# CGS SRO Written Exam Answer Key
# September, 2004

The reactor is at 330 degrees, the mode switch is in STARTUP, and power is mid-scale on IRM range 2. The reactor operator tasked with withdrawing control rods reports Source Range Monitor (SRM) Channel A is reading higher than the other three channels. The followup investigation finds SRM Channel A is operating correctly, but the pulse height discriminator on SRM channels B, C, and D was set higher than desired due to an outage maintenance procedure deficiency.

The technical specification/LCS action statement (attached) for this situation requires:

A. one rod block monitor channel be placed in a tripped condition within 7 days.

B. one rod block monitor channel be placed in a tripped condition within 1 hour.

C. control rod withdrawal be suspended immediately.

D. suspending all rod movement except by scram.

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| ANSWER: | B |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 215004A3.04 & 3.6 - Source Range Monitor/Control rod block Status |
| REFERENCE: | Tech Spec 3.3.1.2, 3.3.2.1, Tech Spec Bases B 3.3.1.2, B 3.3.2.1, Licensee Controlled Specifications 1.3.2.1, LO000132, Source Range Monitor, Revision 9, Page 24, and LO000148, RMCS, Revision 10, Page 15. |
| SOURCE: | New Question – SRO Tier 2, Group 1 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 3 |
| ATTACHMENTS: | Tech Specs 3.3.1.2, 3.3.2.1, and LCS 1.3.2.1. |
| JUSTIFICATION: | There are 3 inoperable SRMs in Mode 2 of a startup with power at mid-scale of IRM range 2. TS 3.3.1.2 requires 3 operable SRMs in this mode. Since one of three required SRMs is operable, action 3.3.1.2.A.1 applies (restore the required SRMs within 4 hours) making answer C (applicable for 3.3.1.2.B.1) incorrect. SRM inoperability does not affect TS RBM operability and therefore TS 3.3.2.1 does not apply. This eliminates answer D. LCS 1.3.2.1 (Rod Block Monitor) states that with one or more functions with two or more required channels inoperative, action statement A.1 is |

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| | N/A and action statement B.1 is in effect. Therefore the correct answer is B. |
| 10CFR55 BASIS: | 10CFR55.43 (2) |
| COMMENTS: | Q 1 |

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With the plant in a refueling outage, a control rod is to be withdrawn for testing. When the control rod is selected for withdrawal, a rod block is activated. This rod block would be the result of which of the following sets of conditions?

A. The mode switch is in START/HOT STANDBY, a control rod is selected, and the Fuel Hoist Interlock light is lit on the refueling bridge.

B. The mode switch is in START/HOT STANDBY, a control rod is selected, and the Hoist Loaded light is lit on the refueling bridge.

C. The mode switch is in REFUEL, the refueling bridge is over the core, and the Hoist Loaded light is lit on the refueling bridge.

D. The mode switch is in REFUEL, the refueling bridge is over the core, and the Fuel Hoist Interlock light is lit on the refueling bridge.

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 259023/AK3.02 & 3.8 – Refueling Accidents/Interlocks associated with fuel handling equipment. |
| REFERENCE: | SD000207, Revision 10, Page 29. Procedure 2.14.1, Refueling Bridge Operation, Revision 32, Sections 5.10 and 5.11. |
| SOURCE: | New Question – SRO Tier 1, Group 1 |
| LEARNING OBJECTIVE: | LO-5795 – Identify the conditions that will cause rod blocks. |
| RATING: | 2 |
| ATTACHMENTS: | None |
| JUSTIFICATION: | Answers A and D are incorrect because the Fuel Hoist Interlock prevents renders the fuel hoist inoperative but does not input to the rod block logic. Answer B is incorrect because the Hoist Loaded light does not input into the rod block logic with the mode switch in startup. With the mode switch in refuel, the bridge over the core, and the hoist loaded a rod block will be applied. Therefore the correct answer is C. |
| 10CFR55 BASIS: | 10CFR55.43 (7) |
| COMMENTS: | Q 2 |

# CGS SRO Written Exam Answer Key
# September, 2004

During a post-maintenance test, it is discovered Standby Gas Treatment System (SGTS) train A will not startup on an auto initiation signal and is declared inoperable at 1200 on July 1. At 1600, SGT train B is declared inoperable due to a common cause failure. At 2100, SGTS A is restored to an operable status.

With the plant currently in Mode 1, and based on this sequence of events, when do the attached station technical specifications require the reactor be in Mode 3?

A. July 1 at 2300

B. July 8 at 1600

C. July 9 at 0000

D. July 9 at 0400

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| ANSWER: | D |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 261000K4.01 & 3.8 - Standby Gas Treatment/Automatic system initiation. |
| REFERENCE: | LO000144, Standby Gas Treatment, Revision 11, Section 5. |
| SOURCE: | Bank Question – SRO Tier 2, Group 1 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 3 |
| ATTACHMENTS: | Technical Specification 3.6.4.3. |
| JUSTIFICATION: | Answer A is incorrect because it is based on the entry into TS 3.0.3 at 1600, however 3.0.3 is exited at 2100. Answer B is incorrect because it does not take into account the completion of Action Statement B. Answer C is incorrect because it is based on the initial entry of train A into the TS which was exited at 2100. The correct answer is D because the action statements for the inoperable train are 7 days 12 hours from 1600 on July 1. |

10CFR55 BASIS: 10CFR55.43 (2)

COMMENTS: Q 3

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# CGS SRO Written Exam Answer Key
# September, 2004

The reactor is operating at 95 percent power when a rod drift alarm occurs due to peripheral rod 06-35 slowly withdrawing from the reactor core without being selected. The CONTINUOUS INSERT Pushbutton is depressed and the control rod continues to drift outward.

Which of the following is correct with regard to this situation?

A. Procedure ABN-POWER should be entered as this procedure addresses a potential fluid flashing phenomenon than can slow the scram time for multiple control rods.

B. Procedure ABN-POWER should be entered because this procedure addresses uncontrolled rod withdrawals at a high reactor power level.

C. Procedure ABN-ROD should be entered and the reactor scrammed because there is the potential for fuel damage as unpreconditioned fuel nodes are exposed.

D. Procedure ABN-ROD should be entered and the reactor scrammed because a mispositioned control rod could place the reactor in an unanalyzed control rod pattern.

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 201002A2.02 &3.3 - RMCS/Rod Drift Alarm. |
| REFERENCE: | ABN-ROD, Control Rod Faults, Revision 6, Step 3.1.2. |
| SOURCE: | New Question – SRO Tier 2, Group 2 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 2 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | The correct procedure to enter for a drifting rod is ABN-ROD making answer A and B incorrect. According to ABN-ROD, the bases for this action is to prevent fuel damage as stated in answer C. |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 4 |

# CGS SRO Written Exam Answer Key
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The plant is at 80 percent power when a malfunction results in the Main Turbine Bypass System being declared inoperable.

What is the technical specification design bases event requiring this system to be operable at this power level?

A. A turbine runback due a loss of load.

B. A turbine trip.

C. A feedwater controller failure resulting in a maximum feedwater demand event.

D. A reactor recirc controller failure resulting in a rapid increase in reactor recirc pump speed.

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 241000K6.10 & 3.7 - Reactor/Turbine Pressure Regulating System/Bypass Valves. |
| REFERENCE: | Technical Specification Bases B 3.7.6 - Main Turbine Bypass System. |
| SOURCE: | New Question – SRO Tier 2, Group 2 |
| LEARNING OBJECTIVE: | CAF. |
| RATING: | 1 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | The referenced document states the design bases event is a feedwater controller failure resulting in a maximum feedwater demand event. Therefore the correct answer is C. |
| 10CFR55 BASIS: | 10CFR55.43 (2) |
| COMMENTS: | Q 5 |

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# CGS SRO Written Exam Answer Key
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# CGS SRO Written Exam Answer Key
# September, 2004

On May 2, Emergency Diesel Generator 1 (EDG 1) has been paralleled to SM-7 and is operating at full load. At 0900, the SM-7 bus lockout relay (86) actuates leaving SM-7 deenergized and the EDG trips on overspeed. At 1100 the cause of the lockout is discovered to be a ground on SL-73 and SL-73 is declared Inoperable. At 1400, SM-7 and SL-73 Operability is restored and at 1500, EDG 1 Operability is restored. At 2200, SL-71 is declared Inoperable due to several non-qualified relays that were recently installed. There is no common cause failure/issue and all required technical specification surveillances have been performed.

With this sequence of events, at what time do Technical Specifications 3.8.7 and 3.8.1 (attached) require the plant to be in Mode 3?

A. May 5 at 2100

B. May 3 at 0700

C. May 3 at 1500

D. May 3 at 1800

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295003AA1.02 & 4.3* - Partial or Complete Loss of AC/Emergency Diesel Generators |
| REFERENCE: | Technical Specifications 3.8.1, AC Sources - Operating, and 3.8.7, Distribution Systems - Operating. |
| SOURCE: | New Question – SRO Tier 1, Group 1 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 3 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | Answer A is incorrect because it is based on TS 3.8.1 which was exited at 1400. Answer B is incorrect because it is based using the 3.8.7.A.1 8 hour completion time and this condition was exited at 1400. Answer D is incorrect because it is based on the Inoperability |

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| | declared at 2200. However, the clock started at 1100 with the first 3.7.1 entry and does not reset for 16 hours. (Action statement C provides an additional 12 hours to reach Mode 3.) Thus, the second entry is a multiple entry and therefore the correct answer is C |
| 10CFR55 BASIS: | 10CFR55.43 (2) |
| COMMENTS: | Q 6 |

The plant is at Rated Thermal Power (RTP) when an electrical transient results in a partial loss of forced core flow through the reactor. Following the transient, plant conditions are as follows:

- APRM indicated power is 75 percent RTP.
- All bypass valves are closed.
- Reactor Recirc flow is 20 percent of rated flow.
- Reactor pressure and level are stable in the normal band.
- One reactor recirc pump is running

What procedure should the control room crew enter to mitigate this event?

A. ABN-CORE, Unplanned Core Operating Conditions.

B. ABN-RRC-LOSS, Loss of Reactor Recirc Flow.

C. PPM 5.1.1, RPV Control.

D. ABN-POWER, Unplanned Reactor Power Change.

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295001AA1.02 & 3.3 - Partial or Complete Loss of Forced Core Flow Circulation |
| REFERENCE: | PPM 5.1.1, RPV Control and Procedure 1.3.1, Conduct of Operations, step 4.11, Procedure Usage. |
| SOURCE: | New Question – SRO Tier 1, Group 1 |
| LEARNING OBJECTIVE: | CAF. |
| RATING: | 3 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | With reactor recirc flow at 20 percent a power to flow reactor scram |

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| | should have occurred at 70.6 percent APRM power. Therefore, the plant is in an ATWS condition which requires entering PPM 5.1.1, RPV Control. Therefore the correct answer is C. |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 7 |

# CGS SRO Written Exam Answer Key
# September, 2004

Following a 40 day refueling outage, the plant is being returned to Rated Thermal Power (RTP). When reactor power reaches 50 percent RTP, a DEHC common mode failure results in 3 bypass valves that will not open on demand (failed closed) should there be a high reactor pressure condition. The procurement engineer reports the repair parts “will not arrive on site for at least 3 days.” Current plant conditions are:

- Both reactor recirc loops are in operation
- The core MCPR is 1.48
- All control rods are operable
- RRC flow is 60%

Unless the bypass valves are repaired, the station technical specifications require:

A. reactor power be ≤25% RTP within 2 hours.

B. reactor power be ≤25% RTP within 4 hours.

C. reactor power be limited to less than 30% RTP.

D. reactor power be limited to less than 65% RTP.

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| ANSWER: | D |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295025AK2.08 & 3.7 - High Reactor Pressure/Reactor/Turbine pressure regulating system. |
| REFERENCE: | Technical Specifications 3.2.2, 3.7.6, and the COLR. |
| SOURCE: | New Question – SRO Tier 1, Group 1 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 3 |
| ATTACHMENTS: | Technical Specifications sections 3.2.2, 3.7.6, and COLR pages 12 through 26. |

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| JUSTIFICATION: | The technical specification for the bypass valves states the bypass valves must be operable OR the MCPR be ≥ to that specified in the COLR. With an MCPR of 1.48, the COLR allows the reactor be operated up to 65% RTP making answer D correct. |
| 10CFR55 BASIS: | 10CFR55.43 (2) |
| COMMENTS: | Q 8 |

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The plant is in Mode 4 and a cable tray fire has been burning for 20 minutes below the control room floor. The fire has resulted in heavy smoke coming out from below the floor and the Shift Manager has just announced his decision to abandon the control room.

The Emergency Plan requires which of the following be declared?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

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| ANSWER: | B |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295016G2.4.29 & 4.0 - Control Room Abandonment/Knowledge of the Emergency Plan |
| REFERENCE: | Procedure *13.1.1, Classifying the Emergency, Revision 32, Attachment 5.1. |
| SOURCE: | New Question – SRO Tier 1, Group 1 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 1 |
| ATTACHMENTS: | Procedure *13.1.1 or EP Implementing Flowchart. |
| JUSTIFICATION: | The control room fire places the station in an unusual event status and the decision to abandon the control room places the station in an alert status per section 7.2 of the referenced procedure. Therefore the correct answer is B. |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 9 |

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After a Temporary Modification Request (TMR) had been reviewed and approved, the initiator arrives in the Control Room to discuss the TMR with the CRS. He explains that he needs to add a circuit breaker to the original TMR due to an industrial safety concern identified by the maintenance electricians.

The CRS should:

A. approve the change if there are no additional operational or nuclear safety concerns.

B. explain to the initiator the Shift Manager is procedurally required to approve the change.

C. explain to the initiator the system engineer must initial the change before the CRS can approve it.

D. explain to the initiator a new TMR must be submitted for approval.

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| ANSWER: | D |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 2.2.11 & 3.4 - Knowledge of the process for controlling temporary changes. |
| REFERENCE: | Procedure 1.3.9, Temporary Modifications, Revision 35, Step 2.13. |
| SOURCE: | New Question – SRO Generic Knowledge and Abilities. |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 1 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | The referenced procedure states the approved TMR must be closed and a new request be submitted if a technical change is being made. Therefore the correct answer is D |
| 10CFR55 BASIS: | 10CFR55.43 (3) |
| COMMENTS: | Q 10 |

A Nuclear Component Transfer List would be used for:

A. transferring a spent fuel cask containing irradiated fuel.

B. removing fuel from the reactor.

C. the initial placement of new fuel in the spent fuel pool.

D. moving a control rod blade.

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| ANSWER: | B |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 2.2.26 & 3.7 - Knowledge of refueling administrative requirements. |
| REFERENCE: | Procedure 1.3.59, Reactivity Management Program, Revision 7, Step 2.9.2.c. |
| SOURCE: | New Question – SRO Generic Knowledge and Abilities |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 1 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | The Nuclear Component Transfer List is the administrative tool for controlling the movement of fuel once it is installed in the spent fuel pool. Therefore the answer is B. |
| 10CFR55 BASIS: | 10CFR55.43 (7) |
| COMMENTS: | Q 11 |

The plant has entered the Emergency Plan and the distribution of Thyro-Block (Potassium Iodide) tablets is being considered.

Which of the following individuals can authorize the distribution of Thyro-Block?

A. Radiation Protection Manager

B. Radiological Emergency Manager

C. Technical Support Center Manager

D. Plant General Manager

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 2.4.44 & 4.0 - Knowledge of emergency plan protective action recommendations. |
| REFERENCE: | Procedure 13.2.1, Emergency Exposure Levels/Protective Action Guides, Revision 15, Attachment 5.4, Step D.3. |
| SOURCE: | New Question – Generic Knowledges and Abilities |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 1 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | The issuance of KI can be authorized by either the TSC Manager or Emergency Director. From the list of answers provided, the correct answer is C. |
| 10CFR55 BASIS: | 10CFR55.43 (4) |
| COMMENTS: | Q 12 |

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# CGS SRO Written Exam Answer Key
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# CGS SRO Written Exam Answer Key
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The plant experienced a transient that has resulted in the following plant conditions:

- Drywell temperature is 310 degrees
- All control rods are fully inserted
- Reactor pressure is 50 psig
- wetwell temperature is 170 degrees
- Reactor water level Narrow Range Indicators have been erratic for the last 30 minutes

Referring to the attached Caution 1 from the Emergency Operating Procedures, which of the following RPV level indicators and procedures should be used?

A. Level should be considered unknown and P.P.M. 5.1.4, RPV Flooding, should be entered.

B. The Upset Range indication of +7 inches should be used and level should be recovered using P.P.M 5.1.1, RPV Control.

C. The Shutdown Flooding Range indication of +7 inches should be used and level should be recovered using P.P.M 5.1.1, RPV Control..

D. Narrow Range indication of +7 inches should be used and level should be recovered using P.P.M 5.1.1, RPV Control..

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| ANSWER: | A |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295028EA2.03 & 3.9 - High Drywell Temperature/Reactor Water Level |
| REFERENCE: | Procedure 5.0.10, Flowchart Training Manual, Revision 7, Page 64. |
| | EOP Figure A, RPV Saturation Temperature. |
| SOURCE: | New Question – SRO Tier 1, Group 1 |

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| LEARNING OBJECTIVE: | CAF |
| RATING: | 2 |
| ATTACHMENTS: | EOP Figure A, RPV Saturation Temperature. |
| JUSTIFICATION: | The RPV pressure/drywell temperature is outside the saturation limit shown by Figure A making any erratic indicator unusable (answer D). Answers B and C are incorrect because the indicators do not meet the minimum usable levels prescribed by Caution 1. Therefore level cannot be determined and RPV Flooding is required (Answer A). |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 13 |

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The bases of the ATWS-RPT being set at the Low-Low RPV setpoint (Level 2) is:

A. there is no threat to the fuel when RPV water level is above Level 2.

B. to prevent the system initiating at Level 3 with feedwater available.

C. to reduce RPV power and pressure thereby maximizing ECCS injection flowrates.

D. to allow the operators time to mitigate the ATWS prior to tripping the RPTs.

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| ANSWER: | B |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295037EK2.03 & 4.2 - SCRAM Condition Present and Power Above APRM Downscale or Unknown/RPT |
| REFERENCE: | Technical Specification Bases B 3.3.4.2 |
| SOURCE: | New Question – SRO Tier 1, Group 1 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 1 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | The bases is to allow the feedwater system to recover level if available. Therefore, the correct answer is B. |
| 10CFR55 BASIS: | 10CFR55.43 (2) |
| COMMENTS: | Q 14 |

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# CGS SRO Written Exam Answer Key
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The station has been operating at power with several suspected leaking fuel elements. Offgas activity has been steadily increasing. A leak in the supply line to Offgas Pre-Treatment Monitor OG-RIS-612 requires isolation. The operator assigned to isolate this monitor is projected to receive 3.4 rem TEDE.

Which ONE of the following has the final review and approval of this Planned Special Exposure?

A. The Plant General Manager.

B. The Operations Manager.

C. The Shift Manager.

D. The Radiation Protection Manager.

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| ANSWER: | A |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | G2.3.2 & 2.9 - Knowledge of facility ALARA Program. |
| REFERENCE: | CAF |
| SOURCE: | Bank Question LO00257 – SRO Generic Knowledge and Abilities |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 1 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | Answer is per the facility exam bank. |
| 10CFR55 BASIS: | 10CFR55.43 (4) |
| COMMENTS: | Q 15 |

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At 0300 the plant is in Mode 2 and reactor pressure reaches 150 psig. At 1100, with the plant still in Mode 2 and reactor pressure at 750 psig, it is realized RCIC Operability Surveillance SR 3.5.3.4 has not been performed. Subsequently, the surveillance is completed at 1145 with reactor pressure reaching 800 psig by the end of the surveillance. It is then determined RCIC does not meet the Technical Specification minimum flow requirement and the CRS declares it inoperable effective at 1145. HPCS is confirmed Operable at 1155.

Based on this sequence of events and the Columbia Technical Specifications (attached), the plant:

A. must reperform SR 3.5.3.4 before RCIC can be declared Operable.

B. may remain in Mode 2, but RCIC must be restored to an Operable status within 14 days beginning at 0300.

C. may remain in Mode 2, but RCIC must be restored to an Operable status within 14 days beginning at 1145 AM.

D. may enter into Mode 1, but RCIC must be restored to an Operable status within 14 days beginning at 1145 AM.

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 217000G2.1.12 & 4.0 - RCIC/2.1.12 - Ability to apply technical specifications for a system. |
| REFERENCE: | Columbia Generating Station Technical Specification 3.5.3 and 3.0.4 |
| SOURCE: | New Question – SRO Tier 2, Group 1 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 3 |
| ATTACHMENTS: | Tech Spec 3.5.3 |
| JUSTIFICATION: | SR 3.5.3.4 was missed and per TS SR 3.0.3, this surveillance is not required to be performed for 24 months making answer A incorrect. RCIC is declared Inoperable at 1145 based on inadequate flow. This enters TS 3.5.3.A.1/2 which requires RCIC be restored within 14 days starting at 1145 making B incorrect. TS 3.0.4 prohibits entry into Mode 1 unless RCIC is restored making answer D incorrect. The |

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| | plant may remain in Mode 2 for 14 days from the time of the inoperability declaration making answer C correct. |
| 10CFR55 BASIS: | 10CFR55.43 (2) |
| COMMENTS: | Q 16 |

# CGS SRO Written Exam Answer Key
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The reactor is operating at 100 percent RTP when a reactor scram occurs due to an electrical power loss that results in reactor water level decreasing to -20 inches. Currently, plant conditions are as follows:

- Indicated reactor power is 50 on IRM Range 8 and stable
- The SRMs and IRMs fully inserted
- All APRM downscale lights are lit
- Reactor pressure is 975 psig and slowly decreasing
- The MSIVs are open
- Indicated reactor period is slightly positive
- Reactor level is 30 inches and slowly rising
- All containment parameters are normal
- Both Reactor Recirc Pumps are tripped

Based on this information, what is the primary procedure that should now be in use?

A. PPM 3.3.1, Reactor Scram

B. PPM 5.1.2, RPV Control-ATWS

C. PPM 5.1.1, RPV Control

D. ABN-RRC-LOSS, Loss of Reactor Recirculation Flow

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| ANSWER: | B |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 2.1.7 & 4.4 - Ability to evaluate plant performance and make operational judgements based on operating characteristics / reactor behavior / and instrument interpretation. |
| REFERENCE: | EOP PPM 5.1.1 |
| SOURCE: | New Question – SRO Generic Knowledge and Abilities |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 1 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | PPM 5.1.1 would be entered based on the initial low level condition in the reactor. Once 5.1.1 is entered, 5.1.2 would be entered based on |

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| | step Q-1 of the power control leg. PPM 5.1.2 would remain in effect until safe shutdown is assured under all conditions. Therefore the answer is B. |
| 10CFR55 BASIS | 10CFR55.43 (5) |
| COMMENTS: | Q 17 |

# CGS SRO Written Exam Answer Key
# September, 2004

The station experienced a loss of off-site power (LOOP) following a small break LOCA. Following the LOOP, the Division 1 and 3 Emergency Diesel Generators failed to start. The EOP related situation in the reactor building has continued to degrade since then and consideration is being given to powering SM-3 from SM-8.

According to part 50.54 (x and y) of the Code of Federal Regulations, this action is:

A. not permitted because it is a violation of the station operating license issued by the NRC.

B. permitted if approved by a licensed senior operator.

C. permitted if approved by the NRC.

D. permitted if approved by the TSC Manager.

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| ANSWER: | B |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 2.1.10 & 3.9 - Knowledge of conditions and limitations in the facility license. |
| REFERENCE: | 10CFR50.54(x and y) |
| SOURCE: | New Question – SRO Generic Knowledge and Abilities |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 2 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | According to the referenced regulation, intentional deviation from the station technical specifications requires the approval of an SRO. Therefore the correct answer is B. |
| 10CFR55 BASIS: | 10CFR55.43 (1) |
| COMMENTS: | Q 18 |

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Which of the following identifies when PPM 5.4.1, Radioactivity Release Control, must be entered?

A. When the exclusion area boundary radioactivity release rate exceeds the ALERT emergency values.

B. Prior to venting the primary containment from the drywell.

C. When transuranic materials are confirmed in the containment atmosphere.

D. When there is known fuel damage and a loss of primary containment.

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| ANSWER: | A |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 2.4.1 & 4.6 - Knowledge of EOP entry conditions and immediate action steps. |
| REFERENCE: | |
| SOURCE: | Bank Question LR00590 – SRO Generic Knowledge and Abilities |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 2 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 19 |

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# CGS SRO Written Exam Answer Key
# September, 2004

The plant was operating at 95% power when a transient occurred. The following conditions now exist:

- Drywell pressure is 1.59 psig and up slow.
- Suppression Pool level is –1.25 inches and down slow.
- All control rods are fully inserted except control rod 30-31 is at position 48.
- Area Radiation Monitor RIS-4 (East CRD Area) indicates 2.3E4 mr/hr.
- Reactor Building differential pressure is -.05 inches of water.
- Reactor Building Exhaust Plenum is 12 mr/hr

Based on these conditions, the procedure that should be entered is:

A. PPM 5.3.1 Secondary Containment Control.

B. PPM 5.1.2 RPV Control ATWS.

C. PPM 5.2.1 Primary Containment Control.

D. PPM 5.1.1 RPV Control.

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| ANSWER: | A |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295033G2.4.1 & 4.6 - High Secondary Containment Radiation Levels/Knowledge of EOP entry conditions and immediate action steps. |
| REFERENCE: | PPMs 5.1.1, 5.1.2, 5.2.1, and 5.3.1. |
| SOURCE: | Bank Question LO01149 – SRO Tier 1, Group 2. |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 2 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | A secondary radiation level of >10E4 requires entry into PPM 5.3.1. None of the other conditions listed require meet the EOP entry |

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| | requirements. Therefore, the correct answer is A. |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 20 |

Procedure PPM 6.3.2, Fuel Shuffling and/or Core OffLoading and Reloading, states that if the steam line plugs are to be installed during a refueling then a vent line for the main steam lines (MSLs) should be supplied.

What is the intent of this precaution?

A. Water coming from the vent line provides an indication of a leaking MSL plug.

B. In the event of a leaking plug, the vent line will prevent filling the MSL with water and damaging MSL hangers and/or snubbers.

C. The vent line will prevent pressurizing an MSL and the potential expulsion of the plug.

D. The vent line provides some protection to maintenance personnel performing work downstream of the plug (MSIV or SRV).

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295008AA1.03 & 3.1 - High Reactor Water Level/Main Steam System |
| REFERENCE: | Procedure PPM 6.3.2, Fuel Shuffling and/or Core Offloading and Reloading |
| SOURCE: | New Question – SRO Tier 1, Group 2 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 1 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | The intent of the vent line is to prevent MSL pressurization and expulsion of the plug. Therefore the correct answer is C. |
| 10CFR55 BASIS: | 10CFR55.45 (7) |
| COMMENTS: | Q 21 |

# CGS SRO Written Exam Answer Key
# September, 2004

# CGS SRO Written Exam Answer Key
# September, 2004

While performing a maintenance activity, a series of procedural errors cause a severe reactor pressure transient. Reactor pressure peaks at 1343 psig before the reactor scram and bypass valves reestablish pressure in the normal operating band. The plant is now in Mode 3, all control rods are fully inserted, and reactor pressure is being maintained at 800 psig.

Which of the following is required prior to restarting the plant?

A. The post scram review is approved by the Plant General Manager.

B. The engineering transient analysis is approved by the Engineering Manager.

C. NRC approval for restart has been given.

D. The cause of the event is known and corrected.

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295007G2.2.22 & 4.1 - High Reactor Pressure/Knowledge of limiting conditions of operations and safety limits. |
| REFERENCE: | Techical Specification 2.0, Safety Limits, and 10CFR50.36/72. |
| SOURCE: | New Question – SRO Tier 1, Group 2 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 2 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | The pressure transient violated the technical specification primary coolant system pressure safety limit of 1325 psig. Therefore NRC approval (10CFR50.36) prior to restart. |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 22 |

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# CGS SRO Written Exam Answer Key
# September, 2004

PPM 5.3.1, Secondary Containment Control, was initially entered based on high RB area temperatures due to a primary system discharging into Secondary Containment. The reactor has been scrammed and all rods are fully inserted.

According to PPM 5.3.1, Secondary Containment Control (attached), which of the following indicates PPM 5.1.3, Emergency Depressurization, should be entered?

A. RCIC Pump Room - Temp. 210 ˚F, Radiation 5,000 mR/hr, area water level 5 ft
RHR A Pump Room - Temp. 220 ˚F, Radiation 8,000 mR/hr, area water level 38 ft

B. RCIC Pump Room - Temp. 200 ˚F, Radiation 5,000 mR/hr, area water level 5 ft
RHR A Pump Room - Temp. 200 ˚F, Radiation 9,000 mR/hr, area water level 25 ft

C. RCIC Pump Room - Temp. 220 ˚F, Radiation 5,000 mR/hr, area water level 4 ft
RHR A pump Room - Temp. 200 ˚F, Radiation 8,000 mR/hr, area water level 38 ft

D. RCIC Pump Room - Temp 180 ˚F, Radiation 9,000 mR/hr, area water level 5 ft
RHR A Pump Room - Temp 190 ˚F, Radiation 8,000 mR/hr, area water level 30 ft

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295036EK3.01 & 2.8 - Secondary Containment High Sump/Area Water Level/Emergency Depressurization |
| REFERENCE: | PPM 5.3.1, Secondary Containment Control, Revision 13, Step SC-15 |
| SOURCE: | Bank Question – SRO Tier 1, Group 2 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 2 |
| ATTACHMENTS: | PPM 5.3.1, Secondary Containment Control. |
| JUSTIFICATION: | According to the referenced procedure, if a safe operating value is |

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| | exceeded in two areas, an emergency depressurization is required. The only set of conditions listed that meets this criteria is C (RCIC Pump Room Temperature and RHR area water level). |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 23 |

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A reactor transient occurs during which the following conditions exist:

- RPV pressure 1065 psig
- RPV water level +30 inches
- All rods are fully inserted

PPM 5.1.1 "RPV Control" is entered and is being executed. Five minutes later the following conditions are reported:

- RPV water level +10 inches
- RPV pressure 850 psig
- Reactor feedwater pumps have tripped

Which of the following is correct?

A. Re-enter PPM 5.1.1, "RPV Control" at the beginning.

B. Continue in PPM 5.1.1, "RPV Control."

C. Prioritize operator response by performing those steps of PPM 5.1.1 "RPV Control" that would provide control of RPV level.

D. Exit PPM 5.1.1 "RPV Control" since the EOP entry condition is no longer met.

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| ANSWER: | A |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 259002/2.4.1 & 4.6, Reactor Water Level Control System/Knowledge of EOP Entry Conditions and immediate action steps |
| REFERENCE: | Facility Exam Bank |
| SOURCE: | Bank Question LR00976 – SRO Tier 2, Group 1 |
| LEARNING OBJECTIVE: | CAF |

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| RATING: | 1 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | Answer is per the facility exam bank. |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 24 |

# CGS SRO Written Examination Key
# September 2004

The plant had been operating at rated thermal power when a loss of both reactor recirculation pumps occurred. The current plant status is:

- The SCRAM is reset
- Both reactor recirc pumps remain tripped
- RWCU is isolated
- Core cooling is by natural circulation
- Feedwater and the bypass valves are available

Until forced flow can be restored, RPV thermal stratification will by minimized by implementing procedures that will:

A. maximize CRDH flow.

B. minmize RPV cooldown.

C. restore and maximize RWCU flow.

D. minimize feedwater flow.

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| ANSWER: | C |
| QUESTION TYPE: | SRO |
| KA# & KA VALUE: | 295001AK1.03 & 4.1 - Partial or Complete Loss of Forced Core Flow Circulation/Thermal Limits |
| REFERENCE: | ABN-RRC-Loss, Loss of Reactor Recirculation Flow, Revision 1, Section 5.0. |
| SOURCE: | New Question – SRO Tier 1, Group 1 |
| LEARNING OBJECTIVE: | CAF |
| RATING: | 3 |
| ATTACHMENTS: | None. |
| JUSTIFICATION: | One of the specific concerns in this scenario is the cooldown and temperature stratification that will occur in the lower head region with RWCU isolated. Maximizing CRDH flow will add cold water to the lower head region accelerating the temperature stratification. Minimizing cooldown will maintain a higher temperature in most areas of the RPV except the lower head region thus allowing a higher |

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| | differential temperature to develop. Minimum feedwater flow will have the same effect as minimizing cooldown and will promote differential temperatures. Maximizing RWCU flow will provide for a flowpath through the lower head region thereby reducing the temperature differential. Therefore the correct answer is C. |
| 10CFR55 BASIS: | 10CFR55.43 (5) |
| COMMENTS: | Q 25 |

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