UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-269/94-300, 50-270/94-300 and 50-287/94-300

Licensee: Duke Power Company
422 South Church Street
Charlotte, NC 28242

Docket Nos.: 50-269, 50-270, and 50-287 License Nos.: DPR-38, DPR-47, and DPR-55

Facility Name: Oconee 1, 2 and 3

Examination Conducted: March 7-10, 1994

Inspector: Lawrence L. Lawyer for Michael E. Ernstes per e-mail 4/6/94 Date Signed

Accompanying Personnel: R. Baldwin, NRC Region II
K. Faris, PNL

Approved by: Lawrence L. Lawyer 4/6/94 Date Signed
Lawrence L. Lawyer, Chief
Operator Licensing Section
Operations Branch
Division of Reactor Safety

SUMMARY

Scope:

NRC examiners conducted regular, announced operator licensing initial examinations and inspection activities during the period March 7-10, 1994. Examiners administered examinations under the guidelines of the Examiner Standards (ES), NUREG-1021, Revision 7. Three Senior Reactor Operator (SRO) and six Reactor Operator (RO) candidates received written and operating examinations.

Results:

Candidate Pass/Fail:

	SRO	RO	Total	Percent
Pass	3	5	8	84 %
Fail	0	1	1	16 %

Examiners identified an instance of lack of regard for station radiological controls as a weakness (paragraph 2.a(2)).

Examiners identified the use of the "STAR" process as a strength (paragraph 2.a(3)).

Examiners identified the inability of the licensee to effectively train the control board operators to identify the failure of Integrated Control System to runback as an inspector follow-up item, (paragraph 2.a(3)) IFI 50-269/94-300-01.

Examiners identified the training on the definition of Excessive Heat Transfer as a weakness (paragraph 2.a(3)).

Examiners identified the failure to verify that an Operations Management Procedure (OMP) revision did not conflict with other existing OMPs, as a violation, (paragraph 2.d) VIO 50-269/94-300-02. The violation is of concern because changes to the administrative requirements for approval to bypass safety systems without a procedure should be strictly controlled.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*D. Dalton, Regulatory Compliance
*J. Hampton, Vice President, Oconee Nuclear Station
*B. Jones, Training Manager
*P. Mabry, Operations Training Coordinator
*B. Peele, Manager, Oconee Nuclear Station
*J. Price, Shift Operations Manager
*G. Rothenberger, Superintendent of Operations
*P. Stovall, Director of Operator Training
*G. Washburn, Operator Training

Other licensee employees contacted included instructors, engineers, technicians, operators, and office personnel.

NRC Personnel

*P. Harmon, Senior Resident Inspector

*Attended exit interview

2. Discussion

a. Operator Performance

The NRC evaluated the license candidates in accordance with Revision 7 of NUREG-1021. One candidate failed the written examination. Paragraph 2.a(1) lists generic knowledge deficiencies identified by the written examination. These deficiencies were in the Plant Wide Generic area for the SROs and in Plant Systems for the ROs. The NRC examiners evaluated each candidate on a plant walkthrough and two simulator scenarios. All operators passed these parts of the examination. The examiners found a weaknesses in Emergency Plan knowledge and a potential for failure to control exposure during the walkthrough examinations. During the simulator examinations, the examiners found weaknesses in event diagnosis and communications. The team identified a strength in the candidate's use of the STAR process.

(1) Written Examination

Analysis of the written examination results showed generic weaknesses in various knowledge areas. The SRO candidates performed poorly on the Plant Wide Generic questions. They missed an average of 18 percent of these questions. The RO candidates were weak on the Plant Systems questions. They missed an average of 18 percent of the questions in this area.

Failure rates on the following questions identified specific weak areas. Examination question numbers for the area follow in parenthesis.

- Setpoint for the Source Range High Volts power supply cutoff (RO 32, SRO 32)
- Adverse affect of a high Core Flood Tank level (RO 48, SRO 44)
- Adverse effect of S/G overfill during ATWS (RO 73)
- Method of pressure compensation for pressurizer level indications (RO 56, SRO 50)
- Adverse effect of not performing the Abnormal Procedure for dam failure quickly (RO 62, SRO 55)
- Basis for keeping RCPs running when in Section 507, Inadequate Core Cooling (RO 81, SRO 74)
- Major contributor to offsite dose following core damage (RO 85, SRO 78)
- Execution of a "GO TO" statement in the EOP (SRO 16)
- Required actions for an RIA-6 failure (SRO 80)
- Pressurizer level response to a leak in the steam space (SRO 87)

(2) Walkthrough examinations

One candidate demonstrated a significant lack of awareness in complying with the station's procedures for controlling radiation exposure. A Job Performance Measure (JPM) required entry into a Radiation Control Area (RCA) to start the Spent Fuel Pool (SFP) filtered exhaust system. Entry into the RCA required entering on a Standing Radiation Work Permit (SRWP) and obtaining dosimetry. An SRO upgrade candidate attempted to enter the RCA through a posted door without entering on an SRWP. The NRC examiner stopped him before he went through the door. This would have been a violation of the facility's radiation control procedures. Entering an RCA without entering on an SRWP would cause the individual to be unaware of the hazards and limitations of being in the area and increase the probability of an overexposure. Entering without dosimetry would prevent him from complying with ALARA principles and tracking his dose. Examiners identified the lack of regard for station radiological controls as a weakness.

Most candidates demonstrated weaknesses in implementation of the Emergency Plan. One SRO candidate could not determine the correct Emergency Action Level for a simulated event. Several operators did not fully understand the methods used for accountability during an event, such as time limits or offsite assembly locations.

(3) Simulator Examinations

The candidates demonstrated good use of the "STAR" process. The licensee implemented this process to reduce personnel errors by directing the operators to Stop, Think, Act, and Review when performing tasks. The methodical evaluation of their actions aided the operators in accurately performing tasks during the operating examinations. The examiners identified the use of the "STAR" process as a strength.

The control board operators did not successfully identify a failure of the Integrated Control System (ICS) to runback. Two of the three crews did not check reactor power or control rod position to see if actual reactor plant power reduced. This resulted in an unnecessary high pressure reactor trip. Follow-up questioning showed that the candidates saw that load demand to ICS had reduced and concluded that the plant had run back. Examination Report 50-269/93-301 also identified the inability to diagnose a failed runback as an operator performance weakness. The examiners identified the repetitive inability of the licensee to effectively train the control board operators to identify the failure of ICS to runback as IFI 50-269/94-300-01.

The examiners identified a training weakness on the definition of Excessive Heat Transfer. At step 5.15 of the Emergency Operating Procedure, EP/1/A/1800/01, one candidate incorrectly diagnosed excessive heat transfer. None of the five indications of excessive heat transfer listed in this step existed. However, on follow-up questioning the candidate stated that he had been trained that excessive heat transfer existed if Tave was less than 532°F. The training department established this temperature as a benchmark. Temperatures below this value indicated excessive heat transfer. The transition was inappropriate since the cooldown was from a known cause and under control. The performance of the Excessive Heat Transfer section of the EOP, delayed the isolation of a S/G with a tube rupture by an additional seven minutes.

Communications by the crews were informal and at times unclear. This resulted in the need for crew members to repeat some information. Some candidates failed to identify a specific component or parameter when providing information. For example, during a S/G tube leak, when reporting that "A" S/G level was higher than the "B" S/G level, a RO said, "'A' is bouncing around a little higher." When reporting Turbine Bypass Valves had opened to control header pressure, a RO said, "They just came open." He did not report what had opened or the reason for opening. Incomplete information created a potential for the crew to take inappropriate actions based on incorrect information.

b. Plant conditions

During the walkthrough examinations, the NRC examiners noted poor lighting in many of the Auxiliary Building corridors. Several lights were out, creating a safety hazard. The lighting at the entrance to the Unit 1 & 2 SFP Cooler room was so dim that the radiation survey map on the door could not be read without a flashlight.

c. Simulator

Problems with the simulator impacted the development and administration of the examinations. Enclosure two contains simulator deficiencies observed by the examiners. The number and significance of these deficiencies makes realism in simulator training difficult and creates undue potential for negative training. The following paragraphs discuss the impacts of three of these problems.

Examiners need the alarm typer to identify the opening of Main Steam Safety Relief Valves. This can not be evaluated by monitoring the main steam header pressure gauge on the main control board due to the small amount of blowdown. The typer was not operable during the first four scenarios.

The simulator did not have a malfunction to simulate a leak in the letdown system anywhere other than in the Letdown Storage Tank (LDST). Simulation of a leak elsewhere in the letdown system required entering the LDST leak and then overriding the LDST pressure indication. The inability to vary the source of the leak created negative training. Training conditioned the candidates to respond to a lowering LDST level as a rupture in the LDST. When candidates observed the malfunctions to simulate a letdown leak, two of the three crews inappropriately took actions for a LDST leak.

The simulator did not have malfunctions to override the automatic actions associated with an ES actuation. In order to evaluate candidates' ability to verify automatic actions, the simulator operator failed the switch indication. This required the simulator operator to immediately remove the indication malfunction when the operator tried to manually actuate ES equipment. This required the use of hand signals by the examiners for switches in remote areas of the simulator.

d. Procedures

The examiners found two procedural inadequacies during the examination. One with the Operations Management Procedures (OMPs) and the other with Procedure 01/A/1103/06, "Reactor Coolant Pump (RCP) Operation."

The facility had conflicting procedures on the station policy for non-procedural blocking of a safety system actuation. The licensee failed to verify that a change to the OMPs resulted in a conflict between two procedures on the approval needed for non-procedural bypassing of a Safety System. On June 1, 1993, the licensee changed OMP 2-1, "Duties and Responsibilities of On Shift Operations Personnel," so that step 2.3 of Enclosure 4.11, "Guidelines for Bypassing of Safety Systems," allowed a Safety System to be bypassed without specific procedure guidance under direction of the Control Room SRO or Unit Supervisor.

This procedure change responded to an event at another B&W designed facility. The licensee did not change step 4.10 of OMP 1-2, "Rules of Practice," which conflicted with the new change. OMP 1-2 required approval from two licensed personnel, one of whom is a supervisor who holds an SRO license. The approval to bypass safety systems without a procedure is an important policy to the health and safety of the public and was subject to multiple interpretations. Examiners identified the failure to maintain control of the OMPs as VIO 50-269/94-300-02.

Procedure 01/A/1103/06, Enclosure 4.3, "Re-start of RCPs with High Temperatures," contained an error which could have resulted in establishing Component Cooling (CC) to the wrong RCP. The alpha-numeric designation in the procedure for the RCP seal cooler outlet valves did not match the alpha numeric designation on the control board labels. The procedure directed closing valve 1CC-3 for the "1A1 RCP SEAL CLR OUTLET." The control board showed that valve 1CC-3 was for the "1B1 RCP SEAL CLR OUTLET." During JPM validation, a facility instructor operated the wrong valve and re-established CC flow to the wrong RCP.

4. Exit Interview

At the conclusion of the site visit, the examiners met with representatives of the plant staff listed in paragraph 1 to discuss the results of the examinations The licensee did not identify as proprietary any material provided to, or reviewed by the examiners. The examiners further discussed in detail the inspection findings listed below. Dissenting comments were not received from the licensee.

Item Number	Status	Description and Reference
50-269/94-300-01	Open	IFI - Inability of the licensee to effectively train the control board operators to identify the failure of ICS to runback (paragraph 2.a(3)).
50-269/94-300-02	Open	VIO - Failure to maintain control of the Operations Management Procedures (paragraph 2.d).

ENCLOSURE 3

SIMULATOR FACILITY REPORT

Facility Licensee: Duke Power Company

Facility Docket Nos.: 50-269, 50-270, and 50-287

Operating Tests Administered On: March 7-10, 1994

This form is to be used only to report observations. These observations do not constitute, in and of themselves, audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information that may be used in future evaluations. No licensee action is required solely in response to these observations.

While conducting the simulator portion of the operating tests, the following items were observed (if none, so state):

ITEM	DESCRIPTION
1HP-228	Valve did not respond when operated.
Tave meter	Did not respond during a power change. This created an unplanned evolution during the examination.
Computer points	Cannot override computer points. Values had to be artificially supplied by examiner for turbine vibration, RCP temperatures.
HPI flow indicator	Indicated flow when there was none at full power.
MFP oil booster	The reactor tripped when an operator locked out the B MFP auxiliary oil pump.
Alarm typer	This was disabled for the majority of the scenarios. The examiners need this data for tracking the sequence of plant events.
ES actions	Could not override the automatic ES actions. This precluded testing the operators ability to verify automatic actions.
Fire alarm panel	The simulator does not have a fire alarm panel as in the plant. To provide information on the location of a fire, the simulator operator comes out of the control booth and tells the crew what areas on the panel are in alarm.
Letdown system leak	The only location available to simulate a leak in the letdown system was the Letdown Storage Tank.

Duke Power Company
Oconee Nuclear Station
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ENCLOSURE 4

(803) 882-5363

DUKE POWER

March 9, 1994

U.S. Nuclear Regulatory Commission
ATTN: Thomas A. Peebles, Chief
Operations Branch
Region II, Suite #3100
101 Marietta Street, N.W.
Atlanta, GA. 30323

Subject: **Oconee Nuclear Station Facility Comments on RO and SRO License Written Examinations Given March 7, 1994**

Dear Mr. Peebles:

I would like to take this time to thank you for the complimentary comments that Mr. Mike Ernstes gave Mr. Paul Stovall on our pre-examination review conducted in Atlanta during the week of February 22. Improvements in our pre-exam review process were a result of lessons learned from last year's exam process.

The pre-examination review process resulted in only eight post-exam comments and very positive student feedback following the written examination. We hope this pre-exam review will be allowed to continue in future Oconee Initial Licensing Examinations.

Please consider the following eight facility comments, justifications and supporting highlighted documentation.

1. a. #20 RO exam and #20 SRO exam. Answer Key : C
Reference: Oconee Lesson Plan OP-OC-IC-ES, Rev 7, p. 17; Obj. R13 1992 (HNUM 33919), OMP 2.1, 4.11

b. Facility Comment: In addition to choice "C" being correct, choice "D" is also correct.

c. Facility Justification: Refer to attached OMP 1-2, paragraph 4.10.2, which states, "Non-procedural blocking of automatic safety actuations must be approved prior to taking the action by two licensed personnel, one of whom is a supervisor who holds an SRO license." As this is the more conservative of the two procedures, this is the proper way to block safety systems in non-procedural situations, rather than using the guidance given in OMP 2-1, Enclosure 4.11, paragraph 2.3 (attached), which states: "Safety Systems may be bypassed without specific procedure guidance under direction of the CR SRO or Unit Supervisor."

This question received a pre-exam comment in which the wording of answer "D" needed to be changed in order to make the distractor totally wrong. The wording was changed, but not as requested.

2. a. #24 RO Exam and #26 SRO. Answer Key: B
Reference: Oconee Exam Bank IC-59
OP-OC-IC-CRI,pg. 19

b. Facility Comment: Answer "A" is the only correct answer.

c. Facility Justification: The answer key is incorrect. Refer to OP-OC-IC-CRI, pg. 19, paragraph D.1 and pg.20, paragraph E.1 (attached).

3. a. #32 RO Exam and #32 SRO Exam. Answer Key: D
Reference: OC-OP-IC-NI, p. 19, para. 2.2.A.13.d, EO B.11

b. Facility Comment: No answer is correct. The question should be deleted.

c. Facility Justification: The power supply to NI-1 stays energized at all power levels and is not disabled by the Source Range High-voltage cutoff. See Reference OC-OP-IC-NI, pg. 30, paragraph 7 (attached). Student feedback initiated this comment.

4. a. #35 RO Exam. Answer Key: B
Reference: OP-OC-PNS-RBC, p. 15, para. 2.?.B, EO B.10 Jan 93 Exam

b. Facility Comment: Answers "A" and "B" are both correct.

c. Facility Justification: Refer to OP-OC-PNS-RBC, page 15, paragraph B.2 (attached). The RBCU's shift to low speed because of the high density in the Reactor Building. Since the Reactor Building atmosphere will be saturated following an accident that activates ES channels 5 and 6, the temperature and pressure will both increase. Either one would indicate to the operator that a high density exists in the Reactor Building atmosphere. Student feedback generated this comment.

5. a. #45 RO Exam and #41 SRO Exam. Answer Key: D
Reference: OP-OC-RAD-RIA, pp. 29-32, para. 2.5.B.7, EO B.6

b. Facility Comment: Answer "A" and "D" are both correct.

c. Facility Justification: Answer "A" is also correct because a sample pump failure will result in a loss of sample flow condition. This will result in an alarm per OP-OC-RAD-RIA, page 13, paragraph 7.b. and page 14, paragraph 6.d (Attached) Student feedback initiated this comment.

6. a. #62 RO Exam and #55 SRO Exam.
Reference: OP-OC-STG-CCW, pp. 33-34, para. 2.4.K.4, EO B.24 NRC INSPECTION REPORT NO. 50-269/93-25.

b. Facility Comment: No answer is correct. The question should be deleted.

c. Facility Justification: Per NOTICE OF VIOLATION AND NOTICE OF DEVIATION (NRC INSPECTION REPORT NOS. 50-269/93-25, 50-270/93-25, AND 50-287/93-25), dated February 11, 1994, page 20, paragraph 3 (attached), CCW-8 would be submerged in the event of a Keowee Dam failure. The stem of the question does not state whether the Keowee Dam or one of the other Dams has failed. There are several scenarios which would require entry into this AP in which the inundation of CCW-8 is not feasible, including a loss of the Little River Dam and a loss of one of the various dikes on Lake Keowee. In both of these scenarios, there would be no chance of flooding CCW-8.

In AP/2/A/1700/13, Case B, Section 4.0, Immediate Manual Actions, the opening CAUTION statement (attached) reads: "The amount of inventory loss from the Intake Canal is directly related to the amount of time that elapses during the completion of the section." The AP is referenced in the stem of the question. The above CAUTION is the only reference to time constraints in this document.

This question was challenged during the pre-examination review. Student feedback initiated further comment.

7. a. #84 RO Exam and #77 SRO Exam. Answer Key: A
Reference: OP-OC-TA-AM7 Obj. 10, 15 Pg. 16.

b. Facility Comment: Answers "A","B" and "D" are all correct. The question should be deleted.

c. Facility Justification: Refer to OP-OC-TA-AM7, page 16, paragraph C.1 and page 17, paragraph 3.a (attached). The answers "A","B" and "D" all indicate potential leakage pathways to the Auxiliary Building which would be isolated by ES actuation. Valve seat leakage or operator action to unisolate these pathways is neither more nor less probable than the others for any one of these pathways.

This question is from a non-QA Oconee exam bank, and it received pre-exam review comments.

8. a. #87 RO Exam and #80 SRO Exam Answer Key: B
Reference: Oconee Lesson Plan OP-OC-FH-FHS Section 2.4, LO 12.

b. Facility Comment: Answer "D" is also correct.

c. Facility Justification: Refer to Tech. Spec. 3.8.1. and OP/1/A/1502/07, Limit and Precautions 2.3 (Attached). The Tech. Spec. reference requires "portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation...", with no reference to local alarm capability. Therefore, there are conflicting references. These references are so similar in content that without a reference, as requested in pre-exam comments, both choices "B" and "D" are acceptable.

We appreciate your prompt attention to these matters.

Sincerely,

J W Hampton

J.W. Hampton, Vice President
Oconee Site

xc: M. Ernestes, Chief Examiner
B.L. Peele, Station Manager/ONS
J.B. Price, Shift Operations Manager/ONS
P.M. Stovall, Director, Operator Training/ONS
G.E. Rothenberger Superintendent of Operations/ONS
B.K. Jones, Training Manager/ONS

JWH:GCW:et

Attachments

Comment #1

4.10 It is the policy of nuclear stations to not block automatic safety actuations from performing their intended functions.

4.10.1 This policy does permit safety signals to be "blocked" under the direction of an approved station procedure (EOP, AP, OP, PT, etc.) where the consequences have been evaluated.

4.10.2 Non-procedural blocking of automatic safety actuations must be approved prior to taking the action by two licensed personnel, one of whom is a supervisor who holds an SRO license.

4.11 When operating Manual valves:

4.11.1 Unless prohibited by system conditions, manual valves should be moved to positively determine position.

4.11.2 If system conditions prohibit or red tags are in place, manual valves should not be moved to verify position. In this situation, it is acceptable to use the best available means to determine position other than valve movement (e.g., observation of local position indication, remote position indication, stem position, or process parameters).

4.11.3 If a manual valve cannot be moved to determine position because of mechanical problems, consult a supervisor and issue a work request if appropriate.

4.12 When using cheaters:

4.12.1 If a manual valve cannot be operated without the use of a cheater, the following precautions must be observed.

1. Lubricate the valve stem. Reduce system differential pressure if possible.

2. Select a cheater of the appropriate size for the valve to be operated. Never use additional devices to extend a cheater (such as pipes or bars).

3. If items 1 and 2 are not successful in operating the valve, contact the Unit Supervisor for guidance.

4. If the valve is still unable to be operated, Maintenance should be contacted. Any extreme measures to operate a difficult valve should be performed by Mechanical Maintenance.

4.12.2 Cheater devices shall not be used on any motor-operated valves, kerotest valves, reach rods devices, or diaphragm valves.

OMP 2-1
ENCLOSURE 4.11
GUIDELINES FOR BYPASSING OF SAFETY SYSTEMS

1.0 Safety systems (RPS, AMSAC, DSS, ES) must be allowed to perform their automatic function when required for transient mitigation.

2.0 Safety systems must NOT be bypassed prior to automatic actuation except as follows:

2.1 Safety systems may be bypassed when directed by operating procedures for normal plant cooldowns.

2.2 Safety systems may be bypassed when directed by emergency/abnormal operating procedures for specific transients.

2.3 Safety systems <u>may</u> be bypassed <u>without</u> specific procedure guidance under direction of the CR SRO or Unit Supervisor if both the following are true:

2.3.1 The Safety System is not required to perform its intended safety function (i.e., adequate SCM exists, SG pressures within acceptable limits, etc.).

2.3.2 Actuation of the Safety System could <u>increase</u> the severity of the transient, damage equipment, or cause unnecessary Operator Burden.

3.0 If a safety system has been bypassed, the operator assumes the responsibility to reactuate the system if necessary for transient mitigation.

4.0 Equipment automatically actuated by a safety system must NOT be repositioned except as follows:

4.1 Equipment may be overridden and repositioned when directed by emergency/abnormal operating procedures for specific transients.

4.2 Equipment may be overridden and repositioned with concurrence from the CR SRO or Unit Supervisor, if both of the following are true:

4.2.1 The Safety System is not required to perform its intended safety function (i.e., adequate SCM exists, SG pressures within acceptable limits, etc.).

4.2.2 Continued operation of the Safety System could <u>increase</u> the severity of the transient, damage equipment, or cause unnecessary Operator Burden.

Documentation for Comment on RO Question #24

Comment

D. Auxiliary and Regulating Power Supplies

1. The Auxiliary Power Supply can be used to operate up to 12 rods in a Group. There is only one Auxiliary power supply which is used for any of the following purposes:

   a. To pull the Safeties to their full out position prior to startup.

   b. To operate the Regulating rods as needed in case of a loss of one of the normal group power supplies.

   c. To recover an individual dropped safety or regulating rod.

2. Each of the regulating groups 5 thru 7 as well as group 8 have their own Group power supply.

3. The Auxiliary and Group power supplies can be used to move the rods in or out at jog (3" per min) or run (30" per min) speed.

4. Basic Concept of Operation

   A 120 volt DC power supply with program actuated Silicon Controlled Rectifier Diodes provide the proper drive pulses for energizing the 12 CRD stator poles in the proper sequence to rotate the stator magnetic field that holds the CRD segment arms apart to engage the control rod lead screw. The rotating magnetic field rotates the CRD segment arms. The rotating segment arms cause the control rod lead screw and therefore the control rod to move up or down by screw action between control rod lead screw and segment arm roller nuts. (See PNS-CRD lesson)

   Drawing OP-OC-CRI-2 shows how the 12 pole pieces are electrically wound. Only one winding (AA) is shown. Notice that the A/AA, B/BB and C/CC phases are on one pole piece. The direction of direct current flow through the windings will cause the pole piece to be either a north or south magnetic pole. Opposite pole pieces, which are energized simultaneously (All AA etc.. are energized at the same time), due to their windings current flow direction will be of opposite magnetic polarity. Obviously an A and AA, B and BB or C and CC will never be energized simultaneously or their magnetic fields would cancel each other out.

6. Programmer

The Programmer consists of a 15 volt DC power supply to a 5 volt DC regulator, solid state (microprocessor based) programmer to accept operator/ICS input commands (in, out, jog and run) and output drivers to drive (gate) the SCRs. (Refer to OP-OC-CRI-3)

E. DC Hold Power Supply (Refer to OP-OC-CRI-1 and 3)

1. After the safety rod groups are moved to their out limit they will remain in this position until reactor shutdown. It would be extremely wasteful to employ individual sets of power supplies with the associated SCRs, programmers, etc. for holding these groups at their out limits. Instead an auxiliary power supply is used to move the safety groups one at a time to their "full-out" position. After the safety group reaches the out limit position the group is transferred from the auxiliary supply to the DC hold supply.

   The DC Hold Supply is similar to the regulating supply except for two things. It utilizes diodes in place of SCRs, and energizes two coils of the stator, "A" and "CC", rather than all six coils. The supply has no need for a programmer since its function is to hold the latch arms together rather than turn the rotor.

2. DC Hold Power Supplies.

   Two; one is energizing the A phase of the rods in each of the four safety groups and the other is energizing the CC phase. Only one phase energized is sufficient to hold a rod out. Therefore, if one of the DC Hold power supplies is lost the Safeties will remain in the full out position.

Documentation for Comment on Question #32 (RO and SRO)

Comment #3

c. The incoming thermal neutron interacts with the U235 and causes it to fission. The highly ionized fission fragments move through the chamber and ionize the gas. The ions migrate to the electrodes and release a charge, which cause a current pulse. Because of the relatively low voltage, 870 VDC, there is no gas amplification and each fission records a pulse.

d. Because the energy of the fission fragments is very high a pulse height discriminator can be used to distinguish between a pulse from Gamma and Alpha, both of which produce pulses of lower magnitude.

e. In an attempt to make the Source Range more sensitive, two chambers are used and the outputs added together. Only one chamber is used for the wide range output.

f. When operating in the Source Range, the system counts each individual pulse.

g. When power is increased the pulses become too numerous to count individually, the output becomes almost constant. At higher levels a process called Campbelling mathematically derives the power level from the current input. The Campbelling process starts at 4E-3% power. An auctioneering circuit changes the output at this point to ensure a continuous readout.

h. Using these two different counting modes allows one detector to cover the entire range from 1E-8% to 200% power.

i. Another scaler output will feed random pulses to the Refueling and Zero Power Physics areas.

7. The Gamma-Metrics Source Range instruments will not be turned off at power because the signal is coming from the same detector that is sending the Wide Range signal.

8. The signal will be adjusted so that 100% power on the Wide Range will correspond to 100% power on the power range NIs. The new Gamma-Metric system is not as sensitive as the old system and may vary considerably.

9. As it stands today, the SUR inhibit for the source range of 2 DPM will remain the same and the old intermediate range inhibit of 3 DPM will cover the wide range. There are no plans to remove the 2 DPM inhibit but the 3 DPM wide range inhibit may be removed.

Documentation for Comment on RO #35

Comment #4

B. RBCUs

1. Receive signal from ES 5 & 6 ($\leq$ 4# bldg. press)

2. All RBCUs go to low speed

   a. RBCUs are run in low speed due to denser RB atmosphere which could cause fan motors to fail in high speed since higher current would result from moving denser air.

3. Fusible dropout plates would drop if air temperature in ductwork reaches 150°F. This would probably only occur if the discharge ductwork were damaged and impeded the air flow.

C. Associated LPSW valves (OC-PNS-RBC-07)

1. RBCU outlet valves go full open.

2. LPSW-565 closes and LPSW-566 opens restoring full LPSW flow to "B" RBCU.

3. This results in full LPSW flow to all 3 RBCUs.

D. Returning RBCUs to normal following ES actuation. (OC-PNS-RBC-03)

1. Upon ES signal, RBCUs are locked in low speed. Control room switches have no effect.

2. After the digital and analog ES signals have been reset or after "MANUAL" is selected on RZ module, a seal in logic must also be reset prior to regaining control of the RBCUs.

3. Above each damper position indicator on AB3 is a "PUSH TO RETURN TO NORMAL AFTER ES RESET" button.

4. After ES signals are reset or the RZ module is in "MANUAL", the reset button for the associated RBCU is pushed and the RBCU will go to whatever position called for by its switch.

5. LPSW-565, 566, and cooler outlet valves must be repositioned as desired. The reset push buttons have no effect on these valves.

Documentation for Comment on Question #45(RO) AND #41(SRO)

Comment #5

6. The new monitors have an auxiliary sample connection for sampling of various areas in the Auxiliary Building. This enables the operator to monitor potential problem areas by pulling samples through a temporary sample line connected to the auxiliary sample connection.

7. Both Alert and High radiation alarms are annunciated in the Control Room.

   a. If conditions arise which cause the monitor to become inoperable, a fault alarm is also given in the Control Room.

   b. These conditions include loss of sample flow, loss of detector signal, checksource response failure, and loss of power.

   c. If low sample flow conditions exist for more than ~~a short time,~~ one minute the RM-80 will ~~trip~~ stop the sample pump to prevent damaging it.

Pg 5
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11-16-93

C. (1)(2)(3)RIA-35 - ~~Victoreen~~ Sorrento

1. Monitor Low Pressure Service Water effluent from auxiliary building.

2. Monitor gross gamma activity.

3. Supplement RIA-31.

4. Located in basement of turbine building behind air compressors.

5. Detector sodium iodide.

6. NSM-12737, 1RIA-31 and 1RIA-35 modification was performed to correct low flow conditions of LPSW sample points affecting 1RIA-35. These low flow sample points would render the tech spec RIA (1RIA-35) inoperable.

   a. Refer to OC-SSS-IA-5 to view a schematic of the new sample piping arrangement relating to NSM-12737.

   b. This modification rerouted sample tubing for 1RIA-31 and 1RIA-35. In the past the Unit 1 sample points not being selected by 1RIA-31 would be routed through 1RIA- 35. If a low flow condition existed on one of the sample points, 1RIA-35 would not receive a representative sample from that component.

#45

c. 1RIA-35 now has its own sample pump. The RIA will continuously sample the LPSW return lines. 1LPSW-875 is throttled to balance flows between the two sample lines as indicated on the two individual flow guages.

d. Individual flows are monitored in each sample line by flow switches. If a low flow condition (<1 gpm) is sensed in either line, the Process Monitor Fault statalarm will sound in the Control Room. Also, another flow switch on the combined sample line will actuate the same alarm if it senses flow less than 2 gpm.

e. An ENABLE\DISABLE switch has been provided for flow switch 1LPSFS 0015 mounted on the turbine building wall in the basement near column M-24. This switch is used anytime the 1B LPI Cooler must be taken out of service. This prevents the low flow alarm function from actuating the Process Monitor Fault stat alarm unnecessarily.

f. The sample pump is powered from breaker #39 on lighting panel 1L32. If a low flow alarm is received, an Operator must be immediately dispatched to stop the pump because there is no automatic pump cutoff on low suction pressure. However, it will trip on high discharge pressure.

D. (1)(2)(3)RIA-36

1. This monitor has been deleted and is being removed from all units.

E. (1)(3)RIA-37 and 38 - Sorrento

1. Monitor waste gas effluent.

2. RIA-37 detector plastic beta.

a. The low range beta scintillation detector has an integal beta checksource to periodically verify proper channel functioning.

b. The range for RIA-37 has been shifted slightly higher. With the old detectors, the monitor had been offscale high during about 80% of the effluent releases.

FROM. NOTICE OF VIOLATION AND NOTICE OF DEVIATION (NRC INSPECTION RE. ?T NOS. 50-269/93-25, 50-270, ?25, AND 50-287/93-25)

COMMENT #6

degrade. The licensee had no analyses supporting operation in this recirculation condition. Without such an analysis LPSW system operation was inconsistent with system design requirements.

10 CFR 50, Appendix B, Criterion III, "Design Control," requires in part that measures shall be established to assure that design bases are correctly translated into design documents and procedures. This is considered an example of Violation 50-269, 270, 287/93-25-03F, "Failure to Perform Adequate Calculations and Evaluations to Support Facility Design."

(3) Procedure AP/1/A/1700/13, Case B, "Dam Failure Without Loss of Intake Canal", required the operator to actuate ECCW by pressing the "CCW DAM FAILURE" pushbutton. This action tripped the CCW pumps and opened the emergency discharge valve to the Keowee tailrace, CCW-8, to establish the ECCW flow. Upon restoration of power to the CCW pumps the operator was directed to start a CCW pump, verify that CCW-8 closed, and verify the emergency discharge to the CCW intake canal structure valve, CCW-9, opened.

However, as a consequence of the Keowee Dam failure valve CCW-8 would be submerged. Per Calculation FERC Project Number 2503 dated December 18, 1992, "Final Summary of Analysis to Determine the Extent of Inundation Due to Catastrophic Dam Failure for Keowee Hydro Project," for the dam failure event, the water level at the valve would reach a maximum of 776 feet for the "sunny day" failure and 785 feet for the "postulated maximum flood" failure. Valve CCW-8 was located within the flooded area in a concrete and steel enclosure. According to this calculation, the valve would be submerged by as much as 55 feet, and 12 hours would elapse before the water would recede below the valve. Even with the receding water, the water, mud and debris trapped within the enclosure could impact valve operation.

The licensee's analysis had not considered the consequences of the dam failure affecting valve CCW-8. Subsequent licensee review concluded that all operator actions associated with the valve's cycling would be accomplished prior to the flood water reaching the valve. However, the procedure and previous operator training did not indicate that these actions were time dependent. Also, the calculation establishing the time available for operator action was not very precise. Therefore, the ability to perform the necessary actions without such procedural direction/training was questionable.

(4) Procedure AP/1/A/1700/13, "Loss of Condenser Circulating Water Intake Canal/Dam Failure," had other weaknesses including:

Documentation for Comment on RO#62/SRO#55 Comment #6

LOSS OF CONDENSER CIRCULATING WATER
INTAKE CANAL/DAM FAILURE
AP/2/A/1700/13

CASE B

Dam Failure Without Loss Of CCW Intake Canal

Immediate Manual Actions

4.0 Immediate Manual Actions

CAUTION **The amount of inventory loss from the Intake Canal is directly related to the amount of time that elapses during the completion of this Section.**

_____4.1 Monitor lake level for indication of dam failure:

4.1.1 IF a dam failure has occurred,
THEN manually trip the Reactor:

- **REFER** TO EP/2/A/1800/01, EMERGENCY OPERATING PROCEDURE.

_____4.2 Depress the "CCW DAM FAILURE" pushbutton.

_____4.3 Verify the following:

- All CCW pumps tripped
- 2CCW-7 (Waterbox Emergency Disch) valve is open
- All Condenser Outlet valves closed by computer point:

_____2CCW-20 (Condenser '2A1' Outlet) D0273
_____2CCW-21 (Condenser '2A2' Outlet) D0275

_____2CCW-22 (Condenser '2B1' Outlet) D0277
_____2CCW-23 (Condenser '2B2' Outlet) D0279

_____2CCW-24 (Condenser '2C1' Outlet) D0281
_____2CCW-25 (Condenser '2C2' Outlet) D0283.

Documentation for Comment on RO# 84/SRO #77

Comment #7

C. Release Pathways From the RB

1. Aside from a catastrophic failure of the RB, fission product activity can reach the environment by way of the Auxiliary Building through several systems that are interconnected with the RCS. Normally these pathways can be readily isolated following an accident, provided it is recognized that a release is occurring. Some of the more probable escape points into the Auxiliary Building from these systems are:

   a. the letdown line from the RCS, generally due to RV lift following letdown overpressurization.

   b. the LPI System, due to RV lift because of LPI System overpressurization.

   c. RBS System leakage.

   d. LDST RV leakage.

   e. Waste Gas System leakage or overpressurization. [LPRO #7 LPSO #10 PTRQ #4]

2. While these pathways can generally be readily isolated, the Operator may be reluctant to do so, feeling that the system affected is essential for mitigating the accident. At TMI-2, the letdown line was probably the single largest contributor to the releases that occurred to the environment, but this pathway was never intentionally isolated by the operators.

3. Release paths directly to the environment from the RB are few, but those that may occur would be relatively difficult to isolate. Some possible examples of these pathways are:

   a. leakage through the RB Purge System valves into the Auxiliary Building exhaust system.

   b. leakage through closures, like the RB Equipment Hatch.

   c. Main Steam relief valves leaking following a SG tube leak or rupture.

   d. Condenser Air Ejector discharges following a SG tube leak.

   e. Turbine Building Sump releases following a SG tube leak. [LPRO #8 LPSO #11 PTRQ #5]

2.6 Release Pathways Identified at TMI-2

A. The accident at TMI-2 involved an extended loss of coolant (via the PORV) which resulted in a temperature excursion high enough to damage the core. Fuel rods burst, releasing fission activity directly into the coolant; continued temperature excursions caused by Zr-H20 reactions resulted in the melting of core materials, and the expulsion of fuel pellets from the fuel pins. Almost three hours into the accident, the hot, oxidized cladding failed as it was thermally shocked when the core was finally reflooded.

B. The percentage of fission products released from the fuel rods into the coolant during the accident has been estimated to be about 70% of the pre-accident inventory of noble gases in the core, and about 75% of the iodine and cesium inventories.

C. Virtually all of the offsite dose consequences that occurred from the TMI-2 accident was as a result of the noble gases released; about 13 million curies of Xe-133 were released, along with lesser amounts of other noble gases.

Documentation for Comment on RO#87/SRO#80

Comment #8

3.8 FUEL LOADING AND REFUELING

Applicability

Applies to fuel loading and refueling operations.

Objective

To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by RIA-3 and by a portable bridge monitor for each bridge which is being used for fuel handling. Radiation levels in the spent fuel storage area shall be monitored by RIA-6 and by a portable bridge monitor. If any of these required instruments becomes inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3 At least one low pressure injection pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required to shutdown the core to a $k_{eff} \leq 0.99$ if all control rods were removed.

3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the reactor building at least one door on the personnel and emergency hatches shall be closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 Both isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed.

3.8.8 When two irradiated fuel assemblies are being handled simultaneously within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.

Irradiated fuel assemblies may be handled with the Auxiliary Hoist provided no other irradiated fuel assembly is being handled in the fuel transfer canal.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

REFUELING PROCEDURE

1.0 Purpose

To describe the procedure to be followed for the refueling.

2.0 Limits and Precautions

2.1 If any of the specified Limits and Precautions for fuel loading are not met, the SRO in charge shall evaluate and take appropriate action. Action shall be initiated to correct the violated conditions so that the specified limits are met, and no operations that may affect the reactivity of the core shall be conducted until the specified limits are met.

2.2 Radiation levels in the Reactor Building refueling area shall be monitored by 1RIA-49, 1RIA-3, 1RIA-4. Radiation levels in the Spent Fuel Pool area shall be monitored by RIA-41 and RIA-6. All of the above RIAs and their alarms will be checked for operability within one week prior to refueling. When a fuel bridge is being used to handle fuel, radiation levels shall be monitored by an area mointor mounted on the bridge. Should one of the above alarms sound, due to high radiation, evacuate immediately to an Aux. Bldg. Personnel Change - Decon. Room, Hotside (Room 315 for Rx Bldg; Room 615 for Spent Fuel Bldg.). Persons in Spent Fuel Receiving area, shall evacuate to Room 315.

2.3 If any of the following radiation monitors becomes inoperable, portable survey instrumentation must be used. The instrumentation shall have appropriate range and sensitivity to fully protect individuals involved in refueling operation until the permanent instrumentation is returned to service. Contact RP for an appropriate substitute instrument. Substitute instrument for 1RIA-3 or RIA-6 must be a Portable Area Monitor with local alarm capability. Area monitors are required only on bridges which are being used to handle fuel.

- 1RIA-49 - RB Airborne Activity Monitoring.
- RIA-41 - Spent Fuel Building Ventilation Air.
- 1RIA-3 - Fuel Transfer Canal Monitor.
- Main Fuel Bridge Area Monitor.
- Aux Fuel Bridge Area Monitor.
- RIA-6 - Spent Fuel Pool Area Monitor.
- SFP Bridge Area Monitor.

ENCLOSURE 5

NRC RESOLUTION OF FACILITY COMMENTS

1. RO #20, SRO #20

The January 1992 initial examination contained this question and received no comments. The January 1994 facility review team recommended changing the correct answer from "D" to "C" based on a change to the Oconee Operations Management Procedures.

Oconee Nuclear Station recently changed their policy for non-procedural bypassing of a safety system. OMP 2-1, "Duties and Responsibilities of Shift Operations Personnel," Enclosure 4.11, "Guidelines for Bypassing of Safety Systems," documented this new policy through a procedure change on June 1, 1993. Comments during the facility examination pre-review noted the change of policy to allow bypassing of a safety system under the approval of one SRO licensed supervisor. The facility review team stated that the Operations Management Procedures had been changed and that "D" was now the correct answer. The NRC examiner reviewed their references, concurred with their comment, and changed the answer key. The current controversy has been caused by the facility's failure to update a paragraph in OMP 1-2, "Rules of Practice", to reflect this new policy. To argue that the outdated procedure is more conservative and therefore should be followed, is to argue that the changed to OMP 2-1 was never intended to be made. This is without merit. It is important to note however, that the policy examined here was not effectively transmitted to the license candidates and may not be precisely understood by the instructional staff. This is an important policy to the health and safety of the public and should not be subject to multiple interpretations. The only correct answer at the time of the examination was "C."

The facility comments also stated that "the wording of answer 'D' needed to be changed in order to make the distractor totally wrong. The wording was changed but not as requested." The facility requested the addition of the word "only" to make it read:

d. Can <u>only</u> be performed with approval of two license SROs one of whom is a supervisor.

NUREG/BR-0122, Examiners Handbook for Developing Operator Licensing Examinations, Section 4.6.1(12)(h) eliminates any distractor that contains words such as "only" that suggest a wrong option. Therefore the NRC examiner chose to use the word "minimum" to indicate the least restrictive condition. The question distractor on the examination read as follows:

d. Can be performed with approval of <u>a minimum</u> of two licensed SROs, one of whom is a supervisor.

No change to the answer key.

2. RO #24, SRO #26

The answer key listed the wrong answer. Agree with facility comment. **Change answer key from "B" to "A".**

3. RO #32, SRO #32

The question tested knowledge of setpoint for the Source Range high voltage power supply cutoff. The NRC examiner included the instrument numbers parenthetically after the words "Source Range" to give dual identification of the term "Source Range" to the candidates. There is only one of these instruments that has a high voltage power supply. Therefore the question applies only to that instrument and "D" is the only correct answer. The candidates had no questions or comments to indicate any confusion during the examination on this question. **No change to the answer key.**

4. RO #35

The facility supplied reference shows that the RBCUs receive a signal from ES 5 & 6. This reference shows that the ES 5 & 6 setpoint is less than or equal to 4 # bldg. pressure. The facility supplied no reference to indicate that this setpoint is actuated at a particular temperature. Therefore, "B" is the only correct answer. **No change to the answer key.**

5. RO #45, SRO #41

Review of the references shows that "A" and "D" are correct answers. However, the same reference indicates that a loss of power will cause a fault alarm. Therefore "B" is also a correct answer. **Delete question from exam since more than two correct answers.**

6. RO #62, SRO #55

This question is not dependent on which dam has failed. The question tested the basis for getting to and performing step 4.7 quickly. Since they have no alarm to say which dam has failed, actions and procedures must assume worst case for the available symptoms. Urgency is based on a Kewowee dam failure where CCW-8 would be underwater. **No change to the answer key.**

7. RO #84, SRO #77

The facility comment refers to paragraph C.3.a of the lesson plan which lists "release paths directly to the environment from the RB." This question tested knowledge of the likely pathway for release of fission products from "the Reactor Building to the Auxiliary Building." A leak from the PORV would go to Quench Tank and then the Reactor Building through the rupture disk; therefore, answer "B" is not correct. RCP seal leakoff goes to the LDST, or if ES actuates, to the RCDT. Therefore, answer "D" is not correct.

Their comment also stated that none of these pathways is more or less probable than the others. However, in the same lesson plan, paragraph G states:

> "Of the possible leakage pathways identified at TMI, the one most closely associated with Oconee is probably the LDST vent path; vent valve bonnet leaks and high vent header pressures have been experienced at Oconee during LDST venting in the past. [LPSO #15]"

LPSO #15 refers to a training objective to discuss the most closely associated possible release pathway at Oconee with the one identified at TMI-2. **No change to the answer key.**

8. RO #87, SRO #80

Tech. Spec. 3.8.1 requires radiation levels in the spent fuel storage area to be monitored by RIA-6 and a portable bridge monitor. If one of these fails, this Tech. Spec. required additional portable survey instrumentation with "appropriate ranges and sensitivity to fully protect individuals involved in refueling operation." OP/1/A/1502/07, "Refueling Procedure," refers to this equipment to be substituted for the inoperable RIA-6. OP/1/A/1502/07 specifically requires local alarm capability. Since local alarm capability is an aspect required to fully protect individuals involved in refueling operation, there is no conflict between the two procedures. Answer "D" would not satisfy the requirements of the procedures and is not a correct answer.

The NRC examiner added the portion of distractor "B" that referred to the portable area monitor with local alarm capability in response to the facility's pre-examination comment to ensure it was the only correct answer. **No change to the answer key.**

U. S. NUCLEAR REGULATORY COMMISSION
SITE-SPECIFIC
WRITTEN EXAMINATION

APPLICANT INFORMATION	
Name: **NRC MASTER COPY**	Region: II
Date: 03/07/94	Facility/Unit: Oconee 1,2,3
License Level: **SRO**	Reactor Type: BW

INSTRUCTIONS

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80 percent. Examination papers will be picked up 4 hours after the examination starts.

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

____
Applicant's Signature

RESULTS		
Examination Value	________	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)
If you change your answer, write your selection in the blank.

MULTIPLE CHOICE

001 a b c d ____

002 a b c d ____

003 a b c d ____

004 MATCHING

a ____

b ____

c ____

d ____

MULTIPLE CHOICE

005 a b c d ____

006 a b c d ____

007 a b c d ____

008 a b c d ____

009 a b c d ____

010 a b c d ____

011 a b c d ____

012 a b c d ____

013 a b c d ____

014 a b c d ____

015 a b c d ____

016 a b c d ____

017 a b c d ____

018 a b c d ____

019 a b c d ____

020 a b c d ____

021 a b c d ____

022 a b c d ____

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024 a b c d ____

025 a b c d ____

026 a b c d ____

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028 a b c d ____

029 a b c d ____

030 a b c d ____

031 a b c d ____

032 a b c d ____

033 a b c d ____

034 a b c d ____

035 a b c d ____

036 a b c d ____

037 a b c d ____

038 a b c d ____

039 a b c d ____

040 a b c d ____

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)
If you change your answer, write your selection in the blank.

041 a b c d ____

042 a b c d ____

043 a b c d ____

044 a b c d ____

045 a b c d ____

046 a b c d ____

047 MATCHING

a ____

b ____

c ____

d ____

MULTIPLE CHOICE

048 a b c d ____

049 a b c d ____

050 MATCHING

a ____

b ____

c ____

d ____

MULTIPLE CHOICE

051 a b c d ____

052 a b c d ____

053 a b c d ____

054 a b c d ____

055 a b c d ____

056 a b c d ____

057 a b c d ____

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064 a b c d ____

065 a b c d ____

066 a b c d ____

067 a b c d ____

068 a b c d ____

069 a b c d ____

070 a b c d ____

071 a b c d ____

072 a b c d ____

073 a b c d ____

074 a b c d ____

075 a b c d ____

076 a b c d ____

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)
If you change your answer, write your selection in the blank.

077 a b c d ____
078 a b c d ____
079 a b c d ____
080 a b c d ____
081 a b c d ____
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092 a b c d ____
093 a b c d ____
094 a b c d ____
095 a b c d ____
096 a b c d ____
097 a b c d ____

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

1. Cheating on the examination will result in a denial of your application and could result in more severe penalties.

2. 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.

3. To pass the examination, you must achieve a grade of 80 percent or greater.

4. The point value for each question is indicated in parentheses after the question number.

5. There is a time limit of 4 hours for completing the examination.

6. Use only black ink or dark pencil to ensure legible copies.

7. Print your name in the blank provided on the examination cover sheet and the answer sheet.

8. Mark your answers on the answer sheet provided and do not leave any question blank.

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

10. 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.

11. When you complete the examination, assemble a package including the examination questions, examination aids, and answer sheets and give it to the examiner or proctor. Remember to sign the statement on the examination cover sheet.

12. After you have turned in your examination, leave the examination area as defined by the examiner.

QUESTION: 001 (1.00)

Unit 3 is operating at 100% rated power when 3CC-8, Component Cooling Water Containment Return Valve, inadvertently closes and can not be re-opened locally. Which one of the following states the IMMEDIATE MANUAL operator actions?

a. Trip the reactor when the pressurizer low level alarm is received.

b. Trip the reactor when two CRD stator temperatures exceed 180° F.

c. Trip individual RCPs when the motor stator temperature exceeds 185° F.

d. Trip the reactor and trip the RCPs when letdown isolates.

QUESTION: 002 (1.00)

Plant conditions on Unit 3 are as follows:

- Control Room has been evacuated due to a fire.
- SSF pressurizer level indicates 50 inches.
- SSF RCS pressure indicates 1600 psig.

Which one of the following is the ACTUAL pressurizer level? (Refer to attached figure as required.)

a. 25 inches

b. 43 inches

c. 53 inches

d. 63 inches

QUESTION: 003 (1.00)

Which one of the following will result from the failure of Reactor Building Sprays during a LOCA that results in partial fuel failure and a Reactor Building (RB) pressure of 12 psig?

a. Increased hydrogen gas generation in the steam environment and increased radioactive iodine in the RB atmosphere.

b. Increased hydrogen gas generation in the steam environment and increased radioactive krypton in the RB atmosphere.

c. Reduced hydrogen gas generation in the steam environment and increased radioactive iodine in the RB atmosphere.

d. Reduced hydrogen gas generation in the steam environment and increased radioactive krypton in the RB atmosphere.

QUESTION: 004 (2.00)

Unit 1 is operating at 100% rated power.

MATCH each of the components listed in Column A with its operational condition from Column B upon a COMPLETE LOSS of the 1KI Bus. (NOTE: Each response in Column B may be used once, more than once, or not at all, and only a single answer may occupy each answer space in Column A.) (4 answers required, 0.5 each)

Column A (COMPONENT)	Column B (CONDITION)
_____a. Motor driven EFDW pump	1. Remains in standby available for AUTOMATIC start ONLY
_____b. HP-31, RCP Seal Flow Control valve	2. Remains in standby available for MANUAL start ONLY
_____c. HP-120, RC Volume Control valve	3. Starts automatically
_____d. Main FDW pump	4. CANNOT be stopped automatically or manually
	5. Continues to run and can only be MANUALLY controlled
	6. Trips AUTOMATICALLY
	7. Must be MANUALLY tripped locally
	8. AUTOMATICALLY controls at setpoint
	9. Can ONLY be MANUALLY operated to control flow

QUESTION: 005 (1.00)

Which one of the following is the reason for maintaining the Letdown Storage Tank (LDST) below the Maximum Pressure for the Indicated LDST level?

a. To prevent hydrogen gas from entering the BWST when the HPI to BWST suction valves open on an ES actuation

b. To prevent pumping the LDST dry and gas binding of the HPI pumps following an ES actuation

c. To prevent closing the HPI to BWST suction check valves and injecting gas into the RCS following an ES actuation

d. To prevent gas binding of the RCP seal package due to injecting gas into the RCP seals upon an ES actuation

QUESTION: 006 (1.00)

Immediately following a COMPLETE Loss of Offsite Power, ES Channels 1, 2, 3, and 4 initiated. Which one of the following actions will reset the Load Shed Circuits on Unit 3 when offsite power is restored?

a. Reset the MFBMPs by simultaneously depressing both MFBMP reset switches on the MFBMP in the cable room OR simultaneously depressing both MFBMP reset switches on panel AB1.

b. Place the "Load Shed & Standby Breaker 1" RZ module OR "Load Shed & Standby Breaker 2" RZ module to MANUAL, then simultaneously depress both MFBMP reset switches on panel AB1.

c. Place the "Load Shed & Standby Breaker 1" RZ module AND "Load Shed & Standby Breaker 2" RZ module to MANUAL on panel VB2.

d. Place the "Load Shed & Standby Breaker 1" RZ module AND "Load Shed & Standby Breaker 2" RZ module to MANUAL, then simultaneously depress both MFBMP reset switches on panel UB1.

QUESTION: 007 (1.00)

Upon a loss of 1KI, AP/1/A/1700/23, "Loss of 1KI Bus," directs an operator to the Aux Shutdown Panel to perform various actions. Which one of the following is an action the operator can perform from this panel?

a. Bypass 1KI Inverter.

b. Re-energize Turbine Bypass Valves.

c. Bypass Turbine Stop Valve controls.

d. Control RCS pressure with Heater Banks 1 and 3.

QUESTION: 008 (1.00)

Which one of the following is the MINIMUM exposure which may allow waiving of the Independent Verification of a single valve?

a. 10 mrem

b. 50 mrem

c. 100 mrem

d. 1000 mrem

QUESTION: 009 (1.00)

Which one of the following describes the process of Double Verification?

a. Some time interval (less than one shift) is required between the actions of the "Doer" and the verification process of the "Verifier".

b. "Verifier" relies upon observation of actions of the "Doer" and ensures correct component identification, position or condition.

c. "Doer" determines required actions and performs any manipulation then the "Verifier" independently determines required actions and verifies it was conducted properly.

d. "Doer and "Verifier" independently determine required component and action prior to "Doer" performing manipulation.

QUESTION: 010 (1.00)

Which one of the following describes an SRWP?

a. A Special Radiation Work Permit is developed for one time use for a specific task.

b. A Special Radiation Work Permit is effective no longer than one shift without approval by RP General Supervisor.

c. A Standing Radiation Work Permit governs routine work such as plant inspections and operator rounds.

d. A Standing Radiation Work Permit is used to control specific tasks not covered by a RWP.

QUESTION: 011 (1.00)

Which one of the following defines Total Effective Dose Equivalent (TEDE) which has a 10 CFR 20 annual limit of 5.0 rem?

a. External whole body exposures.

b. A combination of deep dose equivalent (external exposures) and committed effective dose equivalent (internal exposures).

c. A combination of deep dose equivalent and shallow dose equivalent (extremity exposure).

d. A combination of deep dose equivalents, lens of the eye, skin and extremities exposures.

QUESTION: 012 (1.00)

Which one of the following is the Duke Power Administrative limit for lifetime dose? (N is the age of the employee on his/her last birthday)

a. 1.5 X (N - 17) Rem

b. 0.5 X (N-18) Rem

c. 5 X (N-18) Rem

d. There is no longer a limit for lifetime dose.

QUESTION: 013 (1.00)

Which one of the following is the LOWEST dose area which always requires continuous RP coverage for the entry?

a. Radiation Area

b. High Radiation Area

c. Extra High Radiation Area

d. Very High Radiation Area

QUESTION: 014 (1.00)

A fire has occurred on an Appendix R switchgear 4160 VAC breaker. Which one of the following describes the method that an operator should use on the FIRST attempt to close the breaker MANUALLY? The closing spring is charged.

a. Open the cubicle door and PUSH the manual close lever by hand.

b. Open the cubicle door and PULL the manual close lever by hand.

c. Secure the breaker cubicle door closed and PULL the manual close lever using a pull cord.

d. Secure the breaker cubicle door closed and PUSH the manual close lever using the breaker jack.

QUESTION: 015 (1.00)

Site Directive 3.2.3, "Personnel Safety Area Access", requires individuals entering containment to stay within sight of another individual ("Buddy System") when RCS temperature is above a certain value. Which one of the following is that temperature limit?

a. 180° F.

b. 212° F.

c. 300° F.

d. 525° F.

QUESTION: 016 (1.00)

Which one of the following actions describes the required actions when a "GO TO" statement is encountered in the EOP?

a. Transfer out of the current EOP Section into the referenced procedure or section.

b. Use the referenced procedure while continuing to progress through the steps of the current EOP.

c. Go to the referenced procedure to determine applicability to the plant status once the current EOP Section is completed.

d. Go to the referenced procedure if conditions are met otherwise continue in the current procedure until transfer conditions are met.

QUESTION: 017 (1.00)

Which one of the following describes the proper actions if a "WHEN" conditional statement in the EOP is NOT met at the time that it is read?

a. Ignore the "WHEN" statement for the duration of the transient.

b. Perform this step if the "WHEN" conditions are met anytime prior to transitioning to another procedure or Section.

c. Return to this step if the "WHEN" conditions are met anytime in this procedure or after transitioning to another procedure.

d. Return to the "WHEN" step after the remainder of this section of the EOP has been completed.

QUESTION: 018 (1.00)

Which one of the following describes the MINIMUM time requirement for the on-shift duties that an operator must satisfy to maintain an active NRC license?

a. Five 8 hour shifts per calendar quarter.

b. Five 12 hour shifts per calendar quarter.

c. A total of 40 hours over any number of shifts per calendar quarter.

d. A total of 60 hours over any number of shifts per calendar quarter.

QUESTION: 019 (1.00)

Which one of the following is the Control Room SRO authorized to do without being relieved?

a. Provide relief for the Control Room operators.

b. Prepare Removal and Restorations (R&Rs).

c. Designate another SRO as reader of the EOP.

d. Prepare procedure changes.

QUESTION: 020 (1.00)

Which one of the following is the Station's Policy concerning the non-procedural blocking of an automatic safety actuation? The determination has been made that the safety system is not required to perform its function and the actuation of the safety system could increase the severity of the transient?

a. Can be performed at any time but must be reported to the Shift Supervisor immediately.

b. Can be performed with approval of two licensed Reactor Operators.

c. Can be performed with approval of one SRO licensed supervisor.

d. Can be performed with approval of a minimum of two licensed SROs one of whom is a supervisor.

QUESTION: 021 (1.00)

Which one of the following lists the MINIMUM APPROVAL(S) required to extend an individual's dose limits to the planned EMERGENCY exposure limit for preventing the loss of essential equipment during an emergency?

a. Verbal approval of the Emergency Coordinator.

b. Verbal approval of the Emergency Coordinator AND the Radiological Protection Manager.

c. Written approval of the Site Vice President.

d. Written approval of the Radiological Protection Manager AND the Emergency Coordinator.

QUESTION: 022 (1.00)

Which one of the following is allowed secondary system pH range?

a. 6.9 - 7.2

b. 7.9 - 8.2

c. 9.3 - 9.6

d. 10.3 - 10.6

QUESTION: 023 (1.00)

The Emergency Plan has been implemented for a fire that could affect Control Rooms 1&2. Which one of the following would be the location of the Technical Support Center?

a. Operations Center area of Control Room 3

b. Oconee Office Building

c. Administrative Building

d. Oconee Training Center

QUESTION: 024 (1.00)

The Emergency Coordinator declared and approved an upgrade of an emergency classification from a Notification of Unusual Event to an Alert. Which one of the following is the amount of time allowed by RP/O/1000/15, "Offsite Communications", for making the offsite notification message to the state and county?

a. Immediately

b. 15 minutes

c. 30 minutes

d. one hour

QUESTION: 025 (1.00)

During a reactor coolant heatup Pressurizer Level is maintained less than or equal to 220 inches until RCS temperature is above 325° F. Which one of the following is the reason for this?

a. To provide an adequate surge volume to prevent the PZR from going solid in the event of a continuous control rod withdrawal accident.

b. To provide operators time to take action to prevent an RCS overpressurization upon failure of HP-120, PZR Level Control Valve.

c. To provide adequate surge volume for the RCS during the heatup as the density of the primary coolant decreases.

d. To provide operators time to take action to prevent the PZR from going solid upon an inadvertent initiation of HPI.

QUESTION: 026 (1.00)

For which one of the following evolutions is the CRD System Auxiliary Power Supply the only power source available?

a. Withdraw of the Safety Groups to their Out-Limits.

b. Withdraw of Group 8.

c. Operation of the Regulating Groups out of sequence.

d. Latch and PI alignment of the Regulating Groups.

QUESTION: 027 (1.00)

The reactor is operating at 100% rated power with ICS in AUTOMATIC, when a reactor trip and a trip of all RCPs occurs. Main Feedwater remains in operation and NO operator actions are taken.

Which one of the following describes the feedwater injection path to the B OTSG? (Refer to the Feedwater System Drawing attached.)

a. Single flow path through FDW-42, FDW-44, and FDW-45.

b. Single flow path through FDW-42, FDW-44, and FDW-47.

c. Single flow path through FDW-40, and FDW-41.

d. FDW-40 to FDW-41 in parallel with FDW-42, FDW-44, and FDW-47.

QUESTION: 028 (1.00)

Unit 1 had been operating for 211 days. After a trip, an ECP is to be calculated. An ECP for a restart at which one of the following times would have the highest rod position due to Xenon.

a. one hour after the trip

b. four hours after the trip

c. twenty hours after the trip

d. forty hours after the trip

QUESTION: 029 (1.00)

Which one of the following describes a design function of the warming lines that bypass HP-120, Pressurizer Level Control, in the normal injection line of the HPI system?

a. Provides a continuous flow path for RCS purification during reactor startup; when makeup may not be in service.

b. Prevents thermal stress of the "A" Loop injection nozzles from cold BWST water during operation in the normal makeup mode.

c. Provides a method for warming the normal injection lines during reactor startup to prevent thermal stress.

d. Prevents thermal stress of the "A" Loop injection nozzles from cold BWST water if the HPI system is operated in the ES mode.

QUESTION: 030 (1.00)

Which of the following aspects of HPI interlocks associated with the Seal Injection and Seal Return is true on Unit 1?

a. If RCS pressure is ≥ 400 psig and seal supply as well as CC flow is lost, all four seal return valves will go closed.

b. When all four seal return valves close, HP-21, Seal Return Stop, will close.

c. If seal injection flow drops < 4 gpm/pump on each pump for greater than one minute, HP-31, Seal Injection Flow Control, will automatically close.

d. If seal injection flow drops ≤ 30 gpm, the Standby HPI pump will Auto Start (if selected to "AUTO").

QUESTION: 031 (1.00)

Which one of the following is correct concerning the response of the Engineered Safeguards System to a loss of power to more than one analog channel?

a. On the loss of KVIA and KVIC power, analog channels A & C will send trip signals to only EVEN numbered digital channels causing channels 2, 4, 6, and 8 to actuate.

b. On the loss of KVIB and KVIC power, analog channels B & C will send trip signals to only ODD numbered digital channels causing channels 1, 3, 5, and 7 to actuate.

c. On the loss of KVIA and KVIB power, analog channels A & B will send trip signals to all digital channels causing all channels to actuate.

d. On the loss of KVIA and KVIB power, analog channels A & B will send trip signals to all digital channels, but no digital channels will actuate.

QUESTION: 032 (1.00)

Which one of the following sets of conditions describes the setpoint of the Source Range (NI-1 and NI-2) high voltage power supply cutoff?

a. At least two power range channels ≥ 10% power, either NI-5 and NI-6 OR NI-7 and NI-8.

b. At least two power range channels ≥ 10% power, either NI-5 or NI-6 AND NI-7 or NI-8.

c. At 10-9 amps on NI-3 AND NI-4.

d. At 10-3% power on NI-3 AND 10-9 amps on NI-4.

QUESTION: 033 (1.00)

Upon receipt of a Reactor Building Cooling Unit (RBCU) rupture alarm, it is important to secure the affected RBCU and to isolate Low Pressure Service Water (LPSW) in order to prevent which one of the following?

a. Runout of the LPSW pumps.

b. LPSW from collecting in the Reactor Building sump.

c. Radioactive contamination of the LPSW system.

d. Overheating of the RBCU fan.

QUESTION: 034 (1.00)

The Hotwell Pump (HWP) discharge controller (C-10) is interlocked such that it must be ≤ 10% open before starting the first HWP. This interlock was installed to prevent which one of the following?

a. The HWP from tripping on overcurrent while it is coming up to full speed.

b. Backflow through the idle HWPs until the minimum flow recirculation line opens.

c. Rupturing of the generator hydrogen cooler gaskets.

d. Flow pressure shock from flushing demineralizer resins out of the Powdex cells.

QUESTION: 035 (1.00)

The following Unit 2 conditions exist:

- Plant startup and power increase are in progress.
- Reactor power is 26%
- The Reactor Operator reports the Steam Generator Load Ratio (ΔTc) station is in MANUAL.

Which one of the following conditions must be met before the ΔTc station should be placed in AUTO?

a. Both OTSGs are on level control.

b. The unit's ΔTc instrument must be within +/- 2°F.

c. Neither OTSG is on level control.

d. Both Feedwater Masters are in MANUAL.

QUESTION: 036 (1.00)

The Unit 2 TDEFDWP has started and is supplying the OTSGs following an automatic initiation signal.

Which one of the following describes the response of FDW-315 and FDW-316 (SG EFDW Control Valves) controllers when the initiation signal clears on Unit 2?

a. Continue to operate on level control until the MANUAL pushbutton is depressed.

b. Continue to operate on level control until the AUTO initiation signal is reset.

c. Shifts to MANUAL as soon as the AUTO initiation signal clears.

d. Shifts to MANUAL after the TDEFDW pump control switch is placed to OFF (spring returns to AUTO).

QUESTION: 037 (1.00)

Which one of the following describes the operation of the AMSAC and the DSS during an ATWS with a complete loss of Main Feedwater?

a. AMSAC trips the main turbine while DSS trips the regulating rods and starts the EFDWPs.

b. AMSAC trips the regulating rods while DSS trips the main turbine and starts the EFDWPs.

c. AMSAC trips the main turbine and starts the EFDWPs while DSS trips the regulating rods.

d. AMSAC trips the regulating rods and starts the EFDWPs while DSS trips the main turbine.

QUESTION: 038 (1.00)

A complete loss of DC power to the Turbine Driven EFWP has occurred.

Which one of the following describes the response of the TDEFWP if an automatic initiation signal is received?

a. Will automatically start because the steam supply valve, MS-93, will be opened when its pilot solenoid deenergizes, and the operating valve, MS-95, fails open on loss of oil pressure.

b. Due to a loss of auto initiation logic control power, must be manually started by placing the Control Room control switch to RUN which will open MS-93.

c. Will not start locally unless started by pulling up on the local hand starting lever, which physically opens the operating valve, MS-95, admitting steam to roll the turbine.

d. Can not be started because the DC auxiliary oil pump is not available.

QUESTION: 039 (1.00)

Which one of the following completes the statement on Process Radiation Monitors.

A high radiation trip of the Reactor Building (RB) Radiation Monitor RIA-49 will ___(1)___ and a high radiation trip of the Turbine Building (TB) Sump Monitor RIA-54 will ___(2)___.

a. (1) close RB normal sump isolation, (2) trip the TB sump pumps.

b. (1) close RB normal sump isolation, (2) close TB sump isolation.

c. (1) trip RB sump pump, (2) trip the TB sump pumps.

d. (1) trip RB sump pump, (2) close the TB sump isolation.

QUESTION: 040 (1.00)

Which one of the following statements is correct concerning the Core Exit Thermocouples (CETCs)?

a. If a CETC is in scan lockout, an inserted value can be used in the calculation.

b. If power is ≥ 2%, it is assumed that a hostile environment exists in the Reactor Building.

c. When power is < 2% for about 45 seconds, the SCM program takes the average of the 24 highest operable qualified CETCs (12 from each train).

d. When power ≥ 2%, the Subcooling Margin (SCM) program takes the average of the 47 operable CETCs not being used by the Safe Shutdown Facility.

QUESTION: 041 (1.00) DELETE 3-24-94 ME

~~Which one of the following describes the meaning of a fault alarm received in the Control Room from a Sorrento monitor skid via the System Control and Data Acquisition (SCADA) system?~~

a. ~~The monitor's sample pump has failed.~~

b. ~~Power to one of the Sorrento detectors has failed.~~

c. ~~A fault has occurred on one of the control boards in the RM-80 microprocessor.~~

d. ~~A Sorrento detector channel has failed to respond properly to its check source.~~

QUESTION: 042 (1.00)

Which one of the following is the automatic action associated with a "GWD DISCHARGE RAD INHIBIT" alarm (1SA-8/C-10) from RIA 37 (Normal WD Gas) or RIA-38 (Hi WD Gas)?

a. Trips and isolates the Waste Gas Supply (WGS) fan.

b. Closes the discharge valves from all GWD tanks and all GWD interim tanks.

c. Closes GWD-100 (Decay Tank Discharge Header Block) to stop all gas tank discharge flow through the Unit 1 stack.

d. Closes GWD-132 and GWD-134 (Unit 2 & 3 Vent Header Cross Connects) in the vent header to split Unit 3 from Units 1 and 2.

QUESTION: 043 (1.00)

Which one of the following is correct regarding the "Piggyback" mode of Low Pressure Injection (LPI)?

a. Emergency injection is automatically realigned to the Piggyback Mode when BWST level decreases to 10 feet.

b. Emergency injection is manually realigned to the Piggyback Mode when BWST level ≤ 6 feet and the RBNS level is off scale high.

c. Following a Large Break LOCA, the Piggyback Mode provides a long term source of water to the HPI pump suction from the RBES through the LPI pumps.

d. The Piggyback Mode provides a long term source of water from the RBES through the LPI pumps for additional cooling or extended operation of the Reactor Building Spray system.

QUESTION: 044 (1.00)

Which one of the following is the adverse effect of an abnormally high Core Flood Tank (CFT) level during a large LOCA event?

a. The CFT may dump gas into the RCS.

b. The CFT may dump into the RCS too soon.

c. The CFT may dump into the RCS too late.

d. The CFT may dump insufficient water to re-cover the core hot spot.

QUESTION: 045 (1.00)

Which one of the following describes the basis for limiting the pressurization rate of the Core Flood Tanks (CFTs) to < 100 psig per 15 minute time interval?

a. To prevent thermal shock of the CFT.

b. To prevent lifting the CFT relief valves.

c. To prevent a pressure surge from unseating the check valves.

d. To prevent exceeding the CFT pressure Technical Specification limit.

QUESTION: 046 (1.00)

Through which one of the following methods does the Diverse Scram System trip the reactor?

a. Interrupts power to the SCR gate drives.

b. Provides a direct trip signal to the CRD breakers.

c. Interrupts power to the normal and alternate CRD power supply sources.

d. Provides a redundant signal to deenergize the CRDM motor generators.

QUESTION: 047 (2.00)

Match each component listed in Column A with its correct RCS penetration location from Column B. (NOTE: Each response in Column B may be used once, more than once, or not at all, and only a single answer may occupy one answer space.) [0.5 pt. each]

Column A (EQUIPMENT)	Column B (Location)
_____ a. Unit 1 PZR Spray	1. A1 RCP Suction
_____ b. Unit 3 Normal HPI Line	2. B1 RCP Suction
_____ c. Unit 2 Letdown Line	3. A1 RCP Discharge
_____ d. Unit 2 Decay Heat Removal Line	4. B1 RCP Discharge
	5. B2 RCP Discharge
	6. A Hot Leg
	7. B Hot Leg

QUESTION: 048 (1.00)

Which one of the following describes how to manually open the Unit 3 PORV (3RC-66) from the control room with RCS pressure at 2000 psig.?

a. Select to low setpoint, using the setpoint key 3RC-66 selector switch on UB1.

b. Select OPEN using the spring loaded keyswitch on the back of ICS Cabinet 13.

c. Select OPEN on the setpoint selector switch, and push the OPEN-PERMIT pushbutton.

d. Select LOCAL-ASP on the Auxiliary Shutdown Panel control switch, and push the OPEN pushbutton.

QUESTION: 049 (1.00)

Which one of the following describes the reason that Emergency Feedwater (EFW) can have a proportionally larger cooling effect on the RCS than Main Feedwater? Assume feedwater flow rates are the same and all RCPs are in operation.

a. EFW is injected into the steam space.

b. Main Feedwater is NOT counterflow to the OTSG tubes.

c. OTSG pressure is higher during EFW flow conditions.

d. EFW is injected into the outer wrapper area of the OTSG.

QUESTION: 050 (2.00)

Match the pressurizer level instrument in Column A with its correct type of level compensation from Column B. (NOTE: Each response in Column B may be used once, more than once, or not at all, and only a single answer may occupy one answer space.) [0.5 pt. each]

Column A PZR INSTRUMENT	Column B TYPE COMPENSATION
_____ a. Pzr Level Channel 1	1. Uncompensated
_____ b. Pzr Level Channel 3 (on Dixon indication on UB1)	2. Temperature Compensated by RCS WR RTD
_____ c. Pzr Level on ASP	3. Temperature Compensated by Pzr RTD
_____ d. Pzr Level on SSF	4. Pressure Compensated

QUESTION: 051 (1.00)

Which one of the following describes the recommended Reactor Building Spray Flow when in the recirculation mode from the RBES and the basis for the recommended flow?

a. 1500 gpm to meet Environmental Qualification analysis.

b. 1500 gpm to prevent pump runout.

c. 1000 gpm to keep within available NPSH.

d. 1000 gpm to meet pump minimum flow requirements.

QUESTION: 052 (1.00)

Which one of the following describes a valid interlock associated with the Component Cooling (CC) System?

a. A RCP will trip if CC flow is less than 575 gpm.

b. CC-1 or CC-2 (CC to letdown coolers) must be open to open CC-7 or CC-8 (CC Return Penetration Blocks).

c. If CC-7 closes, the CC pumps will trip AND automatically restart when reopened.

d. CRDs cannot be energized unless CC flow is greater than 138 gpm OR RCS temperature is less than 190° F.

QUESTION: 053 (1.00)

Keowee Hydro Unit (KHU) 1 is generating to the grid and KHU 2 is in standby dedicated to the underground power path. A LOCA occurs on Oconee Unit 2 and shortly thereafter a Switchyard Isolation Signal is generated. Due to a fault, PCB-8 fails to open. Which one of the following describes the emergency power lineup following these events?

a. KHU-1 continues operating at rated speed, ACB-1 recloses and supplies the Yellow Bus via the overhead power path and PCB-9. KHU-2 emergency starts, and immediately supplies the Standby Busses via the underground power path and CT-4.

b. KHU-1 continues operating at rated speed but not tied to the overhead power path. The Yellow Bus will remain de-energized. KHU-2 emergency starts, and immediately supplies the Standby Busses via the underground power path and CT-4.

c. KHU-1 trips due to overspeed and receives a Normal Lockout. KHU-2 emergency starts and upon reaching rated speed, ACB-2 closes supplying the Yellow Bus via the overhead power path and PCB-9. The Lee Steam Station will supply the Standby Busses via CT-5.

d. KHU-1 trips due to the fault on PCB-8 but restarts on the Keowee Emergency Start signal. KHU-1 will accelerate to rated speed but not tie to the overhead power path. The Yellow Bus will remain de-energized. KHU-2 automatically starts and upon reaching rated speed, ACB-4 closes supplying the Standby Busses via the underground path and CT-4.

QUESTION: 054 (1.00)

Both Keowee Hydro Units have received an Emergency Start signal. KHU-1 is aligned to the underground power path and KHU-2 is aligned to the overhead power path. The Keowee Operator reports that KHU-1 has received an Emergency Lockout statalarm. Which one of the following describes the impact of this statalarm on Keowee operations?

a. KHU-1 will continue to operate.

b. ACB-2 will trip and be blocked from reclosing.

c. ACB-3 will trip and be blocked from reclosing.

d. Both KHUs Field, Field Flashing and Supply breakers will trip.

QUESTION: 055 (1.00)

A "CCW LAKE LEVEL LOW" statalarm has been received. Plant operators pressed the "CCW DAM FAILURE" pushbutton and followed the Immediate Manual Actions of AP/2/A/1700/13, "CASE B, Dam Failure Without Loss of CCW Intake Canal". Step 4.7 has the operator close or verify closed CCW-8, EMERG CCW DISCH TO TAILRACE. Which one of the following describes the reason why this and the previous steps must be performed quickly?

a. CCW-8 would soon be under several feet of water and may not be operable.

b. The ECCW siphon would be broken unless CCW-8 was closed shortly after dam failure.

c. The "Dam Failure" logic would prevent CCW-8 from closing after a 15 minute time delay.

d. The actions to realign the CCW system for recirculation to the intake canal are time dependent. After 15 minutes, this lineup would be ineffective.

QUESTION: 056 (1.00)

The fire suppression spray/sprinkler system is declared inoperable for the Unit 2 Turbine Driven Emergency FDW pump. Assume the Fire Detection system is inoperable. Which one of the following actions is required by SLC 16.9, Auxiliary Systems?

a. Commence a unit shutdown within one hour.

b. Establish, within one hour a continuous fire watch for the affected area with backup fire suppression equipment.

c. Establish, within one hour an hourly fire watch patrol for the affected area with backup water fire suppression equipment.

d. Log ambient temperature readings for the affected area hourly and route an equivalent capacity fire hose to the area.

QUESTION: 057 (1.00)

Which one of the following describes the condition that exists if the Sync Verification Indicator lamp on a Vital Bus Inverter stays on continuously at half brightness?

a. 125V DC voltage supplying the inverter is low.

b. AC line voltage is low if inverter voltage is normal.

c. Voltage and frequency of the AC line and the inverter are out of phase.

d. Voltage and frequency are properly matched between the inverter output and the 120V AC Regulated Power System.

QUESTION: 058 (1.00)

Which one of the following is correct concerning interlocks for operation of the Standby Busses from emergency power sources? (Assume normal operating lineup.)

a. The Lee Feeder Breakers (SL1 & SL2) may automatically shut if the Keowee Feeder Breakers (SK1 & SK2) are shut.

b. The Lee Feeder Breakers (SL1 & SL2) may automatically shut if the Keowee Feeder Breakers (SK1 & SK2) are open.

c. The Keowee Feeder Breakers (SK1 & SK2) may automatically shut if the Lee Feeder Breakers (SL1 & SL2) are shut.

d. The Keowee Feeder Breakers (SK1 & SK2) may automatically shut if the Lee Feeder Breakers (SL1 & SL2) are open.

QUESTION: 059 (1.00)

Which one of the following lists the unit(s) which have bypass valves (LP-92 and LP-93) on the Low Pressure Injection cooler?

a. Unit 2 only.

b. Unit 3 only.

c. Both Units 1 and 2.

d. Both Units 2 and 3.

QUESTION: 060 (1.00)

Which one of the following is an automatic action associated with a HIGH alarm on Reactor Building monitor RIA-49?

a. Trips Reactor Building Sump Pump.

b. Trips Reactor Building Main and Mini-purge fans.

c. Closes valves PR-2 through PR-5.

d. Closes valve LWD-2, "RB Normal Sump Isolation".

QUESTION: 061 (1.00)

Which one of the following, describes the basis for the design of the spent fuel storage racks?

This design keeps Keff _____ with ________, high density 1.0 gm/cubic cm water.

a. < 0.95 / unborated

b. < 0.95 / borated

c. < 0.99 / unborated

d. < 0.99 / borated

QUESTION: 062 (1.00)

Which one of the following statements is correct with respect to containment integrity if a motor operated containment isolation valve fails in its ES position?

a. Containment integrity is maintained ONLY if power is removed from the valve.

b. Containment integrity is maintained since the valve is considered to be operable in the context of containment integrity.

c. Containment integrity is lost until the affected penetration is isolated by use of either a closed manual or automatic backup isolation valve located inside containment only.

d. Containment integrity is lost until the affected penetration is isolated by use of either a closed manual or automatic backup isolation valve located inside or outside containment.

QUESTION: 063 (1.00)

According to AP/1/A/1700/17, "Loss of Containment Integrity", which one of the following is the correct operator action if a pneumatically-operated valve has been found FAILED OPEN (and its ES position is SHUT)?

In order to re-establish containment integrity, the valve must be shut and:

a. De-energize its solenoid.

b. Blocked with an appropriate blocking device.

c. Tagged shut at all of its operating stations.

d. Operating air isolated with the valve operator bled off.

QUESTION: 064 (1.00)

According to OP/0/A/1105/09, "Control Rod Drive System", which one of the following is the basis for the below precaution?

PRECAUTION: Attempts to operate a partially withdrawn control rod, which is stuck or jammed, may only be accomplished in RUN speed.

a. JOG speed would cause the motor coils to heat above design temperature limits.

b. Attempts to operate in JOG speed may overload the spider.

c. JOG speed is unavailable to Regulating Rods.

d. Operating in JOG speed would be too slow to meet Technical Specification time limits.

QUESTION: 065 (1.00)

According to AP/1/A/1700/21, "High Activity in the RC System", if the unit is shutdown and RCS cooldown is to be performed during operations with failed fuel, Reactor Building air temperatures are to be maintained greater than 100° F. Which one of the following is the reason for this requirement?

a. To prevent Iodine from being absorbed in the concrete of the RB.

b. To reduce the amount of fission product gas released from the RCS coolant.

c. To increase the relative humidity in the RB which enhances the removal of fission product particulates from the atmosphere.

d. To prevent Cobalt-60 from being absorbed in the concrete of the RB.

QUESTION: 066 (1.00)

The following conditions exist:

- Rx Power 100 %
- RCS Pressure is 2100 # and decreasing
- Pressurizer Level is 220 inches
- Pressurizer Temperature is decreasing
- LDST Level is constant

Which one of the following is the cause for these indications.

a. A Pressurizer steam leak is occurring.

b. An RCS leak in the Letdown System is occurring.

c. The Pressurizer level is below the heater cutoff level.

d. The Pressurizer Spray Valve is stuck open.

QUESTION: 067 (1.00)

The following plant conditions exist:

- A steam line rupture has occurred and Rx is tripped
- Section 503 has been implemented (Excessive Heat Transfer)
- Operators are attempting to isolate OTSG "B"
- RCS Temperature and Pressure are decreasing

Which one of the following describes the reason for securing one of the four RCPs when RCS temperature is about 325° F ?

a. Minimizes RCS depressurization by reducing spray flow.

b. Reduces heat transfer to secondary system to control cooldown.

c. Reduces dynamic head to increase margin of safety to PTS.

d. Prevents damage to fuel assembly spacer grids due to core lift.

QUESTION: 068 (1.00)

The following plant conditions exist:

- RCS Pressure and Temperature rapidly decreased
- Only One HPI Pump is injecting
- Both OTSGs have just been isolated
- "B" OTSG Pressure is rapidly decreasing.
- "A" OTSG Pressure is slowly decreasing.
- RCS Pressure is holding constant
- CETCs holding constant

Section 503, Excessive Heat Transfer, is in use, step 10.0 is in progress to re-establish FDW to the intact OTSG, when Subcooling Margin becomes less than zero. Which one of the following is the correct operator action to take?

a. Transfer to Section 502, Loss of Heat Transfer.

b. Transfer to Section 501, Loss of Subcooling.

c. Continue to re-establish FDW flow to the "A" OTSG.

d. Trip one RCP per loop.

QUESTION: 069 (1.00)

Which one of the following describes the correct method of bumping a RCP when attempting to recouple an OTSG during a loss of heat transfer event?

a. Start pump, allow starting current to decay to normal running amps, then secure pump.

b. Start pump, run for 10 seconds, then secure pump.

c. Start pump, run for 1 minute, then secure pump.

d. Start pump, run until Th in affected loop stabilizes, then secure pump.

QUESTION: 070 (1.00)

The following plant conditions exist:

- The Rx had been at 60 % power for 2 days
- Rx Tripped 35 minutes ago due to loss of FDW
- Section 502, Loss of Heat Transfer, is in use and HPI Cooling was initiated
- Present SCM > 20° F.

The operators are in the process of recovering from HPI cooling and have re-established FDW flow in accordance with Enclosure 7.6, Total Feedwater Flow Required To Match Decay Heat, (attached) at a rate of 135 gpm per OTSG. Which one of the following effects on the RCS will this have when the PORV and High Point Vents are closed?

a. The RCS will continue to heat up with a reduction in SCM.

b. The RCS temperature and pressure will remain the same.

c. The RCS will overcool resulting in a loss of SCM.

d. The RCS will continue to cool with an increase in SCM.

QUESTION: 071 (1.00)

Which one of the following conditions requires entry into AP/1/A/1700/12, "Loose Parts in Reactor Coolant System"?

a. Several SG tube plugs are discovered missing during a SG tube inspection.

b. Excessive vibration is detected on one RCP.

c. Results of chemistry samples indicate an increase of failed fuel activity.

d. A sustained low magnitude noise is detected by the Loose Parts Monitoring System.

QUESTION: 072 (1.00)

Which one of the following describes the reason for draining a ruptured OTSG using the Hot Blowdown drain flow path to the condenser in accordance with Section 504, SG Tube Leak.

a. To maintain tube to shell delta T < 100° F.

b. To maintain affected OTSG pressure < 950 psig.

c. To reduce the chance of a release of primary water to atmosphere.

d. To reduce OTSG activity concentration to minimize release rate.

QUESTION: 073 (1.00)

Which one of the following plant conditions requires entry into Section 507, Inadequate Core Cooling?

a. A large break LOCA has occurred. Section 501, Loss of Subcooling, is in use. RCS pressure is 75 psia five minutes after the break occurred. CETCs are 340° F.

b. A steam generator tube rupture has occurred. Section 504, SG Tube Leak, in use. RCS pressure is 900 psia. CETCs are 531° F. and increasing.

c. A load reject/reactor trip resulted in a stuck open PORV. Section 501, Loss of Subcooling, is in use. RCS pressure is 1700 psia. CETCs are 670° F.

d. A loss of main and auxiliary feedwater has occurred. Section 502, Loss of Heat Transfer, is in use. RCS pressure is 2350 psia. CETCs are 640° F.

QUESTION: 074 (1.00)

Which one of the following describes the bases for leaving RCPs running if they are operating when Section 507, Inadequate Core Cooling, is entered ?

a. Forced steam cooling will provide adequate core cooling if a level exists in a S/G.

b. Core will become uncovered if RCPs are secured at this point.

c. RCP discharge pressure will minimize voiding of the core and increase heat transfer.

d. Water trapped in the loops and lower region of the vessel will be used for core cooling.

QUESTION: 075 (1.00)

Which one of the following control room indications is evidence that core damage has occurred following a LOCA ?

a. Incore thermocouples read approximately 620° F.

b. Self Powered Neutron Detector Chart recorders are spiking full scale.

c. SRM Count rate decreases at -1/3 DPM for about 20 minutes then levels out and remains constant.

d. Key Nuclide Ratios from RCS sample are slightly higher than normal.

QUESTION: 076 (1.00)

Which one of the following will be the first to terminate the effects of a rod ejection accident from rated power? Assume highest worth rod ejected.

a. Doppler Coefficient of reactivity.

b. RPS High pressure trip.

c. Integrated Control System.

d. RPS High Flux trip.

QUESTION: 077 (1.00)

Which one of the following is a likely pathway for release of fission products from the Reactor Building to the Aux Building following an accident that results in fuel damage? (Assume SG tube leakage within TS limits)

a. LDST relief valve leakage.

b. PORV

c. Main Steam Isolation Valve leakage

d. RCP Seal leakoff

QUESTION: 078 (1.00)

Which one of the following nuclides would be the most significant contributor to offsite dose consequences following an accident that resulted in core damage ?

a. Xenon 131

b. Cesium 136

c. Argon 41

d. Cobalt 51

QUESTION: 079 (1.00)

Which one of the following Radiation Monitors is potentially unreliable under accident conditions?

a. RIA - 4, Reactor Building Hatch Area Radiation Monitor

b. RIA- 56, High Range Stack Radiation Monitor

c. RIA- 57, High Range Containment Radiation Monitor

d. RIA- 58, High Range Containment Radiation Monitor

QUESTION: 080 (1.00)

Fuel movement is in progress. RIA-6, SFP Area Monitor, just failed. Which one of the following actions is required?

a. Verify RIA-3, Refueling Canal Wall Area Monitor, is functioning properly.

b. Direct RP to set up a portable area monitor with local alarm capability.

c. Continue fuel movement, no action required.

d. Place a portable radiation detection instrument on the affected bridge.

QUESTION: 081 (1.00)

HP-120, pressurizer level controller, has failed. Pressurizer level is being maintained MANUALLY. Assume no operator action. Which one of the following describes the response of pressurizer level for raising Reactor power from 25% to 50%?

a. Pressurizer level will decrease and stabilize at a lower level.

b. Pressurizer level will increase and stabilize at a higher level.

c. Pressurizer level will initially decrease and then return to the original level.

d. Pressurizer level will initially increase and then return to the original level.

QUESTION: 082 (1.00)

Given the following conditions:

- Primary IA compressor:	Running
- Backup IA compressors "A" and "C":	Standby 1
- Backup IA compressor "B":	Standby 2
- Auxiliary IA compressor:	Auto
- IA-2718 (Air Supply to Radwaste Facility):	Open

A large break has occurred in the Instrument Air (IA) system. IA pressure has decreased to 92 psig. Which one of the following is the expected response of the IA system?

a. Backup IA compressors "A" and "C" start.

b. All Backup IA compressors start.

c. All Backup IA compressors start, IA-2718, Air Supply to Radwaste Facility, shuts.

d. All Backup IA compressors start, Auxiliary IA compressor starts.

QUESTION: 083 (1.00)

Given the following conditions for Unit 1:

- Reactor is being refueled.
- Keowee 1 is out of service for maintenance.

A loss of Off-site power (Unit 1) occurs and Keowee 2 failed to start. Which one of the following is the correct section of AP/1700/11, Loss of Power, to be INITIALLY used by the operator from Step 5.0 Subsequent Actions ?

a. Section 501, Power From Standby Bus

b. Section 502, Manually energizing MFB's

c. Section 503, Unit Status Assessment

d. Section 504, Blackout

QUESTION: 084 (1.00)

Which one of the following describes the reason for maintaining the tube to shell ΔT of the isolated SG within limits while performing a cooldown in accordance with section 504, SG Tube Leak?

a. Maintains the isolated SG within NDT limits.

b. Minimizes metal heating of the RCS by the isolated SG.

c. Prevents thermal shock of tubes when feeding with EFW.

d. Minimizes the tensile stress developed across the tubes.

QUESTION: 085 (1.00)

Which one of the following describes the reason for minimizing Subcooling Margin while performing a cooldown with a ruptured SG per Section 504, SG Tube Leak?

a. Promotes reflux boiling/heat removal during natural circulation.

b. Minimizes leakage of RCS water into the affected SG.

c. Ensures sufficient NPSH is available to operate RCPs during cooldown.

d. Maximizes heat transfer between the RCS and SGs to expedite cooldown.

QUESTION: 086 (1.00)

Which one of the following describes the common link for transients resulting in the use of CP-602, "SG Cooldown with Saturated RCS" ?

a. Only one SG is available to remove decay heat.

b. Excessive Heat Transfer.

c. HPI Cooling is required to remove decay heat.

d. SCM is 0° F and SG heat transfer is necessary to depressurize the RCS.

QUESTION: 087 (1.00)

Unit 3 is at 100 % power when a small break in the STEAM SPACE of the pressurizer occurs. No operator actions are taken and HPI stabilizes RCS pressure slightly higher than secondary system pressure. Which one of the following describes the response of the pressurizer level?

a. PZR level initially increases rapidly, then slowly increases until the PZR is completely filled.

b. PZR level initially increases then decreases when a reactor trip occurs, after the RCS hot leg flashes the PZR fills completely.

c. PZR initially decreases then drops off scale low during depressurization until HPI initiates, then returns on scale during repressurization with HPI.

d. PZR level initially decreases slowly, then decreases rapidly when a trip occurs; then returns on scale during repressurization with HPI.

QUESTION: 088 (1.00)

Which one of the following is the reason for maintaining RCS pressure at 2155 psig until T-hot is less than 450° F. during a natural circulation cooldown using CP-605, Subcooled Cooldown?

a. Prevent the stagnant area in the RV head from flashing.

b. Provide positive pressure control of the pressurizer steam space.

c. Prevent shifting of the steam bubble from the PZR to the Hot Legs.

d. Prevent void formation in the non-isolated RCS loops.

QUESTION: 089 (1.00)

Which one of the following is the purpose for maintaining pressurizer level above 80 inches during the RCS cooldown from 579° to 532° F. following a SG Tube Leak?

a. Prevent loss of subcooling and the necessity of tripping of RCPs.

b. Maintain sufficient surge volume in the event of a reactor trip.

c. Ensure level indication is maintained.

d. Ensure that a steam void does not form in the reactor vessel.

QUESTION: 090 (1.00)

While operating in CP-602, "SG Cooldown With Saturated RCS", HPI piggyback alignment is started when BWST level indicates 10 feet.

Which one of the following describes the reason for commencing alignment at this time?

a. BWST level instrument errors can result in suction vortexing before indicated level reaches 6 feet.

b. Waiting longer would result in radiation fields in LPI Pump rooms that prohibit local operation of the required valves.

c. Sufficient level still remains to ensure adequate suction if manual alignment of the necessary valves is required.

d. Adequate NPSH for piggyback operation CANNOT be assured below 10 feet in the BWST.

QUESTION: 091 (1.00)

Following a loss of Main Feedwater (MFDW), OTSG "B" has boiled DRY. Both RCPs in Loop "B" have tripped and cannot be started.

Which one of the following describes the limitations on feeding OTSG "B" with EFDW and/or MFDW?

a. EFDW or MFDW must be fed slowly to the OTSG to prevent a RCS overcooling and to limit unanalyzed OTSG stresses.

b. EFDW or MFDW must be fed via the Aux nozzles to prevent excessive stresses on the lower tube sheet.

c. Neither EFDW nor MFDW may be fed to the OTSG due to inducing unanalyzed OTSG tube-to-shell differential temperatures.

d. Neither EFDW nor MFDW may be fed to the OTSG due to inducing excessive RCS loop differential temperatures.

QUESTION: 092 (1.00)

AP/1/A/1700/21, "High Activity in RC System", requires determining the source of high activity (failed fuel or corrosion products). Which one of the following is the reason for this requirement?

a. Extended cleanup actions must commence within one hour, if the activity is due to a crud burst.

b. The reactor must be shutdown within 24 hours, if there is any confirmed indication of failed fuel.

c. While operating with failed fuel, normal power level changes must be limited to less than 3% FP/hr.

d. If activity due to nuclides with half-lives greater than 30 minutes approaches 165/E-bar and the reactor is critical, the reactor must be shutdown.

QUESTION: 093 (1.00)

According to AP/1/A/1700/11, "Loss of Power (Blackout)", within how many minutes of the event must RCS makeup and Primary to Secondary heat transfer be regained to prevent core damage?

a. 10

b. 20

c. 30

d. 40

QUESTION: 094 (1.00)

Which one of the following correctly describes the Standby Shutdown Facility RC Makeup Pump?

a. A centrifugal pump designed to maintain flow to the RCP seals during the approach to Cold Shutdown.

b. A positive displacement pump designed to maintain flow to the RCP seals during the approach to Cold Shutdown.

c. A centrifugal pump designed to recover the decrease in the RCS water volume due to shrinkage during the approach to Hot Shutdown.

d. A positive displacement pump designed to recover the decrease in the RCS water volume due to shrinkage during the approach to Hot Shutdown.

QUESTION: 095 (1.00)

Which one of the following is the minimum amount of time each of the 125V DC batteries (DCSF, DCSFS) is designed to supply the required SSF and security loads?

a. One hour.

b. Two hours.

c. Four hours.

d. Eight hours.

QUESTION: 096 (1.00)

Which one of the following is the MINIMUM RCP shaft vibration level in which the Alarm Response Procedure requires the affected RCP to be tripped? Assume four RCPs are in operation.

a. 3 mils

b. 15 mils

c. 20 mils

d. 35 mils

QUESTION: 097 (1.00)

Which one of the following is true concerning the ASW pump controls?

a. The pump can be stopped locally if it was started from the SSF Control Room panel.

b. The pump can be stopped at the SSF Control Room panel if it was started locally.

c. The pump can be started from the SSF Control Room panel if an ESG signal is present and OTS1-4, D/G breaker, is open.

d. The pump can be started from the SSF Control Room panel if an ESG signal is present and OTS1-1, 4KV switchgear feeder breaker, is closed.

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

ANSWER: 001 (1.00)

b.

REFERENCE:

AP/1/A/1700/20, LOSS OF COMPONENT COOLING, p. 3
OP-OC-PNS-CC, pg 13; Obj. R6
JAN 92 Exam
[3.9/4.1]

000026G011 ..(KA's)

ANSWER: 002 (1.00)

b.

REFERENCE:

ATTACHED FIGURE REQUIRED: Fig. OP-OC-EAP-SSF
OP-OC-IC-ASP; Obj. R2
JAN 92 Exam
[4.1/4.3]

000068A207 ..(KA's)

ANSWER: 003 (1.00)

c.

REFERENCE:

Oconee Lesson Plan OP-OC-PNS-BS, pg 7, 13; Obj. 1
JAN 92 Exam
[3.1/3.6]
026000K402 ..(KA's)

ANSWER: 004 (2.00)

a. 3
b. 8
c. 9
d. 6 (0.5 each)

REFERENCE:

AP/1/1/1700/23, LOSS OF 1KI BUS, pg 1
OP-OC-EL-VPC; Obj. R13 and R16
JAN 92 Exam
[4.0/4.3]

000057A219 ..(KA's)

ANSWER: 005 (1.00)

b.

REFERENCE:

OP-OC-PNS-HPI, pg 35, Obj. R25
JAN 1992 exam
[3.2/3.7]

000022G007 ..(KA's)

ANSWER: 006 (1.00)

d.

REFERENCE:

OP-OC-EL-EPD, Para J.4, pg 80, EO 22 (R22)
JAN 92 Exam
[4.3/4.5]
000055A107 ..(KA's)

ANSWER: 007 (1.00)

b.

REFERENCE:

AP/1/A/1700/23, Section 5.4, pg 3
OP-OC-IC-ASP, p.15
JAN 92 Exam
[3.5/3.5]

000057A106 ..(KA's)

ANSWER: 008 (1.00)

a.

REFERENCE:

Site Directive 4.1.6 rev 3/30/93, pg 10
[3.6/3.7]
194001K101 ..(KA's)

ANSWER: 009 (1.00)

d.

REFERENCE:

Station Directive 4.1.6 rev 3/30/93, pg 6
[3.6/3.7]
194001K101 ..(KA's)

ANSWER: 010 (1.00)

c.

REFERENCE:

Radiation Protection Directive III-1 Rev 3, p. 2
OP-OC-RAD-RPP, Rev. 1, p. 65
weakness in ER 93-300
[2.8/3.4]
194001K103 ..(KA's)

ANSWER: 011 (1.00)

b.

REFERENCE:

OP-OC-RAD-RPP, Rev 1, p. 23
[2.8/3.4]
194001K103 ..(KA's)

ANSWER: 012 (1.00)

a.

REFERENCE:

OP-OC-RAD-RPP, Rev 1, p. 29
[2.8/3.4]
194001K103 ..(KA's)

ANSWER: 013 (1.00)

c.

REFERENCE:

OP-OC-RAD-RPP, Rev 1, p. 39
[2.8/3.4]
194001K103 ..(KA's)

ANSWER: 014 (1.00)

c.

REFERENCE:

OP-OC-CP-AFF, Rev 1, p. 25; Enabling objective 22
1993 SRO #84
[3.6/3.7]
194001K107 ..(KA's)

ANSWER: 015 (1.00)

c.

REFERENCE:

Site Directive 3.2.3, paragraph 4.3.2, page 8
[3.3/3.6]
194001K114 ..(KA's)

ANSWER: 016 (1.00)

a.

REFERENCE:

Oconee Lesson Plan OP-OC-EAP-E11, pg 12; Obj. 7, R11
[4.1/3.9]
194001A102 ..(KA's)

ANSWER: 017 (1.00)

b.

REFERENCE:

Oconee Lesson Plan OP-OC-EAP-E11, Rev. 5, p. 17; Obj. R9
modified JAN 1992 (HNUM 33917)
[4.1/3.9]
194001A102 ..(KA's)

ANSWER: 018 (1.00)

b.

REFERENCE:

OMP 1-12, Rev 7, Section 3, pg 1
1992 & 1993 #99 modified HNUM 35650
[2.5/3.4]
194001A103 ..(KA's)

ANSWER: 019 (1.00)

c.

REFERENCE:

OMP 2-1 Rev 10/15/93, Encl. 4.5
EAP-E11, Obj. R7
[2.7/3.9]
194001A109 ..(KA's)

ANSWER: 020 (1.00)

c.

REFERENCE:

Oconee Lesson Plan OP-OC-IC-ES, Rev 7, p. 17; Obj. R13
1992 (HNUM 33919), OMP 2.1, 4.11
[2.8/4.1]
194001A111 ..(KA's)

ANSWER: 021 (1.00)

a.

REFERENCE:

RB/O/B/1000/11, Sec 2.2
1993 SRO #88 slightly modified
[3.1/4.1]
194001A112 ..(KA's)

ANSWER: 022 (1.00)

c.

REFERENCE:

OP-OC-CH-SC, Rev 2, p. 12; LRO 8,
[2.5/2.9]
194001A114 ..(KA's)

ANSWER: 023 (1.00)

b.

REFERENCE:

RP/0/B/1000/02 p. 2
ONS Emergency Plan Rev 9/19/91, pg H-2
[3.1/4.4]
194001A116 ..(KA's)

ANSWER: 024 (1.00)

b.

REFERENCE:

RP/0/B/1000/15, Sec 3.3
EAP-SEP, Obj. R5
ONS Emergency Plan Rev 9/19/91, pg H-2
[3.1/4.4]
194001A116 ..(KA's)

ANSWER: 025 (1.00)

b.

REFERENCE:

OP-OC-PNS-HPI, Section 2.5.K, pg 37, R26 and OP/1104/02, Limit and Precaution 2.5, OP-OC-CP-017, Section 2.3.B.2, p. 16, R8
JAN 92 EXAM
[3.2/3.2]

011000A202 ..(KA's)

ANSWER: 026 (1.00)

~~b.~~ A. 3-24-94 ME

REFERENCE:

Oconee Exam Bank IC-59
OP-OC-IC-CRI, pg 19
[3.6/4.1]

001000G007 ..(KA's)

ANSWER: 027 (1.00)

b.

REFERENCE:

OP-OC-CF-FDW, Fig. FDW-1, Para 5, pg 21, Para 3, pg 20, EO 12 (R11)
92 & 93 EXAMS
[3.1/3.2]

059000K105 ..(KA's)

ANSWER: 028 (1.00)

b.

REFERENCE:

Oconee Bank # RT-74 modified
OP-OC-RT-RBC Obj. B.3c.
PT/1/A/1103/15, "Reactivity Balance Procedure-Unit 1".
Duke Power Co. Fundamentals of Nuclear Reactor Engineering p. 297
[3.2/3.6]

001010K518 ..(KA's)

ANSWER: 029 (1.00)

d.

REFERENCE:

OP-OC-PNS-HPI, p. 26, para. 2.1.D.2.e, EO B.15
DWG NO. OSFD-101A-1
[3.7/4.2]

004000K509 ..(KA's)

ANSWER: 030 (1.00)

a.

REFERENCE:

OP-OC-PNS-HPI, EO B.12 pp. 24-25,sec. 2.1.C; pp. 26-27, sec.
2.1.D.3
[2.8/3.2]

004020K402 ..(KA's)

ANSWER: 031 (1.00)

d.

REFERENCE:

OP-OC-IC-ES, p. 18, para. 2.2.B.2, EO R7
[3.6/3.8]

013000K201 ..(KA's)

ANSWER: 032 (1.00)

d.

REFERENCE:

OC-OP-IC-NI, p. 19, para. 2.2.A.13.d, EO B.11
[3.1/3.2]

015000K604 ..(KA's)

ANSWER: 033 (1.00)

b.

REFERENCE:

OP-OC-PNS-RBC, pp. 16-17, EO B.6, R12
[3.1/3.5]

022000A205 ..(KA's)

ANSWER: 034 (1.00)

c.

REFERENCE:

OP-OC-CF-C, p. 24, para. 2.3.B.13, EO B.5
[2.6/2.8]

056000G001 ..(KA's)

ANSWER: 035 (1.00)

c.

REFERENCE:

STG-ICS, pp. 56-60, para. 2.6.B, pp. 83-85, para. 2.10, EO B.65.
OP/1/A/1102/04, pp. 6.
[3.2/3.2]

059000K107 ..(KA's)

ANSWER: 036 (1.00)

a.

REFERENCE:

OP-OC-CF-EF, pg 48, para. 2.4.C.2.c), EO B.27
JAN 92 exam modified
[4.2/4.2]

061000A301 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

OP-OC-CF-EF, p. 45, para. 2.4.A.2, EO B.22.
JAN 92 Exam modified
[4.0/4.2]

061000K406 ..(KA's)

ANSWER: 038 (1.00)

c.

REFERENCE:

OP-OC-CF-EF, p. 41 & 58, paras. 2.3.B.6.c & 2.5.F.1.c, EO R26
[3.5/3.7]
063000K302 ..(KA's)

ANSWER: 039 (1.00)

a.

REFERENCE:

OP-OC-RAD-RIA Obj. R2
JAN 92 Exam
[3.8/3.7]
068000A404 ..(KA's)

ANSWER: 040 (1.00)

d.

REFERENCE:

OP-OC-IC-RCI, p. 50, para. 2.2.C.11, EO B.35, B.36.
[3.4/3.7]
017020K401 ..(KA's)

ANSWER: 041 (1.00)

d. DELETE 3-24-94 ME

REFERENCE:

OP-OC-RAD-RIA, pp. 29-32, para. 2.5.B.7, EO B.6.
[3.0/3.3]
072000A401 ..(KA's)

ANSWER: 042 (1.00)

b.

REFERENCE:

OP-OC-WE-GWD, p. 18, para. 2.3.B; Figs. 2, 3, & 4; EO B.4
[3.1/3.1]

071000K106 ..(KA's)

ANSWER: 043 (1.00)

d.

REFERENCE:

OP-OC-PNS-LPI, pp. 36-37, para. 2.6.C, EO B.22
[3.1/3.5]
005000K408 ..(KA's)

ANSWER: 044 (1.00)

d.

REFERENCE:

OP-OC-PNS-CF, p. 22, para. 2.4.A, EO B.17.c
[3.5/3.7]
006020A107 ..(KA's)

ANSWER: 045 (1.00)

a.

REFERENCE:

OP-OC-PNS-CF, p. 18, para. 2.2.E.4, EO B.9
[3.4/3.7]

006000G010 ..(KA's)

ANSWER: 046 (1.00)

a.

REFERENCE:

OP-OC-IC-CRI, p. 45, para. 2.6.D, EO B.31
[3.4/4.3]

012000G005 ..(KA's)

ANSWER: 047 (2.00)

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

REFERENCE:

OP-OC-PNS-RCS, pp. 12-13, para. 2.4, EO B.7
[4.5/4.6]

002000K108 ..(KA's)

ANSWER: 048 (1.00)

c.

REFERENCE:

OP-OC-SPS-CM-PZR, p. 14, para. , EO B.
[4.0/3.8]

010000A403 ..(KA's)

ANSWER: 049 (1.00)

a.

REFERENCE:

OP-OC-CF-EF, pp. 32-33, para.2.2.B, EO B.1
B&W Abnormal Transient Operator Guidelines Tech. Basis Document.
[4.2/4.5]

035010K101 ..(KA's)

ANSWER: 050 (2.00)

a. 3 (0.5 pts each)
b. 3
c. 3
d. 1

REFERENCE:

OP-OC-IC-RCI, pp. 38-39, para. , EO R27
OP-OC-IC-ASDP, p.13
[3.7/3.8]
011000K102 ..(KA's)

ANSWER: 051 (1.00)

c.

REFERENCE:

Oconee Lesson Plan OP-OC-PNS-BS, p. 12; Obj. R5
[3.3/3.5]
026000G010 ..(KA's)

ANSWER: 052 (1.00)

c.

REFERENCE:

OP-OC-PNS-CC, Section 2.4, p. 12; Obj. 2
[3.2/3.0]
008000A301 ..(KA's)

ANSWER: 053 (1.00)

b.

REFERENCE:

OP-OC-EL-KHG, pp. 17-18 & 37, paras. 1.3, 1.4, 2.3.E, EO B.4.d
[3.8/4.0]

064000G015 ..(KA's)

ANSWER: 054 (1.00)

c.

REFERENCE:

OP-OC-EL-KHG, p. 45, para. 2.5.B.2.b, EO B.9
[3.9/4.2]
064000K402 ..(KA's)

ANSWER: 055 (1.00)

a.

REFERENCE:

OP-OC-STG-CCW, pp. 33-34, para. 2.4.K.4, EO B.24
NRC INSPECTION REPORT NO. 50-269/93-25,
[3.5/3.5]
076000A201 ..(KA's)

ANSWER: 056 (1.00)

b.

REFERENCE:

SLC 16.9; OC Nuclear Site Directive 3.2.9, p. 6, para. 5.5.1.1
[3.0/3.6]
086000G005 ..(KA's)

ANSWER: 057 (1.00)

b.

REFERENCE:

OP-OC-EL-VPC, p. 13, para. 2.1.4.h, EO B.3
[3.1/3.1]

062000A407 ..(KA's)

ANSWER: 058 (1.00)

d.

REFERENCE:

OP-OC-EL-EPD, pp. 81-82, EO B.25
[3.7/4.1]
062000A211 ..(KA's)

ANSWER: 059 (1.00)

b.

REFERENCE:

OP-OC-PNS-LPI, p. 23, para. 2.1.G.4, EO B.3
[2.9/3.2]
005000K403 ..(KA's)

ANSWER: 060 (1.00)

d.

REFERENCE:

OP-OC-RAD-RIA, p. 18-19, para. 2.2.K.5, EO B.2.c
[4.0/4.3]
073000K401 ..(KA's)

ANSWER: 061 (1.00)

a.

REFERENCE:

OP-OC-FH-FHS, p.14, EO B.10
(3.1/3.3)
033000K405 ..(KA's)

ANSWER: 062 (1.00)

b.

REFERENCE:

AP/1/A/1700/17, p. 4, Sec. 5.0 NOTES
(3.7/4.3)

000069A201 ..(KA's)

ANSWER: 063 (1.00)

d.

REFERENCE:

AP/1/A/1700/17, p. 4, Sec. 5.0 NOTES
[3.9/4.4]

000069A202 ..(KA's)

ANSWER: 064 (1.00)

b.

REFERENCE:

OP/0/A/1105/09, p. 3, Precaution 2.10
[3.9/4.2]

000005K306 ..(KA's)

ANSWER: 065 (1.00)

a.

REFERENCE:

AP/1/A/1700/21, p.5
[2.9/3.6]
000076K305 ..(KA's)

ANSWER: 066 (1.00)

d.

REFERENCE:

OP-OC-PNS-PZR Obj 17
[4.0/3.9]
000027A101 ..(KA's)

ANSWER: 067 (1.00)

d.

REFERENCE:

OP-OC-EAP-EP23 Obj B2
P&I 9/16/93
[4.5/4.7]
000040K304 ..(KA's)

ANSWER: 068 (1.00)

c.

REFERENCE:

OP-OC-EAP-EP23 Obj. 3, 6 pg 15
[4.1/4.3]

000040G011 ..(KA's)

ANSWER: 069 (1.00)

b.

REFERENCE:

OC-OP-EAP-E22 Obj 14 pg 21
[3.2/3.2]

000054G012 ..(KA's)

ANSWER: 070 (1.00)

c.

REFERENCE:

OP-OC-EAP-E22 Obj 11 pg 18
Enclosure 7.6 of EOPs required in handout.
[4.4/4.6]

000054K304 ..(KA's)

ANSWER: 071 (1.00)

d.

REFERENCE:

OP-OC-EAP-APG Obj 2
AP/1/A/1700/12
[3.5/3.6]

000015G011 ..(KA's)

ANSWER: 072 (1.00)

c.

REFERENCE:

OP-OC-EAP-E24 Obj 10
[4.2/4.6]

000038A216 ..(KA's)

ANSWER: 073 (1.00)

c.

REFERENCE:

OP-OC-EAP-E27 Obj. 2 pg 6
[4.5/4.6]

000074G011 ..(KA's)

ANSWER: 074 (1.00)

a.

REFERENCE:

OP-OC-EAP-E27 Obj. 3 pg 6
[4.0/4.4]

000074K311 ..(KA's)

ANSWER: 075 (1.00)

b.

REFERENCE:

OP-OC-TA-AM3 Obj 2
[4.1/4.7]

000074A207 ..(KA's)

ANSWER: 076 (1.00)

d.

REFERENCE:

OP-OC-TA-SA, p. 42, Obj 4.1
[3.4/3.7]
000001K101 ..(KA's)

ANSWER: 077 (1.00)

a.

REFERENCE:

OP-OC-TA-AM7 Obj. 10, 15 PG 16
[3.1/4.0]
000060A202 ..(KA's)

ANSWER: 078 (1.00)

a.

REFERENCE:

OP-OC-TA-AM7 Obj. 6 pg 12
[2.4/3.1]

000076K301 ..(KA's)

ANSWER: 079 (1.00)

a.

REFERENCE:

OP-OC-TA-AM6 Obj. 2
[3.1/3.5]

000061A204 ..(KA's)

ANSWER: 080 (1.00)

b.

REFERENCE:

Oconee Lesson Plan OP-OC-FH-FHS Section 2.4, LO 12
[3.4/3.6]

000036K202 ..(KA's)

ANSWER: 081 (1.00)

a.

REFERENCE:

Oconee Lesson Plan OP-OC-PNS-PZR Section 2.0
[2.9/3.0]

000028K304 ..(KA's)

ANSWER: 082 (1.00)

a.

REFERENCE:

Oconee Lesson Plan OP-OC-SSS-IA Section 2.1.B, Obj. R8
[3.5/3.4]

000065A104 ..(KA's)

ANSWER: 083 (1.00)

b.

REFERENCE:

AP/1A/1700/11
Oconee Question Bank #298
[3.5/3.6]

000056G012 ..(KA's)

ANSWER: 084 (1.00)

d.

REFERENCE:

OP-OC-EAP-E24 Obj. 9 pg 19
[4.2/4.4]
000037K307 ..(KA's)

ANSWER: 085 (1.00)

b.

REFERENCE:

OP-OC-EAP-E24 Obj 6 pg 13
[3.6/4.1]
000037K306 ..(KA's)

ANSWER: 086 (1.00)

d.

REFERENCE:

OP-OC-EAP-E32 Obj 1
Bank EAP #1
[4.2/4.4]
000009G011 ..(KA's)

ANSWER: 087 (1.00)

b.

REFERENCE:

OP-OC-TA-AT Obj 1 pg 14
Bank TA # 40
[3.7/4.4]

000008K301 ..(KA's)

ANSWER: 088 (1.00)

a.

REFERENCE:

OP-OC-EAP-E35 Obj 3 pg 13
Facility Question Bank EAP #35
[4.2/4.5]

000038K306 ..(KA's)

ANSWER: 089 (1.00)

a.

REFERENCE:

OP-OC-EAP-E24 Obj 4
[4.1/4.3]

000037K308 ..(KA's)

ANSWER: 090 (1.00)

c.

REFERENCE:

OP-OC-EAP-E32 Obj 20
Facility Question Bank EAP #4
[4.2/4.5]

000009K321 ..(KA's)

ANSWER: 091 (1.00)

b.

REFERENCE:

OP-OC-CF-FDW Obj 12 pg 22
Facility Question Bank EAP #16
[4.4/4.6]
000054K304 ..(KA's)

ANSWER: 092 (1.00)

c.

REFERENCE:

AP/1/A/1700/21, p. 2-6, Sec. 4.0 & 5.0 & Encl. 6.1
[3.2/3.8]

000076K306 ..(KA's)

ANSWER: 093 (1.00)

d.

REFERENCE:

AP/1/A/1700/11, p. 10, Sec. 502, CAUTION
[3.6/3.7]

000055G007 ..(KA's)

ANSWER: 094 (1.00)

d.

REFERENCE:

OP-OC-EAP-SSF, p. 52-53, EO B.35
[3.9/4.4]

000068K309 ..(KA's)

ANSWER: 095 (1.00)

a.

REFERENCE:

OP-OC-EAP-SSF, p. 48, para. 2.2.C.3.b.3).c),
[2.7/3.4]
000055K301 ..(KA's)

ANSWER: 096 (1.00)

c.

REFERENCE:

ARP 1SA-9/D-2
AP/1/A/1700/16, p. 19, CASE "E"
[3.7/4.0]

000015K303 ..(KA's)

ANSWER: 097 (1.00)

b.

REFERENCE:

OP-OC-EAP-SSF, p. 64, EO B.37
[4.3/4.5]
000068A102 ..(KA's)

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

TEST CROSS REFERENCE

SRO Exam PWR Reactor
Organized by Question Number

QUESTION	VALUE	REFERENCE
001	1.00	33869
002	1.00	33871
003	1.00	33882
004	2.00	33908
005	1.00	33958
006	1.00	35584
007	1.00	35653
008	1.00	9001101
009	1.00	9001102
010	1.00	9001105
011	1.00	9001106
012	1.00	9001107
013	1.00	9001108
014	1.00	9001111
015	1.00	9001112
016	1.00	9001113
017	1.00	9001114
018	1.00	9001115
019	1.00	9001117
020	1.00	9001118
021	1.00	9001119
022	1.00	9001120
023	1.00	9001121
024	1.00	9001122
025	1.00	9001123
026	1.00	9001127
027	1.00	9001129
028	1.00	9001130
029	1.00	9001131
030	1.00	9001132
031	1.00	9001134
032	1.00	9001135
033	1.00	9001139
034	1.00	9001140
035	1.00	9001142
036	1.00	9001143
037	1.00	9001144
038	1.00	9001145
039	1.00	9001146
040	1.00	9001147
~~041~~	~~1.00~~	~~9001148~~ DELETED
042	1.00	9001149
043	1.00	9001150
044	1.00	9001151
045	1.00	9001152
046	1.00	9001154
047	2.00	9001155

048	1.00	9001156
049	1.00	9001157
050	2.00	9001159
051	1.00	9001161
052	1.00	9001163
053	1.00	9001165
054	1.00	9001166
055	1.00	9001168
056	1.00	9001169
057	1.00	9001170
058	1.00	9001171
059	1.00	9001172
060	1.00	9001173
061	1.00	9001175
062	1.00	9001176
063	1.00	9001177
064	1.00	9001179
065	1.00	9001181
066	1.00	9001183
067	1.00	9001184
068	1.00	9001185
069	1.00	9001188
070	1.00	9001189
071	1.00	9001190
072	1.00	9001191
073	1.00	9001198
074	1.00	9001199
075	1.00	9001200
076	1.00	9001201
077	1.00	9001202
078	1.00	9001203
079	1.00	9001204
080	1.00	9001205
081	1.00	9001206
082	1.00	9001207
083	1.00	9001208
084	1.00	9001209
085	1.00	9001210
086	1.00	9001211
087	1.00	9001212
088	1.00	9001214
089	1.00	9001215
090	1.00	9001216
091	1.00	9001217
092	1.00	9001218
093	1.00	9001219
094	1.00	9001220
095	1.00	9001221
096	1.00	9001222
097	1.00	9001223

	100.00	

SRO Exam PWR Reactor
Organized by KA Group

PLANT WIDE GENERICS

017	1.00	194001A102
016	1.00	194001A102
018	1.00	194001A103
019	1.00	194001A109
020	1.00	194001A111
021	1.00	194001A112
022	1.00	194001A114
023	1.00	194001A116
024	1.00	194001A116
009	1.00	194001K101
008	1.00	194001K101
012	1.00	194001K103
011	1.00	194001K103
010	1.00	194001K103
013	1.00	194001K103
014	1.00	194001K107
015	1.00	194001K114

PWG Total	17.00	

PLANT SYSTEMS

Group I

026	1.00	001000G007
028	1.00	001010K518
029	1.00	004000K509
030	1.00	004020K402
031	1.00	013000K201
032	1.00	015000K604
040	1.00	017020K401
033	1.00	022000A205
051	1.00	026000G010
003	1.00	026000K402
034	1.00	056000G001
027	1.00	059000K105
035	1.00	059000K107
036	1.00	061000A301
037	1.00	061000K406
038	1.00	063000K302
039	1.00	068000A404
042	1.00	071000K106
~~041~~	~~1.00~~	~~072000A401~~ DELETED

PS-I Total	~~19.00~~ 18.00	

Group II

047	2.00	002000K108
045	1.00	006000G010
044	1.00	006020A107
048	1.00	010000A403
025	1.00	011000A202
050	2.00	011000K102
046	1.00	012000G005
061	1.00	033000K405
049	1.00	035010K101
058	1.00	062000A211
057	1.00	062000A407
053	1.00	064000G015
054	1.00	064000K402
060	1.00	073000K401
056	1.00	086000G005

PS-II Total	17.00	

Group III

059	1.00	005000K403
043	1.00	005000K408
052	1.00	008000A301
055	1.00	076000A201

PS-III Total	4.00	

PS Total	~~40.00~~ 39.00	

EMERGENCY PLANT EVOLUTIONS

Group I

076	1.00	000001K101
064	1.00	000005K306
071	1.00	000015G011
096	1.00	000015K303
001	1.00	000026G011
068	1.00	000040G011
067	1.00	000040K304
006	1.00	000055A107
093	1.00	000055G007
095	1.00	000055K301
007	1.00	000057A106
004	2.00	000057A219
097	1.00	000068A102
002	1.00	000068A207
094	1.00	000068K309
062	1.00	000069A201
063	1.00	000069A202
075	1.00	000074A207
073	1.00	000074G011
074	1.00	000074K311
078	1.00	000076K301
065	1.00	000076K305
092	1.00	000076K306

EPE-I Total	24.00	

Group II

087	1.00	000008K301
086	1.00	000009G011
090	1.00	000009K321
005	1.00	000022G007
066	1.00	000027A101
085	1.00	000037K306
084	1.00	000037K307
089	1.00	000037K308
072	1.00	000038A216
088	1.00	000038K306
069	1.00	000054G012
091	1.00	000054K304
070	1.00	000054K304
077	1.00	000060A202
079	1.00	000061A204
082	1.00	000065A104

EPE-II Total	16.00	

Group III			
	081	1.00	000028K304
	080	1.00	000036K202
	083	1.00	000056G012

EPE-III Total		3.00	

EPE Total		43.00	

SRO Test Total		~~100.00~~ 94.00	