vessel level will continue to lower until power is reduced to the lower feed capacity. Scram prior to auto scram at level 3. Lowering power would not be an option since it would require a R/R pump downshift.</p> <p>A – incorrect – this will have no effect as the RFPT recirc valve fails open on a loss of air</p> <p>B – incorrect – motor feed pump will start but control valves will not open to provide flow to vessel due to the loss of air</p> <p>D – incorrect – RFPT B will respond in auto but it is already operating at 95%, will add recirc flow of 7000+ gpm, vessel level will lower</p>			
Technical Reference(s): ONI-P52 rev 14, ARI-H13-P680-003 rev 11, & SOI-C34 rev 26		Reference Attached: ONI-P52 p 30, ARI-H13-P680-003 p 103, & SOI-C34 p 4	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-N27-K.4			
Question Source:	Bank # Modified Bank # New	RQL-16886	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 073**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	271000	A4.02
	Importance Rating	2.9	
K&A: Ability to manually operate and/or monitor in the control room: System flows			
<b>Offgas</b>			
<p>Explanation: <b>Answer C</b> – a low level in the condenser seal troughs can cause increased air in-leakage into the condensers. Increased air leakage will result in higher Offgas system flow.</p> <p>A – incorrect – moisture in the adsorber trains can actually lower after filter flow by blocking the flowpath</p> <p>B – incorrect – excessive steam flow to the SJAE can cause high cooler condenser temperatures, but not a change in Offgas flow</p> <p>D – incorrect – a prefilter line drain loop seal low level can cause a spread of contamination, but not an increase in after filter flow</p>			
Technical Reference(s): ARI-H13-P845-001 rev 9 & ARI-H13-P680-002 rev 8		Reference Attached: ARI-H13-P845-001 p 107 & ARI-H13-P680-002 p 3	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-N62-K.2			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	x	
	55.43		
Comments: Level of Difficulty = x			



**QUESTION 074**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	290003	K6.01
	Importance Rating	2.7	
K&A: Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC: Electrical power			
<b>Control Room HVAC</b>			
<p>Explanation: <b>Answer D</b> – Both trains initiate in ER on loss of radiation monitor signal.</p> <p>A – incorrect – plausible – loss of power will not have any affect</p> <p>B&amp;C – incorrect – both trains initiate in ER</p>			
Technical Reference(s): SOI-M25/26 Rev 18 & dwg 208-117 sh 11 rev S		Reference Attached: SOI-M25/26 p 5 & dwg 208-117 sh 11	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-M25/26 F & G			
Question Source:	Bank # Modified Bank # New	Perry 2009	
Question History:	Previous NRC Exam: Perry 2009		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis      x		
10 CFR Part 55 Content:	55.41      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 075**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	290002	2.2.12
	Importance Rating	3.7	
K&A: Knowledge of surveillance procedures.			
<b>Reactor Vessel Internals</b>			
<p>Explanation: <b>Answer C</b> – With the Plant computer out of service, pressure readings must be obtained from installed instrumentation. Tech Spec and the SVI requires cooldown at &lt;100°F, cooldown from 1000-425 psig is approximately 92°F, from 425-100 psig is approximately 116°F, and 100-25 psig is approximately 71°F per ABB steam tables. Similar readings were obtained using RPV saturation curve of IOI-11.</p> <p>A – incorrect – cooldown exceeded 100°F/hr between 1300 and 1400</p> <p>B &amp; D – incorrect – cooldown rate not exceeded</p>			
Technical Reference(s): SVI-B21-T1176 rev 12, TS 3.4.11, and TS Bases 3.4.11 rev 4		Reference Attached: SVI-B21-T1176 p 6, TS 3.4.11 pp 3.4-26&27, and TS Bases 3.4.11 pp B 3.4-54&55	
Proposed references to be provided to applicants during examination: Steam Tables			
Learning Objective (As available): OT-3035-05(LP)-A.10 & OT-3037-08-B			
Question Source:	Bank # Modified Bank # New	Brunswick 2008	
Question History:	Previous NRC Exam: Brunswick 2008		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41                      x 55.43		
Comments: Level of Difficulty = x			

**QUESTION 076**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.1.1	
	Importance Rating		4.2
K&A: Knowledge of conduct of operations requirements.			
<b>Generic</b>			
<p>Explanation: <b>Answer D</b> – The SRO is required to stand 5 12-hour watches per calendar quarter in a SRO position to maintain proficiency. Alternatively, he may stand 1 SRO watch and the remaining 4 may be in a Tech Spec required RO or SRO position.</p> <p>A – incorrect – plausible if candidate uses quarter rather than calendar quarter</p> <p>B – incorrect – the Field Supervisor is not a required TS position – only 4 RO watches can be credited</p> <p>C – incorrect – the Shift Engineer must also maintain an Active RO or SRO license per PYBP-POS-1-5</p>			
Technical Reference(s): TMA-4206 rev 12 & PYBP-POS-1-5 rev 3		Reference Attached: TMA-4206 p 17 & PYBP-POS-1-5 p 5	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-2600-01-F			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 55.43b.1 x		
Comments: Level of Difficulty = x			

**QUESTION 077**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.1.36	
	Importance Rating		4.1
K&A: Knowledge of procedures and limitations involved in core alterations.			
<b>Generic</b>			
<p>Explanation: <b>Answer A</b> – per TS 3.9.6, RPV water level needs to be <math>\geq 22'9"</math> above the RPV flange while moving irradiated fuel assemblies. The basis of the action is to have sufficient water above the fuel to scrub any iodine generated during a fuel handling accident.</p> <p>B &amp; D – incorrect – this is the bases for TS 3.9.2 which is plausible due to misconception between control rod withdrawal and control rod handling</p> <p>C &amp; D – incorrect – this is the action while moving new fuel or CRB's with water <math>&lt;23'</math> above the fuel (TS 3.9.7) - this is plausible due to misconception of required water above the RPV flange vs. above irradiated fuel</p>			
Technical Reference(s): TS 3.9.6 & TS Bases 3.9.6 rev 4 Lesson Plan OT-360201 PowerPoint rev 10		Reference Attached: TS 3.9.6 p 3.9-8, TS Bases 3.9.6 p B 3.9-19, & LP OT-360201 PowerPoint slide 87	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3037-13-A, B, & C			
Question Source:	Bank # Modified Bank # New x		
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 55.43b.2 x		
Comments: Level of Difficulty = x			

**QUESTION 078**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.2.23	
	Importance Rating		4.6
K&A: Ability to track Technical Specifications limiting conditions for operations			
<b>Generic</b>			
<p>Explanation: <b>Answer C</b> – per TS 1.3, this qualifies for a 24 hour completion time extension – the plant is required to be in Mode 3 by June 9, 1700</p> <p>A – incorrect – this is the original completion time for RHR A inop</p> <p>B – incorrect – this is the time to enter TS 3.5.1 condition D – not Mode 3</p> <p>D – incorrect – this is the time to enter Mode 4</p>			
Technical Reference(s): TS 3.5.1 and TS 1.3		Reference Attached: TS 3.5.1 pp 3.5-1—3.5-3 and TS 1.3 pp. 1.0-11&12	
Proposed references to be provided to applicants during examination: Tech Spec 3.5.1			
Learning Objective (As available): OT-3037-02-F			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		
			x
10 CFR Part 55 Content:	55.41		
	55.43b.2	x	
Comments: Level of Difficulty = x			

**QUESTION 079**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.2.36	
	Importance Rating		4.2
K&A: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.			
<b>Generic</b>			
<p>Explanation: <b>Answer B</b> – with RCS temperature &gt; 250 deg. F, TS 3.8.4 is applicable. LCO Bases states either the Unit 1 or Unit 2 battery.</p> <p>A – incorrect – while this cycles the pump breaker, this is not the minimum requirement</p> <p>B – incorrect – this also cycles the pump breaker, but is not the minimum requirement</p> <p>C – incorrect – this is performed if keep-fill is lost or voiding is suspected</p>			
Technical Reference(s): Tech Spec 3.8.4 & 3.8.5 and TS Bases 3.8.4 rev 1		Reference Attached: Tech Spec 3.8.4 & 3.8.5 and TS Bases 3.8.4 p B 3.8-53	
Proposed references to be provided to applicants during examination: Technical Specifications 3.8.4 & 3.8.5			
Learning Objective (As available): OT-3037-12-C			
Question Source:	Bank # Modified Bank # New	Perry 2009 Audit	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41 55.43b.2   x		
Comments: Level of Difficulty = x			

**QUESTION 080**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.3.4	
	Importance Rating		3.7
K&A: Knowledge of radiation exposure limits under normal or emergency conditions.			
<b>Generic</b>			
<p>Explanation: <b>Answer A</b> – based on the 3 rem exposure estimate and the 500 mr dose received to date, it is not expected to exceed federal Dose limits, but it does exceed Perry admin dose limits. Since the TSC has not been activated, only the Shift Managers approval is necessary for the increased exposure.</p> <p>B – incorrect – the federal dose limit will not be exceeded</p> <p>C – incorrect – the Rad Pro Manager &amp; Tech's approval is not required</p> <p>D – incorrect – the federal dose limit will not be exceeded and Plant &amp; RP managers approval is not necessary</p>			
Technical Reference(s): HPI-B-003 rev 26 & NOP-OP-4201 rev 1		Reference Attached: HPI-B-003 pp. 4-5 & NOP-OP-4201 pp 15 & 20	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3037-03-I			
Question Source:	Bank # Modified Bank # New	Perry Audit 2003	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41 55.43b.4   x		
Comments: Level of Difficulty = x			

**QUESTION 081**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.3.15	
	Importance Rating		3.1
K&A: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.			
<b>Generic</b>			
<p>Explanation: <b>Answer D</b> – TS SR 3.0.3 allows up to 24 hours to complete a missed surveillance. TS SR 3.0.2 - 25% extension is not allowed for a missed surveillance.</p> <p>A &amp; B – incorrect –TS SR 3.0.3 allows up to 24 hours to complete a missed surveillance – 3.0.2 - 25% extension is not allowed for a missed surveillance</p> <p>A &amp; C – incorrect –TS SR 3.0.3 states if a surveillance is missed then immediate entry into the LCO is not required</p>			
Technical Reference(s): Tech Spec 3.0.3, Tech Spec Bases 3.0.3 rev 4, and Tech Spec 3.3.7.1		Reference Attached: Tech Spec 3.0.3 p 3.0-4, Tech Spec Bases 3.0.3 pp. B 3.0-12 to 14, and Tech Spec 3.3.7.1 p 3.3-70 to 73	
Proposed references to be provided to applicants during examination: Tech Spec 3.3.7.1			
Learning Objective (As available): OT-3037-04-J & K			
Question Source:	Bank # Modified Bank # New	Grand Gulf 2009	
Question History:	Previous NRC Exam: Grand Gulf 2009		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41 55.43b.4   x		
Comments: Level of Difficulty = x			



**QUESTION 082**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.4.40	
	Importance Rating		4.5
K&A: Knowledge of SRO responsibilities in emergency plan implementation.			
<b>Generic</b>			
<p>Explanation: <b>Answer D</b> – these duties are allowed to be delegated by the Emergency Coordinator who, in this case, is the Shift Manager</p> <p>A &amp; B – incorrect – Classification of the event can not be delegated</p> <p>B &amp; C – incorrect – Protective Action Recommendations can not be delegated</p>			
Technical Reference(s): EPI-A2 rev 14		Reference Attached: EPI-A2 pp 3-5	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): EPL-0801-01-3			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam: x		
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43b.5		x
Comments: Level of Difficulty = x			

**QUESTION 083**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295003	AA2.05
	Importance Rating		4.2
<p>K&amp;A: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Whether a partial or complete loss of A.C. power has occurred</p>			
<b>Partial or Complete Loss of AC / 6</b>			
<p>Explanation: <b>Answer B</b> – With HPCS pump and Div 1 ECCS pumps out of service, having the Div 1 and Div 3 DGs start is irrelevant. The plant is still in a TLAC not an SBO.</p> <p>A – incorrect – SBO would be entered if HPCS pump was available</p> <p>C – incorrect – LOOP would be entered if either Bus EH11 or EH12 with associated ECCS pumps available</p> <p>D – incorrect – ONI-R22-1 directs entry into ONI-R10 (and exit of ONI-R22-1)</p>			
Technical Reference(s): ONI-R10 rev 9 & ONI-R22-1 rev 8		Reference Attached: ONI-R10 p 3 & ONI-R22-1 p 5	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-18 (LP) A-4			
Question Source:	Bank # Modified Bank # New <span style="float: right;">x</span>		
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis <span style="float: right;">x</span>		
10 CFR Part 55 Content:	55.41 55.43b.5 <span style="float: right;">x</span>		
Comments: Level of Difficulty = x			

**QUESTION 084**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295004	2.4.11
	Importance Rating		4.2
K&A: Knowledge of abnormal condition procedures.			
<b>Partial or Total Loss of DC Pwr / 6</b>			
<p>Explanation: <b>Answer D</b> – the Entry Condition for all R42 ONI's is Bus Voltage zero. The other 'entry conditions' in the ONI describe the effects of the loss of the DC bus. Therefore, only ONI-R42-2 should be entered. Additionally, Since the Plant is in Mode 4, EU-1 is the correct E-plan classification.</p> <p>A &amp; B – incorrect – voltage on ED-1-C is greater than the ONI-R42-3 Entry Condition voltage</p> <p>A &amp; C – incorrect – ES-1 would be correct if plant was in Mode 1, 2, or 3.</p>			
Technical Reference(s): ONI-R42-2 rev 6 & EPI-A001 rev 21		Reference Attached: ONI-R42-2 p 3 & EPI-A001 p 14	
Proposed references to be provided to applicants during examination: EPI-A1 Attachments 1 and 2			
Learning Objective (As available): OT-3035-05(LP)-A.2			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		
			x
10 CFR Part 55 Content:	55.41		
	55.43b.5		
			x
Comments: Level of Difficulty = x			

**QUESTION 085**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295005	AA2.03
	Importance Rating		3.1
K&A: Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Turbine valve position			
<b>Main Turbine Generator Trip / 3</b>			
<p>Explanation: <b>Answer C</b> – On a turbine trip with Rx power &gt; 38%, the Turbine Stop valves generate a Rx scram signal when they go closed. Testing of the Stop Valves includes cycling the TSV to ensure a ½ scram signal is generated when each valve is &lt; 93% open. Time of Discovery of an INOP condition starts the clock for Required Actions. In this case, Time of Discovery was 16:00. RA is to place channel in trip within 12 hours. The bases of the TSV closure scram is to prevent exceeding the MCPR safety limit.</p> <p>A &amp; B – incorrect – the bases of the TSV scram is to not exceed MCPR</p> <p>B &amp; D – incorrect – the Required Actions must be completed within 12 hours of <u>time of discovery</u> – this answer is plausible if candidate believes that the 6-hour note for surveillance testing applies to completing the Required Actions</p>			
Technical Reference(s): Tech Spec 3.3-1, Tech Spec 1.3, & Tech Spec Bases B 3.3.1.1 rev 0		Reference Attached: Tech Spec 3.3-1 pp. 3.3-1&3, Tech Spec 1.3 p 1.0-11, & Tech Spec Bases B 3.3.1.1 pp. B 3.3-14&16	
Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.1 – partial – modified			
Learning Objective (As available): OT-3037-02-E			
Question Source:	Bank # Modified Bank # New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41 55.43b.2		x
Comments: Level of Difficulty = x			

**QUESTION 086**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	295006	2.4.41
	Importance Rating		4.6
K&A: Knowledge of the emergency action level thresholds and classifications.			
<b>SCRAM / 1</b>			
<p>Explanation: <b>Answer B</b> – this is a CA1 – failure to auto scram and power below 4%</p> <p>A – incorrect – this should be a CS1</p> <p>C – incorrect – no auto scram setpoint exceeded – Rx is S/D under all conditions without boron</p> <p>D – incorrect – no auto scram setpoint exceeded – no EAL entry</p>			
Technical Reference(s): EPI-A1 rev 21		Reference Attached: EPI-A1 pp. 14 & 23	
Proposed references to be provided to applicants during examination: EPI-A1 Attachments 1 and 2			
Learning Objective (As available): EPL-0804-01-4			
Question Source:	Bank # Modified Bank # New	Grand Gulf 2004	
Question History:	Previous NRC Exam: Grand Gulf 2004		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis      x		
10 CFR Part 55 Content:	55.41 55.43b.5   x		
Comments: Level of Difficulty = x			

**QUESTION 087**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295023	AA2.02
	Importance Rating		3.7
K&A: Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Fuel pool level			
<b>Refueling Acc / 8</b>			
<p>Explanation: <b>Answer B</b> – &gt; The stem contains entry conditions described for ONI-E12-2 entry. The ONI also describes Safe Conditions. FTI-D009 further restricts placing a bundle back into the Rx after it has been off-loaded.</p> <p>A – incorrect – Moving the bundle back to the core is an unsafe act/condition</p> <p>C – incorrect – An IFTS transfer to the FHB is an unsafe act/condition</p> <p>D – incorrect – An IFTS transfer to the FHB is an unsafe act/condition - moving the bundle back to the core is an unsafe act/condition</p>			
Technical Reference(s): ONI-E12-2 rev 25 & FTI-D009 rev 13		Reference Attached: ONI-E12-2 pp. 3-7 & FTI-D009 p 9	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-11(LP)-A.2 & OT-3602-01-D.4 & E.2			
Question Source:	Bank # Modified Bank # New	RQL-0748	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis      x		
10 CFR Part 55 Content:	55.41 55.43b.7   x		
Comments: Level of Difficulty = x			

**QUESTION 088**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295026	2.1.19
	Importance Rating		3.8
K&A: Ability to use plant computers to evaluate system or component status.			
<b>Suppression Pool High Water Temp. / 5</b>			
<p>Explanation: <b>Answer B</b> – With a suppression pool average temperature &gt; 105 deg F, testing adding heat to the pool must be suspended immediately (trip the RCIC turbine). Once testing is stopped, TS Bases states that the SP temperature must be lowered to ≤95°F within 24 hours.</p> <p>A – incorrect – Verifying SP temp ≤110°F is once per hour</p> <p>C &amp; D – incorrect – Initiating SP Cooling is not required</p> <p>D – incorrect – lowering Rx power is not a correct requirement</p>			
Technical Reference(s): Tech Spec 3.6.2.1 & TS 3.6.2.1 Bases rev 1		Reference Attached: Tech Spec 3.6.2.1 pp 36-38 & TS 3.6.2.1 Bases pp 70-71	
Proposed references to be provided to applicants during examination: SPDS screen print			
Learning Objective (As available): 3402-01-C.1, C.30, C.1.c			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 55.43b.5 x		
Comments: Level of Difficulty = x			

**QUESTION 089**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295030	EA2.02
	Importance Rating		3.9
K&A: Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool temperature.			
<b>Low Suppression Pool Wtr Lvl</b>			
<p>Explanation: <b>Answer C</b> – Eight of sixteen SP level instruments are uncovered and exposed to the containment atmosphere when SP level drops below 17.33'. In order for the SP level instruments to perform their intended function (Operable) they must be submerged.</p> <p>A – incorrect – the instruments not submerged are not operable</p> <p>B – incorrect – these instruments are reading containment air temperature and are not operable</p> <p>D – incorrect – the instruments still submerged are operable – plausible is candidate is confused about which sets of instruments are reading accurately</p>			
Technical Reference(s): Tech Spec 1.0, Tech Spec Bases 3.3 rev 1, & Dwg 240-082 rev K		Reference Attached: Tech Spec 1.0 p 1.0-5, Tech Spec Bases 3.3 p B 3.3-54 & Dwg 240-082	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-D23.C & D			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 55.43b.2 x		
Comments: Level of Difficulty = x			



**QUESTION 090**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295010	AA2.05
	Importance Rating		3.3
K&A: Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell air cooler drain flow: BWR-6			
<b>High Drywell Pressure / 5</b>			
<p>Explanation: <b>Answer A</b> – &gt; A Condensing Pot leak would be a mixture of hot water and steam, which results in High Drywell Pressure, Drywell Air Cooler Drain Flow, and Unidentified Leakage, E-plan is AU2 Unusual Event</p> <p>B &amp; D – incorrect – an Alert is incorrect classification for leakage greater than 10 gpm</p> <p>C &amp; D – incorrect – NCC cools the DW coolers and an NCC leak will not cause Drywell pressure rise and no –E-plan entry not a reactor coolant leak</p>			
Technical Reference(s): EPI-A1 rev 21		Reference Attached: EPI-A1 p 17	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-E31-C.1, EPL-0804			
Question Source:	Bank # Modified Bank # New	Perry 2005	
Question History:	Previous NRC Exam: Perry 2005		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis      x		
10 CFR Part 55 Content:	55.41 55.43b.5   x		
Comments: Level of Difficulty = x			

**QUESTION 091**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295012	AA2.01
	Importance Rating		3.9
K&A: Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell temperature.			
<b>High Drywell Temperature / 5</b>			
<p>Explanation: <b>Answer D</b> – &gt; Per TS 3.6.5.5 Bases, initial DW average air temperature is limited to ≤145°F to ensure that peak LOCA DW temperature does not exceed 330°F.</p> <p>A &amp; B - incorrect – initial temperature is assumed to be ≤145°F not 135°F</p> <p>A &amp; C - incorrect – the DBA LOCA for DW temperature is a small steam line break not FW line break, also, the peak temperature is 330°F not 300°F</p>			
Technical Reference(s): Tech Spec Bases 3.6.5.5 rev 7		Reference Attached: Tech Spec Bases 3.6.5.5 p B 3.6-148	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3037-10-B			
Question Source:	Bank # Modified Bank # New	Hope Creek 2005	
Question History:	Previous NRC Exam: Hope Creek 2005		
Question Cognitive Level:	Memory or Fundamental Knowledge    x Comprehension or Analysis		
10 CFR Part 55 Content:	55.41 55.43b.2    x		
Comments: Level of Difficulty = x			

**QUESTION 092**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295029	2.1.32
	Importance Rating		4
K&A: Ability to explain and apply system limits and precautions.			
<b>High Suppression Pool Wtr Lvl / 5</b>			
<p>Explanation: <b>Answer C</b> – &gt; The SRV Tail Pipe Level Limit is 24.5'. Operation of SRV's with SP level above this could result in damage to the SRV discharge lines and direct pressurization of containment. Since the reactor is shutdown, Anticipate ED would be an option. However, since SP level is above the level that ED is required, AED is not permitted. ED by SRV is required.</p> <p>A &amp; B – incorrect – using the turbine bypass valves to anticipate ED is not allowed when a requirement to Ed has been met</p> <p>D – incorrect – the high SP level can affect the MSIV drain valves not the SRV solenoids</p>			
Technical Reference(s): EOP- Bases rev 0		Reference Attached: EOP Bases p 38	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3402-05-C			
Question Source:	Bank # Modified Bank # New	Perry 2003 Audit	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41 55.43b.5   x		
Comments: Level of Difficulty = x			

**QUESTION 093**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	215003	A2.05
	Importance Rating		3.5
<p>K&amp;A: Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty or erratic operation of detectors/system</p>			
<b>IRM</b>			
<p>Explanation: <b>Answer D</b> – IRM channels C, G, &amp; H are out of spec. Channels G &amp; H are on trip system B. TS require 3 of 4 channels per trip system. TS 3.3.1.1 Condition A is entered since only Trip System B is below minimum number of channels. The channel is placed in trip IAW the Trip Instructions. Or the trip system is placed in trip (half scram inserted)</p> <p>A – incorrect – This the action from Condition H which is not required at this time - incorrect use of the table</p> <p>B – incorrect – This is the action from Condition B - it is a misconception that both trip systems are affected; Trip System B still has three channels</p> <p>D – incorrect – This is the action from Condition C, which is not entered since no loss of function has occurred</p>			
Technical Reference(s): IOI-1 rev 30 & Tech Spec 3.3.1.1		Reference Attached: IOI-1 28-29 & Tech Spec 3.3.1.1p 3.3-1	
Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.1 – Partial (Modified)			
Learning Objective (As available): OT-COMBINED-C51_IRM-J.1			
Question Source:	Bank # Modified Bank # New	RQL-0934	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41 55.43b.2   x		
Comments: Level of Difficulty = x			

**QUESTION 094**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	217000	2.1.27
	Importance Rating		4
K&A: Knowledge of system purpose and/or function.			
<b>RCIC</b>			
<p>Explanation: <b>Answer D</b> – The function of RCIC is to provide water for adequate core cooling if feedwater is isolated. The Suppression Pool suction is INOP because it is down powered. The Tech Spec required suction source is the Suppression Pool. With RCIC aligned to the CST and the SP suction INOP, the TS required source is unable to support the TS required function and RCIC is INOP.</p> <p>A &amp; C – incorrect – RCIC is INOP due to not being on the SP</p> <p>A &amp; B – incorrect – this is the Tech Spec function of HPCS</p>			
Technical Reference(s): Tech Spec Bases 3.5.3 rev 5 and Tech Spec Bases 3.5.1 rev 0		Reference Attached: Tech Spec Bases p B 3.5-21 & B 3.5-1	
Proposed references to be provided to applicants during examination: PMT data			
Learning Objective (As available): OT-3037-09-B			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 55.43b.2 x		
Comments: Level of Difficulty = x			

**QUESTION 095**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	223002	A2.08
	Importance Rating		3.1
<p>K&amp;A: Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Surveillance testing</p>			
<b>PCIS/Nuclear Steam Supply Shutoff</b>			
<p>Explanation: <b>Answer B</b> – per Tech Spec Bases, sufficient channels remain Operable such that a valid signal will isolate the MSL's. Therefore, the isolation function is maintained. The SR's are modified by a Note stating that entry into the Conditions and Required Actions may be delayed for up to 6 hrs. therefore entry into the Spec is 24 hours from time of discovery rather than from the US signing channel inoperability.</p> <p>A – incorrect – this is 24 hours from the US signing channel inoperability</p> <p>C &amp; D – incorrect — Function is not lost – MODE 2 is a RA if function is lost.</p>			
Technical Reference(s): Tech Spec 3.3.6.1 & Tech Spec Bases B 3.3.6.1 rev 4, SVI-B21-T0072-C rev 5		Reference Attached: Tech Spec 3.3.6.1 pp. 48 & 54 and Tech Spec Bases p B 3.3-142 & 164	
Proposed references to be provided to applicants during examination: Technical Specification 3.3.6.1 – partial – modified & PDB-I-5 partial			
Learning Objective (As available): OT-COMBINED-B21(NS4)-I			
Question Source:	Bank #		
	Modified Bank #		
	New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41		
	55.43b.2	x	
Comments: Level of Difficulty = x			

**QUESTION 096**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	239002	2.4.46
	Importance Rating		4.2
K&A: Ability to verify that the alarms are consistent with the plant conditions.			
<b>SRVs</b>			
<p>Explanation: <b>Answer C</b> – &gt; In order to ED, the US is in EOP 4-1. With ED occurring 20 minutes ago and the only injection source being CTS, RPV pressure will lower to &lt; 30 psig. This will cause the SRV OPEN annunciator to reset and the SRV Open/Close lights to change state. The SRVs are verified open by observing SRV tailpipe temperature of 250°F which corresponds to reactor pressure of ~25 psig. The US will be in EOP-4-1 and direct the RO to restore and maintain using CTS.</p> <p>A &amp; B incorrect – SRVs are still open – actions for additional vent paths is inappropriate</p> <p>D – incorrect – tailpipe temperature of 330°F corresponds to normal reactor pressure – action to open additional SRV's is inappropriate</p>			
Technical Reference(s): ARI-H13-P601-019 rev 11, EOP 4-1 rev o & ABB Steam Tables		Reference Attached: ARI-H13-P601-019 p 17 & EOP 4-1 chart	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-B21_N11-H			
Question Source:	Bank # Modified Bank # New	Perry 2005	
Question History:	Previous NRC Exam: Perry 2005		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41 55.43b.5   x		
Comments: Level of Difficulty = x			

**QUESTION 097**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	261000	A2.07
	Importance Rating		2.8
<p>K&amp;A: Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical failure</p>			
<b>SGTS</b>			
<p>Explanation: <b>Answer B</b> – The loss of Bus EH12 INOPs the 2<sup>nd</sup> AEGTS fan. With 2 AEGTS systems INOP entry into TS 3.0.3 is required. Entry into TS 3.0.3 is required at 08:15 on June 10<sup>th</sup>. Mode 3 is required in 13 hours and Mode 4 is required in 37 hours.</p> <p>A – incorrect – plausible is student believes that TS 3.03 entry is requires Mode 3 in 7 hours (this is Mode 2 requirement) and Mode 4 in 13 hours (this is Mode 4 requirement)</p> <p>C – incorrect – this is the TS 3.6.4.1 requirement but not the most restrictive condition</p> <p>D – incorrect – this is the TS 3.6.4.3 requirement but not the most restrictive condition</p>			
Technical Reference(s): Tech Spec 3.0.3, Tech Spec 3.6.4.3 and Tech Spec Bases B 3.6.4.3 rev 7		Reference Attached: Tech Spec 3.0.3 p 3.0-1, 3.6.4.3 p 3.6.56-57 and Tech Spec 3.6.4.1 p 3.6-51	
Proposed references to be provided to applicants during examination: TS 3.6.4.1 & TS 3.6.4.3			
Learning Objective (As available): x			
Question Source:	Bank # Modified Bank # New	x	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 55.43b.2 x		
Comments: Level of Difficulty = x			



**QUESTION 098**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	201005	2.1.32
	Importance Rating		4.0
K&A: 2.1.32 Ability to explain and apply system limits and precautions.			
<b>RCIS</b>			
<p>Explanation: <b>Answer B</b> – Per IOI-3, P&amp;L 2.2, if FMEOD is &gt; 0.990, then insert control rods to restore FMEOD to &lt;0.990 per Reactor Engineering direction. It is the Unit Supervisor's responsibility to determine method to insert control rods.</p> <p>A – incorrect – IOI-3, P&amp;L 2.2 does not direct use of Cram Rods</p> <p>C &amp; D – incorrect – per ONI-C51 Entry Conditions and IOI-3, P&amp;L 2.2, ONI-C51 is only entered when FMEOD is &gt; 1.000</p>			
Technical Reference(s): IOI-3 rev 24 & ONI-C51 Chart rev I		Reference Attached: ONI-C51 p 5 & ONI-C51 flowchart	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3035-08(LP)-B.4.B			
Question Source:	Bank # Modified Bank # New	Perry 2007-1	
Question History:	Previous NRC Exam: Perry 2007-1		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis                      x		
10 CFR Part 55 Content:	55.41 55.43b.1   x		
Comments: Level of Difficulty = x			

**QUESTION 099**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	202001	A2.06
	Importance Rating		3.8
<p>K&amp;A: Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Inadvertent recirculation flow decrease</p>			
<b>Recirculation System</b>			
<p>Explanation: <b>Answer A</b> – FCV operability is based on FCV open/close speed (&lt;11%/sec) and the ability to successfully lockup. Both FCV's are Operable. Operating in the Backup Stability Protection P/F Map indicates the OPRM's are not Operable. With OPRM's INOP and core flow &lt;42 Mlbm/hr, a reactor scram is required. Entry into ONI-C71 is required for a reactor scram.</p> <p>B &amp; D – incorrect – insertion of scram rods would be appropriate if OPRM's were Operable – However, since the plant is operating using the Backup Stability P/F map, the OPRM's are not operable and this is the incorrect action.</p> <p>C &amp; D – incorrect – it is a common misconception that if the FCV's are locked up, they are not operable, however, FCV operability is based on FCV speed and ability to lock them up</p>			
Technical Reference(s): Tech Spec 3.4.2, Tech Spec 3.4.2 Bases rev 7, PDB-A6 rev 14, & ONI-C51 rev 24		Reference Attached: Tech Spec 3.4.2 p 3.4-6, Tech Spec 3.4.2 Bases pp B 3.4 9 & 10, & ONI-C51 p 6	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-COMBINED-B33-C.3, L.4, & M			
Question Source:	Bank # Modified Bank # New	X	
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis x		
10 CFR Part 55 Content:	55.41 55.43b.5 x		
Comments: Level of Difficulty = x			

**QUESTION 100**

<b>Examination Outline Cross-Reference</b>	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	216000	2.1.9
	Importance Rating		4.5
K&A: 2.1.9 Ability to direct personnel activities inside the control room.			
<b>Nuclear Boiler Inst.</b>			
<p>Explanation: <b>Answer D</b> – With CRDH shutdown, reference leg purge is out of service. With reference leg purge out of service, accuracy of the RPV water level instruments is questioned. OAI-0201 requires compensatory measures when performing activities with a potential to drain the vessel with the reference leg purge is out of service while in Modes 1, 2, or 3. With RHR in SDC, valve manipulations on any part of the RHR system must be supervised by the SM or US.</p> <p>A – incorrect – with plant not in Mode 5, adjusting upper pool flow does not affect RPV level</p> <p>B – incorrect – OAI-0201 specifies RWCU system manipulations in Mode 3 must be supervised – not Mode 4</p> <p>C – incorrect – per Risk Management, if the plant was in Mode 5 and NCC was affecting Decay Heat removal, this would require supervisors oversight</p>			
Technical Reference(s): OAI-0201 rev 20		Reference Attached: OAI-0201 pp. 13-14	
Proposed references to be provided to applicants during examination: None			
Learning Objective (As available): OT-3039-01-A			
Question Source:	Bank #		
	Modified Bank #		
	New		x
Question History:	Previous NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43b.5		x
Comments: Level of Difficulty = x			

## RO Question 18

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0016	
Title:  Decay Heat Curve		Use Category: In-Field Reference	
		Revision: 9	Page: 1 of 7

# DECAY HEAT CURVE

Functional Location (1J11)

Plant Data Book

Effective Date: 1-29-09

Preparer: Patrick Curran / 11-24-08  
Date

## RO Question 18

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0016	
Title:  Decay Heat Curve		Use Category: In-Field Reference	
		Revision: 9	Page: 2 of 7

### 1.0 REFERENCES

#### 1.1 Discretionary

Calculation FM-060, Decay Heat Analysis for Perry RFO12

#### 1.2 Obligations

None

Commitments addressed in this document:

None

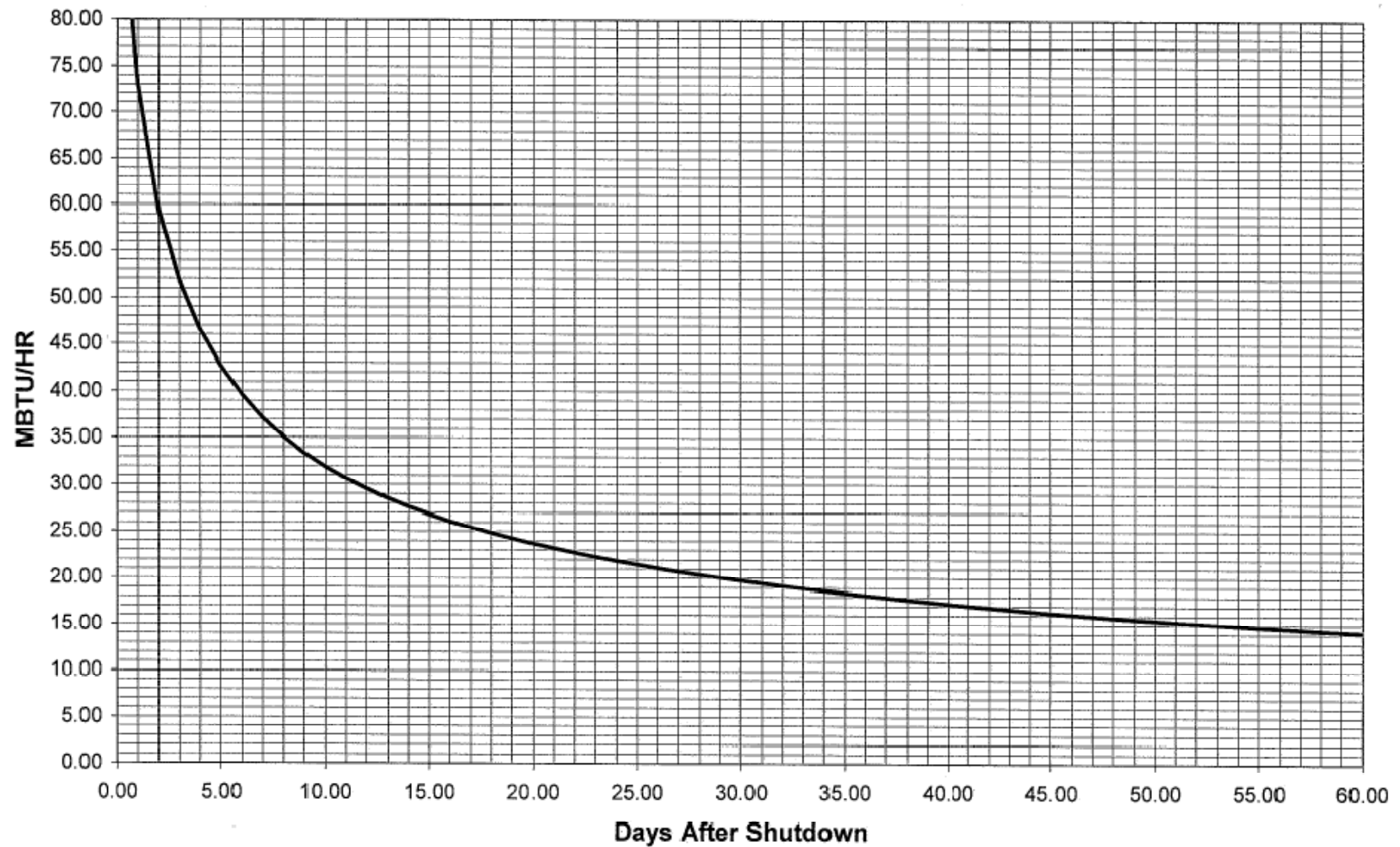
### 2.0 SCOPE OF REVISION

Rev. 9      1.      Update Decay Heat Curves for RFO12.

# RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0016	
Title: Decay Heat Curve		Use Category: In-Field Reference	
		Revision: 9	Page: 3 of 7

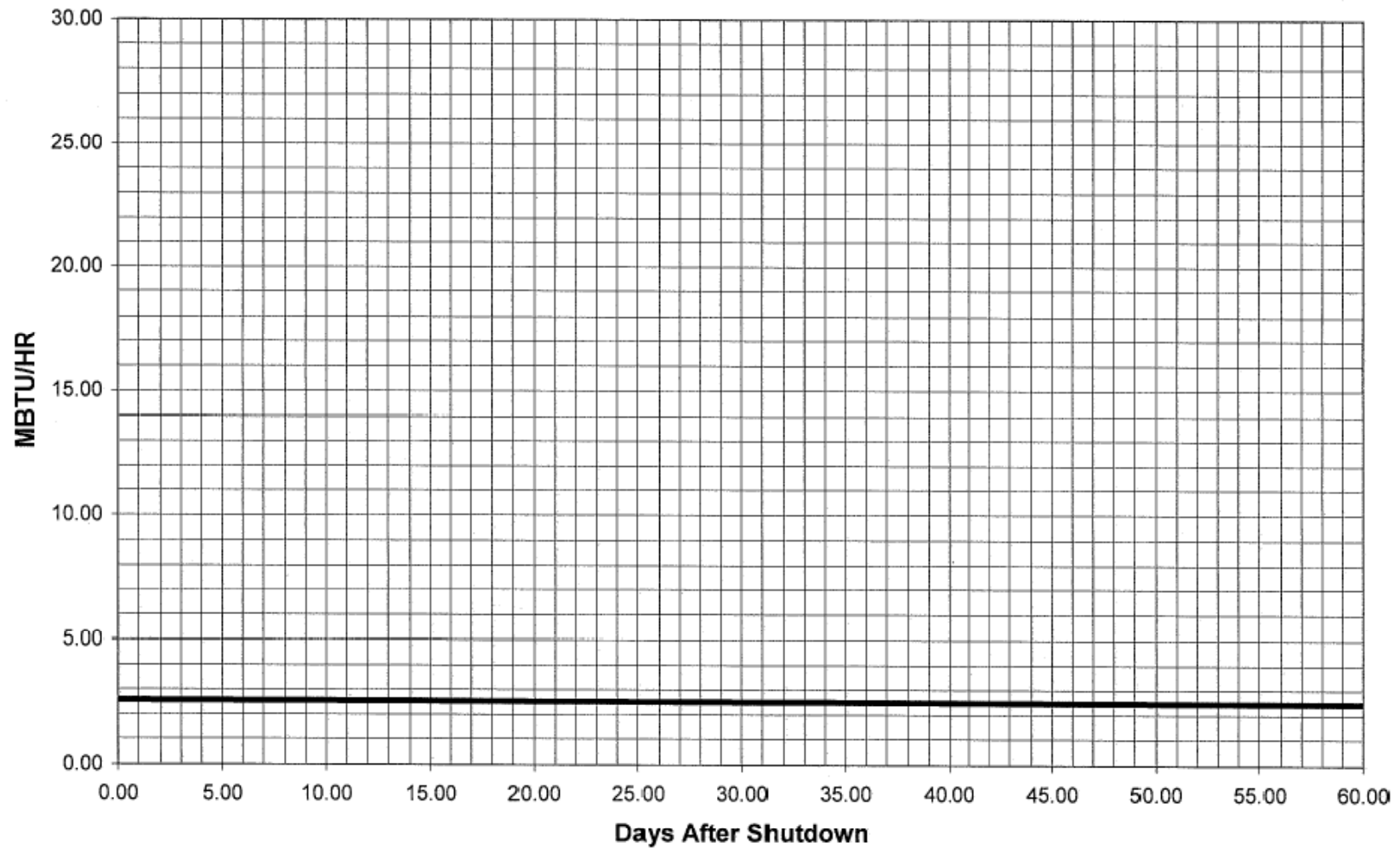
## RFO12 In-Vessel Decay Heat Before Fuel Discharge



## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0016	
Title: Decay Heat Curve		Use Category: In-Field Reference	
		Revision: 9	Page: 4 of 7

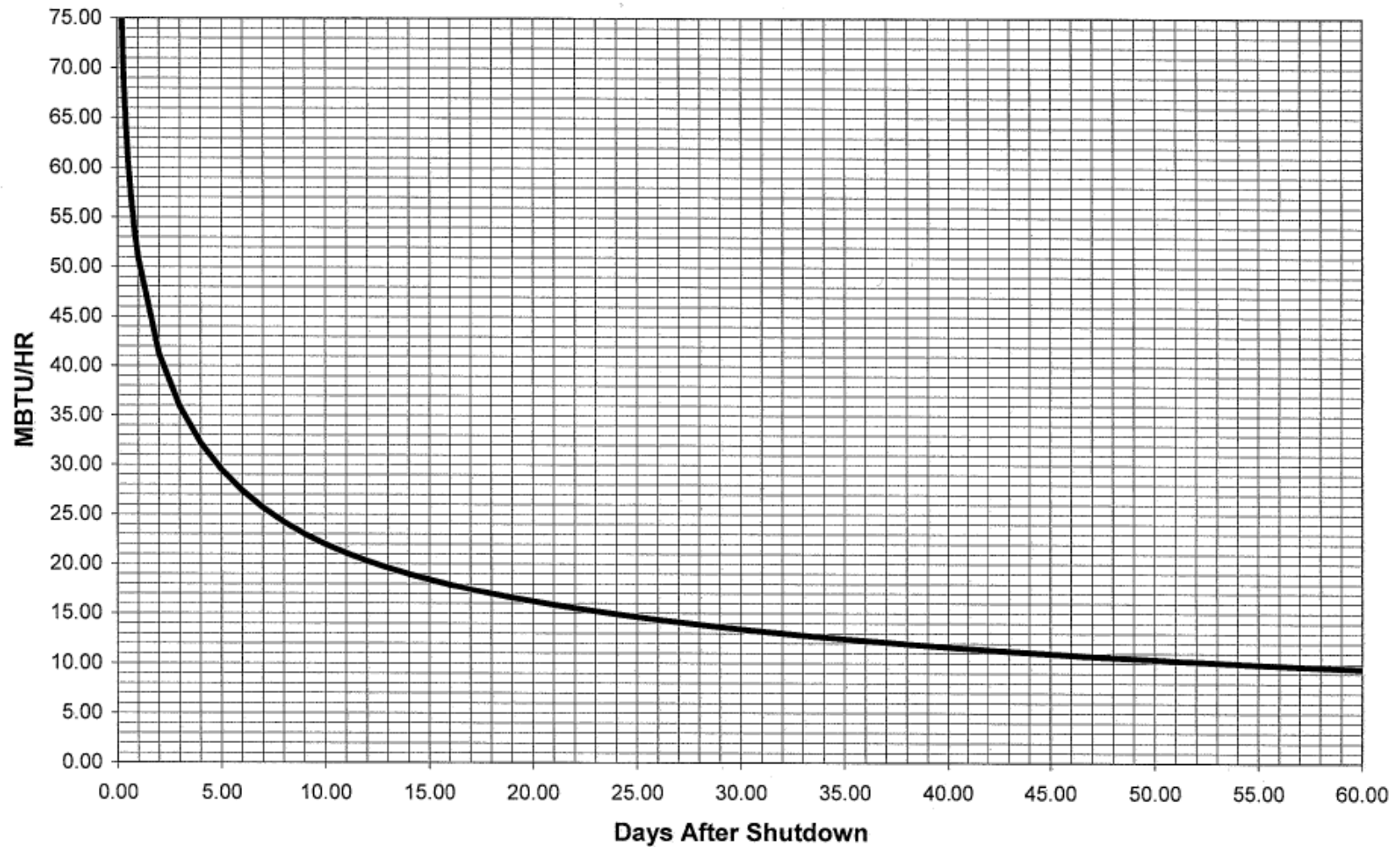
### RFO12 Fuel Pool Decay Heat Before Fuel Discharge



# RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0016	
Title: Decay Heat Curve		Use Category: In-Field Reference	
		Revision: 9	Page: 5 of 7

RFO12 In-Vessel Decay Heat After 280 Fuel Bundle Discharge

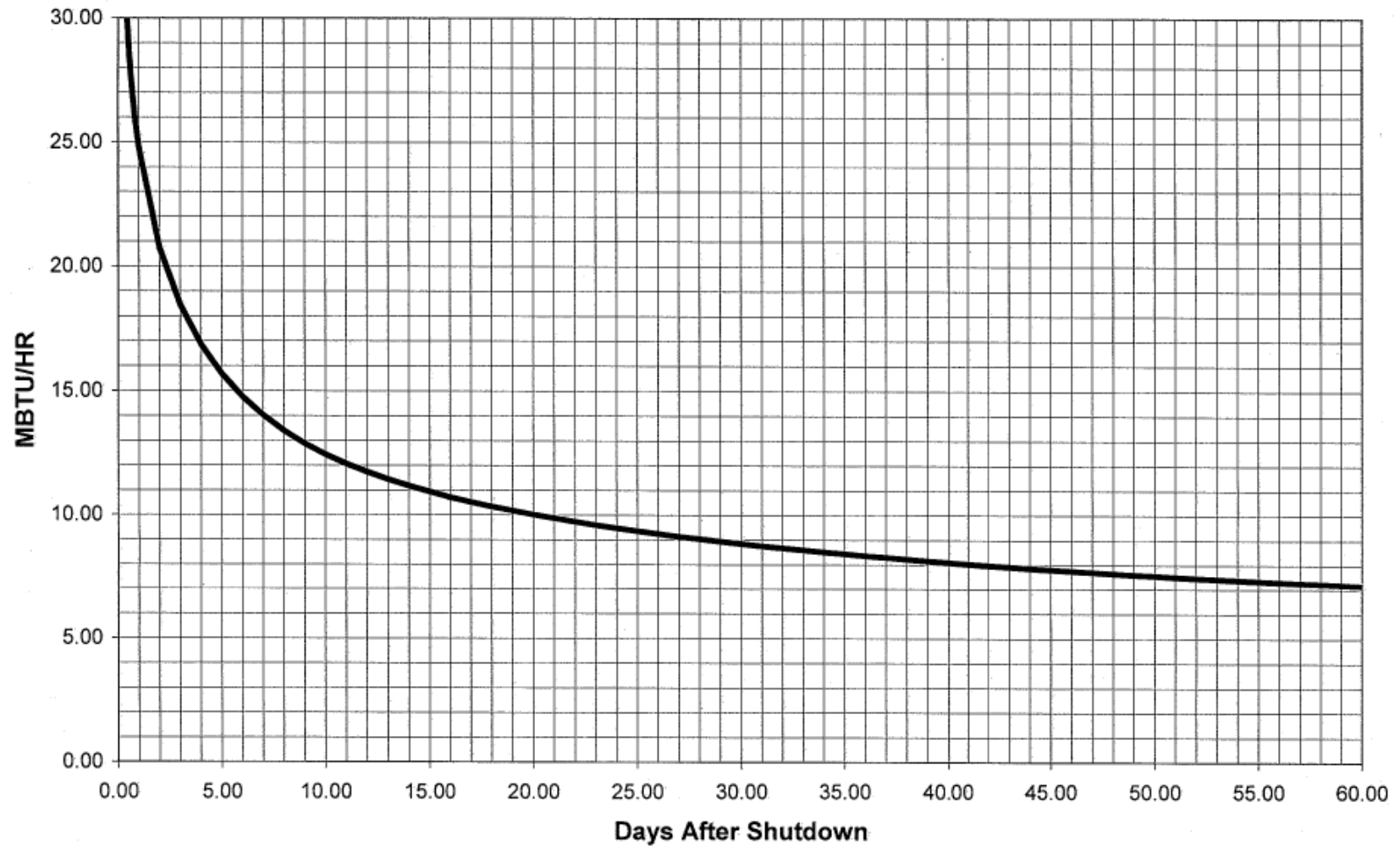




# RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0016	
Title: Decay Heat Curve		Use Category: In-Field Reference	
		Revision: 9	Page: 6 of 7

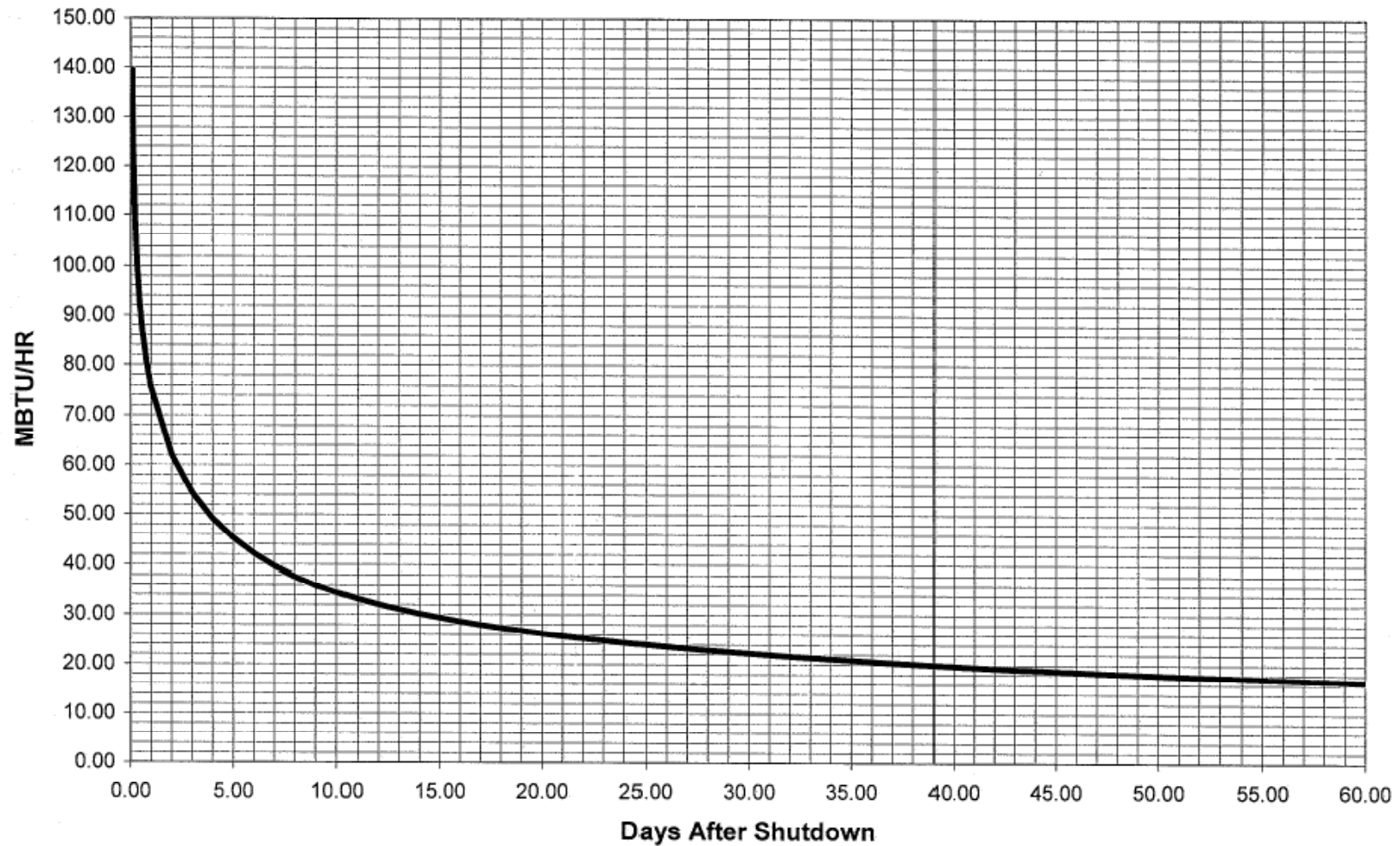
RFO12 Fuel Pool Decay Heat After Adding 280 Fuel Discharged Bundles



# RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0016	
Title: Decay Heat Curve		Use Category: In-Field Reference	
		Revision: 9	Page: 7 of 7

RFO12 Fuel Pool Decay Heat After Adding 748 Discharged Fuel Bundles



## RO Question 18

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0017	
Title:  Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 1 of 19

# POOL HEATUP CURVES

Functional Location (J11)

Plant Data Book

Effective Date: 2-18-09

Preparer: A. Widmer / 1-18-09  
Date

## RO Question 18

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0017	
Title:  Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 2 of 19

### 1.0 PURPOSE

Provide plant personnel the information with the time for either the reactor or fuel handling building pools to reach a specified temperature. Given the plant configuration, decay heat load, and location of the fuel from the previous operating cycle, the time it would take to reach bulk coolant saturation temperature with **no** Decay Heat Removal (or Spent Fuel Pool Cooling) systems in operation. In the case of the Fuel Handling Building pools, information is provided regarding time to reach 150°F as well as 212°F.

### 2.0 REFERENCES

#### 2.1 Discretionary

Calculation G41-038 Revision 10

Calculation G41-042 Revision 10

Plant Data Book PDB-A0016

PAP-1925 Shutdown Defense In Depth Assessment and Management

NSAC/175L - EPRI Outage Risk Assessment and Management (ORAM) Program "Safety Assessment of BWR Risk During Shutdown Operations"

#### 2.2 Obligations

None

Commitments addressed in this document:

None

## RO Question 18

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0017	
Title:  Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 3 of 19

### 3.0 DETAILS

In the event that Decay Heat Removal system is lost, the following graphs can be utilized to determine the time it takes to get to a specified temperature. The graphs are decay heat vs. time. The decay heat value may be determined from Plant Data Book PDB-A0016 for a given plant configuration and day after shutdown. The decay heat value can be used herein to determine the applicable heat-up time given the initial water temperature is known.

### 4.0 SCOPE OF REVISION

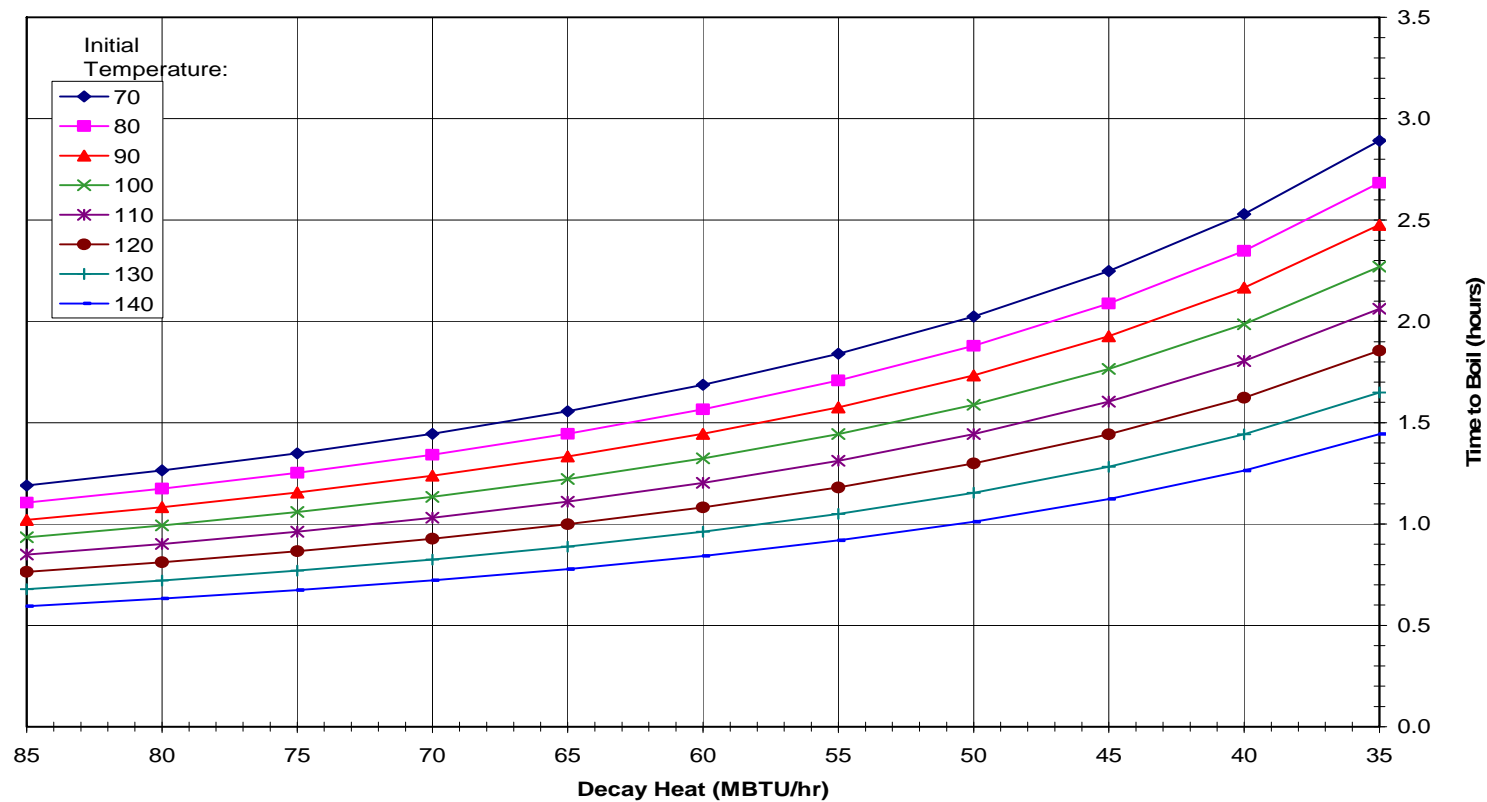
- |         |    |  |
|---------|----|--|
| Rev. 10 | 1. | Updated curves to reflect decay heat vs. time. |
|         | 2. | Revised format in accordance with NOP-SS-3007. |

## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 4 of 19

### TIME-TO-BOIL CURVES FOR REACTOR VESSEL

Reactor Vessel @ Normal Water Level

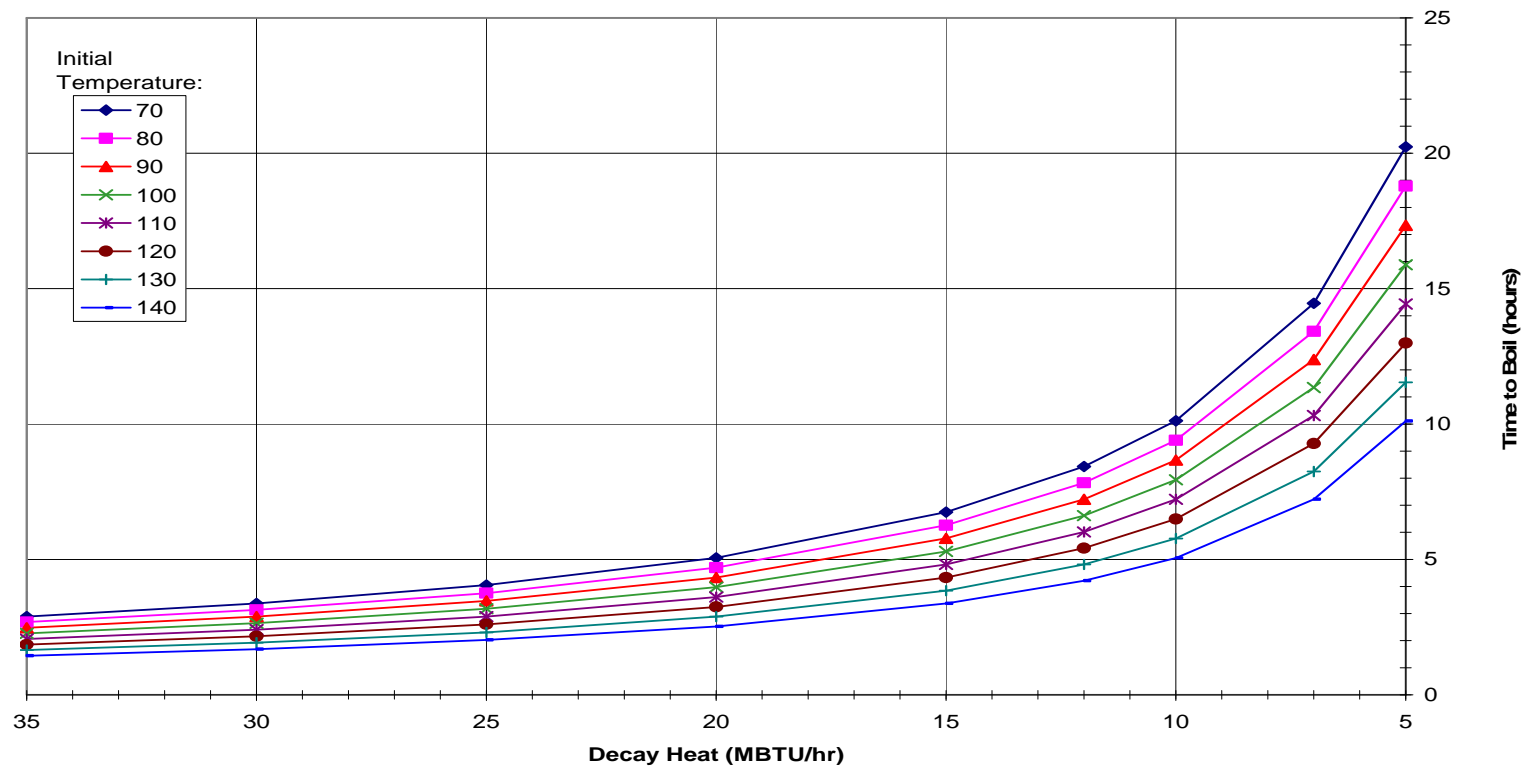


## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 5 of 19

### TIME-TO-BOIL CURVES FOR REACTOR VESSEL

Reactor Vessel @ Normal Water Level

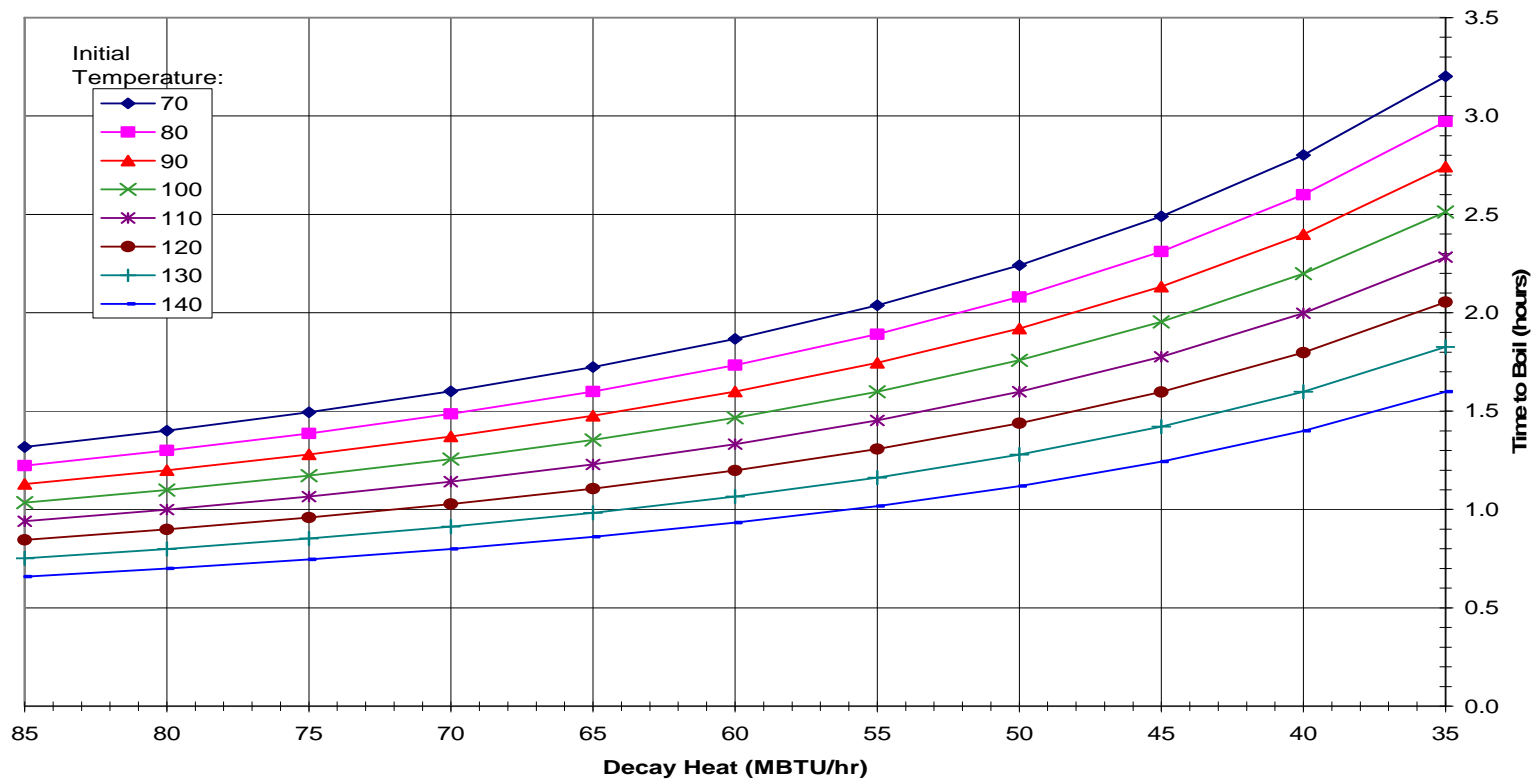


## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 6 of 19

### TIME-TO-BOIL CURVES FOR REACTOR VESSEL

Water Level at 250" above TAF



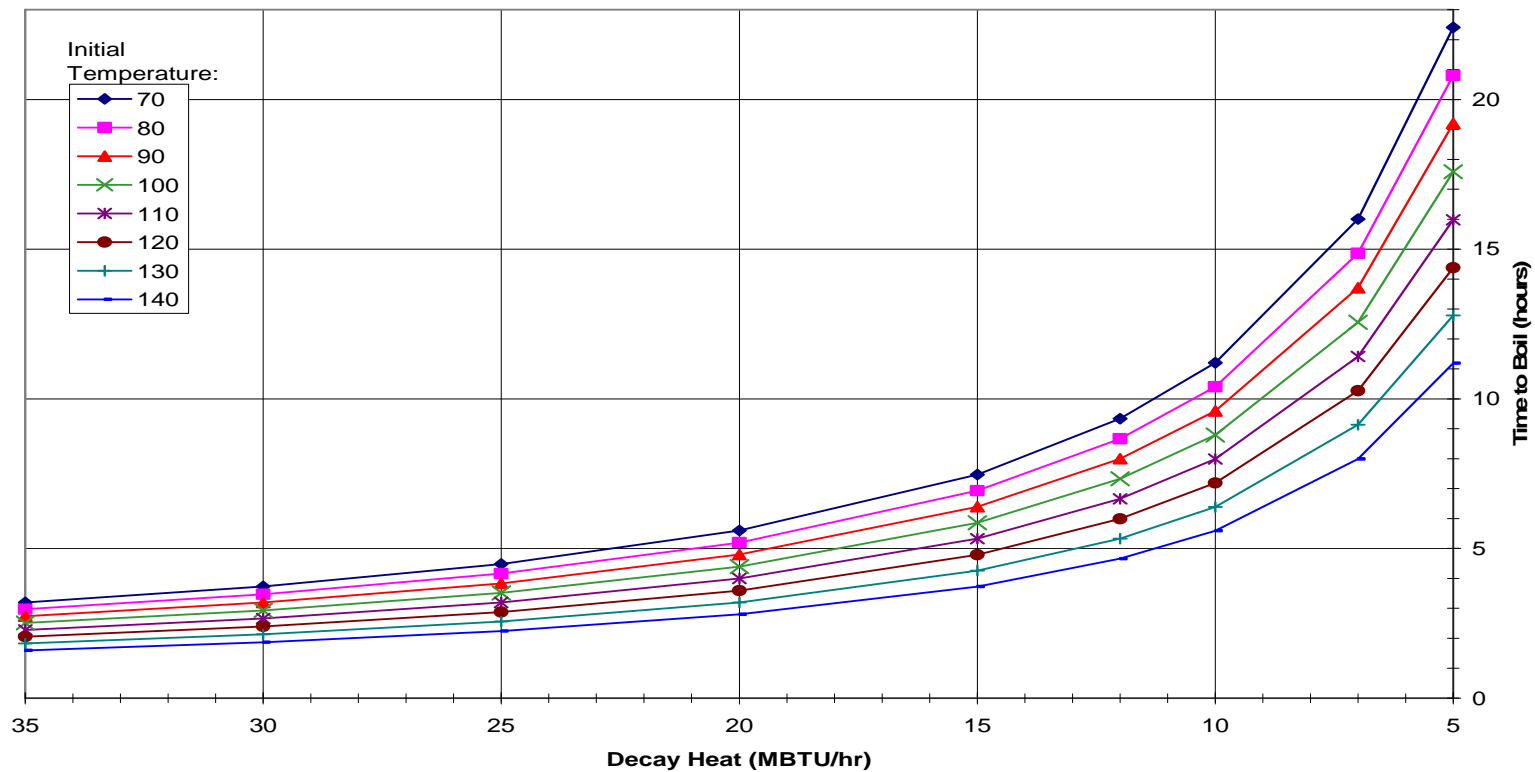


## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 7 of 19

### TIME-TO-BOIL CURVES FOR REACTOR VESSEL

Water Level at 250" above TAF

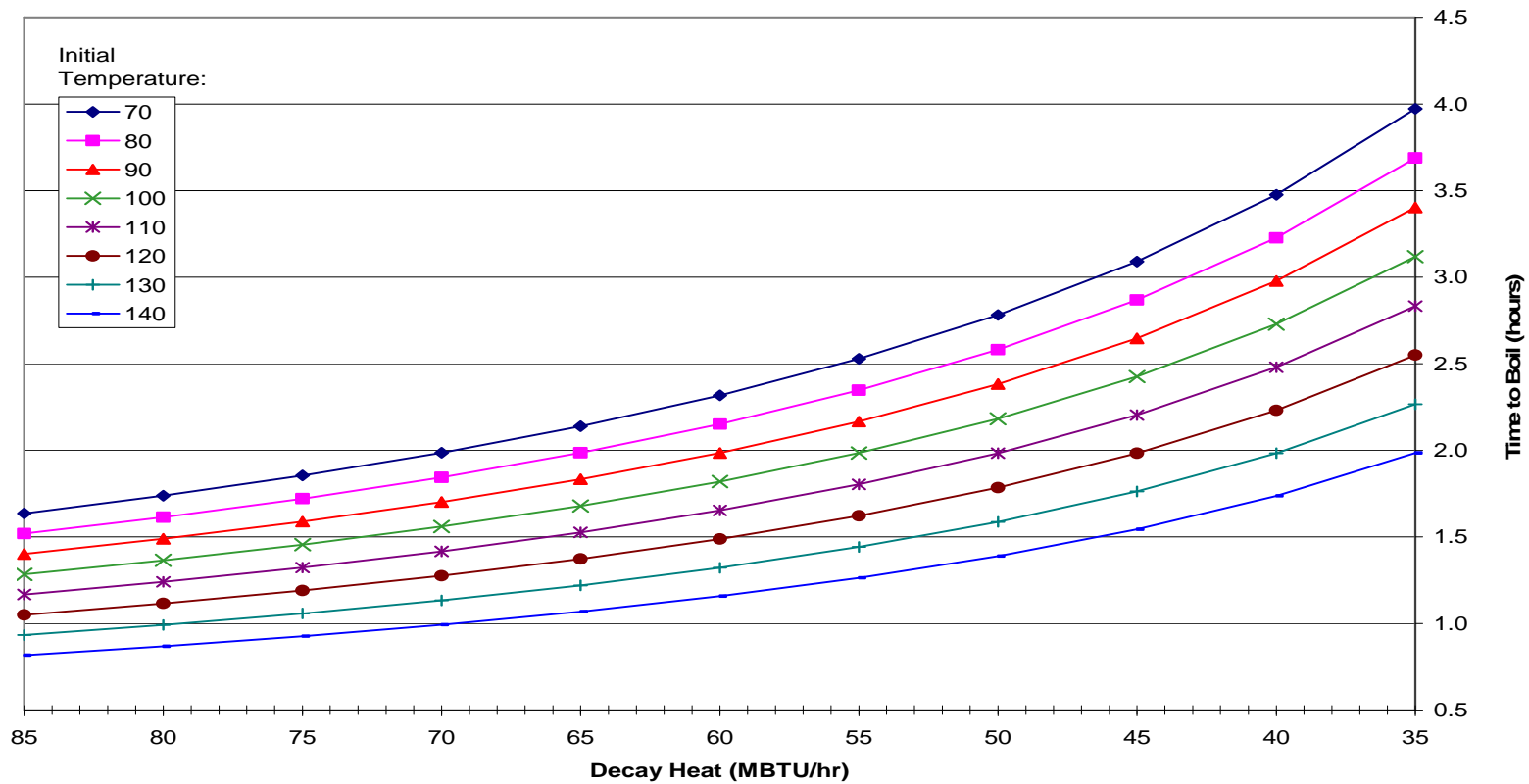


# RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 8 of 19

## TIME-TO-BOIL CURVES FOR REACTOR VESSEL

Water Level at Reactor Flange

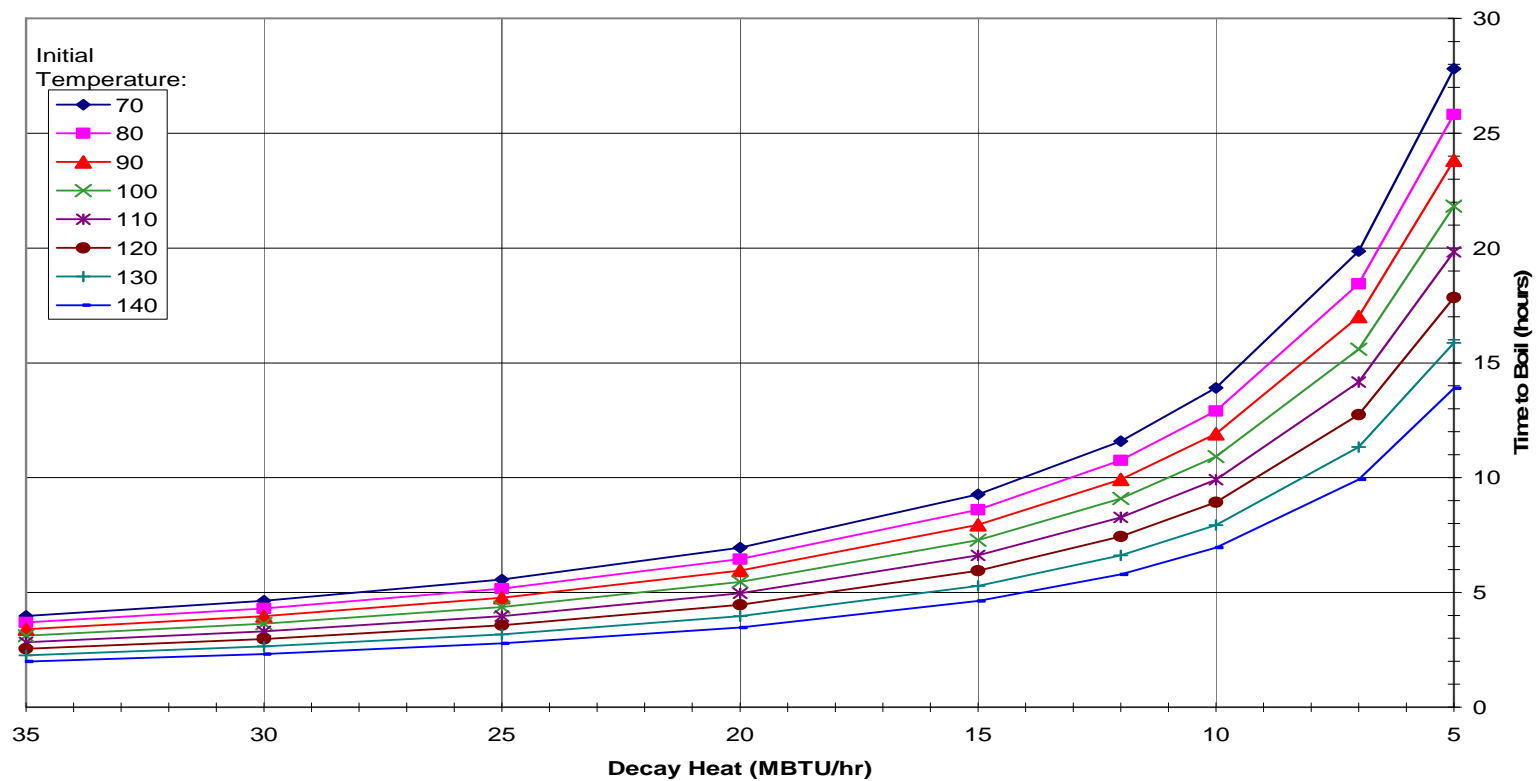


## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 9 of 19

### TIME-TO-BOIL CURVES FOR REACTOR VESSEL

Water Level at Reactor Flange

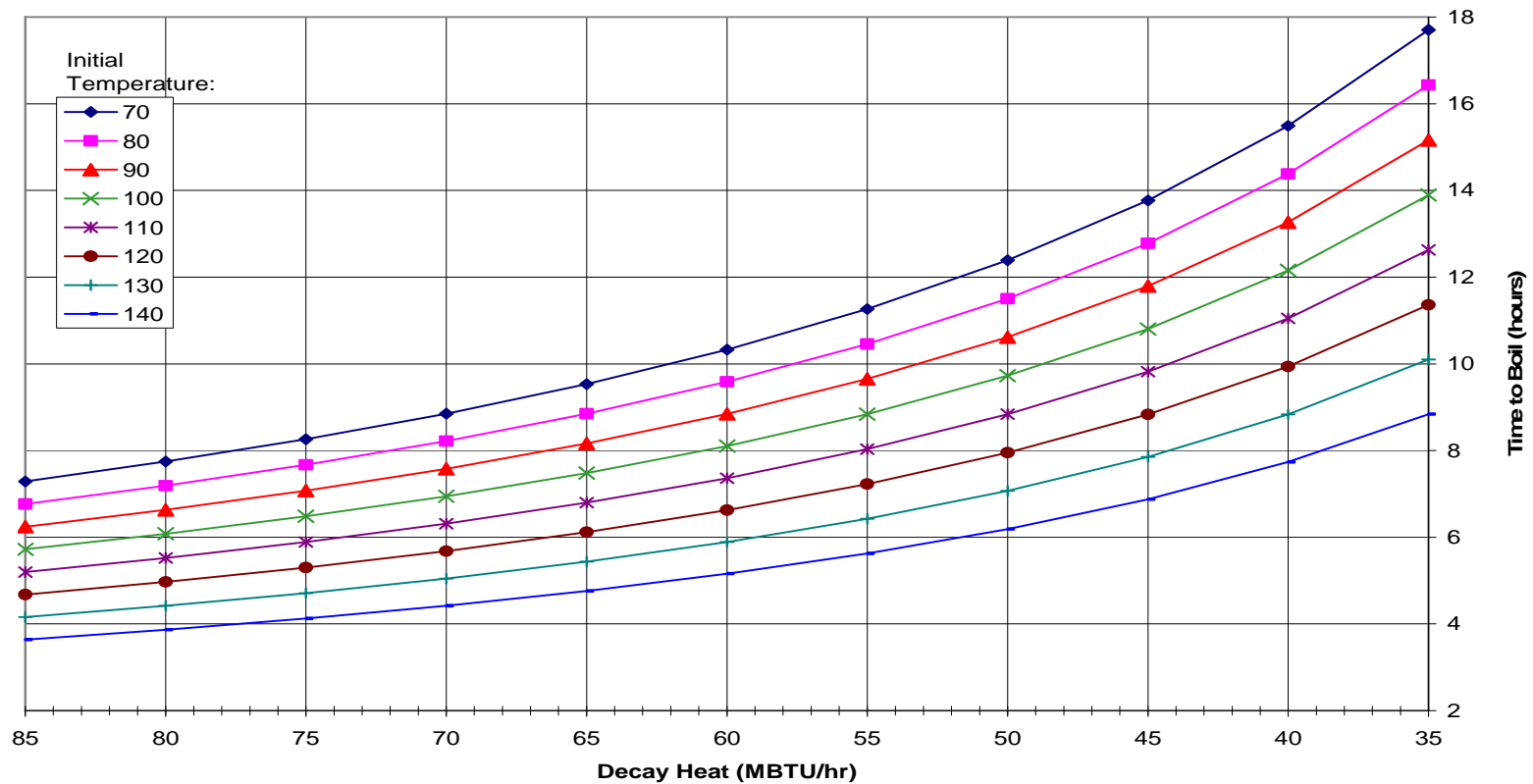


# RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 10 of 19

## TIME-TO-BOIL CURVES FOR REACTOR VESSEL

Water Level 23' above Reactor Flange

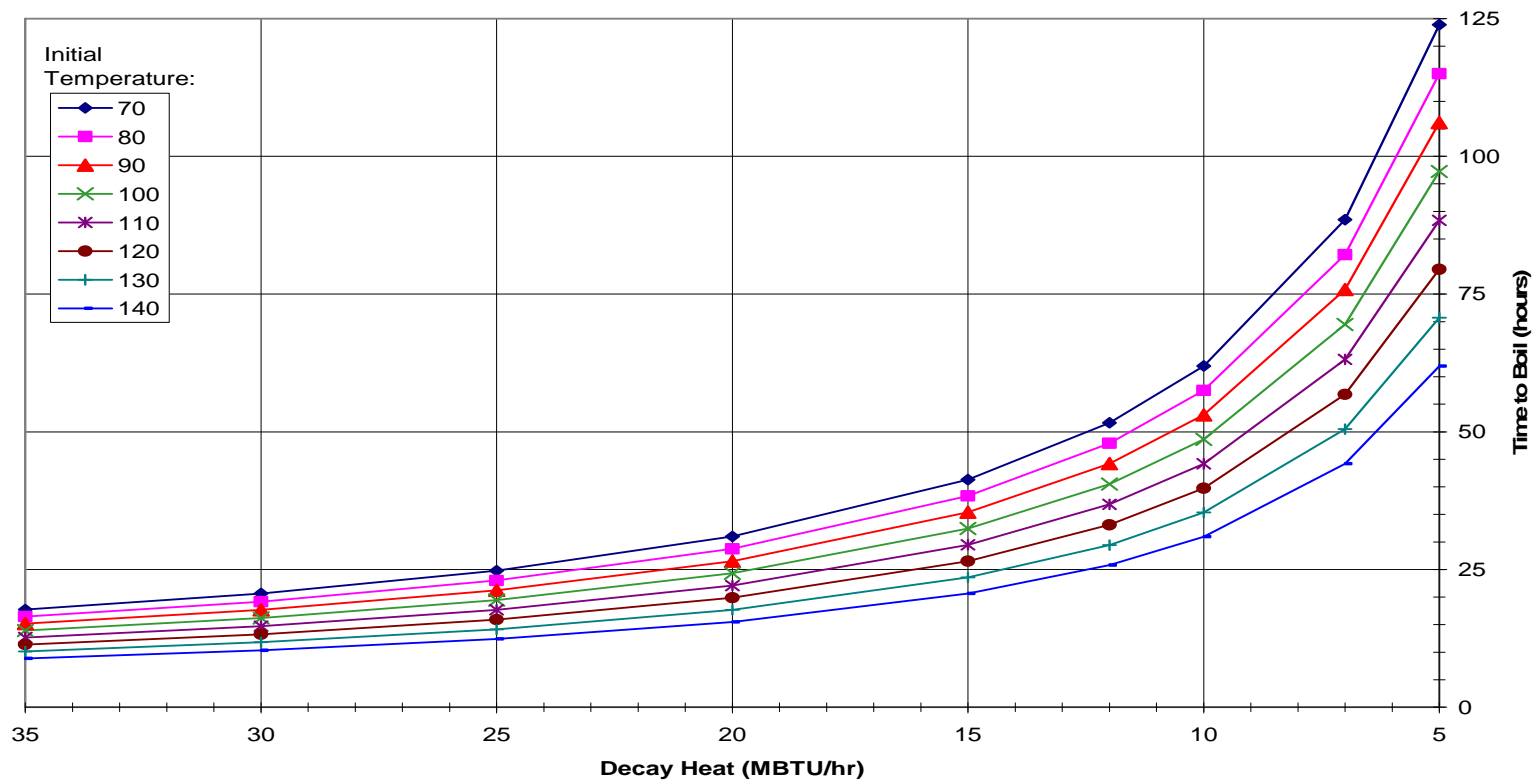


## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 11 of 19

### TIME-TO-BOIL CURVES FOR REACTOR VESSEL

Water Level 23' above Reactor Flange

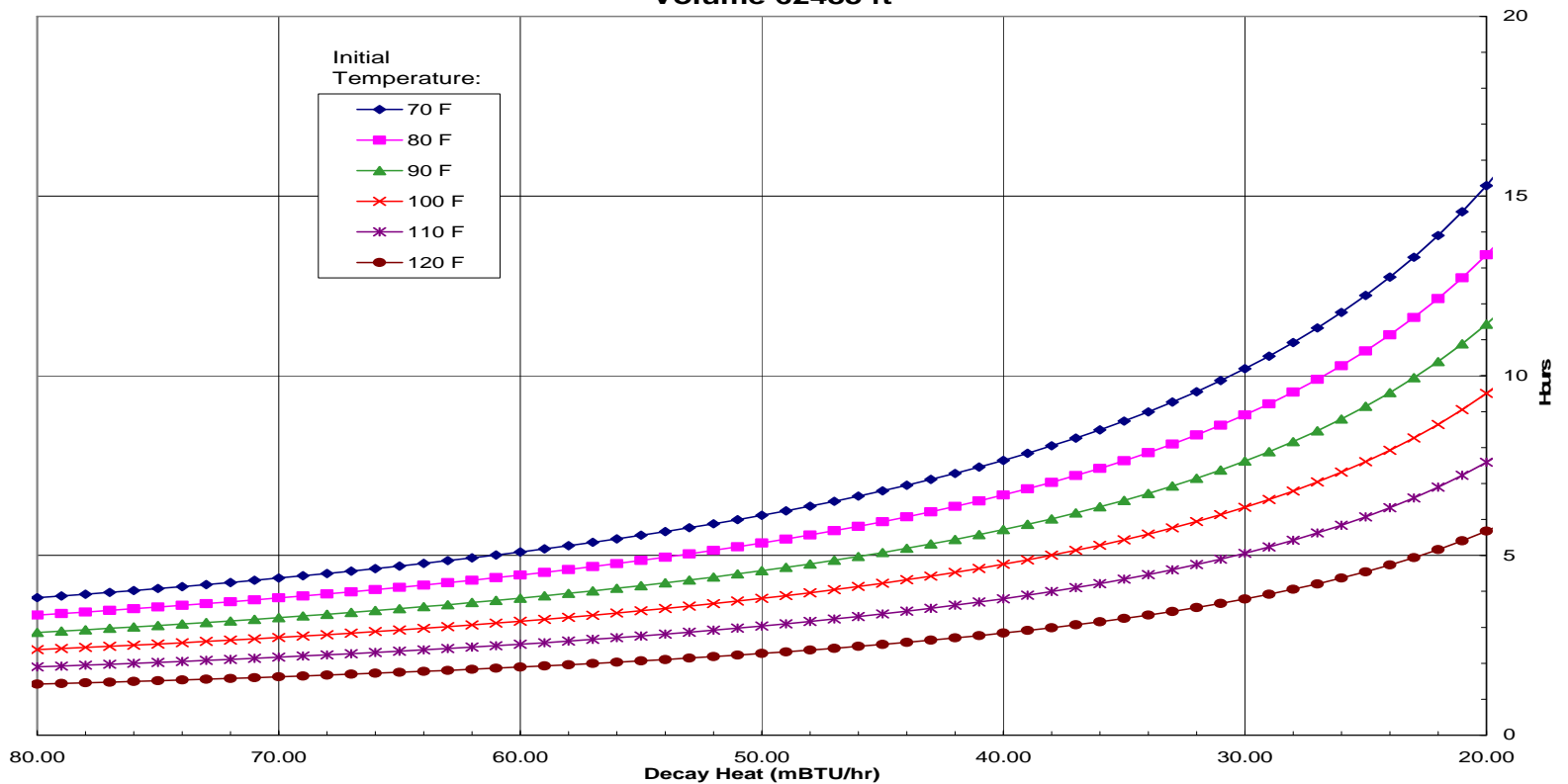


## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 12 of 19

### TIME TO HEAT FUEL HANDLING BUILDING POOLS TO 150°F

**Time Needed to Heat FHB Pools in Communication to 150 °F**  
**Volume 62485 ft<sup>3</sup>**

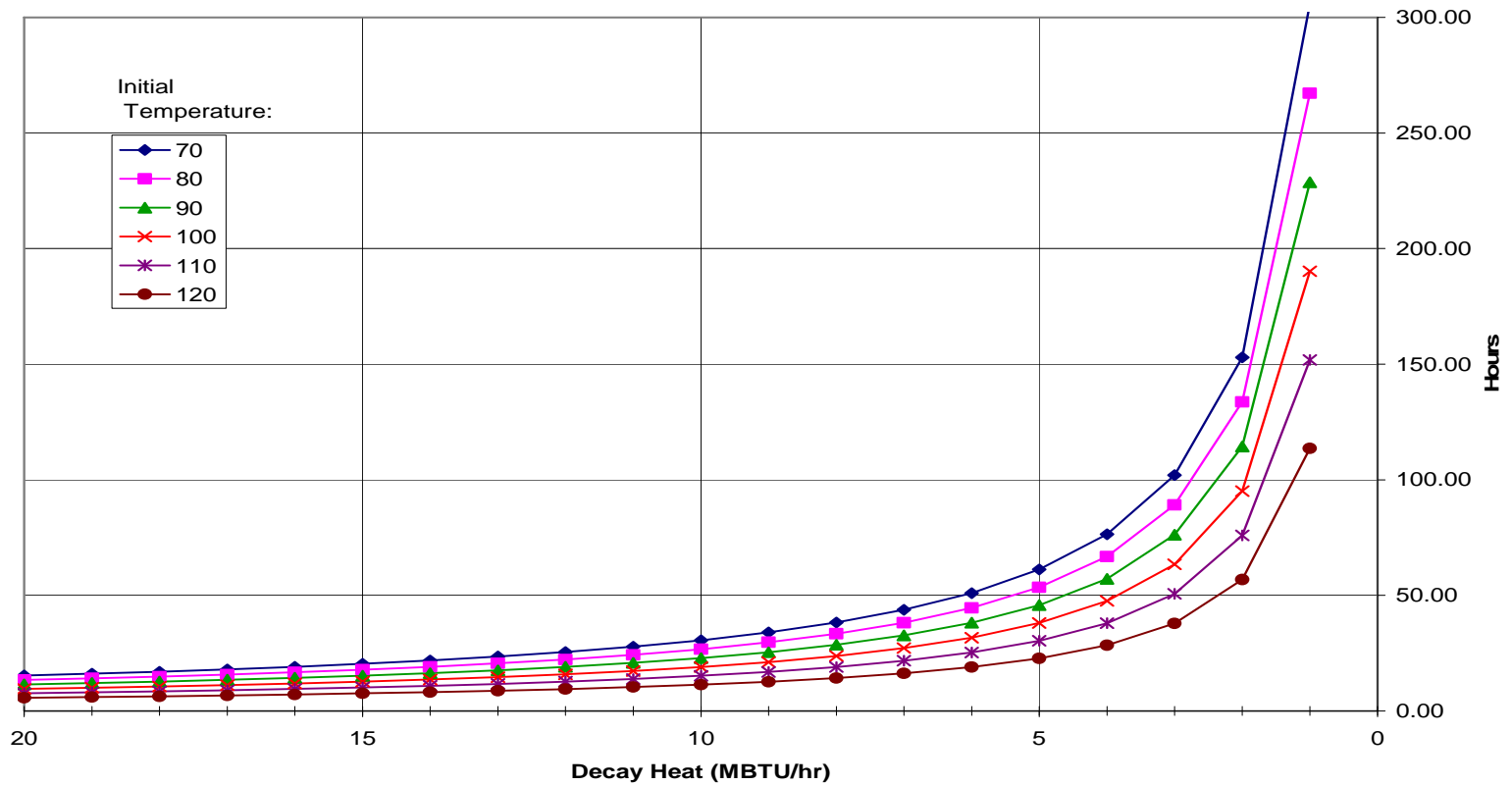


## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 13 of 19

### TIME TO HEAT FUEL HANDLING BUILDING POOLS TO 150°F

Time to Heat Fuel Pools in Communication to 150°F  
Volume 62485 ft<sup>3</sup>

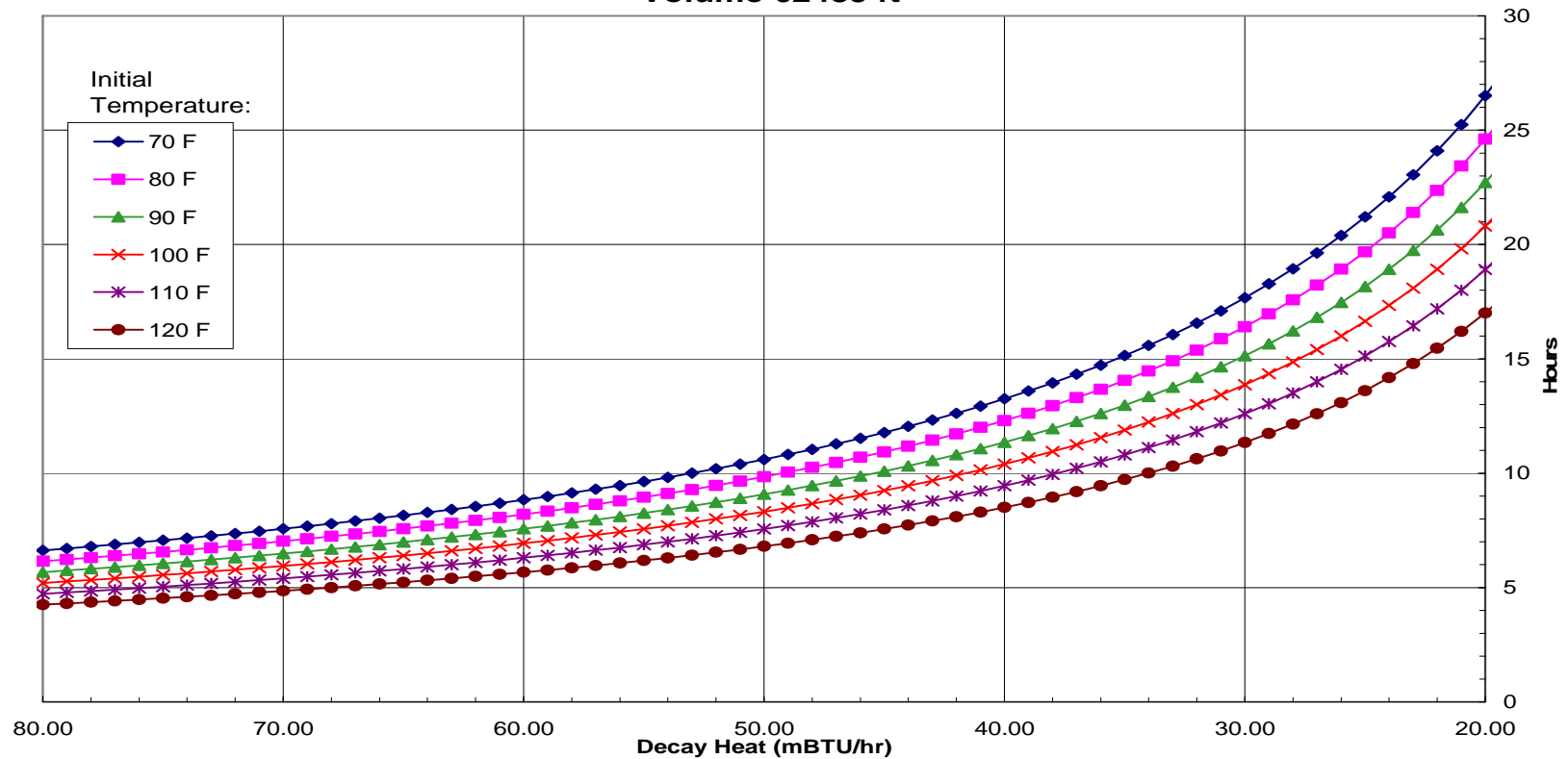


# RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 14 of 19

## TIME TO HEAT FUEL HANDLING BUILDING POOLS TO 212°F

**Time Needed to Heat FHB Pools in Communication to 212 °F**  
**Volume 62485 ft<sup>3</sup>**



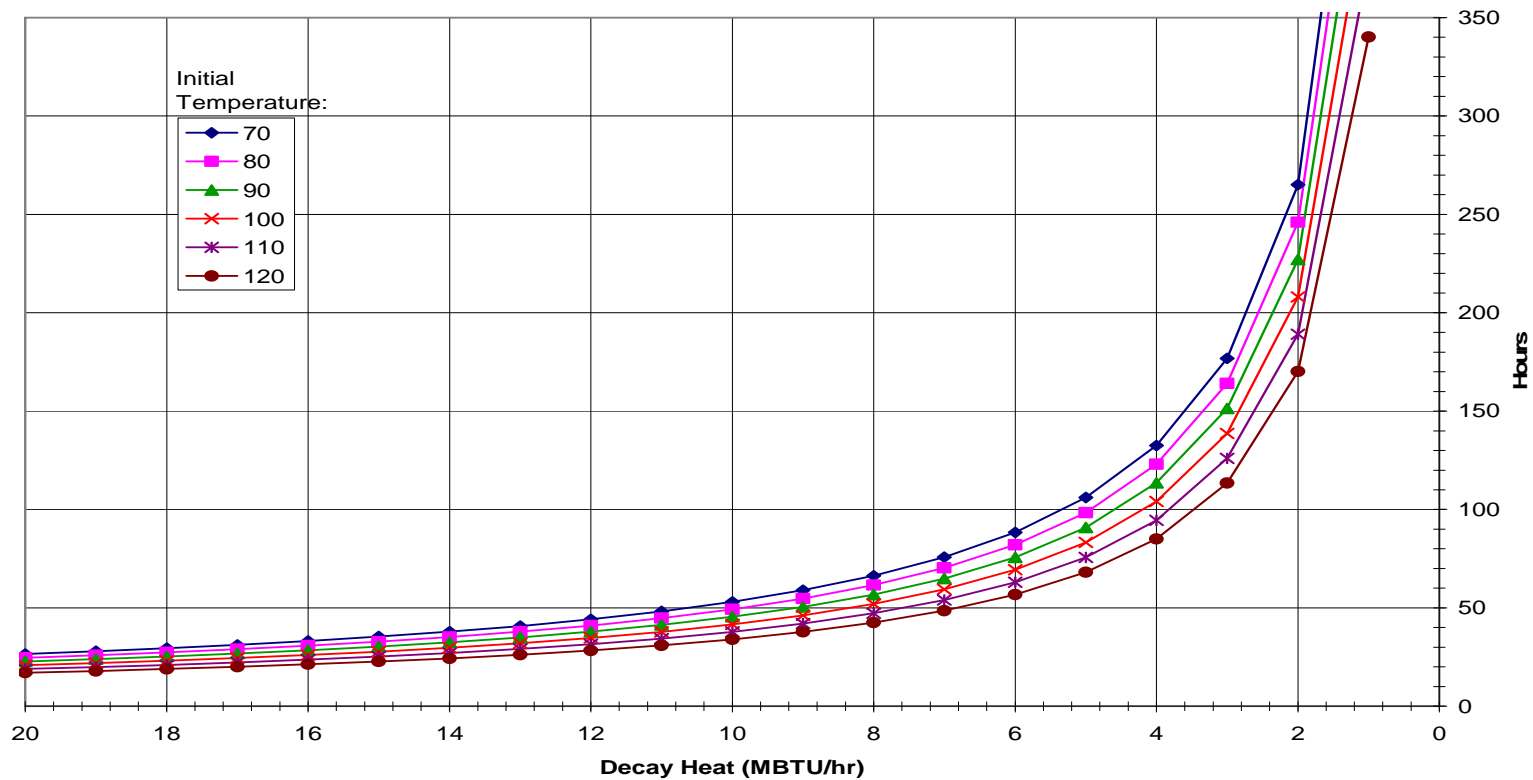


## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 15 of 19

### TIME TO HEAT FUEL HANDLING BUILDING POOLS TO 212°F

Time to Heat FHB Pools in Communication to 212°F  
Volume 62485 ft<sup>3</sup>

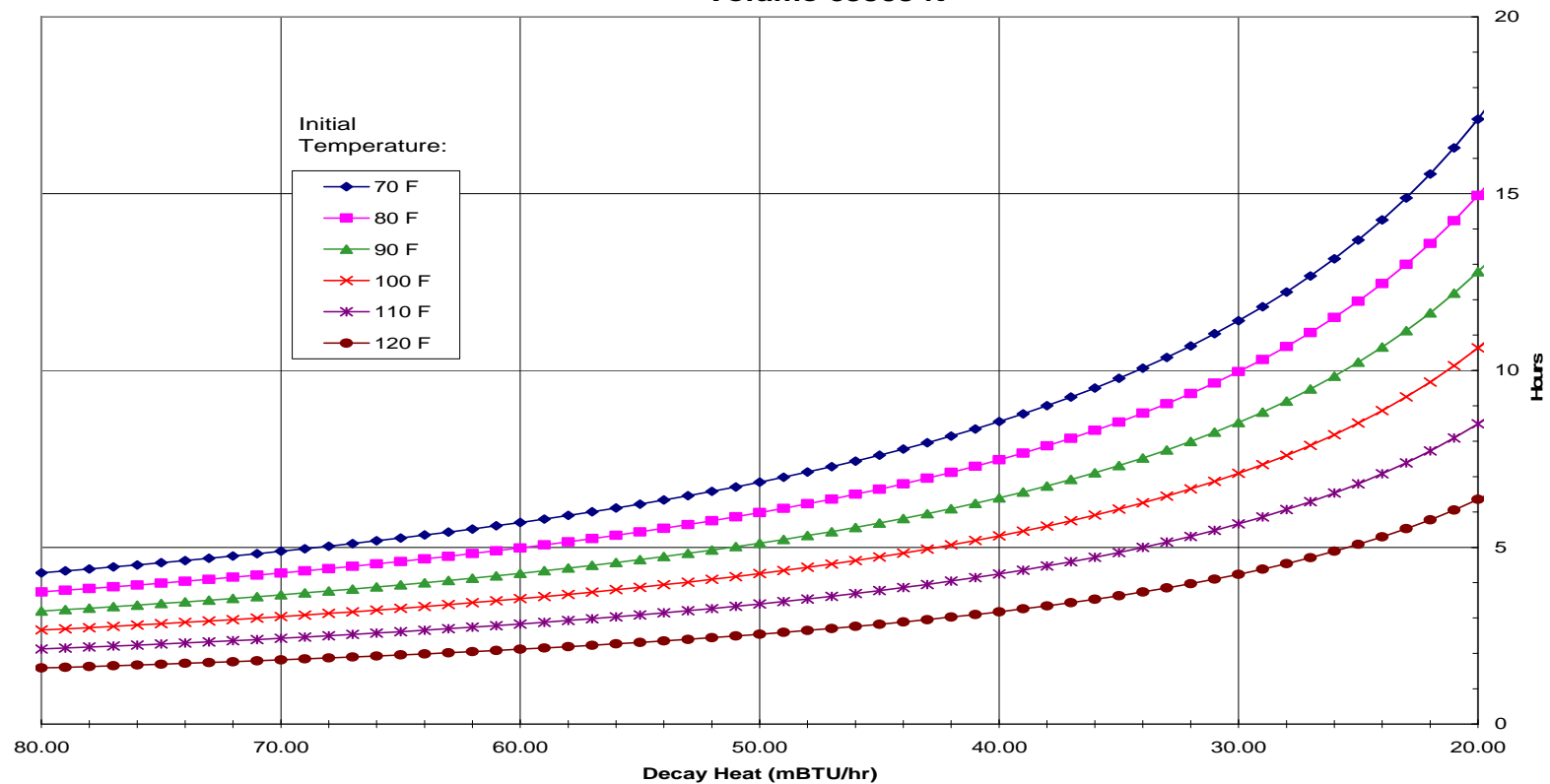


# RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 16 of 19

## TIME TO HEAT FUEL HANDLING BUILDING POOLS (INCLUDING CASK POOL) TO 150°F

Time Needed to Heat FHB Pools Including the Cask Pool to 150 °F  
Volume 69905 ft<sup>3</sup>

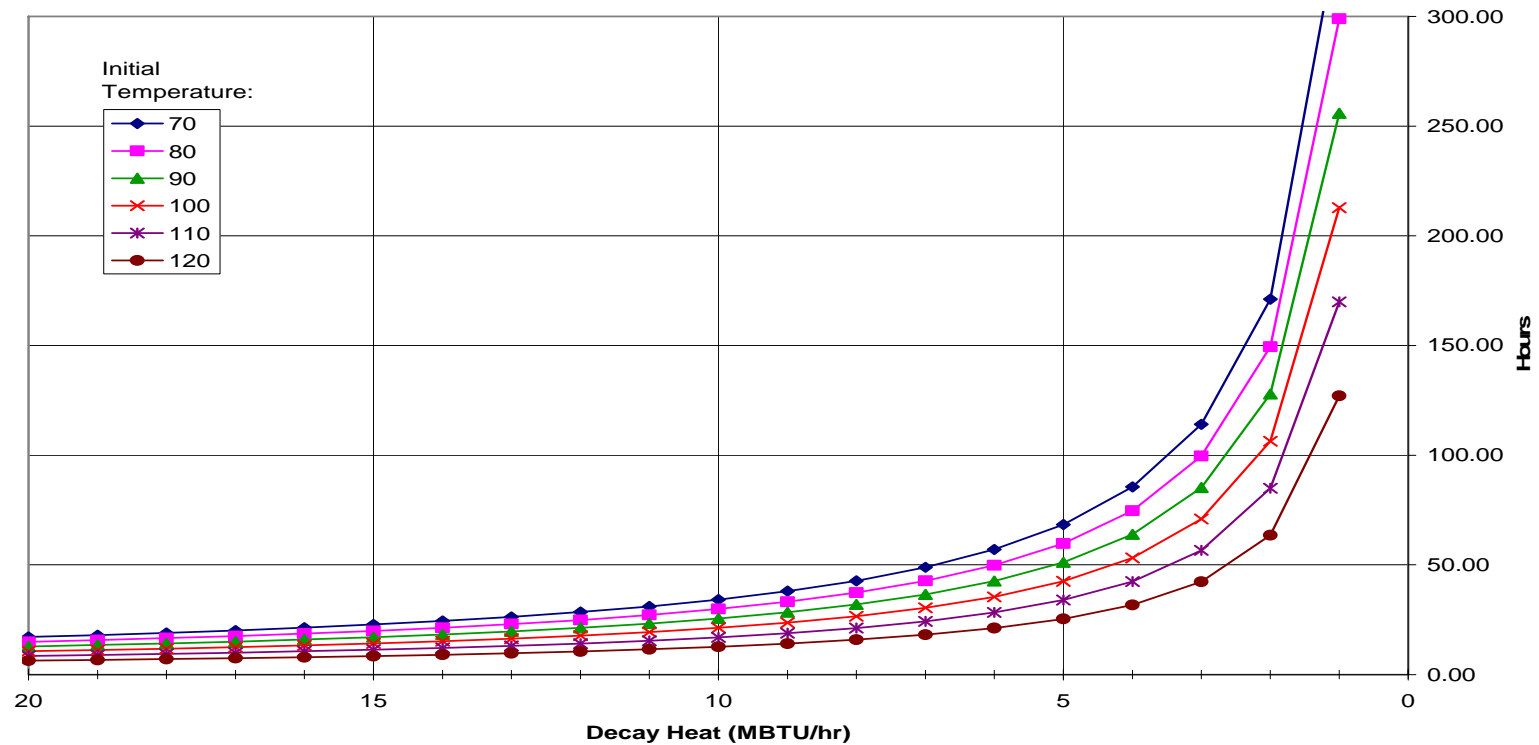


# RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 17 of 19

## TIME TO HEAT FUEL HANDLING BUILDING POOLS (INCLUDING CASK POOL) TO 150°F

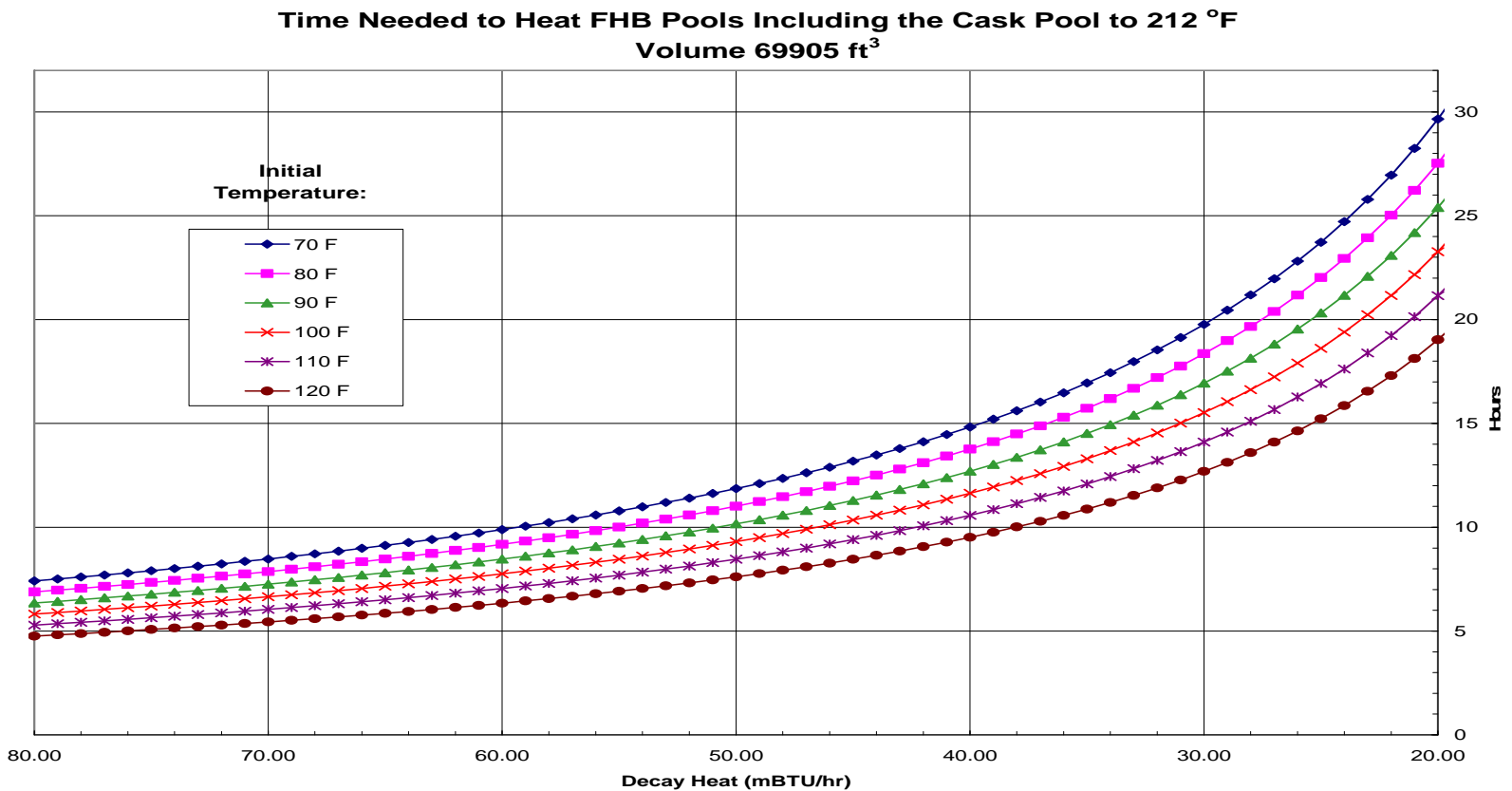
Time needed to Heat FHB Pools Including Cask Pool to 150°F  
Volume 69905 ft<sup>3</sup>



## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 18 of 19

### TIME TO HEAT FUEL HANDLING BUILDING POOLS (INCLUDING CASK POOL) TO 212°F

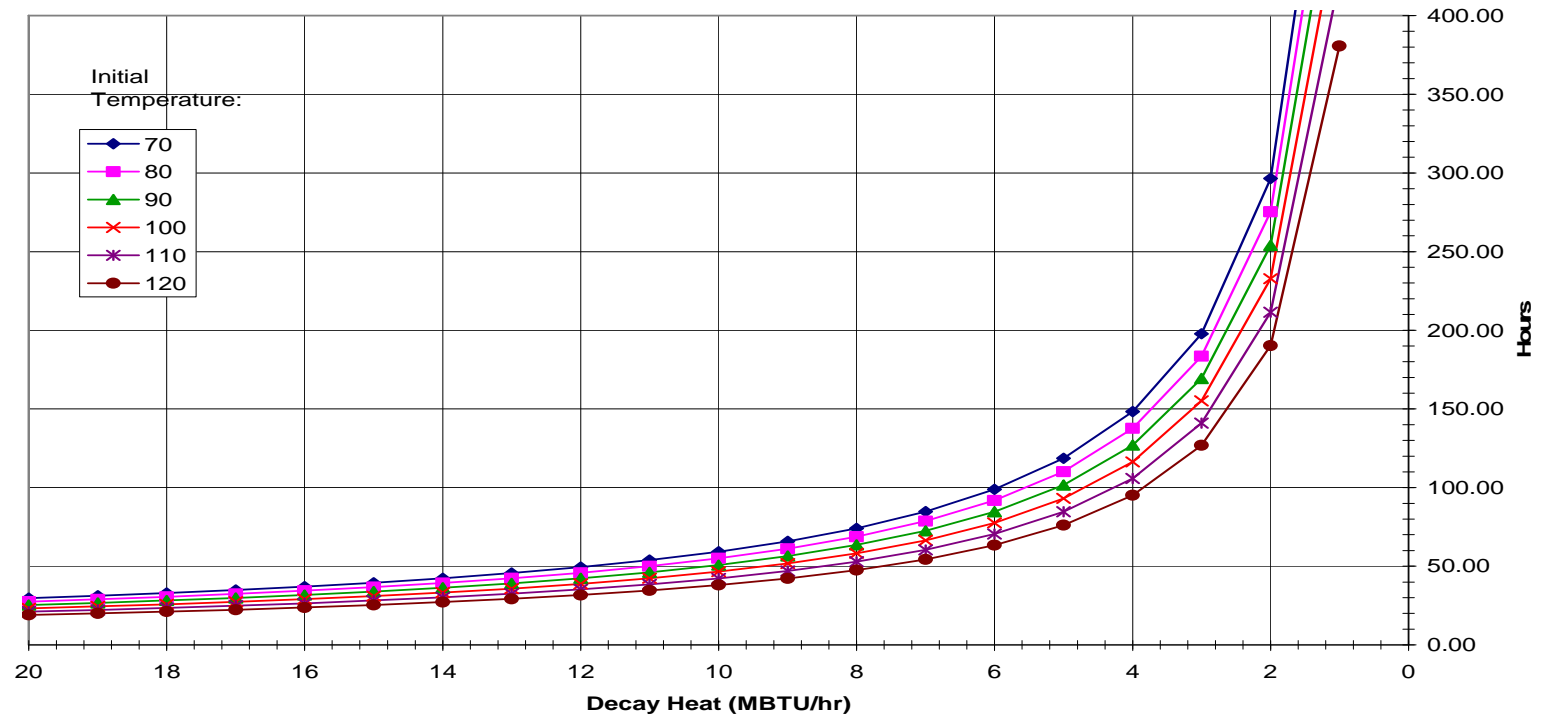


## RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017	
Title: Pool Heatup Curves		Use Category: In-Field Reference	
		Revision: 10	Page: 19 of 19

### TIME TO HEAT FUEL HANDLING BUILDING POOLS (INCLUDING CASK POOL) TO 212°F

Time needed to Heat FHB Pools Including the Cask Pool to 212°F  
Volume 69905 ft<sup>3</sup>



## RO Question 48

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0012	
Title: Recirc Drive Flow vs. Total Core Flow		Use Category: General Skill Reference	
		Revision: 15	Page: 1 of 6

# RECIRC DRIVE FLOW VS. TOTAL CORE FLOW

Functional Location (B33)

Plant Data Book

Effective Date: 9-20-07

Preparer: P. Curran / 8-24-07  
Date

## RO Question 48

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0012	
Title: Recirc Drive Flow vs. Total Core Flow		Use Category: General Skill Reference	
		Revision: 15	Page: 2 of 6

### 1.0 REFERENCES

#### 1.1 Discretionary

None

#### 1.2 Obligations

SVI-F41-T3008, Reactor Recirculation System Flow Data Verification performed May 2005 and June/July 2007.

SVI-B33-T1160, Jet Pump Operability

Technical Specification SR 3.4.3.1.b

ICS/SDS Displays RFPERF and RSPERF.

Commitments addressed in this document:

None

### 2.0 SCOPE OF REVISION

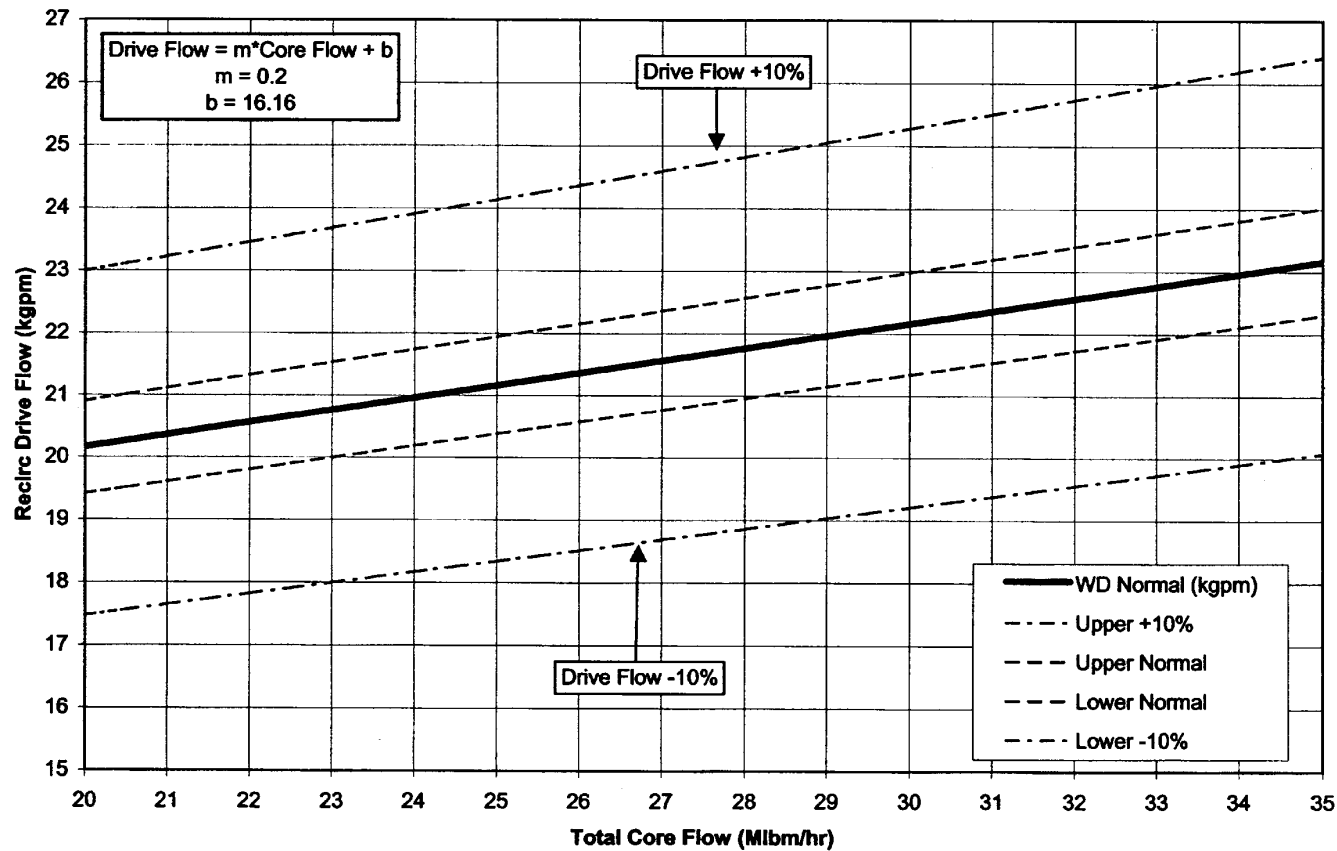
- |         |  |
|---------|--|
| Rev. 15 | <ol style="list-style-type: none"><li>1. Revised format in accordance with NOP-SS-3007.</li><li>2. Updated references to include the June/July 2007 performance of SVI-F41-T3008. There were no changes to the graphs.</li></ol> |
|---------|--|

# RO Question 48

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0012	
Title: Recirc Drive Flow vs. Total Core Flow		Use Category: General Skill Reference	
		Revision: 15	Page: 3 of 6

A0012 Slow Speed (Kgpm)

Recirc Drive Flow vs Total Core Flow Both Loops Operating in Slow Speed (kgpm)



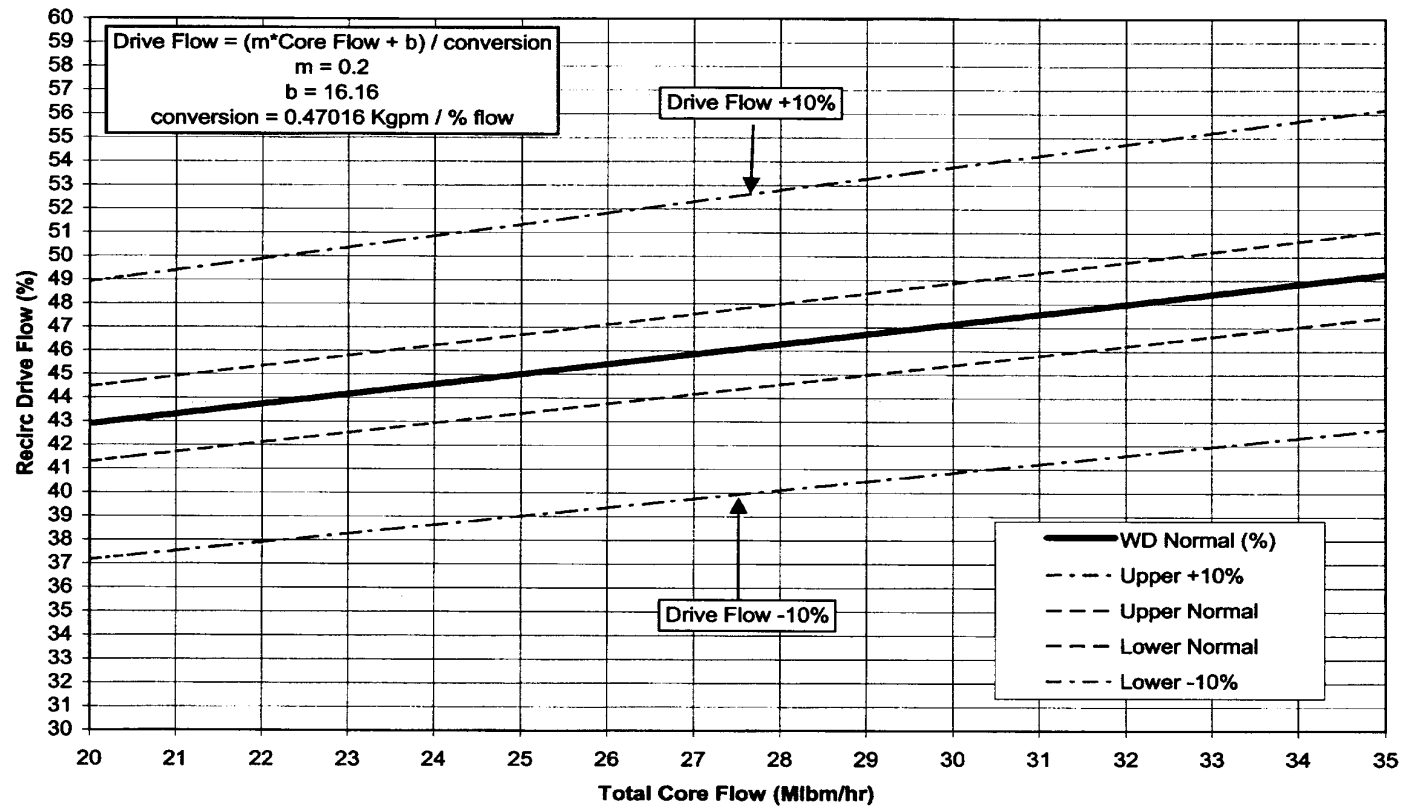


# RO Question 48

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0012	
Title: Recirc Drive Flow vs. Total Core Flow		Use Category: General Skill Reference	
		Revision: 15	Page: 4 of 6

A0012 Slow Speed (%)

**Recirc Drive Flow vs Total Core Flow Both Loops Operating in Slow Speed (%)**

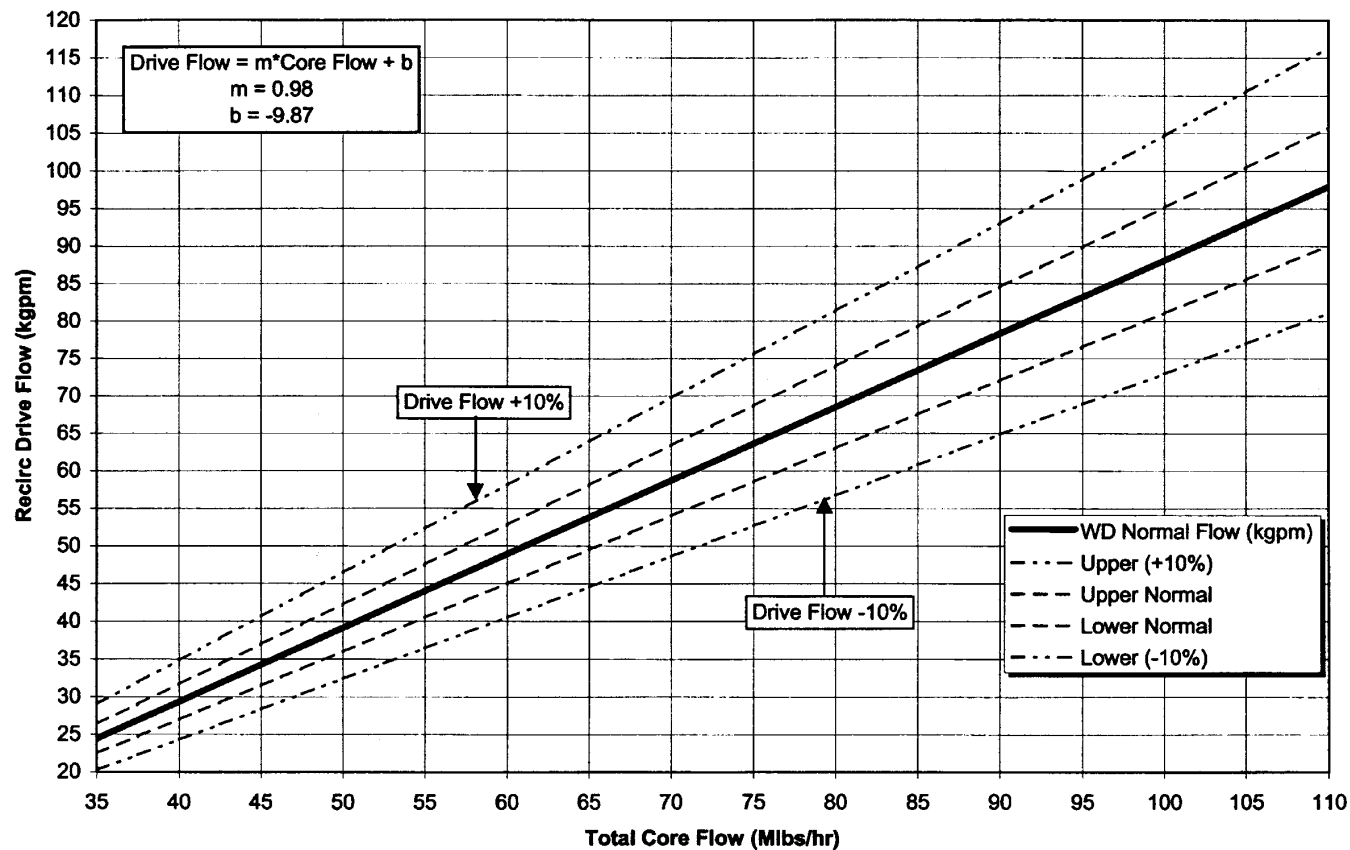


## RO Question 48

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0012	
Title: Recirc Drive Flow vs. Total Core Flow		Use Category: General Skill Reference	
		Revision: 15	Page: 5 of 6

A0012 Fast Speed (Kgpm)

**Recirc Drive Flow vs Total Core Flow, Both Loops Operating in Fast Speed (kgpm)**

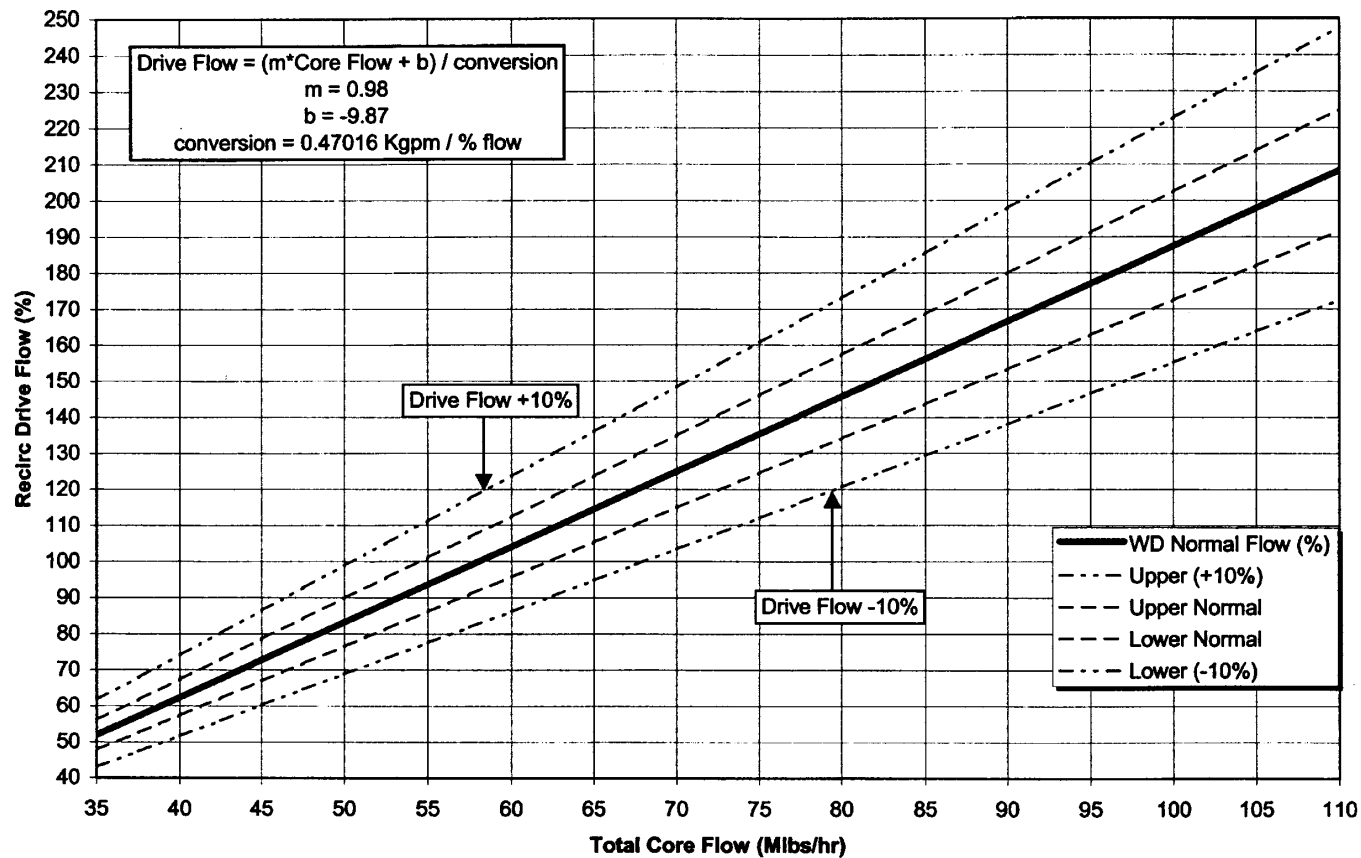


## RO Question 48

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0012	
Title: Recirc Drive Flow vs. Total Core Flow		Use Category: General Skill Reference	
		Revision: 15	Page: 6 of 6

A0012 Fast Speed (%)

**Recirc Drive Flow vs Total Core Flow, Both Loops Operating in Fast Speed (%)**





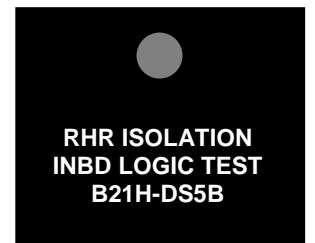
Light on



Light on



Light off



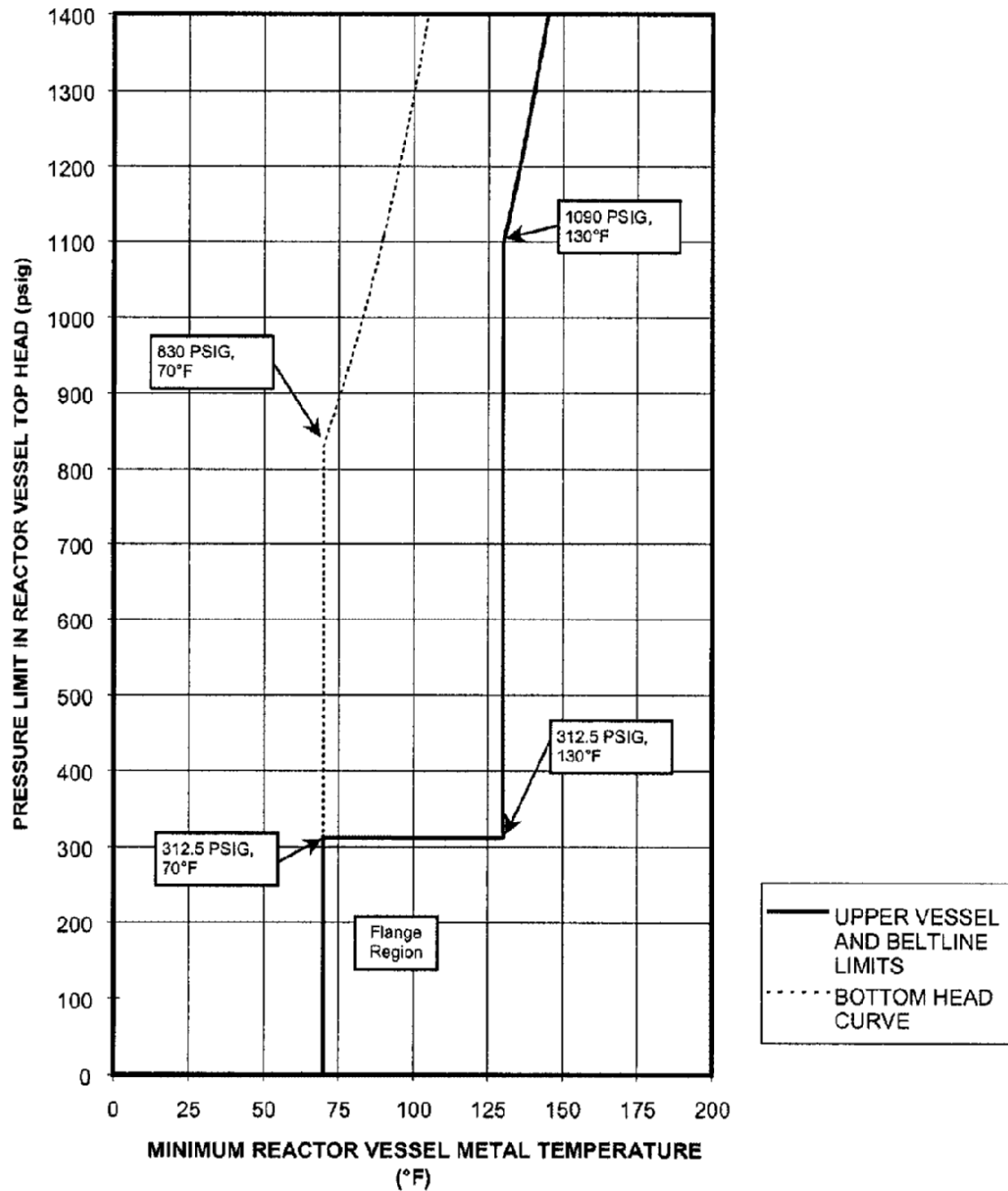
Light off

## RO Question 75

Attachment 2 (Cont.)  
Sheet 2 of 3

SVI-B21-T1176  
Page: 18  
Rev.: 12

P/T Limits - Non-Nuclear Heatup/Cooldown (Curve B) Valid up to 22 EFPY  
Technical Specification Figure 3.4.11-1(b)



### SRO Question 3

ECCS-Operating  
3.5.1

#### 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

##### 3.5.1 ECCS-Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of eight safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,  
MODES 2 and 3, except ADS valves are not required to be  
OPERABLE with reactor steam dome pressure  $\leq 150$  psig.

##### ACTIONS

-----NOTE-----  
LCO 3.0.4.b is not applicable to HPCS.  
-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC is required to be OPERABLE.	1 hour
	<u>AND</u> B.2 Restore HPCS System to OPERABLE status.	14 days

(continued)

# SRO Question 3

ECCS—Operating  
3.5.1

## ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two ECCS injection subsystems inoperable.  <u>OR</u>  One ECCS injection and one ECCS spray subsystem inoperable.	C.1 Restore one ECCS injection/spray subsystem to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	12 hours  36 hours
E. One ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days
F. One ADS valve inoperable.  <u>AND</u>  One low pressure ECCS injection/spray subsystem inoperable.	F.1 Restore ADS valve to OPERABLE status.  <u>OR</u>  F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours  72 hours
G. Two or more ADS valves inoperable.  <u>OR</u>	G.1 Be in MODE 3. <u>AND</u>	12 hours  (continued)

# SRO Question 3

ECCS—Operating  
3.5.1

ACTIONS		
CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. (continued)</p> <p>Required Action and associated Completion Time of Condition E or F not met.</p>	<p>G.2 Reduce reactor steam dome pressure to <math>\leq 150</math> psig.</p>	<p>36 hours</p>
<p>H. HPCS and Low Pressure Core Spray (LPCS) Systems inoperable.</p> <p><u>OR</u></p> <p>Three or more ECCS injection/spray subsystems inoperable.</p> <p><u>OR</u></p> <p>HPCS System and one or more ADS valves inoperable.</p> <p><u>OR</u></p> <p>Two or more ECCS injection/spray subsystems and one or more ADS valves inoperable.</p>	<p>H.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>



## SRO Question 4

DC Sources—Operating  
3.8.4

### 3.8 ELECTRICAL POWER SYSTEMS

#### 3.8.4 DC Sources—Operating

LC0 3.8.4 The Division 1, Division 2, and Division 3 DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Division 1 or 2 DC electrical power subsystem inoperable.	A.1 Restore Division 1 and 2 DC electrical power subsystems to OPERABLE status.	2 hours
B. Division 3 DC electrical power subsystem inoperable.	B.1 Declare High Pressure Core Spray System inoperable.	Immediately
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours
	<u>AND</u> C.2 Be in MODE 4.	36 hours

## SRO Question 4

DC Sources - Operating  
3.8.4

## SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.4.1    Verify battery terminal voltage is $\geq 129$ V on float charge.	7 days
SR 3.8.4.2    Verify no visible corrosion at battery terminals and connectors.  <u>OR</u>  Verify battery connection resistance is  $\leq 5.0 \text{ E-5 ohm}$ for inter-cell connections, $\leq 5.0 \text{ E-5 ohm}$ for inter-rack connections, $\leq 5.0 \text{ E-5 ohm}$ for inter-tier connections, $\leq 5.0 \text{ E-5 ohm}$ for terminal connections; for Div 1 and Div 2  and $\leq 1.0 \text{ E-4 ohm}$ for inter-cell connections, $\leq 1.0 \text{ E-4 ohm}$ for inter-rack connections, $\leq 1.0 \text{ E-4 ohm}$ for inter-tier connections, $\leq 1.0 \text{ E-4 ohm}$ for terminal connections. for Div 3.	92 days
SR 3.8.4.3    Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	24 months
SR 3.8.4.4    Remove visible corrosion, and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	24 months

(continued)

# SRO Question 4

DC Sources—Operating  
3.8.4

## SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.4.5    Verify battery connection resistance is</p> <p>≤ 5.0 E-5 ohm for inter-cell connections,  ≤ 5.0 E-5 ohm for inter-rack connections,  ≤ 5.0 E-5 ohm for inter-tier connections,  ≤ 5.0 E-5 ohm for terminal connections;  for Div 1 and Div 2</p> <p>and</p> <p>≤ 1.0 E-4 ohm for inter-cell connections,  ≤ 1.0 E-4 ohm for inter-rack connections,  ≤ 1.0 E-4 ohm for inter-tier connections,  ≤ 1.0 E-4 ohm for terminal connections  for Div 3.</p>	24 months
<p>SR 3.8.4.6    Verify each required Division 1 and 2  battery charger supplies ≥ 400 amps at  ≥ 125 V for ≥ 8 hours; and each required  Division 3 battery charger supplies  ≥ 50 amps at ≥ 125 V for ≥ 8 hours.</p>	24 months
<p>SR 3.8.4.7    -----NOTE-----  SR 3.8.4.8 may be performed in lieu of  SR 3.8.4.7 once per 60 months.  -----</p> <p>Verify battery capacity is adequate to  supply, and maintain in OPERABLE status,  the required emergency loads for the design  duty cycle when subjected to a battery  service test.</p>	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)		
	SURVEILLANCE	FREQUENCY
SR 3.8.4.8	Verify battery capacity is $\geq$ 80% of the manufacturer's rating when subjected to a performance discharge test.	60 months
		AND  -----NOTE----- Only applicable when battery shows degradation or has reached 85% of expected life. ----- 18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

- LCO 3.8.5 The following DC electrical power subsystems shall be OPERABLE:
- a. One Class 1E DC electrical power subsystem capable of supplying one division of the Division 1 or 2 onsite Class 1E electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems - Shutdown";
  - b. One Class 1E battery or battery charger, other than the DC electrical power subsystem in LCO 3.8.5.a, capable of supplying the remaining Division 1 or Division 2 onsite Class 1E DC electrical power distribution subsystem when required by LCO 3.8.8; and
  - c. The Division 3 DC electrical power subsystem capable of supplying the Division 3 onsite Class 1E DC electrical power distribution subsystem, when the Division 3 onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,  
During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building.

# SRO Question 4

DC Sources — Shutdown  
3.8.5

## ACTIONS

-----NOTE-----  
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the primary containment and fuel handling building.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

# SRO Question 4

DC Sources – Shutdown  
3.8.5

SURVEILLANCE REQUIREMENTS	
SURVEILLANCE	FREQUENCY
<p>SR 3.8.5.1</p> <p>----- NOTE -----</p> <p>The following SRs are not required to be performed: SR 3.8.4.4, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8.</p> <p>-----</p> <p>For DC sources required to be OPERABLE, the following SRs are applicable:</p> <p>SR 3.8.4.1    SR 3.8.4.4    SR 3.8.4.7  SR 3.8.4.2    SR 3.8.4.5    SR 3.8.4.8  SR 3.8.4.3    SR 3.8.4.6</p>	<p>In accordance with applicable SRs</p>

## SRO Question 6

### CRER System Instrumentation 3.3.7.1

#### 3.3 INSTRUMENTATION

##### 3.3.7.1 Control Room Emergency Recirculation (CRER) System Instrumentation

LCD 3.3.7.1 The CRER System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7.1-1.

#### ACTIONS

-----NOTE-----  
Separate Condition entry is allowed for each channel.  
-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	B.1 Declare associated CRER subsystem inoperable.	1 hour from discovery of loss of CRER initiation capability in both trip systems
	<u>AND</u> B.2 Place channel in trip.	24 hours

(continued)



## SRO Question 6

### CRER System Instrumentation 3.3.7.1

#### ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1 Provide alternate method of control room radiation monitoring.	24 hours
	<u>AND</u> C.2 Restore the inoperable monitor to an OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Place the associated CRER subsystem in the emergency recirculation mode of operation.	1 hour
	<u>OR</u> D.2 Declare associated CRER subsystem inoperable.	1 hour

## SRO Question 6

CRER System Instrumentation  
3.3.7.1

### SURVEILLANCE REQUIREMENTS

#### NOTES

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CRER initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Calibrate the trip unit.	92 days
SR 3.3.7.1.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.7.1.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

# SRO Question 6

## CRER System Instrumentation 3.3.7.1

Table 3.3.7.1-1 (page 1 of 1)  
Control Room Emergency Recirculation System Instrumentation

FUNCTION	APPLICABLE NODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level — Low Low Low, Level 1	1,2,3, (a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≥ 14.3 inches
2. Drywell Pressure — High	1,2,3	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 1.88 psig
3. Control Room Ventilation Radiation Monitor	1,2,3, (b)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 800 cpm

(a) During operations with a potential for draining the reactor vessel.

(b) During operations with a potential for draining the reactor vessel, and movement of recently irradiated fuel assemblies in the primary containment or fuel handling building.

## SRO QUESTIONS 9 & 11

# INITIATING CONDITION INDEX

PNPP No. 8852 Rev. 3/26/10

EPI-A1

EVENT CATEGORY	MODE UNUSUAL EVENT	MODE ALERT	MODE SITE AREA EMERGENCY	MODE GENERAL EMERGENCY
A: FISSION PRODUCT BARRIER DEGRADATION	1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32 33 34 35 36 37 38 39 40 41 42 43 44 45 46 47 48 49 50 51 52 53 54 55 56 57 58 59 60 61 62 63 64 65 66 67 68 69 70 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90 91 92 93 94 95 96 97 98 99 100 101 102 103 104 105 106 107 108 109 110 111 112 113 114 115 116 117 118 119 120 121 122 123 124 125 126 127 128 129 130 131 132 133 134 135 136 137 138 139 140 141 142 143 144 145 146 147 148 149 150 151 152 153 154 155 156 157 158 159 160 161 162 163 164 165 166 167 168 169 170 171 172 173 174 175 176 177 178 179 180 181 182 183 184 185 186 187 188 189 190 191 192 193 194 195 196 197 198 199 200 201 202 203 204 205 206 207 208 209 210 211 212 213 214 215 216 217 218 219 220 221 222 223 224 225 226 227 228 229 230 231 232 233 234 235 236 237 238 239 240 241 242 243 244 245 246 247 248 249 250 251 252 253 254 255 256 257 258 259 260 261 262 263 264 265 266 267 268 269 270 271 272 273 274 275 276 277 278 279 280 281 282 283 284 285 286 287 288 289 290 291 292 293 294 295 296 297 298 299 300 301 302 303 304 305 306 307 308 309 310 311 312 313 314 315 316 317 318 319 320 321 322 323 324 325 326 327 328 329 330 331 332 333 334 335 336 337 338 339 340 341 342 343 344 345 346 347 348 349 350 351 352 353 354 355 356 357 358 359 360 361 362 363 364 365 366 367 368 369 370 371 372 373 374 375 376 377 378 379 380 381 382 383 384 385 386 387 388 389 390 391 392 393 394 395 396 397 398 399 400 401 402 403 404 405 406 407 408 409 410 411 412 413 414 415 416 417 418 419 420 421 422 423 424 425 426 427 428 429 430 431 432 433 434 435 436 437 438 439 440 441 442 443 444 445 446 447 448 449 450 451 452 453 454 455 456 457 458 459 460 461 462 463 464 465 466 467 468 469 470 471 472 473 474 475 476 477 478 479 480 481 482 483 484 485 486 487 488 489 490 491 492 493 494 495 496 497 498 499 500 501 502 503 504 505 506 507 508 509 510 511 512 513 514 515 516 517 518 519 520 521 522 523 524 525 526 527 528 529 530 531 532 533 534 535 536 537 538 539 540 541 542 543 544 545 546 547 548 549 550 551 552 553 554 555 556 557 558 559 560 561 562 563 564 565 566 567 568 569 570 571 572 573 574 575 576 577 578 579 580 581 582 583 584 585 586 587 588 589 590 591 592 593 594 595 596 597 598 599 600 601 602 603 604 605 606 607 608 609 610 611 612 613 614 615 616 617 618 619 620 621 622 623 624 625 626 627 628 629 630 631 632 633 634 635 636 637 638 639 640 641 642 643 644 645 646 647 648 649 650 651 652 653 654 655 656 657 658 659 660 661 662 663 664 665 666 667 668 669 670 671 672 673 674 675 676 677 678 679 680 681 682 683 684 685 686 687 688 689 690 691 692 693 694 695 696 697 698 699 700 701 702 703 704 705 706 707 708 709 710 711 712 713 714 715 716 717 718 719 720 721 722 723 724 725 726 727 728 729 730 731 732 733 734 735 736 737 738 739 740 741 742 743 744 745 746 747 748 749 750 751 752 753 754 755 756 757 758 759 760 761 762 763 764 765 766 767 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# INITIATING CONDITION INDEX

PNPP No. 8852 Rev. 3/26/10

EPI-A1

EVENT CATEGORY	MODE 1 2 3 4 5 6	UNUSUAL EVENT	MODE 1 2 3 4 5 6	ALERT	MODE 1 2 3 4 5 6	SITE AREA EMERGENCY	MODE 1 2 3 4 5 6	GENERAL EMERGENCY
<b>H: INCREASED RADIATION RELEASE TO THE ENVIRONMENT</b>	1 2 3 4 5 6	Any unplanned release of gaseous radioactivity to the environment that exceeds two times the ODCM Control limit for 60 minutes or greater.  Page 46 - HU1	1 2 3 4 5 6	Any unplanned release of gaseous radioactivity to the environment that exceeds 200 times the ODCM Control limit for 15 minutes or greater.  Page 48 - HA1	1 2 3 4 5 6	Site Boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mRem TEDE dose OR 500 mRem CDE Child Thyroid dose for the actual OR projected duration of the release.  Page 50 - HS1	1 2 3 4 5 6	Site Boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mRem TEDE OR 5000 mRem CDE Child Thyroid dose for the actual or projected duration of the release.  Page 51 - HG1
	1 2 3 4 5 6	Any unplanned release of liquid radioactivity to the environment that exceeds two times the ODCM Control limit for 60 minutes or greater.  Page 47 - HU2	1 2 3 4 5 6	Any unplanned release of liquid radioactivity to the environment that exceeds 200 times the ODCM Control limit for 15 minutes or greater.  Page 49 HA2				
<b>I: CONTROL ROOM EVACUATION</b>		NOT APPLICABLE	1 2 3 4 5 6	Control Room Evacuation has been initiated.  Page 52 - IA1	1 2 3 4 5 6	Control Room evacuation has been initiated, AND plant control <u>CANNOT</u> be established within 15 minutes.  Page 53 - IS1		NOT APPLICABLE
<b>J: LOSS OF ANNUNCIATORS OR INDICATIONS</b>	1 2 3 4 5 6	Loss of most annunciators or indication in the Control Room for greater than 15 minutes.  Page 54 - JU1	1 2 3 4 5 6	Loss of most annunciators or indication in the Control Room with <u>either</u> : (1) a significant transient in progress; OR (2) compensatory indications are <u>NOT</u> available.  Page 55 - JA1	1 2 3 4 5 6	Inability to monitor a significant transient in progress.  Page 56 - JS1		NOT APPLICABLE
<b>K: LOSS OF COMMUNICATIONS</b>	1 2 3 4 5 6	Loss of onsite OR in-plant communications capabilities.  Page 58 - KU1		NOT APPLICABLE		NOT APPLICABLE		NOT APPLICABLE
	1 2 3 4 5 6	Significant degradation of offsite communications capabilities.  Page 59 - KU2						
<b>L: NATURAL OR DESTRUCTIVE PHENOMENA</b>	1 2 3 4 5 6	Natural OR destructive phenomena affecting the Protected Area boundary.  Page 60 - LU1	1 2 3 4 5 6	Natural OR destructive phenomena affecting Safe Shutdown Buildings.  Page 61 - LA1		NOT APPLICABLE		NOT APPLICABLE
<b>M: RELEASE OF TOXIC OR FLAMMABLE GAS</b>	1 2 3 4 5 6	Release of toxic OR flammable gasses affecting the Protected Area boundary deemed detrimental to the safe operation of the plant.  Page 62- MU1	1 2 3 4 5 6	Release of toxic OR flammable gases within a Safe Shutdown Building which jeopardizes operation of systems required to maintain safe operations OR to establish or maintain COLD SHUTDOWN.  Page 63- MA1		NOT APPLICABLE		NOT APPLICABLE
<b>N: SECURITY EVENTS</b>	1 2 3 4 5 6	Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.  Page 64 - NU1		NOT APPLICABLE		NOT APPLICABLE	1 2 3 4 5 6	HOSTILE ACTION resulting in loss of physical control of the facility.  Page 70 - NG1
			1 2 3 4 5 6	Notification of an airborne attack threat.  Page 66 - NA2	1 2 3 4 5 6	HOSTILE ACTION within the PROTECTED AREA  Page 69 - NS2		
			1 2 3 4 5 6	HOSTILE ACTION within the OWNER CONTROLLED AREA  Page 67 - NA3				
<b>O: EMERGENCY COORDINATOR'S JUDGEMENT</b>	1 2 3 4 5 6	Other conditions existing, which in the judgement of the Emergency Coordinator, warrant declaration of an Unusual Event.  Page 72- OU1	1 2 3 4 5 6	Other conditions existing, which in the judgement of the Emergency Coordinator, warrant declaration of an Alert.  Page 73 - OA1	1 2 3 4 5 6	Other conditions existing, which in the judgement of the Emergency Coordinator, warrant declaration of a Site Area Emergency.  Page 74- OS1	1 2 3 4 5 6	Other conditions existing, which in the judgement of the Emergency Coordinator, warrant declaration of a General Emergency.  Page 75 - OG1

Category A: Fission Product Barrier Degradation

Initiating Conditions							Entry Criteria	
<p><b>AU1</b></p> <p><b>Fuel clad degradation</b></p> <p><b>Applicable Modes:</b></p>							<p>High Offgas pretreatment air activity greater than the Technical Specification 3.7.5.</p>	<p>Reactor Coolant System sample indicates activity greater than Technical Specification 3.4.8 limits.</p>
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>				

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NOTE

Fuel clad degradation is NOT an issue when the Reactor is defueled. Damage to fuel in spent fuel pools is addressed in GU1.

Initiating Conditions						Entry Criteria		
<p><b>AU2</b></p> <p><b>Reactor Coolant System leakage.</b></p> <p><b>Applicable Modes:</b></p>						<p>Greater than 10 gpm unidentified leakage in Drywell.</p>	<p>Greater than 30 gpm total leakage in Drywell averaged over the previous 24 hour period.</p>	<p>Greater than 30 gpm total leakage in Drywell.</p>
								<p>Greater than 2 gpm increase in unidentified leakage within the previous 24 hour period.</p>
<b>1</b>	<b>2</b>	<b>3</b>						

<p><b><u>AU2</u></b></p> <p><b>U</b></p> <p><b>N</b></p> <p><b>U</b></p> <p><b>S</b></p> <p><b>U</b></p> <p><b>A</b></p> <p><b>L</b></p> <p><b>E</b></p> <p><b>V</b></p> <p><b>E</b></p> <p><b>N</b></p> <p><b>T</b></p>
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Initiating Conditions						Entry Criteria	
<p style="text-align: center;"><b>AS1</b></p> <p><b>Loss of RPV water level that has or will uncover fuel</b></p>						RPV water level <u>CANNOT</u> be maintained greater than 0 inches.	
						Reactor is “shutdown under all conditions without boron”.	
Applicable Modes:							
1	2	3	4	5			

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NOTE

AS1 is applicable only to non-ATWS situations in which RPV level was NOT intentionally lowered per <EOP-01A> as a means of power control. Refer to Event Category ”C” for classification under ATWS conditions in which RPV water level is intentionally lowered.

# FISSION PRODUCT BARRIER MATRIX

PNPP No. 9855 Rev. 8/26/08

EPLAN/EPI-A1/PSI-0019

FUEL CLAD BARRIER	LOSS CRITERIA	REACTOR PRESSURE VESSEL LEVEL	DRYWELL RADIATION	REACTOR COOLANT SYSTEM ACTIVITY	EMERGENCY COORDINATOR JUDGEMENT
	CHALLENGE CRITERIA	REACTOR PRESSURE VESSEL LEVEL	DRYWELL RADIATION	REACTOR COOLANT SYSTEM ACTIVITY	EMERGENCY COORDINATOR JUDGEMENT
		Entry into SAG-1, Primary Containment Flooding. <sup>3</sup>	Drywell radiation monitor reading greater than 4,000 Rem/hr.	Sample activity is equal to or greater than 300 uCi/gm dose equivalent iodine-131. <sup>5</sup>	Any condition that, in the judgment of the Emergency Coordinator indicates loss of the Fuel Clad barrier. <sup>1</sup>
		RPV level is less than 0 inches. <sup>2</sup>	RPV level <b>CANNOT</b> be determined.	NOT APPLICABLE	Any condition that, in the judgment of the Emergency Coordinator indicates a challenge to the Fuel Clad barrier. <sup>1</sup>

## INSTRUCTIONS

- For each of the three barriers, determine if any **LOSS** or **CHALLENGE** criteria have been met.
- Compare the barrier **LOSS(es)** and **CHALLENGE(s)** to the initiating conditions listed, and make the appropriate event declaration.

## INITIATING CONDITIONS

### UNUSUAL EVENT

AU3 Any loss or challenge to the Containment barrier.

Modes: ☐ 1 ☐ 2 ☐ 3 ☐ 4

### ALERT

AA1 Any loss or challenge to Fuel Clad barrier.

Modes: ☐ 1 ☐ 2 ☐ 3 ☐ 4

AA2 Any loss or challenge to the Reactor Coolant System barrier.

Modes: ☐ 1 ☐ 2 ☐ 3 ☐ 4

### SITE AREA EMERGENCY

AS2 Either a challenge or loss of both the Fuel Clad barrier AND Reactor Coolant System barrier.

Modes: ☐ 1 ☐ 2 ☐ 3 ☐ 4

AS3 Challenge to either the Fuel Clad barrier OR Reactor Coolant System barrier, AND the loss of any additional barrier.

Modes: ☐ 1 ☐ 2 ☐ 3 ☐ 4

### GENERAL EMERGENCY

AG1 Loss of two barriers, AND a loss or challenge to the third barrier.

Modes: ☐ 1 ☐ 2 ☐ 3 ☐ 4

REACTOR COOLANT SYSTEM BARRIER	LOSS CRITERIA	REACTOR PRESSURE VESSEL LEVEL	DRYWELL RADIATION	REACTOR PRESSURE CONTROL	DRYWELL PRESSURE	REACTOR COOLANT SYSTEM BYPASS	EMERGENCY COORDINATOR JUDGEMENT
	CHALLENGE CRITERIA	REACTOR PRESSURE VESSEL LEVEL	DRYWELL RADIATION	REACTOR PRESSURE CONTROL	DRYWELL PRESSURE	REACTOR COOLANT SYSTEM BYPASS	EMERGENCY COORDINATOR JUDGEMENT
		RPV level less than 0 inches. <sup>2</sup>	Drywell radiation monitor reading greater than 135 Rem/hr.	SRV stuck open. An SRV is being cycled to control RPV pressure. Emergency Depressurization is required.	Drywell pressure greater than 1.68 psig. Indication of RCS leakage inside the Drywell.	MSL break outside containment exceeding <u>one or more</u> MSIV Tech. Spec. allowable values. Containment penetration does <b>NOT</b> isolate on a valid closure signal. Immediate Operator actions in the Control Room are <b>NOT</b> successful in isolating affected penetration.	Any condition that, in the judgment of the Emergency Coordinator, indicates loss of the RCS barrier. <sup>1</sup>
		NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE	Unisolable primary system discharging outside Containment per either EOP-03 or EOP-05. <sup>4</sup>	Any condition that, in the judgment of the Emergency Coordinator, indicates a challenge to the RCS barrier. <sup>1</sup>

CONTAINMENT BARRIER	LOSS CRITERIA	REACTOR PRESSURE VESSEL LEVEL	CONTAINMENT RADIATION	CONTAINMENT HYDROGEN	CONTAINMENT PRESSURE	CONTAINMENT ISOLATION	EMERGENCY COORDINATOR JUDGEMENT
	CHALLENGE CRITERIA	REACTOR PRESSURE VESSEL LEVEL	CONTAINMENT RADIATION	CONTAINMENT HYDROGEN	CONTAINMENT PRESSURE	CONTAINMENT ISOLATION	EMERGENCY COORDINATOR JUDGEMENT
		Entry into SAG-1, Primary Containment Flooding <sup>3</sup>	NOT APPLICABLE	Intentional venting of Containment required per SAG-2	Intentional venting of Containment required per EOP-02	Containment penetration does <b>NOT</b> isolate on a valid closure signal. Immediate Operator actions in the Control Room are <b>NOT</b> successful in isolating affected penetration. Pathway to the environment exists via penetration.	Any condition that, in the judgment of the Emergency Coordinator, indicates loss of the Containment barrier. <sup>1</sup>  (Loss of the Containment barrier may include a rapid unexplained decrease in Containment pressure following an initial increase.)
		NOT APPLICABLE	Containment radiation monitor reading greater than 20,000 Rem/hr.	NOT APPLICABLE	In the UNSAFE region on the HCL figure. Containment pressure is greater than 15 psig and increasing.	NOT APPLICABLE	Any condition that, in the judgment of the Emergency Coordinator, indicates a challenge to the Containment barrier. <sup>1</sup>

NOTE: If the criteria for loss has been met, then by definition, the challenge criteria has been met for that same fission product barrier.

FOOTNOTES:

- Those thresholds for which a LOSS or CHALLENGE is determined to be IMMINENT (i.e., within the next 3 hours), classify as though the threshold(s) has been exceeded.
- RPV level is less than 0 inches is both a FUEL CLAD BARRIER CHALLENGE CRITERIA and a REACTOR COOLANT SYSTEM BARRIER LOSS CRITERIA.
- Entry into SAG-1, Primary Containment Flooding is both a FUEL CLAD BARRIER LOSS CRITERIA and a CONTAINMENT BARRIER LOSS CRITERIA.
- Unisolable primary system discharging outside containment per EOP-03 and/or EOP-05 is both a REACTOR COOLANT SYSTEM BARRIER CHALLENGE CRITERIA and a CONTAINMENT BARRIER LOSS CRITERIA.
- Sample activity is equal to or greater than 300 uCi/gm dose equivalent Iodine-131 is both a FUEL CLAD BARRIER LOSS CRITERIA and a contributor to a REACTOR COOLANT SYSTEM BARRIER LOSS CRITERIA.

Category B: Loss of Decay Heat Removal Functions

Initiating Conditions						Entry Criteria	
<b>BA1</b>  <b>Inability to maintain plant in COLD SHUTDOWN</b>  <b>Applicable Modes:</b>						Loss of Shutdown Cooling Mode function for RHR loop A.	
						Loss of Shutdown Cooling Mode function for RHR loop B.	
						RCS temperature exceeds COLD SHUTDOWN limit of 200°F per Technical Specification Table 1.1-1.	Uncontrolled temperature rise approaching 200°F RCS temperature.
			4	5			

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NOTES

- The IC remains applicable for situations in which an increase in RCS temperature greater than 200°F results in a change to Mode 3.
- The above criteria is met as soon as it becomes known that sufficient cooling CANNOT be restored to maintain temperature below 200°F regardless of the current temperature. The intent of BA1 is NOT to classify based on an unplanned excursion above 200°F when heat removal capability is available.
- “Uncontrolled” means that RCS temperature increase is NOT the result of planned actions by plant staff.

Initiating Conditions						Entry Criteria					
<p><b>BS1</b></p> <p><b>Complete loss of functions needed to achieve COLD SHUTDOWN</b></p> <p><b>Applicable Modes:</b></p>						RHR Loops A and B are <u>NOT</u> capable of lowering RPV temperature.					
						The plant is operating in the UNSAFE Region on the HCL figure.					
<b>1</b>	<b>2</b>	<b>3</b>									

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Category C: Loss of Shutdown Functions or Failure to Shutdown

Initiating Conditions						Entry Criteria					
<p style="text-align: center;"><b>CU1</b></p> <p style="text-align: center;"><b>Inability to reach required shutdown within Technical Specification limits</b></p> <p style="text-align: center;"><b>Applicable Modes:</b></p>						<p>Plant is <u>NOT</u> brought to the required operating mode within the Technical Specification Required Action Completion Time following entry into an LCO.</p>					
<b>1</b>	<b>2</b>	<b>3</b>									

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NOTE

Declaration should be made because of equipment failures that prevent the performance of an orderly shutdown or failure to meet the shutdown completion time from the time discovered and a required action being entered. Declaration of an Unusual Event is based on the time at which the specified completion time period elapses and is NOT related to how long a condition may have existed before it was discovered.

Initiating Conditions						Entry Criteria	
<b>CA1</b>  <b>Failure to initiate or complete an automatic Reactor Scram once an RPS function is required</b>  <b>Applicable Modes:</b>						Actuation of RPS has occurred or should have occurred.	Actuation of RRCS has occurred or should have occurred.
						The reactor is <u>NOT</u> “shutdown under all conditions without boron.”	
<b>1</b>	<b>2</b>						

**CA1**

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### NOTES

- CA1 is applicable if either Mode 1 or 2 existed when the transient started and NOT the mode which exists at the time of classification.
- Entry criteria is applicable for actions taken by an Operator to manually initiate either RPS or RRCS prior to or after exceeding an automatic actuation setpoint.

Initiating Conditions						Entry Criteria		
<p style="text-align: center;"><b>CS1</b></p> <p><b>Failure to initiate or complete an automatic Reactor Scram once an RPS function is required, AND a manual Scram was <u>NOT</u> successful</b></p> <p><b>Applicable Modes:</b></p>						Actuation of RPS has occurred or should have occurred.		Actuation of RRCS has occurred or should have occurred.
						The reactor is <u>NOT</u> “shutdown under all conditions without boron.”		
						Manual operator actions taken at 1H13-P680 to insert control rods were <u>NOT</u> successful in lowering Reactor power to less than 4%.	Reactor power <u>CANNOT</u> be determined.	Suppression Pool temperature is greater than 110°F.
<b>1</b>								

**CS1**

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NOTE

Refer to next page.

### NOTES

- CS1 is applicable if Mode 1 existed when the transient started and NOT the mode which exists at the time of classification. Refer to CA1 for Mode 2 applicability.
- “Manual Operator actions” are defined as any set of actions by the Reactor Operator at 1H13-P680 which results in a scram signal. These actions include placing the Reactor Mode Switch in the SHUTDOWN position, arming and depressing the RPS Manual Scram push buttons, and arming and depressing the RRCS Manual ARI push buttons. Injection of boron is NOT considered in reducing reactor power below 4%.
- A concurrent challenge to the ability to cool the core would escalate this event to General Emergency per CG1.



Initiating Conditions						Entry Criteria		
<b>CG1</b>  <b>Failure to initiate or complete a successful shutdown, AND indication of an extreme challenge to the ability to cool the core</b>						Actuation of RPS has occurred or should have occurred.		Actuation of RRCS has occurred or should have occurred.
						The reactor is <u>NOT</u> “shutdown under all conditions without boron.”		
						Manual operator actions taken at 1H13-P680 to insert control rods were <u>NOT</u> successful in lowering Reactor power to less than 4%.	Reactor power <u>CANNOT</u> be determined.	Suppression Pool temperature is greater than 110°F.
						Any of the following conditions exist: <ul style="list-style-type: none"> <li>• Entry into &lt;SAG-1&gt;, Primary Containment Flooding.</li> <li>• In the UNSAFE region on the HCL figure.</li> </ul>		
Applicable Modes:								
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## CG1

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## NOTES

- CG1 is applicable if Mode 1 existed when the transient started and NOT the mode which exists at the time of classification. Refer to CA1 for Mode 2 applicability.
- Entry criteria is applicable for actions taken by an Operator to manually initiate either RPS or RRCS prior to or after exceeding an automatic actuation setpoint.

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Category D: A. C. Power Loss

Initiating Conditions							Entry Criteria						
<p style="text-align: center;"><b>DU1</b></p> <p><b>Loss of all offsite power to Division 1 and 2 EH Essential Busses for greater than 15 minutes</b></p>							ONI-R10 entered for a Loss of Off-site Power (LOOP).						
							<p><u>Either</u> of the following power sources <u>CANNOT</u> be made available within 15 minutes for energizing bus EH11:</p> <ul style="list-style-type: none"> <li>• Normal Preferred</li> <li>• Alternate Preferred</li> </ul>						
							<p><u>Either</u> of the following power sources <u>CANNOT</u> be made available within 15 minutes for energizing bus EH12:</p> <ul style="list-style-type: none"> <li>• Normal Preferred</li> <li>• Alternate Preferred</li> </ul>						
<p><b>Applicable Modes:</b></p>													
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>								

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NOTE

Failure of either bus EH11 or EH12 to be supplied from its respective diesel generator is evaluated for escalation to an Alert under DA1 for Modes 1, 2 and 3. Failure of both busses EH11 and EH12 to be supplied from their respective diesel generators (Station Black Out) is evaluated for escalation to an Alert under DA2 for Modes 4 and 5 and to a Site Area Emergency under DS1 for Modes 1, 2 and 3.

Initiating Conditions						Entry Criteria					
<p style="text-align: center;"><b>DA1</b></p> <p><b>Power capability to Division 1 and 2 EH Essential Busses reduced to a single power source for greater than 15 minutes, such that any additional single failure would result in Station Blackout</b></p> <p><b>Applicable Modes:</b></p>						<p>Essential AC power reduced to only <u>one</u> of the following power sources for greater than 15 minutes:</p> <ul style="list-style-type: none"> <li>• Normal Preferred</li> <li>• Alternate Preferred</li> <li>• Division 1 Diesel Generator</li> <li>• Division 2 Diesel Generator</li> </ul>					
						<p>Loss of the single remaining power source will result in a loss of AC power to <u>both</u> busses EH11 and EH12.</p>					
<b>1</b>	<b>2</b>	<b>3</b>									

**DA1**

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**NOTES**

- Escalation to a Site Area Emergency is evaluated under DS1, for Operating Modes 1, 2 and 3, based on a total loss of AC power to both busses EH11 and EH12.
- A total loss of AC power to busses EH11 and EH12 while in Operating Modes 4 and 5 is classified as an Alert under DA2. No escalation path exists to a Site Area Emergency for Operating Modes 4 and 5.

Initiating Conditions						Entry Criteria					
<b>DA2</b>  <b>Loss of all offsite power AND onsite power to Division 1 and 2 EH Essential Busses for greater than 15 minutes.</b>  <b>Applicable Modes:</b>						Both busses EH11 and EH12 <u>CANNOT</u> be energized from the Normal Preferred source within 15 minutes.					
						Both busses EH11 and EH12 <u>CANNOT</u> be energized from the Alternate Preferred source within 15 minutes.					
						Both busses EH11 and EH12 <u>CANNOT</u> be energized from the Associated Diesel Generator source within 15 minutes.					
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Initiating Conditions						Entry Criteria					
<p style="text-align: center;"><b>DS1</b></p> <p><b>Loss of all offsite power AND onsite power to Division 1 and 2 EH Essential Busses for greater than 15 minutes</b></p>						Both busses EH11 and EH12 <u>CANNOT</u> be energized from the Normal Preferred source within 15 minutes.					
						Both busses EH11 and EH12 <u>CANNOT</u> be energized from the Alternate Preferred source within 15 minutes.					
						Both busses EH11 and EH12 <u>CANNOT</u> be energized from the Associated Diesel Generator source within 15 minutes.					
Applicable Modes:											
1	2	3									

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**NOTE**

Escalation to a General Emergency is evaluated under DG1 for Modes 1, 2 and 3, based on a continuing degradation of core cooling capability.

Initiating Conditions						Entry Criteria		
<p align="center"><b>DG1</b></p> <p><b>Prolonged loss of all offsite power AND onsite power to Division 1 and 2 EH Busses, AND continuing degradation of core cooling capability</b></p> <p><b>Applicable Modes:</b></p>						Both busses EH11 and EH12 <u>CANNOT</u> be energized from the Normal Preferred source.		
						Both busses EH11 and EH12 <u>CANNOT</u> be energized from the Alternate Preferred source.		
						Both busses EH11 and EH12 <u>CANNOT</u> be energized from the Associated Diesel Generator source.		
						Restoration of power to either of the following busses is <u>NOT</u> likely in less than four hours: <ul style="list-style-type: none"> <li>• EH11</li> <li>• EH12</li> </ul>	RPV water level less than 0 inches.	RPV water level <u>CANNOT</u> be determined.
1	2	3						

**DG1**

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Category E: D. C. Power Degradation

Initiating Conditions						Entry Criteria					
<p><b>EU1</b></p> <p><b>Degradation of Division 1 and 2 essential DC power for greater than 15 minutes</b></p> <p><b>Applicable Modes:</b></p>						Voltage on ED-1-A buss is less than 105 VDC for greater than 15 minutes.					
						Voltage on ED-1-B buss is less than 105 VDC for greater than 15 minutes.					
			4	5							

**EU1**

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NOTE

The same set of conditions as described in this EAL would be classified as Site Area Emergency under ES1 if they occurred during Modes 1, 2, or 3.

Initiating Conditions						Entry Criteria					
<p style="text-align: center;"><b>ES1</b></p> <p style="text-align: center;"><b>Degradation of Division 1 and 2 essential DC power for greater than 15 minutes</b></p>						Voltage on ED-1-A buss is less than 105 VDC for greater than 15 minutes.					
						Voltage on ED-1-B buss is less than 105 VDC for greater than 15 minutes.					
<b>Applicable Modes:</b>											
<b>1</b>	<b>2</b>	<b>3</b>									

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Category F: Fire or Explosion

Initiating Conditions							Entry Criteria	
<b>FU1</b>  <b>Fire within a Safe Shutdown Building <u>NOT</u> extinguished within 15 minutes</b>  <b>Applicable Modes:</b>							Fire within any Safe Shutdown Building.	
							Fire <u>CANNOT</u> be extinguished within 15 minutes of the verification of alarm.	Fire <u>CANNOT</u> be extinguished within 15 minutes of the notification received in the Control Room from plant personnel that a fire exists.
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>			

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NOTE

Verification in this context means those actions taken in the Secondary Alarm Station (SAS) to determine that the alarm is NOT spurious. Verification includes the receipt of multiple or independent alarms or confirmation of a single detector by visual inspection of the affected area by a first responder. List of Safe Shutdown Buildings is found in Section 3 “Definitions”, sub-step 3.10.

Initiating Conditions							Entry Criteria						
<p style="text-align: center;"><b>FU2</b></p> <p style="text-align: center;"><b>Explosion affecting a Safe Shutdown Building</b></p> <p style="text-align: center;"><b>Applicable Modes:</b></p>							<p>Report by plant personnel confirming the occurrence of an explosion within the Protected Area resulting in visible damage to a Safe Shutdown Building.</p>						
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>								

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**NOTE**

No attempt is made to assess the magnitude of the damage. The occurrence of the explosion with reports of damage (deformation/scorching) is sufficient for declaration. Actual damage to safe shutdown equipment is covered under Alert FA1. List of Safe Shutdown Buildings is found in Section 3 “Definitions”, sub-step 3.10.

Initiating Conditions						Entry Criteria	
<b>FA1</b>  <b>Fire OR explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown</b>  <b>Applicable Modes:</b>						Either of the following has been confirmed: <ul style="list-style-type: none"> <li>• Fire in a Safe Shutdown Building.</li> <li>• Explosion in a Safe Shutdown Building.</li> </ul>	
						Plant personnel at the scene report visible damage to safe shutdown equipment or components.	Affected safe shutdown system indicates degraded performance.
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>		

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**NOTE**

The inclusion of a “report of visible damage” should NOT be interpreted as mandating a lengthy damage assessment prior to classification. NO attempt is made in this EAL to assess the actual magnitude of damage beyond the immediate area. The occurrence of the explosion or fire with reports of evidence of damage (e.g., deformation, scorching) is sufficient for declaration. List of Safe Shutdown Buildings is found in Section 3 “Definitions”, sub-step 3.10.

### NOTE

Safe Shutdown System/Equipment refers to equipment identified in the Safe Shutdown Capability Report. This is the minimum list of equipment required to achieve and maintain COLD SHUTDOWN (including all auxiliary equipment such as AC/DC power, cooling water and instrumentation). A detailed list is provided in the <Appendix R Evaluation - Safe Shutdown Capability Report>.

Safe Shutdown System/Equipment list: (Division 1 and 2 only)

- Reactor Protection System
- Control Rod Drive Hydraulics
- Automatic Depressurization System/SRV
- Reactor Core Isolation Cooling
- Low Pressure Core Spray
- Low Pressure Coolant Injection - A/B/C
- Suppression Pool Cooling
- Shutdown Cooling
- Safety-Related Instrument Air
- Emergency Service Water
- Emergency Service Water Screen Wash
- Emergency Service Water Pump House Ventilation
- ECCS Pump Room Cooling Systems
- Diesel Generator Building Ventilation
- Stand-by Diesel Generator (DG)
- DG Fuel Oil Storage/Transfer
- Electrical Power Distribution
- Emergency Closed Cooling Pump Area Cooling
- Emergency Closed Cooling
- Control Complex Chilled Water
- MCC, Switchgear and Miscellaneous Electrical Equipment Areas HVAC System
- Battery Room Exhaust
- Control Room HVAC and Emergency Recirculation System

(Reference: <NUMARC/NESP-007> (Rev. 2), Unusual Event HA2)

Category G: Increased Plant Radiation Levels

Initiating Conditions							Entry Criteria	
<p style="text-align: center;"><b>GU1</b></p> <p><b>Unexpected increase in plant radiation levels</b></p>							<p>A valid area radiation monitor (D21) reading increases by a factor of 1000 over normal levels.</p>	<p>Health Physics surveys indicate an increase by a factor of 1000 times over normally expected area radiation levels.</p>
							<p>In-plant radiation level increase <u>CANNOT</u> be attributed to <u>any</u> of the following:</p> <ul style="list-style-type: none"> <li>the start-up and operation of plant equipment or systems within design parameters.</li> <li>the planned movement of radioactive materials.</li> <li>the planned movement of shielding (i.e., plugs, lead shot, etc.)</li> </ul>	
Applicable Modes:								
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>			

**GU1**

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NOTE

“Normal” area radiation levels can be considered as the highest reading in the past 24-hour period, excluding the current peak value.

Initiating Conditions							Entry Criteria						
<p style="text-align: center;"><b>GU2</b></p> <p><b>Uncontrolled fuel pool or Reactor Cavity water level decrease with irradiated fuel remaining covered.</b></p> <p><b>Applicable Modes:</b></p>							<p>Uncontrolled decrease in <u>one or more</u> of the following fuel pools containing irradiated fuel:</p> <ul style="list-style-type: none"> <li>• Reactor Cavity</li> <li>• FHB Fuel Storage and Preparation Pool</li> <li>• FHB Fuel Transfer Pool</li> <li>• FHB Spent Fuel Storage Pool</li> <li>• FHB Cask Pit</li> <li>• CNTMT Fuel Storage Pool</li> <li>• CNTMT Fuel Transfer Pool</li> </ul>						
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>								

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Initiating Conditions							Entry Criteria	
<p style="text-align: center;"><b>GA1</b></p> <p style="text-align: center;"><b>Increases in radiation levels within Safe Shutdown Buildings that impede operation of systems required to maintain safe operations OR to establish or maintain COLD SHUTDOWN</b></p> <p style="text-align: center;"><b>Applicable Modes:</b></p>							<p>Area radiation levels of greater than 15 mRem/hr in any of the following areas:</p> <ul style="list-style-type: none"> <li>Control Room</li> <li>Central Alarm Station</li> </ul>	<p>Area radiation levels of greater than 6000 mRem/hr in a Safe Shutdown Building, as determined by <u>either</u>:</p> <ul style="list-style-type: none"> <li>area radiation surveys</li> <li>installed or portable radiation monitors</li> </ul>
							<p>Access is required to maintain safe operation or perform a safe shutdown, as determined by the Shift Manager.</p>	
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>			

NOTE

This IC addresses increased radiation levels that impede necessary access to operating stations or other areas containing equipment that must be operated manually in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant.

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Initiating Conditions							Entry Criteria	
<b>GA2</b>  <b>Damage to irradiated fuel</b>  <b>Applicable Modes:</b>							Damage to irradiated fuel	
							A valid HIGH alarm on <u>one or more</u> of the following radiation monitors: <ul style="list-style-type: none"> <li>• SPENT FUEL POOL D21-K332</li> <li>• UPPER POOL AREA 1D21-K083</li> <li>• FUEL PREP POOL D21-K322</li> <li>• FHB VENT EXH GAS D17-K716</li> <li>• CNTMT ATMOS GAS 1D17-K686</li> </ul>	Water level observed to be below top of the gate sill separating <u>any</u> of the following containing irradiated fuel: <ul style="list-style-type: none"> <li>• Reactor Cavity</li> <li>• FHB Fuel Storage and Preparation Pool</li> <li>• FHB Fuel Transfer Pool</li> <li>• FHB Spent Fuel Storage Pool</li> <li>• FHB Cask Pit</li> <li>• CNTMT Fuel Storage Pool</li> <li>• CNTMT Fuel Transfer Pool</li> </ul>
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>			

**GA2**

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**NOTE**

The intent of this EAL is to allow observations from plant personnel to be factored into the declaration decision and is not intended to direct an entry into an area solely to observe pool level. The gate sill is the lip between the pools where the bottom of the gate would sit if installed.

NOTE

This EAL is only applicable to emergency conditions and is NOT applicable to pre-planned evolutions such as a reactor drain down to remove or re-install the reactor head during refueling operations.

NOTE

Damage to irradiated fuel is defined as; a degraded fuel bundle that results in the release of fission product gasses normally present in the fuel rod gap to the surrounding environment.

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Category H: Increased Radiation Release to the Environment

Initiating Conditions							Entry Criteria							
<div><b>HU1</b></div> <div>Any unplanned release of gaseous radioactivity to the environment that exceeds two times the ODCM Control limit for 60 minutes or greater.</div> <div>Applicable Modes:</div> <table><tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr></table>							1	2	3	4	5	D	A valid reading greater than TWO times the HIGH alarm setpoint on <u>one or more</u> of the following plant gaseous effluent monitors lasting greater than or equal to 60 minutes:	
							1	2	3	4	5	D		
							<ul style="list-style-type: none"><li>• PLANT VENT GAS 1D17-K786</li><li>• OG VENT PIPE GAS 1D17-K836</li><li>• TB/HB VENT GAS 1D17-K856</li><li>• PLANT VENT GAS 2D17-K786</li></ul>							
Chemistry sample analysis methods <u>CANNOT</u> confirm within 60 minutes of receipt of the HIGH alarm, on <u>one or more</u> of the plant gaseous effluent monitors, that effluent levels are less than two times ODCM 3.11.2.1 limits.														
							Routine or as required sample analysis indicates a release rate greater than two times ODCM 3.11.2.1 limits.							
							The release lasts for greater than or equal to 60 minutes.							

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**NOTE**

It is NOT intended that the release be averaged over 60 minutes. Further, the Emergency Coordinator should NOT wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release will exceed TWO times the ODCM Control 3.11.2.1 limit for greater than 60 minutes.

Initiating Conditions							Entry Criteria			
<div>HU2</div> <div>Any unplanned release of liquid radioactivity to the environment that exceeds two times the ODCM Control limit for 60 minutes or greater.</div>							<div>A valid reading greater than 1.2E3 cpm above background for <u>one or more</u> of the following liquid process monitors lasting greater than or equal to 60 minutes:</div> <div><div><div>•</div><div>EMERGENCY SERVICE WATER LOOP A      D17-K604</div></div><div><div>•</div><div>EMERGENCY SERVICE WATER LOOP B      D17-K605</div></div></div>	<div>A valid reading greater than 20 times the HIGH-HIGH alarm setpoint on NCC HX and RW EFF Common Process Radiation Recorder 0D17-R0170.</div>	<div>Routine or as required sample analysis indicates a release rate greater than two times ODCM Control 3.11.1.1 limits.</div>	
							<div>Chemistry sample analysis methods <u>CANNOT</u> confirm within 60 minutes of receipt of the HIGH-HIGH alarm, on <u>either</u> ESW Loop A or B radiation monitors, that liquid release levels are less than two</div>	<div>Release CANNOT be terminated within 60 minutes of exceeding NCC HX and RW EFF Common Process Radiation Recorder HIGH-HIGH alarm setpoints.</div>	<div>The release lasts for greater than or equal to 60 minutes.</div>	
<div>Applicable Modes:</div>							<div>times the ODCM Control 3.11.1.1 limits.</div>			
1	2	3	4	5	D					

**HU2**

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**NOTE**

It is NOT intended that the release be averaged over 60 minutes. Further, the Emergency Coordinator should NOT wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release will exceed TWO times the ODCM Control 3.11.1.1 limit for greater than 60 minutes.

Initiating Conditions							Entry Criteria				HA1   
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\* Perform an emergency dose calculation (i.e., CADAP run) within 15 minutes to determine if the Site Area Emergency entry criteria is met.

**NOTE 1**

ODCM [coolant activity] methodology is used to determine the threshold for this EAL. The Site Area Emergency threshold for EAL, HS1, is established using a clad damage source term. As a result, if the EAL threshold for the Alert is met AND there is clad damage, then the Site Area Emergency thresholds may be exceeded.

**NOTE 2**

It is NOT intended that the release be averaged over 15 minutes. Rather, the Emergency Coordinator should declare the event as soon as it is determined that the release will exceed 200 times the ODCM Control 3.11.2.1 limit for greater than 15 minutes.

Initiating Conditions							Entry Criteria			HA2   <
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**HA2**

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**NOTE**

It is NOT intended that the release be averaged over 15 minutes. Rather, the Emergency Coordinator should declare the event as soon as it is determined that the release will exceed 200 times the ODCM Control 3.11.1.1 limit for greater than 15 minutes.



Initiating Conditions							Entry Criteria				HS1  SITE  AREA  EMERGENCY
<div>HS1</div> <div>Site Boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mRem TEDE dose OR 500 mRem CDE Child Thyroid dose for the actual or projected duration of the release</div>							<div>A valid indication greater than the listed reading for <u>one or more</u> of the following plant gaseous effluent monitors:</div> <div><div><div>• PLANT VENT GAS 1D19-K300 3.8E-1 μCi/cc</div><div>• OG VENT GAS 1D19-K400 2.2E0 μCi/cc</div><div>• TB/HB VENT GAS 1D17-K856 1.6E4 cpm</div><div>• PLANT VENT GAS 2D19-K300 6.0E-1 μCi/cc</div></div></div>	<div>Emergency dose calculations, using actual meteorology indicate that <u>one or more</u> of the following are met at the Site Boundary:</div> <div><div>• Greater than 100 mRem TEDE</div><div>• Greater than 500 mRem CDE Child Thyroid</div></div>	<div>Field survey results indicate that <u>one or more</u> of the following have been met at the Site Boundary:</div> <div><div>• Greater than 100 mRem/hr Whole Body</div><div>• Greater than 500 mRem CDE Child Thyroid</div></div>		
							<div>Emergency dose calculations <u>CANNOT</u> confirm, within 15 minutes of exceeding limit, that levels at the Site Boundary are less than 100 mRem TEDE and 500 mRem CDE Child Thyroid dose using actual meteorology.</div>		<div>Dose rates are expected to continue for greater than or equal to 1 hour.</div>		
<div>Applicable Modes:</div> <div><div>1</div><div>2</div><div>3</div><div>4</div><div>5</div><div>D</div></div>											

**HS1**

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# HG1

## GENERAL EMERGENCY

Exceeding the entry criteria for HG1 may require the initiation of an RPV emergency depressurization per <EOP-05>. Ensure Shift Manager is notified immediately whenever the above entry criteria for a General Emergency is met.

Category I: Control Room Evacuation

Initiating Conditions							Entry Criteria						
<b>IA1</b>  <b>Control Room evacuation has been initiated</b>  <b>Applicable Modes:</b>							Entry into <ONI-C61>.						
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>								

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NOTE

An inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency per IS1.

Initiating Conditions							Entry Criteria						
<p style="text-align: center;"><b>IS1</b></p> <p style="text-align: center;"><b>Control Room evacuation has been initiated, AND plant control <u>CANNOT</u> be established within 15 minutes.</b></p> <p style="text-align: center;"><b>Applicable Modes:</b></p>							Entry into <ONI-C61>.						
							<p>Within 15 minutes of entry into &lt;ONI-C61&gt;, Operator(s) located at the remote shutdown controls <u>CANNOT</u> establish control of <u>one or more</u> of the following parameters per &lt;IOI-11&gt;:</p> <ul style="list-style-type: none"> <li>• RPV level</li> <li>• RPV pressure</li> <li>• Suppression Pool temperature</li> <li>• Reactor power</li> <li>• Decay heat removal, if required</li> </ul>						
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>								

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**NOTE**

A maximum 15 minute time frame for the physical transfer of control of “required” systems was established by <NUMARC/NESP-007>. Control at the Remote Shutdown Areas is accomplished by the repositioning of control transfer switches per <IOI-11>. Control is assumed unless indication of the absence of control is present.

Category J: Loss of Annunciators or Indication

Initiating Conditions							Entry Criteria	
<b>JU1</b>  <b>Loss of most annunciators or indication in the Control Room for greater than 15 minutes</b>  <b>Applicable Modes:</b>							Unplanned loss of most Control Room annunciators for greater than 15 minutes.	Unplanned loss of most Control Room indication for greater than 15 minutes.
							In the Shift Manager's opinion, increased surveillance is warranted to safely operate the plant.	
<b>1</b>	<b>2</b>	<b>3</b>						

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NOTE

Quantification of "most" is left to the Shift Manager. It is NOT intended that plant personnel perform a detailed count of the instrumentation lost, but rather make a judgment call with approximately 75% being the threshold.

Initiating Conditions						Entry Criteria	
<p align="center"><b>JA1</b></p> <p align="center"><b>Loss of most annunciators or indication in the Control Room with <u>either</u>: (1) a significant transient in progress, OR (2) compensatory indicators are <u>NOT</u> available.</b></p> <p align="center"><b>Applicable Modes:</b></p>						Unplanned loss of most Control Room annunciators for greater than 15 minutes.	Unplanned loss of most Control Room indication for greater than 15 minutes.
						In the Shift Manager's opinion, increased surveillance is warranted to safely operate the plant.	
						A significant plant transient is in progress.	Compensatory indications, i.e., Integrated Computer System (ICS), are <u>NOT</u> available.
<b>1</b>	<b>2</b>	<b>3</b>					

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**NOTES**

- Quantification of “most” is left to the Shift Manager. It is NOT intended that plant personnel perform a detailed count of the instrumentation lost, but rather make a judgment call with approximately 75% being the threshold.
- A “significant transient” includes response to automatic OR manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injection, or thermal power oscillations of 10% or greater.

Initiating Conditions						Entry Criteria	
<b>JS1</b>  <b>Inability to monitor a significant transient in progress</b>						Loss of most Control Room annunciators.	Loss of most Control Room indication.
						Compensatory indicators, i.e., Integrated Computer System (ICS), are <u>NOT</u> available.	
						A significant transient is in progress.	
						Sufficient indication is <u>NOT</u> available to directly monitor plant critical safety parameters for EOPs entered due to the transient.	
Applicable Modes:							
1	2	3					

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Category K: Loss of Communications

Initiating Conditions						Entry Criteria	
<b>KU1</b>  <b>Loss of onsite OR in-plant communications capabilities</b>						Loss of <u>all</u> five Plant Public Address System channels.	
						Loss of <u>all</u> of the following Plant Radio System channels: <ul style="list-style-type: none"> <li>• Channel 1</li> <li>• Channel 2</li> <li>• Channel 3</li> </ul>	
Applicable Modes:							
1	2	3	4	5	D		

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Initiating Conditions						Entry Criteria					
<b>KU2</b>  <b>Significant degradation of offsite communications capabilities</b>						Loss of the State and County Notification Circuit (5-way) reported to the Control Room.					
						Loss of offsite long distance calling capability on <u>three or more</u> of the following systems circuits for greater than 15 minutes: <ul style="list-style-type: none"> <li>• Control Room private (259-) lines</li> <li>• Private Branch Exchange, Service Building (“5000”) Switch</li> <li>• Private Branch Exchange, Warehouse Building (“6000”) Switch</li> <li>• Company Off-Premise Exchange</li> </ul>					
Applicable Modes:											
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>						

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## NOTES

- A loss of the “5 Way” Circuit refers to the inability to contact one or more of the four offsite contacts: the State of Ohio, and the counties of Ashtabula, Geauga, and Lake. Testing to determine “5-Way” operability or to initiate circuit restoration actions are governed under <PSI-0007>.
- Direct (259-) off-site calling capability from the Control Room via private lines refer to: autodialer at the US console, and private (259-) lines on the superphones and line at the SAS console.

Category L: Natural or Destructive Phenomena  
<P00067>

Initiating Conditions							Entry Criteria					LU1  <
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Initiating Conditions							Entry Criteria					
<div>LA1</div> <div>Natural OR destructive phenomena affecting Safe Shutdown Buildings</div> <div>Applicable Modes:</div> <div><div>1</div><div>2</div><div>3</div><div>4</div><div>5</div><div>D</div></div>							Control Room receives report from plant personnel who felt an earthquake.	Report of visible damage to any Safe Shutdown Building caused by <u>any</u> of the following: <ul style="list-style-type: none"><li>tornado</li><li>aircraft, barge or train crash</li><li>other natural or destructive phenomena</li></ul>	Sustained high winds with a velocity greater than 90 mph for 15 minutes or longer.	Greater than <EOP-03> Maximum Safe Operating Value for Area Water Level (internal flooding)	Report by plant personnel confirming a turbine failure which results in penetration of the turbine casing.	
							<u>Either</u> of the following indications present: <ul style="list-style-type: none"><li>YELLOW seismic switch indicator light on local Seismic Monitoring Panel OH51-P021.</li><li>RED light on Seismic Monitoring Panel OH13-P969.</li></ul>				Missile generated from the turbine failure resulting in <u>either</u> : <ul style="list-style-type: none"><li>damage to Safe Shutdown equipment</li><li>penetration of a Safe Shutdown Bldg.</li></ul>	

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Category M: Release of Toxic or Flammable Gases

Initiating Conditions							Entry Criteria	
<b>MU1</b>  <b>Release of toxic OR flammable gases affecting the Protected Area boundary deemed detrimental to the safe operation of the plant</b>  <b>Applicable Modes:</b>							Toxic or flammable gas concentrations detected within the Protected Area.	Report by local, county, or State officials for a potential evacuation of site personnel based on an offsite event.
							Normal operation of the plant is impeded due to access restrictions.	
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>			

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NOTE

A toxic or flammable gas release is considered to be impeding normal operations due to access restrictions if it is of sufficient magnitude that access to areas normally accessed to plant operator rounds is restricted. It also includes releases where access to these areas is possible only through the use of protective equipment, such as respirators since this limits the operators visibility and mobility thereby affecting “normal” plant operations.

Initiating Conditions							Entry Criteria		
<b>MA1</b>  <b>Release of toxic OR flammable gases within a Safe Shutdown Building which jeopardizes operation of systems required to maintain safe operations OR to establish or maintain COLD SHUTDOWN</b>  <b>Applicable Modes:</b>							Entry of toxic or flammable gases into Safe Shutdown Buildings or Areas.		
							Toxic gas in concentrations considered life-threatening	Flammable gas estimated or determined to be in explosive concentrations	Plant personnel <u>NOT</u> able to perform actions necessary to establish and maintain Mode 4 while utilizing appropriate protective equipment.
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>				

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**NOTE**

This IC addresses increased toxic or flammable gas levels that impede necessary access to operating stations or other areas containing equipment that must be operated manually in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant.

Category N: Security Events

Initiating Conditions							Entry Criteria			
<p><b>NU1</b></p> <p><b>Confirmed SECURITY CONDITION OR threat which indicates a potential degradation in the level of safety of the plant</b></p> <p><b>Applicable Modes:</b></p>										
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>		SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor.	A credible site specific security threat notification.	A validated notification from NRC providing information of an aircraft threat.	

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Initiating Conditions							Entry Criteria						
<p><b>NA2</b></p> <p><b>Notification of an Airborne Attack Threat</b></p> <p><b>Applicable Modes:</b></p>							A validated notification from the NRC of an airliner attack threat less than 30 minutes away.						
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>								

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**NOTE**

Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant. The status and size of the plane may be provided by NORAD through the NRC.

Initiating Conditions							Entry Criteria						
<p style="text-align: center;"><b>NA3</b></p> <p><b>HOSTILE ACTION within the Owner Controlled Area</b></p> <p><b>Applicable Modes:</b></p>							<p>A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.</p>						
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>								

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### NOTES

- HOSTILE ACTION is defined as an act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.
- HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

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Initiating Conditions						Entry Criteria					
<p style="text-align: center;"><b>NS2</b></p> <p style="text-align: center;"><b>HOSTILE ACTION within the PROTECTED AREA</b></p> <p><b>Applicable Modes:</b></p>						<p>A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.</p>					
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>						

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**NOTES**

- HOSTILE ACTION is defined as an act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.
- HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

Initiating Conditions							Entry Criteria	
<p style="text-align: center;"><b>NG1</b></p> <p style="text-align: center;"><b>HOSTILE ACTION</b> resulting in loss of physical control of the facility</p> <p style="text-align: center;"><b>Applicable Modes:</b></p>							<p>A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.</p>	<p>A HOSTILE ACTION has caused failure of spent fuel cooling systems and imminent fuel damage is likely for freshly off-loaded reactor core in the pool.</p>
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>			

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**NOTES**

- HOSTILE ACTION is defined as an act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.
- HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

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Category O: Emergency Coordinator's Judgment

Initiating Conditions						Entry Criteria					
<p><b>OU1</b></p> <p><b>Other conditions existing, which in the judgment of the Emergency Coordinator, warrant declaration of an Unusual Event</b></p> <p><b>Applicable Modes:</b></p>						<p>Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant.</p>					
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>						

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**NOTE**

For those cases where the degradation in the level of safety of the plant is tied to equipment or system malfunctions, the decision that the component is degraded should be based upon its functionality and **NOT** its operability.

Initiating Conditions						Entry Criteria
<p><b>OA1</b></p> <p><b>Other conditions existing, which in the judgment of the Emergency Coordinator, warrant declaration of an Alert</b></p> <p><b>Applicable Modes:</b></p>						Events are in progress or have occurred which indicate an actual or potential degradation of systems needed for the protection of the public and which warrant increased monitoring of plant functions.
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>	

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**NOTE**

This IC 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 Emergency Coordinator to fall under the Alert emergency class. This includes a determination by the Emergency Coordinator that additional assistance similar to that provided by the TSC and OSC staffs, including a transfer of the Emergency Coordinator responsibilities to the TSC, is necessary for the event to be effectively mitigated. Transfer of Emergency Coordinator duties for classification, offsite notifications and PAR decisions, is used as an initiator since an event significant enough to warrant transfer of command and control is a substantial reduction in the level of safety of the plant. It is not intended to declare an Alert if Emergency Coordinator responsibilities are transferred out of the Control Room for convenience.



Initiating Conditions							Entry Criteria						
<p style="text-align: center;"><b>OS1</b></p> <p style="text-align: center;"><b>Other conditions existing, which in the judgment of the Emergency Coordinator, warrant declaration of a Site Area Emergency</b></p>							<p>Other conditions exist which indicate an actual or likely major failure of plant functions needed for protection of the public.</p>						
Applicable Modes:													
1	2	3	4	5	D								

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Initiating Conditions						Entry Criteria		
<p style="text-align: center;"><b>OG1</b></p> <p style="text-align: center;"><b>Other conditions existing, which in the judgment of the Emergency Coordinator, warrant declaration of a General Emergency</b></p> <p style="text-align: center;"><b>Applicable Modes:</b></p>								
<b>1</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>5</b>	<b>D</b>	Other conditions exist which indicate an actual or imminent substantial core degradation with the potential loss of Containment integrity.	Potential for an uncontrolled release which can reasonably be expected to be greater than 1 Rem TEDE at the Site Boundary.	Potential for an uncontrolled release which can reasonably be expected to be greater than 5 Rem CDE Child Thyroid at the Site Boundary.

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## SRO Question 10

### RPS Instrumentation 3.3.1.1

#### 3.3 INSTRUMENTATION

##### 3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

#### ACTIONS

-----NOTE-----  
Separate Condition entry is allowed for each channel.  
-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u> A.2 Place associated trip system in trip.	12 hours

#### SURVEILLANCE REQUIREMENTS

- NOTES-----
1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
  2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.
-



## SRO Question 17

RPS Instrumentation  
3.3.1.1

### 3.3 INSTRUMENTATION

#### 3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

#### ACTIONS

-----NOTE-----  
Separate Condition entry is allowed for each channel.  
-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u> A.2 Place associated trip system in trip.	12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip.	6 hours
	<u>OR</u> B.2 Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

## SRO Question 17

RPS Instrumentation  
3.3.1.1

### ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Be in MODE 3.	12 hours
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

## SRO Question 17

RPS Instrumentation  
3.3.1.1

### SURVEILLANCE REQUIREMENTS

- NOTES-----
1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
  2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.
- 

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.4	<p>-----NOTE----- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -----</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.7	<p>-----NOTE----- Only required to be met during entry into MODE 2 from MODE 1. -----</p> <p>Verify the IRM and APRM channels overlap.</p>	7 days

(continued)

## SRO Question 17

RPS Instrumentation  
3.3.1.1

### SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.13	<p>-----NOTES-----</p> <p>1. Neutron detectors are excluded.</p> <p>2. For IRMs, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.</p> <p>-----</p> <p>Perform CHANNEL CALIBRATION.</p>	24 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.19	Perform CHANNEL FUNCTIONAL TEST	31 days

Table 3.3.1.1-1 (page 1 of 3)  
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.19	≤ 122/125 divisions of full scale
b. Inop	2	3	H	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	3	I	SR 3.3.1.1.15 SR 3.3.1.1.19	NA



## SRO Question 18

RPS Instrumentation  
3.3.1.1

### 3.3 INSTRUMENTATION

#### 3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

#### ACTIONS

-----NOTE-----  
Separate Condition entry is allowed for each channel.  
-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u> A.2 Place associated trip system in trip.	12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip.	6 hours
	<u>OR</u> B.2 Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

## SRO Question 18

RPS Instrumentation  
3.3.1.1

### ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Be in MODE 3.	12 hours

RPS Instrumentation  
3.3.1.1

Table 3.3.1.1-1 (page 1 of 3)  
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	
b. Inop	2	3	H	SR 3.3.1.1.4 SR 3.3.1.1.15	

## SRO Question 19

### PMT Data

1. Suction Valve Post Maintenance Tests: 1E51-F010 (EO, EC, ST), 1E51-F031 (EO, EC, ST).

- a. Close 1E51-F010, RCIC PUMP CST SUCTION VALVE, measure stroke time and record the following:

- 1) Exercise Closed Results:  
☒ SAT    ☐ UNSAT

- 2) Stroke Time - Closed direction.

24.8 seconds

Reference Value = 25.0 seconds

Acceptable Range = 21.3 to 28.7 seconds

☒ SAT    ☐ UNSAT

- b. Open 1E51-F031, RCIC PUMP SUPR PL SUCT ISOL, measure stroke time and record the following:

- 1) Exercise Open Results:  
☒ SAT    ☐ UNSAT

- 2) Stroke Time - Open direction.

19.5 seconds

Reference Value = 16.9 seconds

Acceptable Range = 14.4 to 19.4 seconds

☐ SAT    ☒ UNSAT

- c. Close 1E51-F031, RCIC PUMP SUPR PL SUCT ISOL, measure stroke time, and record the following:

- 1) Exercise Closed Results:  
☒ SAT    ☐ UNSAT

- 2) Stroke Time - Closed direction.

18.9 seconds

Reference Value = 17.0 seconds

Acceptable Range = 14.5 to 19.5 seconds

☒ SAT    ☐ UNSAT

## SRO Question 19

### PMT Data

- d. Open 1E51-F010, RCIC PUMP CST SUCTION VALVE, measure stroke time, and record the following:

- 1) Exercise Open Results:

☒ SAT    ☐ UNSAT

- 2) Stroke Time - Open direction.

23.8 seconds

Reference Value = 24.2 seconds

Acceptable Range = 20.6 to 27.8 seconds

☒ SAT    ☐ UNSAT

## SRO Question 20

### Primary Containment and Drywell Isolation Instrumentation 3.3.6.1

#### 3.3 INSTRUMENTATION

##### 3.3.6.1 Primary Containment and Drywell Isolation Instrumentation

LCO 3.3.6.1 The primary containment and drywell isolation instrumentation for each function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

#### ACTIONS

-----NOTE-----  
Separate Condition entry is allowed for each channel.  
-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.b, 5.b, and 5.d  <u>AND</u> 24 hours for Functions other than Functions 2.b, 5.b, and 5.d
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour

(continued)

## SRO Question 20

### Primary Containment and Drywell Isolation Instrumentation 3.3.6.1

#### ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated main steam line (MSL).	12 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	36 hours

## SRO Question 20

### Primary Containment and Drywell Isolation Instrumentation 3.3.6.1

Table 3.3.6.1-1 (page 1 of 6)  
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS
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1. Main Steam Line Isolation

b. Main Steam Line  
Pressure - Low

1

2

E

SR 3.3.6.1.4  
SR 3.3.6.1.5  
SR 3.3.6.1.6

(continued)

## SRO Question 20

### Primary Containment and Drywell Isolation Instrumentation 3.3.6.1

Table 3.3.6.1-1 (page 2 of 6)  
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS
2. Primary Containment and Drywell Isolation				
b. Drywell Pressure — High	1,2,3	2 <sup>(b)</sup>	H	

(continued)

(b) Required to initiate the drywell isolation function.



## SRO Question 20

### Primary Containment and Drywell Isolation Instrumentation 3.3.6.1

Table 3.3.6.1-1 (page 6 of 6)  
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS
5. RHR System Isolation				
b. Reactor Vessel Water Level	1, 2 <sup>(g)</sup> , 3 <sup>(g)</sup>	2	F	
	2 <sup>(e)</sup> , 3 <sup>(e)</sup> , 4, 5	2 <sup>(f)</sup>	J	
d. Drywell Pressure — High	1, 2, 3	2	F	

(e) With reactor vessel steam dome pressure less than the RHR cut in permissive pressure.

(f) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

(g) With reactor vessel steam dome pressure greater than or equal to the RHR cut in permissive pressure.