U.S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING REQUALIFICATION PROGRAM EVALUATION REPORT

REQUALIFICATION PROGRAM EVALUATION REPORT NO. 50-213/90-10(OL-RQ)

FACILITY DOCKET NO.: 50-213

FACILITY LICEN[illegible]: DPR-61

LICENSEE: Connecticut Yankee Atomic Power Company
P. O. Box 270
Hartford, Connecticut 06141

FACILITY: Haddam Neck Power Plant

EXAMINATION DATES: May 24, 1990 (Replacement retakes)
June 5-8, 1990 (Requalification examination)

CHIEF EXAMINER: Paul Bissett, Senior Operations Engineer — 7/22/90 Date

APPROVED BY: for Peter Eselgroth, Chief, PWR Section Chief
Operations Branch, Division of Reactor Safety — 7/24/90 Date

SUMMARY: The licensed operator requalification training program was rated as satisfactory. Written requalification examinations and operating tests were administered to eight senior reactor operators (SROs) and four reactor operators (ROs). The examinations were graded concurrently and independently by the NRC and the facility training staff. As graded by the NRC, all SROs and ROs passed all portions of the examination. Facility grading paralleled that of the NRC in all aspects of the examination.

Also, an SRO simulator and an SRO written replacement retake examination was administered during the requalification examination preparation week. Both individuals successfully passed their respective segment of the administered examination.

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DETAILS

TYPE OF EXAMINATIONS: Requalification

EXAMINATION RESULTS:

NRC Grading	RO Pass/Fail	SRO Pass/Fail	TOTAL Pass/Fail
Written	4/0	8/0	12/0
Simulator	4/0	8/0	12/0
Walk-through	4/0	8/0	12/0
Overall	4/0	8/0	12/0

Facility Grading	RO Pass/Fail	SRO Pass/Fail	TOTAL Pass/Fail
Written	4/0	8/0	12/0
Simulator	4/0	8/0	12/0
Walk-through	4/0	8/0	12/0
Overall	4/0	8/0	12/0

TYPE OF EXAMINATIONS: Replacement

EXAMINATION RESULTS:

	SRO Pass/Fail	TOTAL Pass/Fail
Written	1/0	1/0
Simulator	1/0	1/0
Overall	2/0	2/0

1.0 <u>PERSONNEL CONTACTED DURING THE EXAMINATION/EVALUATION</u>

<u>CHIEF EXAMINER AT SITE</u>:

P. Bissett, Senior Operations Engineer (1,2,3)

<u>OTHER NRC PERSONNEL</u>:

T. Guilfoil, Sonalysts (1,2,3)
F. Victor, Sonalysts (1,2,3)
G. Weale, Sonalysts (1,2,3)

<u>CONNECTICUT YANKEE ATOMIC POWER COMPANY PERSONNEL</u>:

G. Bouchard, Nuclear Unit Director (2,3)
J. Deveau, Training Instructor (1,2)
R. Heidecker, Training Supervisor (1,2,3)
D. Lazarony, Training Instructor (1,2)
R. McBeth, Training Instructor (2)
D. Ray, Operations Manager (2,3)
R. Reeves, Operations Shift Supervisor (1)
J. Rein, Requalification Program Coordinator (1,2)
B. Ruth, Manager, NU Operations Training (3)
G. Waig, Training Instructor (2)

LEGEND:
(1) Participated in examination development
(2) Participated in examination administration
(3) Attended exit meeting on June 8, 1990 at the Northeast Utilities Training Center

2.0 PROGRAM EVALUATION RESULTS

Overall rating: Satisfactory

The program for licensed operator requalification training at Haddam Neck was rated as satisfactory in accordance with the criteria established in the revision 5 of NUREG-1021, ES-601. Those criteria are:

a. A pass/fail decision agreement between the NRC and facility grading of 90% for the written and operating examinations, with the licensee not being penalized for holding a higher standard of operator performance.

   NRC grading resulted in twelve operators passing the written examination. Facility grading also resulted in twelve operators passing the written examination. This satisfies criterion a.

   NRC grading resulted in twelve operators passing the job performance measures of the examination. Facility grading resulted in twelve operators passing the job performance measures of the examination. This also satisfies criterion a.

   NRC grading resulted in twelve operators passing the simulator portion of the examination. Facility grading resulted in twelve operators passing the simulator examination. This also satisfies criterion a.

b. At least 75% of all operators pass the examination.

   NRC grading is the only consideration for this criterion. All twelve operators passed the examination overall. This satisfies criterion b.

c. Failure of no more than one crew during the simulator portion of the operating examination.

   Again, NRC grading is the only consideration for this criterion. Three crews were evaluated and all three crews passed the simulator portion of the operating examination. This satisfies criterion c.

3.0 SCENARIO EVALUATION

The following were noted during the scenario portion of the operating examinations.

A significant improvement was noted in the area of communications during the conduct of the simulator scenarios when compared to communications that was observed during the previous examination administered in May 1989. Continued emphasis, however, is needed in this area, based upon the following observation. During the performance of one scenario, transitioning back to E-0 following an SI resulted in a transition being performed three different ways by three different crews. All three crews performed the transition correctly; however, it was only readily apparent as performed by the crew. Training emphasis is warranted regarding proper communications and procedural step announcements during this particular procedural transition.

Also, during the performance of scenario AE S00016, a procedural weakness was noted with emergency operating procedure E-3, Steam Generator Tube Rupture. At step 15, one is directed to equalize RCS and secondary pressure utilizing pressurizer spray. If pressurizer pressure does not decrease rapidly, one is to go to step 16 and depressurize using the pressurizer PORVs. During the performance of the above, one crew proceeded directly to step 16 once it was determined that the spray valves were not effective. The other two crews shut the spray valve prior to proceeding to step 16. Although not specifically stated in step 15, it is considered prudent action to shut the spray valves prior to proceeding to step 16. The licensee has initiated corrective action that will require shutting of the spray valves if they are found to be ineffective in decreasing pressurizer pressure.

4.0 WRITTEN EXAMINATION EVALUATION

Section A of the written examination was not of sufficient duration. Both static simulator exams took approximately 30 minutes to complete and an additional 15 minutes to review. The licensee had misinterpreted the time requirements for written examinations as stated in ES-601. Each section should take approximately 45 minutes to complete with 15 minutes additional time given for review. Section A should then total approximately 2 hours to complete.

No problems were noted with section B of the written examination.

5.0 JOB PERFORMANCE MEASURES (JPM) EVALUATION

JPM #127

Annunicator 4.9-34, specifically step 5.4.1, utilized during the performance of JPM 127, needs further review in determining the proper method for nulling out the Auxiliary Feedwater Pump pressure controllers.

JPM #134

Additional training needs to be performed in regard to the proper method for keying in the desired core exit thermocouple temperature IDs. Also, question 134-3 is considered a direct lookup since the answer can be found on the placard mounted inside the ICCS cabinet.

JPM #135

JPM needs to be updated as a result of the recent revision to AOP 3.2-50, Operation Outside the Control Room.

A generic concern involved the manner in which the evaluators signaled the end of JPM questioning or step completion. Many evaluators routinely said "OK." This may be construed by the examinee as meaning that their answer or performance was acceptable. Most evaluators stated that "OK" meant only that "I understand what you said or did." The NRC acknowledged the evaluators' interpretation of "OK", however, stated that such frequent usage of the word could possibly be misinterpreted by the examinee and would also be viewed by the NRC as excessive cueing. The licensee acknowledged the NRC concern and agreed to address the problem.

6.0 SUMMARY OF COMMENTS MADE AT EXIT MEETING ON JUNE 8, 1990

a. The NRC expressed appreciation for the level of effort expended by the training department representatives in accommodating the NRC examination team. This level of effort, which included providing an adequate working area, appropriate reference materials, locked storage capabilities, plant access badging, etc., helped in expediting the review process and the conduct of the exam. Appreciation was also expressed for the cooperation and level of effort expended by all those involved in the process, especially the facility team members who administered the examination.

b. The NRC discussed the topics addressed in Paragraphs 2 thru 5 above.

c. Examination scheduling was excellent. The length of examination days were reasonable and no undue delays were experienced.

d. The licensee needs to continue to apply Quality Control examination techniques to written examination and JPM questions in much the same manner that QC techniques were applied during the written exam review performed by the NRC during the exam preparation review week.

e. Although not discussed at the exit meeting, the reference material supplied by the licensee to the NRC for examination preparation was more than adequate. All material was well indexed and tabbed which allowed rapid access to specific topics and component information.

Attachments:
1. Written Examination (Initial - SRO) and Answer Key
2. NRC Response to Licensee Comments
3. Requalification Examination Test Items
4. Connecticut Yankee Atomic Power Company Letter (E. DeBarba to R. M. Gallo) Dated June 25, 1990

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CHIEF EXAMINER: Paul d. Bissett 8/30/90
Paul Bissett, Senior Operations Engineer Date

APPROVED BY: Peter W Eselgroth 8/30/90
Peter Eselgroth, Chief, PWR Section Chief Date
Operations Branch, Division of Reactor Safety

SUMMARY: The licensed operator requalification training program was rated as satisfactory. Written requalification examinations and operating tests were administered to eight senior reactor operators (SROs) and four reactor operators (ROs). The examinations were graded concurrently and independently by the NRC and the facility training staff. As graded by the NRC, all SROs and ROs passed all portions of the examination. Facility grading paralleled that of the NRC in all aspects of the examination.

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EXAMINATION RESULTS:

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Simulator	1/0	1/0
Overall	2/0	2/0

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Attachment 1

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION
REGION 1

FACILITY: Haddam Neck

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 90/05/24

CANDIDATE:

INSTRUCTIONS TO CANDIDATE:

Points for each question are indicated in parentheses after the question.

pass this examination, you must achieve an overall grade of at least 80%. Examination papers will be picked up four and one half (4 ~~1/2~~) hours after the examination starts.

NUMBER QUESTIONS	TOTAL POINTS	CANDIDATE'S POINTS	CANDIDATE'S OVERALL GRADE (%)
100	100.00		

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

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done after you complete the examination.

3. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

N/A

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

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

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

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

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

QUESTION: 001 (1.00)

Entry into the incore sump requires the prior approval of the:

a. Health Physics Supervisor, Shift Supervisor, and Duty Officer.

b. Health Physics Supervisor, Shift Supervisor, and Operations Supervisor.

c. Health Physics Supervisor, Shift Supervisor, and SCO.

d. Health Physics Supervisor and Shift Supervisor only.

QUESTION: 002 (1.00)

When containment integrity is required, two people may make a containment entry in an EMERGENCY if the containment entry requirements are WAIVED by the:

a. Health Physics supervisor.

b. Shift Supervisor.

c. Duty Officer.

d. Manager Operational Support Center.

QUESTION: 003 (1.00)

When containment integrity is required, the MAXIMUM number of people allowed in containment at one time is:

a. Eight [including the Hatch Watch] in consideration of ALARA concerns.

b. Ten [excluding the Hatch Watch] in consideration of emergency egress requirements.

c. Twelve [including the Hatch Watch] in consideration of ALARA concerns.

d. Fourteen [excluding the Hatch Watch] in consideration of emergency egress requirements.

QUESTION: 004 (1.00)

A change to the watch bill rotation be made provided that:

a. The person relieving has abstained from use of alcohol for four [4] hours prior to assuming the duty.

b. The person relieving has abstained from use of alcohol for five [5] hours prior to assuming the duty.

c. The person relieving has abstained from use of alcohol for four [4] hours prior to assuming the duty and the Station Services Superintendent approves the change.

d. The person relieving has abstained from use of alcohol for five [5] hours and the duty officer approves.

QUESTION: 005 (1.00)

The MINIMUM manning requirements for REFUELING OPERATIONS include:

a. Refueling SRO [Containment] and a Core Physics Monitor when loading fuel in the core [Control Room].

b. Refueling SRO [Control Room] and a core Physics Monitor when loading fuel in the core [Containment].

c. Refueling SRO [Control Room and Containment] and Shift Engineer [Containment] when loading or unloading fuel in the core.

d. Refueling SRO [Control Room and Containment] and Core Physics Monitor [Control Room and Containment] when loading or unloading fuel in the core.

QUESTION: 006 (1.00)

At 0323 this morning a temporary change to the Process Computer data base was made. The individual RESPONSIBLE for approving the temporary change was the:

a. Assistant Engineering Supervisor - Reactor Engineering.

b. Senior Control Operator.

c. Shift Supervisor.

d. Duty Officer.

QUESTION: 007 (1.00)

If the cable vault CO2 System must be placed out of service for modification, then:

a. The modification installation supervisors shall control the key to disable the CO2 system and the shift supervisor shall ensure the insurance company is notified of the CO2 system outage.

b. The senior control operators shall maintain control over the key to disable the CO2 system and the Duty Officer shall ensure the insurance company is notified of the CO2 system outage.

c. The shift supervisor shall maintain control over the key to disable the CO2 system and ensure that the insurance company is notified of the CO2 system outage.

d. The shift supervisor shall maintain control over the key to disable the CO2 system and the Duty Officer shall ensure that the insurance company is notified of the CO2 system outage.

QUESTION: 008 (1.00)

Removal of any PAB floor block requires guarding at all times unless other locked barriers are provided, because PAB floor block removal will constitute unlocking a[n]:

a. High Radiation Area.

b. Extremely High Radiation Area.

c. Protected [security] Area Boundary.

d. Vital [security] Area Boundary.

QUESTION: 009 (1.00)

If you find a plant equipment label in the parking lot, you should:

a. Forward the label to the Label Coordinator.

b. Return the label to the Viewing Gallery.

c. Forward the label to the Station Fire Marshall if the label is a fire protection equipment label.

d. Initiate a work request to have the label installed.

QUESTION: 010 (1.00)

A procedure that could cause an inadvertent load reduction is defined as a High Risk Test. Prior to commencement of any High Risk Test the Shift Supervisor shall NOTIFY:

a. The Duty Officer.

b. CONVEX at least 12 hours in advance of the test.

c. CONVEX at least 4 hours in advance of the test.

d. CONVEX at least 1 hour in advance of the test.

QUESTION: 011 (1.00)

After work has been concluded on the telephone system PBX the Shift Supervisor shall:

a. Notify the Security Shift Supervisor that work is complete and the PBX room is secure.

b. Notify the Security Shift Supervisor and Duty Officer that work is complete and the PBX room is secure.

c. Verify that the TSC [Technical Support Center] and OSC [Operations Support Center] phone lines are in service.

d. Verify that the ENS [NRC Emergency Notification System] and CONVEX telephone lines are in service.

QUESTION: 012 (1.00)

The Emergency Diesel Generator EG-2A Fast Start and Load Test, SUR 5.1-157A, was completed at 0314 this morning. The following results were obtained during the surveillance:

ACCEPTANCE CRITERIA	TEST RESULTS
Engine starting time < 10 seconds	10.3 seconds
Minimum voltage and frequency attained within 10 seconds after Exercise Start button is pressed	11.5 seconds
2750 to 2850 Kw load accepted within 60 seconds	57 seconds

Given these surveillance test results, WHAT ACTION are you required to take as the Shift Supervisor?

a. Since no reportable action is required to be taken, inform the Duty Officer during normal working hours

b. Immediately inform the Duty Officer and Station Superintendent

c. File the Plant Information Report [PIR] and inform the Duty Officer during normal working hours

d. File the Plant Information Report [PIR] and inform the Duty Officer immediately

QUESTION: 013 (1.00)

ACP 1.2-14.1, Jumper, Lifted Lead, and Bypass Control, provides exceptions for Jumper Devices. The determination if a Jumper Device is exempt from ACP 1.2-14.1 requirements is made by the:

a. PORC.

b. Supervising Control Operator.

c. Shift Supervisor.

d. Duty Officer.

QUESTION: 014 (1.00)

During a tour of the plant you observe the use of tags as follows:

Valve 1 is shut with three red tags attached.

Valve 2 is shut with two red tags and one yellow tag attached

Valve 3 is open with one red tag attached

Valve 4 is open with one blue tag attached

Valve 5 is shut with one blue tag and one yellow tag attached

Valve 6 is open with two blue tags attached

Valve 7 is shut with three yellow tags attached

Which valves have tags installed in accordance with the tag usage specified in ACP 1.2-14.2, Equipment Tagging?

a. 1, 3, 5, and 7

b. 1, 3, 4, and 7

c. 1, 2, 3, and 7

d. 2, 3, 4, and 6

QUESTION: 015 (1.00)

Two valve isolation will be attempted or used when isolating systems with:

a. Pressures > 50 psig and temperatures > 100 degrees F.

b. Pressures > 100 psig and temperatures > 100 degrees F.

c. Pressures > 250 psig and temperatures > 200 degrees F.

d. Pressures > 500 psig and temperatures > 200 degrees F.

QUESTION: 016 (1.00)

In emergency situations where it is necessary to enter a hazardous area to protect facilities, eliminate future release of effluent or to control a fire, the whole-body dose should be controlled to:

a. < 5(N-18) Rems [N is the exposed individual's age in years].

b. 25 Rems.

c. 75 Rems.

d. 125 Rems.

QUESTION: 017 (1.00)

In the event of conditions where a station/plant emergency incident/accident is in progress, then:

a. The Shift Supervisor classifies emergencies and makes recommendations for off-site protective actions until relieved by the Duty Officer as DSEO.

b. The Supervising Control Operator performs as the Manager of Control Room Operations until relieved by the Manager of Operations Support Center.

c. The Shift Supervisor performs as the Manager of Control Room Operations until relieve by the Manager of Operations Support Center.

d. The Supervising Control Operator is responsible for notification of off-site and Station Emergency Response Organizations.

QUESTION: 018 (1.00)

If the rod control system flux compensation unit fails, then:

a. The rod control system cannot move rods > 5 in/min.

b. The control rods will not move if AUTOMATIC is selected on the Bank Selector Switch.

c. Automatic rod control may be sluggish.

d. Automatic rod control will not be affected.

QUESTION: 019 (1.00)

If the rod control system Bank A slave cycler relay fuse opens and deenergizes the movable gripper coils, then Bank A rods:

a. Can not be moved in AUTOMATIC.

b. Can be moved only in MANUAL.

c. Will be unmovable.

d. Will drop.

QUESTION: 020 (1.00)

If a rod control system slave cycler fails, then:

a. The fault detection logic circuit will terminate automatic rod motion for the affected subgroup.

b. The fault detection logic circuit will stop all rod motion.

c. The affected subgroup rods will drop.

d. The half-power relays deenergize.

QUESTION: 021 (1.00)

Technical Specification 3.1.3.2 requires that the Digital Rod Position Indication System [group step counters] and the Analog Rod Position Indication System [RPI] shall be OPERABLE. What is the technical specification BASIS for OPERABILITY of the control rod position indicators?

a. Ensure acceptable power distribution limits are maintained and compliance with control rod alignment

b. Ensure acceptable power distribution limits are maintained and the minimum SHUTDOWN MARGIN is maintained

c. To determine control rod positions and thereby ensure compliance with control rod alignment and insertion limits

d. To determine control rod positions and thereby ensure compliance with insertion limits and limit the effects of rod ejection accidents

QUESTION: 022 (1.00)

Following a reactor/turbine trip the reactor coolant pump busses are automatically transferred to offsite power and the number one and three reactor coolant pumps are tripped. A fifty two second time delay is provided between the time of the reactor/turbine trip and the subsequent trip of the generator and bus transfer. The two purposes of the fifty two second time delay are to ensure that the:

a. Reactor coolant pumps will not overspeed due to the blowdown of a large break LOCA, and dissipate decay heat immediately following the trip.

b. Reactor coolant pumps will not overspeed due to the blowdown of a large break LOCA, and prevent turbine overload following the reactor trip.

c. Reactor coolant pumps will not overspeed due to the blowdown of a large break LOCA and prevent the turbine from overspeed on loss of load.

d. Bus transfer completed satisfactorily [ensures power supply for the reactor coolant pumps] and dissipates the steam trapped in the secondary system.

QUESTION: 023 (1.00)

Loss of 125 V DC Bus A will cause:

a. Thermal Barrier Flow Control Valve [CC-FCV-608] to fail open.

b. Thermal Barrier Flow Control Valve [CC-FCV-608] to fail closed.

c. RCP Oil Cooler Return Trip Valve [CC-TV-1411] to fail open.

d. RCP Oil Cooler Return Trip Valve [CC-TV-1411] to fail shut.

QUESTION: 024 (1.00)

If a loss of power to Vital "A" occurs, then:

a. Charging flow control valves FCV-110 and FCV-110A will fail open.

b. Charging flow control valves FCV-110 and FCV-110A will fail shut.

c. Charging flow controller FIC-110 output fails high and charging flow transmitter FT-110 output fails low.

d. Charging flow controller FIC-110A output fails high and charging flow transmitter FT-110 output fails low.

QUESTION: 025 (1.00)

MOV-32, RWST to charging pump suction valve, is normally shut. If MOV-32 automatically opened, what interlock conditions must be satisfied in order to reclose MOV-32?

a. VCT level must be at least 10%

b. VCT level must be at least 22%

c. VCT level must be at least 22% and SI reset

d. None

QUESTION: 026 (1.00)

During testing or calibration of VCT level channel LT-100, the following conditions exist:

* BA automatic makeup controller in MANUAL and set to zero
* PW automatic makeup controller in MANUAL and set to zero
* BA pumps in TRIP-PULLOUT
* PW pumps in TRIP-PULLOUT
* FCV-113C bypass in OFF

Why are the controls aligned as stated?

a. To prevent spurious pump starts and letdown diversion

b. To prevent spurious pump starts and unnecessary VCT makeup attempts

c. To prevent charging pump suction transfer on low level and prevent spurious pump starts

d. To prevent charging pump suction transfer on low level and letdown diversion to the WDS

QUESTION: 027 (1.00)

If energizing core cooling initiation relay 4BX1 will start the Diesel Generator [EG-2B], what component failure could cause actuation of the Bus 9 Low Voltage Interlock and Diesel Generator [EG-2A] to start?

a. Core Cooling Initiation relay 4A

b. Core Cooling Initiation relay 4AX

c. DC Bus A [circuit 15A]

d. DC Bus A [circuit 15B]

QUESTION: 028 (1.00)

A safety injection has occurred with all required emergency core cooling equipment operating except the charging pumps. Why are the charging pumps NOT running?

a. The SI sequence timer failed at 33 seconds

b. Offsite power is not available

c. The SI was an inadvertent actuation

d. The SI was caused by high containment pressure

QUESTION: 029 (1.00)

The reactor is at 100% power with the rod control system in automatic. Power Range Channel P4 fails downscale [low] due to a malfunction in its level amplifier.

What is the effect of the failure of Power Range Channel P4?

a. Rods step in at maximum rate due to the load runback

b. Rods step out because of the rate of power change input to rod speed

c. Rods do not automatically respond

d. Reactor trips due to cooldown caused by the runback

QUESTION: 030 (1.00)

All core exit thermocouples [CETs] can be read from the:

a. ICC Cabinets, SPDS, and A Switchgear Room.

b. SPDS, ICC Cabinets, and B Switchgear Room.

c. ICC Cabinets, SPDS, and Post Accident Monitoring Auxiliary Panel.

d. ICC Cabinets, SPDS, Post Accident Monitoring Auxiliary Panel, and B Switchgear Room.

QUESTION: 031 (1.00)

How will a LOSS OF SERVICE WATER effect the containment air recirculation [CAR] units [fan and motor] performance?

a. Since component cooling water is the cooling supply for the CAR units, loss of service water will have no effect.

b. Containment temperature will continue to be maintained within limits; however, containment pressure will trend towards approximately 3.5 pounds in about two days if service water cannot be restored

c. Under normal containment conditions, the motor's air recirculation cooling system will limit the rate of temperature increase, but motor overheating and damage may eventually result if cooling water cannot be restored

d. Under normal conditions the CAR units will continue to maintain containment temperature and pressure within limits; however, service water must be restored to support acceptable performance under accident conditions

QUESTION: 032 (1.00)

During an event requiring containment spray [containment pressure increasing], under what conditions would Fire Water to Containment Spray Valve [FM-MOV-31] be opened to supply the containment spray header from the fire water system?

a. Only one low pressure safety injection pump is available

b. Only one residual heat removal pump is available

c. Only both low pressure safety injection pumps are available

d. The containment sump isolation valve [RH-MOV-22] can not be opened

QUESTION: 033 (1.00)

How will a running main feed pump respond to a loss of 125 V DC control power?

a. Since all automatic functions will be lost, the pump will trip

b. The pump will continue to operate, but all automatic functions will be lost

c. The recirculation valve for the pump will open and the pump will trip on low suction pressure

d. The pump will trip on mechanical overspeed

QUESTION: 034 (1.00)

Why must the status of the waste evaporator be stabilized prior to starting waste gas releases?

a. To protect the reactor coolant pump seal No. 3 from high backpressure

b. To prevent unplanned waste gas releases

c. To prevent mixing air and hydrogen in the waste gas system

d. To prevent possible over pressure in the waste gas system

QUESTION: 035 (1.00)

The containment hatch area radiation monitor [R-37] yellow low light is illuminated. This indicates that:

a. Approximately a 1% fuel failure exists.

b. The R-37 detector voltage is low.

c. R-37 is inoperable.

d. R-37 power was lost and restored.

QUESTION: 036 (1.00)

If under voltage on 480V AC buses #4, #6, or #7 occurs, an automatic trip of the waste evaporator is actuated. The PURPOSE of this automatic trip is:

a. Minimize the heat load on the Component Cooling Water System and protect the evaporator in the event of a loss of one or more component cooling pumps.

b. Minimize the heat load on the Component Cooling Water System and protect the evaporator on the loss of the reboiler pump.

c. To protect the reboiler pump from loss of suction from the resulting evaporator lo-lo level.

d. To protect the reboiler pump from over pressure and to minimize the heat load on the Component Cooling Water System.

QUESTION: 037 (1.00)

Technical Specification 3.4.3 requires that the Pressurizer be OPERABLE with a water volume of less than or equal to 86% [1089 cubic feet], and at least two groups of Pressurizer heaters each having a capacity of at least 150KW be OPERABLE when in MODES 1, 2, and 3. The Technical Specification BASES for these requirements are:

a. Ensures that the steam bubble is formed and the Reactor Coolant System is not a hydraulically solid system.

b. Enhances the capacity of the plant to control Reactor Coolant System Pressure and establish natural circulation.

c. Ensures that the Reactor Coolant System will be operated with adequate subcooling and not be a solid system.

d. Both a. and b.

QUESTION: 038 (1.00)

Four pressurizer solenoid operated vent valves are provided to ensure:

a. That a single failure will not cause the vent path to be placed in service and cause an inadvertent venting of the Pressurizer.

b. That a single failure will not prevent the vent path from being placed in service or causing an inadvertent venting of the Pressurizer.

c. That a single failure will not cause the vent path to be placed in service and to ensure adequate venting capacity during accident conditions.

d. That a single failure will not prevent the vent path from being placed in service and to ensure adequate venting capacity during accident conditions.

QUESTION: 039 (1.00)

Why are the Pressurizer pressure control channel variable low pressure trip setpoint calculators rate compensated and level limited?

a. Rate compensation provides dynamic response to small changes in the setpoint and level limiting limits the magnitude of Tavg and delta-T inputs to the computing circuit

b. The slow response time of the RTDs used to measure loop temperatures makes the rate compensation necessary to provide dynamic response to small changes in the setpoint and level limiting ensures that the setpoint is not lowered to an unsafe value

c. The slow response time of the RTDs used to measure loop temperatures makes the rate compensation necessary to provide dynamic response to small changes in the setpoint and level limiting limits the magnitude of Tavg and Delta-T inputs to the setpoint calculators

d. Rate compensation ensures dynamic response to large changes in the setpoint but level limiting limits the overall magnitude of large setpoint changes

QUESTION: 040 (1.00)

Given the following indications / conditions:

* Loss of output from the Pressurizer pressure controller
* Loss of signal to automatically energize the backup heaters on low Pressurizer pressure [1950 psig]
* Actuation of the "PRESSURIZER HI PRESS" alarm [C2-2-3(U)]
* Actuation of the "PRESSURIZER LO PRESS" alarm [C2-2-3(L)]

What microprocessor card failure could cause these indications or conditions?

a. Pressure control channel 1 control

b. Pressure control channel 2 control

c. Pressure control channel 3 control

d. Pressure control channel 4 control

QUESTION: 041 (1.00)

If the Pressurizer level programmer [reference level] fails low while at 100% power, then:

a. The charging flow control valves will go to the open position if the Pressurizer level controller was operating in auto-remote control.

b. The charging flow control valves will go to the open position if the charging flow controllers were operating in auto-remote control.

c. The charging flow control valves will go to the closed position if the Pressurizer level controller was operating in auto-remote control.

d. The charging flow control valves will remain as is if the charging flow controllers were operating in auto-remote control.

QUESTION: 042 (1.00)

If the Pressurizer backup heaters fail to energize on high Pressurizer level [54.0%], then you would suspect failure of the level control microprocessor card for:

a. Channel 1.

b. Channel 2.

c. Channel 3.

d. Channel 4.

QUESTION: 043 (1.00)

What is the power supply [including control power] to the A high pressure safety injection pump?

a. 4 KV Bus 8 with control power supplied from 125V DC bus A

b. 4 KV Bus 9 with control power supplied from 125V DC bus A

c. 4 Kv Bus 9 with control power supplied from 125V DC bus B

d. 4 KV Bus 8 with control power supplied from 125V DC bus B

QUESTION: 044 (1.00)

Actuation of the B SI and B HCP relays will be caused by:

a. Loss of DC Bus A.

b. Loss of DC Bus B.

c. Deenergizing the B WL solenoid plunger [main control board B] for automatic initiation.

d. Deenergizing the A WL solenoid plunger [main control board B] for automatic initiation.

QUESTION: 045 (1.00)

The core is directly protected from exceeding DNB by the:

a. Overpower trip [high neutron flux] and low flow trips.

b. Low flow trips and the variable low pressure reactor trip.

c. Overpower trip [high neutron flux] and the variable low pressure reactor trip.

d. Variable low pressure reactor trip and the reactor coolant pump breaker open trip.

QUESTION: 046 (1.00)

The P-8 low flow permissive inputs are:

a. Power range nuclear instruments and reactor coolant pump breaker position.

b. Power range nuclear instruments and turbine first stage pressure.

c. Reactor coolant pump breaker position and turbine first stage pressure.

d. Power range nuclear instruments, turbine first stage pressure and reactor coolant pump breaker position.

QUESTION: 047 (1.00)

How is the desired containment purge air flow rate obtained?

a. The purge supply damper bypass is manually throttled

b. The purge supply and exhaust damper bypasses are manually throttled

c. The purge fan inlet and dilution air dampers on the purge fan intake plenum are manually adjusted

d. By adjusting the service air being "charged" to containment

QUESTION: 048 (1.00)

Which of the following process radiation monitors will terminate a release or isolate a flow path when radiation exceeds a setpoint?

1. Containment air particulate and gas monitors [R-11/12]
2. Vent stack Effluent Monitor [R-14A]
3. Wide Range Gas Monitor [R-14B]
4. Air ejector Effluent monitor [R-15]
5. Steam generator blowdown monitors [R-16A & B]
6. Component cooling water system monitor [R-17]
7. service water system effluent monitor [R-18]
8. Spent fuel pool service water monitor [R-19]
9. Reactor coolant system letdown monitor [R-20]
10. CCW return from the waste gas compressors [R-2209]
11. Waste test tank effluent monitor [R-22]

SELECT the correct answer.

a. 1, 2, 5, 7

b. 2, 3, 5, 7

c. 2, 4, 6, 7

d. 2, 6, 7, 11

QUESTION: 049 (1.00)

Technical Specification 3.9.11, Water Level - Storage Pool reads:

At least 20 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

What is the Technical Specification BASIS for the storage pool water level?

a. To ensure that in the event of a failure of the operating RHR LOOP, adequate time is provided to initiate emergency procedures to cool the core

b. To ensure that in the event of a failure of the operating RHR LOOP, adequate time is provided to initiate emergency procedures to cool the fuel in the storage pool

c. To ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly

d. To ensure that sufficient water depth is available to remove 99% of the assumed 100% iodine gap activity released from the rupture of an irradiated fuel assembly

QUESTION: 050 (1.00)

How is the main control board alarm BUS OVERVOLTAGE 1-3/1-9 cleared after bus voltage has been restored to normal?

a. Press the overvoltage relay [59A/1-9] reset button located next to the relay on the auxiliary board in control

b. Press the overvoltage relay [59A/1-9] reset button located inside the auxiliary board in control

c. By acknowledging the alarm at the main control board

d. The alarm clears automatically after the bus voltage returns to n 'mal

QUESTION: 051 (1.00)

In the event of a loss of the spent fuel pool [SFP] cooling pumps, spent fuel pool cooling circulation is maintained by:

a. Connecting the firemain to the SFP heat exchangers.

b. Connecting a temporary pump through SW-V-736 at the SFP heat exchangers.

c. Hooking up a temporary pump and hoses to the SFP cooling pump suction and discharge piping.

d. Hooking up a component cooling water pump via temporary hoses.

QUESTION: 052 (1.00)

A START FAILURE alarm will be received on the emergency diesel engine control panel if the engine fails to:

a. Reach 40 RPM within 2 seconds or 100 RPM within 4 seconds.

b. Reach 40 RPM within 3 seconds or 100 RPM within 4 seconds.

c. Reach 100 RPM within 2 seconds or the Fast Start Relay [FSR] fails to pick up within 1 second of receiving an auto start signal.

d. Reach 100 RPM within 3 seconds or the Fast Start Relay [FSR] fails to pick up within 1 second of receiving an auto start signal.

QUESTION: 053 (1.00)

If the main steam header pressure transmitter output fails low, then the:

a. Main steam header pressure indicates 0 psig.

b. The steam dump pressure controller will not be affected.

c. Main steam header pressure indicates 1000 psig.

d. Steam dump pressure controller will command the steam dump pressure control valves to open.

QUESTION: 054 (1.00)

The Core Deluge Isolation Valve [RH-MOV-873] is:

a. Normally open during power operation, and shut when placing the RHR in service.

b. Normally shut during power operation, and opened when placing the RHR in service.

c. Normally open during power operation and supplying flow to the refueling cavity during refueling operations.

d. Normally open during power and RHR operation and closed during refueling operations [except when controlling flow to the temporary cavity fill connection during refueling operations.

QUESTION: 055 (1.00)

If component cooling water system flow cannot be restored due to pump failure, then:

a. Fire water is lined up to provide flow through the CCW system via temporary hoses.

b. Service water is lined up to provide flow through the CCW system via temporary hoses.

c. Fire water is provided via connections on the CCW heat exchangers to allow CCW flow out to the discharge canal.

d. Service water is provided via CCW heat exchanger inlet and outlet valves to allow CCW flow out to the discharge canal.

QUESTION: 056 (1.00)

Following a reactor trip with plant Tavg reduced to 535 degrees F, the High Pressure Steam Dump temperature control valves:

a. Will fully open if steam generator pressure is > 910 psig.

b. Will modulate to maintain 890 - 910 psig.

c. Will not reopen unless RCS Tavg increases to 545 degrees F.

d. Automatic actuation will not occur.

QUESTION: 057 (1.00)

The Van Air Model HB-200 air dryer operation is directed by the Interlock Logic System which ensures that each phase of the cycle [normal operation, represurization, tower change over, and depressurization] occur in the exact sequence. How does this safeguard against accidental blowdown of the process air supply?

a. The purge exhaust valve[s] are maintained fully open until the offgoing tower is completely shut off

b. The purge exhaust valve[s] cannot open until the master transfer valve has fully seated and flow to the offgoing tower is completely shut off

c. The master transfer valve is not fully seated until the purge exhaust valve[s] are fully shut and the oncoming tower is placed in operation.

d. The master transfer valve is maintained fully open until the oncoming tower is placed in operation and the purge exhaust valve[s] is fully seated.

QUESTION: 058 (1.00)

AOP 3.2-23, Malfunction of Rod Control System, Step 4.3.3 states:

IF uncontrolled rod motion continues, PLACE Rod Motion Disconnect Switch in "DISCONNECT" position.

How does this action stop rod motion?

a. Prevents the lift coil relays in each power cabinet from energizing

b. Prevents the slave cyclers from cycling and thus stops all rod motion

c. Deenergizes the master cycler to prevent uncontrolled rod motion

d. Removes power cabinet control power

QUESTION: 059 (1.00)

Given the following indications:

* Tavg decreasing
* Pressurizer pressure decreasing
* Pressurizer level decreasing
* Plant in Mode 1 at 89% power decreasing [proceeding to 100% IAW NOP 2.2-1, Changing Plant Load]
* "ROD OUT OF STEP" alarm on Plant Process Computer
* Rod bottom lights illuminated on 2 rods
* Auto rod control

What action should be taken?

a. Enter AOP 3.2-23, Malfunction of Rod Control System, Step 4.5, Rod Position Indication Malfunction

b. Enter NOP 2.2-5, Control Rod Alignment

c. Manually trip the reactor and enter E-0, Reactor Trip or Safety Injection

d. Remain in NOP 2.2-1, Changing Plant Load, until the plant is stable, then initiate a controlled plant shutdown IAW NOP 2.3-2, Reactor Shutdown

QUESTION: 060 (1.00)

As the Shift Supervisor, in the event of a dropped control rod, you would:

a. Notify the NRC within 1 hour.

b. Notify the Station Director.

c. Notify the Unit Director.

d. Notify the Duty Officer.

QUESTION: 061 (1.00)

Given the following conditions:

* Mode 2 conducting PHYSICS TESTS
* Tavg - 523 degrees F
* Reactor power - 6%
* One rod inoperable - found to be untrippable

What action should be taken?

a. Determine that the Technical Specification SHUTDOWN MARGIN requirement is satisfied within 1 hour

b. Within 1 hour restore the rod to within + or - 24 steps of its group position

c. Immediately open the reactor trip breakers

d. Physics testing may continue provided Tavg is maintained > 515 degrees F

QUESTION: 062 (1.00)

In addition to Complete loss of control air, what are the E-O Series Procedures Foldout RCP TRIP CRITERIA?

a. At least one HPSI pump OR charging pump running and RCS pressure is > 1150 psig

b. At least one HPSI pump OR charging pump running and RCS pressure is > 1380 psig [containment pressure > 5 psig]

c. At least one HPSI pump OR charging pump running and RCS pressure is < 1380 psig [containment pressure < 5 psig]

d. At least one HPSI pump OR charging pump running and RCS pressure is < 1150 psig

QUESTION: 063 (1.00)

A reactor trip has occurred following a turbine trip. You have correctly directed the operators to exit from E-O, Reactor Trip Or Safety Injection, to ES-O.1, Reactor Trip Response. At step 5 VERIFY ALL AC BUSES ENERGIZED BY OFFSITE POWER you note the following:

* RCS pressure 1710 psig decreasing
* Pressurizer level 24% steady
* Core exit T/Cs 575 degrees F increasing
* Wide range 1E -7 amps decreasing
* Containment pressure 4 psig and increasing slowly
* All SGs wide range levels 65% and increasing slowly

What action will you, as the SCO, direct the operators to take?

a. Proceed to ES-O.1 step 6 after verifying all AC buses energized by offsite power

b. Actuate SI and go to E-O, Reactor Trip Or Safety Injection, Step 1

c. Exit to FR-H.1, Response To Loss Of Secondary Heat Sink, Step 1

d. Exit to FR-Z.1, Response To High Containment Pressure, Step 1

QUESTION: 064 (1.00)

Given the following conditions:

* Reactor power 93%

* Chemistry reports that the main coolant activity is 70 uCi/gm dose equivalent I-131

What action(s) are required in response to the given conditions?

a. Operation may continue for up to 48 hours

b. Operation may continue for up to 24 hours

c. Be in at least HOT STANDBY within 6 hours

d. Trip the reactor

QUESTION: 065 (1.00)

Technical Specification 3.4.8 Action b. states:

> With the specific activity of the reactor coolant greater than 68/E bar microCuries per gram of gross radioactivity, be in at least HOT STANDBY with Tavg less than 500 degrees F within 6 hours.

What is the Technical Specification BASIS for reducing Tavg to less than 500 degrees F?

a. Prevent the release of radioactivity should a LOCA occur

b. Prevent the release of radioactivity should a steam generator tube rupture occur

c. Ensure that the 1-hour dose at the SITE BOUNDARY will not exceed an appropriately small fraction of the 10 CFR Part 100 dose guideline values

d. Ensure that the 2-hour dose at the SITE BOUNDARY will not exceed an appropriately small fraction of the 10 CFR Part 100 dose guideline values in the event of a LOCA

QUESTION: 066 (1.00)

Emergency Operating Procedure E-1, Loss Of Reactor Or Secondary Coolant, Step 3.b. Action/Expected Response reads:

Control AUX feedwater flow to maintain wide range levels between 63% and 69% [67% and 69% for adverse Containment]

Why is level maintained in ALL intact steam generators?

a. To satisfy the feed flow requirement of the Heat Sink Status Tree

b. To maintain symmetric cooling of the RCS

c. Ensures adequate inventory with level readings on visible span

d. Ensures increasing level would be observed following a SGTR

QUESTION: 067 (1.00)

Emergency Operating Procedure E-2, Faulted Steam Generator Isolation, Step 1. Action/Expected Response reads:

Check Main Steamline Isolation and Bypass Valves of Affected SG(s) Closed

What is the BASIS for this step?

a. To maintain at least one loop available for cooldown and to isolate the break

b. To maintain at least one loop available for cooldown and to isolate the SGs from each other

c. To isolate the break and to isolate the affected SG from other SGs

d. To isolate the break and prevent an uncontrolled cooldown

QUESTION: 068 (1.00)

Emergency Operating Procedure E-0, Reactor Trip Or Safety Injection, Step 1. Action/Expected Response reads:

Verify Reactor Trip:

What is the BASIS for this step?

a. To ensure that neutron flux is decreasing and only decay heat is being added to the RCS

b. To ensure that scram breakers are open and neutron flux is decreasing

c. To ensure that the only heat being added to the RCS is from decay heat and pump heat

d. To ensure that the reactor has tripped and neutron flux is decreasing

QUESTION: 069 (1.00)

A reactor trip and safety injection have occurred following a small break LOCA. You have correctly transitioned into E-1, Loss Of Reactor Or Secondary Coolant. The operators have satisfactorily completed E-1 through Step 5, Check PZR PORVs And Block Valves. The following conditions exist at the completion of Step 5:

* Total AUX Feedwater flow - 75 GPM
* Wide range SUR - negative
* RCS subcooling based on core exit T/Cs - 43 degrees F
* Wide range level in all S/Gs - 64%
* RCS pressure - 1550 psig increasing
* Containment pressure - 2 psig
* Pressurizer level - 17%
* RWST level - 150,0000 gallons

What action should you direct the operators to take?

a. Go to Step 8, Verify Containment Spray Not Required

b. Go to ES-1.1, SI Termination

c. Go to ES-1.3, Transfer To RHR Recirculation

d. Go to FR-H.1, Response To Loss Of Secondary Heat Sink

QUESTION: 070 (1.00)

You have correctly entered FR-S.1, Response To Nuclear Power Generation/ATWS. Steps 1 through 9 have been completed satisfactorily. While you are directing the operators to perform Step 10, Check Main Steam Line Trip Valves - CLOSED, an SI actuates.

What action(s) is(are) required to be taken?

a. Verify SI actuation has occurred and continue with FR-S.1 Step 10

b. Perform E-0, Reactor Trip Or Safety Injection simultaneously with FR-S.1 Steps 10 through 15

c. Continue with FR-S.1 Steps 10 through 15 and then go to E-0, Reactor Trip Or Safety Injection

d. Immediately go to E-0, Reactor Trip Or Safety Injection, Step 1

QUESTION: 071 (1.00)

A reactor trip has occurred due to a generator trip. While performing E-O, Reactor Trip Or Safety Injection, it has been determined that power CAN NOT be restored to at least one AC emergency bus. The following conditions exist when you direct going to ECA-0.0, Station Blackout:

* Wide range level in all SGs - about 53% and decreasing
* Containment pressure - 0
* RCS pressure - 1650 psig and decreasing
* RCS subcooling margin - 40 degrees F and decreasing
* PZR level - 45% and decreasing
* Both turbine trip valves - CLOSED

What action(s) should be taken?

a. Go to FR-H.1 and restore wide range level in at least one SG > 63%

b. Continue in ECA-0.0

c. Manually initiate SI and go to E-O

d. Locally start turbine driven AFW pumps if AFW flow < 320 GPM

QUESTION: 072 (1.00)

Functional Restoration Procedure, FR-C.1, Response To Inadequate Core Cooling, contains the following note before Step 10:

Partial uncovering of SG tubes is acceptable in the following steps.

What is the BASIS for this note?

a. Maintenance of SG level during the rapid depressurization will be difficult and partial uncovering of SG tubes is anticipated

b. Maximum feedwater mass addition rate exceeds the steam mass removal rate

c. Maintaining the SG tubes covered may require excessive feedwater addition rates leading to potential pressurized thermal shock (PTS)

d. Maximize primary-to-secondary heat transfer

QUESTION: 073 (1.00)

ECA-0.0, Loss Of All Power, contains the following CAUTION before Step 6:

WHEN power is restored to any AC emergency bus, recovery actions should continue starting with Step 23.

What is the BASIS for this CAUTION?

a. Minimize deterioration of plant conditions by terminating plant cooldown

b. Energize instrumentation and control equipment

c. Steps 6 through 23 are contingency steps rather than action steps

d. Prevent performance of unnecessary actions

QUESTION: 074 (1.00)

When inadequate core cooling exists, what is the proper sequence of and major actions/processes to be performed for removing decay heat from the core?

a. Reinitiation of high pressure safety injection; RCP restart; rapid secondary depressurization

b. Rapid secondary depressurization; reinitiation of high pressure safety injection; RCP restart

c. RCP restart; reinitiation of high pressure safety injection; rapid secondary depressurization

d. Reinitiation of high pressure safety injection; rapid secondary depressurization; RCP restart

QUESTION: 075 (1.00)

Following a reactor trip and safety injection, in accordance with E-O, Reactor Trip Or Safety Injection, Step 23 Response Not Obtained (RNO), you direct going to E-1, Loss Of Reactor Or Secondary Coolant, Step 1. The following conditions exist when you enter E-1, Step 1:

* Charging pumps - RUNNING
* HPSI pumps - RUNNING
* RCS subcooling - 0 degrees F
* Containment pressure - 44 PSIG
* RCPs - all STOPPED
* SG pressure - All DECREASING
* Condenser air ejector radiation - NORMAL
* SG wide range level - All 68% to 73%
* Pressurizer level - EMPTY
* Feedwater flow - 220 GPM

What action should you direct to be taken?

a. STOP all RCPs and close CH-TV-240 & 241 and CC-TV-1411

b. Throttle AUX feedwater flow to maintain wide range level between 67% and 69%

c. Go to FR-H.1, Response To Loss Of Secondary Heat Sink

d. Go to FR-Z.1, Response To High Containment Pressure

QUESTION: 076 (1.00)

Emergency Boration is REQUIRED when:

a. An uncontrolled cooldown is occurring and Tave is approaching 50 degrees F less than steady state value and a LOCA [small break] is in progress.

b. An uncontrolled cooldown is occurring and Tave is approaching 50 degrees F less than the steady state value.

c. A positive reactivity addition is occurring from open SG PORVs [PORVs opened to initiate cooldown].

d. One control rod fails to drop after a reactor trip.

QUESTION: 077 (1.00)

Given the following conditions:

* Containment Sump Level increasing slowly
* Reactor Coolant Pump thermal barrier low flow alarms
* Component Cooling Water surge Tank low level alarm

What action(s) should be taken?

a. Start the standby component cooling water pump

b. Immediately trip the reactor coolant pumps

c. Simultaneously trip the reactor and the reactor coolant pumps

d. Enter AOP 3.2-10, Loss Of Component Cooling Water

QUESTION: 078 (1.00)

AOP 3.2-50, Plant Operations Outside the Control Room, includes Step 4.5.6:

PRESS the exercise pushbutton on the EG-2B Excitation Control Panel.

What is the REASON for pressing the exercise pushbutton?

a. To exercise the EG-2B output circuit breaker

b. To accelerate the diesel to the high speed setting

c. To reset the emergency stop and oil pressure trip

d. To shutdown EG-2B

QUESTION: 079 (1.00)

AOP 3.2-50, Plant Operations Outside the Control Room, specifies dispatching designated operators [3] to perform the following:

a. Tripping the reactor, tripping the turbine and shutting down ED-2A.

b. Tripping the reactor, tripping the turbine and shutting down ED-2B.

c. Starting ED-2A, establishing emergency feed to the steam generators and INITIATING the Emergency Plan.

d. Starting ED-2B, establishing emergency feed to the steam generators and INITIATING the Emergency Plan.

QUESTION: 080 (1.00)

Why is the seal return MOV closed if a positive delta P cannot be maintained on a reactor coolant pump seal?

a. Ensures other pumps perform properly

b. Reduces the backup seal delta P to a minimum

c. Increases RCS flow through the seal

d. Reduces seal injection flow

QUESTION: 081 (1.00)

Which of the following liquid radioactive waste releases are reportable in accordance with EPIP 1.5-1, Attachment 12.4, Reportable Liquid Or Gaseous Releases:

1. Planned release within technical specifications
2. Planned release exceeding technical specifications
3. Unplanned release within technical specifications
4. Unplanned release exceeding technical specifications
5. Unmonitored release within technical specifications
6. Unmonitored release exceeding technical specifications

SELECT the correct answer.

a. 1, 2, 3

b. 1, 2, 5

c. 3, 4, 6

d. 2, 4, 6

QUESTION: 082 (1.00)

Given the following indications:

* 98% power
* LOW PRESS CONTROL AIR alarm
* MAIN CONTROL PANEL LO AIR PRESS alarm
* CONTROL AIR COMPRESSOR TROUBLE alarm

What IMMEDIATE ACTION should be performed?

a. Immediate actions of E-0, Reactor Trip or Safety Injection

b. Place standby charging pump in standby

c. Verify control air pressure increasing to 90 psig

d. Locally control SG feedwater bypass AOVs and terry turbine steam inlet control valves

QUESTION: 083 (1.00)

Which of the following are AUTOMATIC ACTIONS on a complete loss of control air?

1. Charging flow control valves fail closed
2. Letdown orifice isolation valves close
3. RCP seal water return trip valves close
4. Atmospheric steam dump valve fails closed
5. Secondary safety valves actuate
6. Feedwater regulating valves close
7. Feedwater bypass valves fail closed

SELECT the correct answer.

a. 1, 3, 5, 7

b. 2, 4, 6, 7

c. 1, 2, 3, 4

d. 3, 4, 5, 6

QUESTION: 084 (1.00)

EOP 3.1-49, Partial Loss of DC, discussion states:

> The DC bus tie breakers should never be closed during this event [a partial loss of DC].

The REASON for prohibiting closure of the DC bus tie breakers is:

a. To prevent inadvertent paralleling of two DC sources.

b. Because closure could cause a loss of another DC bus.

c. To prevent having one train supply all safety related loads.

d. Because closure could cause bus voltage to be reduced and cause equipment overheating from higher currents.

QUESTION: 085 (1.00)

Loss of DC Bus B/BX will require decay heat removal by natural circulation because:

a. Although the RCPs will auto transfer to Buses 1-2 and 1-3, the RCPs will have to be tripped because component cooling water to the thermal barrier trip valve fails closed and RCP seal water return trip valve fails closed.

b. Although component cooling water to the thermal barrier trip valve fails open and RCP seal water return trip valve fails open, the RCPs will fail to auto transfer to Bus 1-2 and Bus 1-3.

c. Component cooling water to the thermal barrier trip valve fails closed and the RCP seal water return trip valve fails closed and the RCPs will have to be tripped.

d. RCP seal water return trip valve fails open, but the component cooling water to the thermal barrier trip valve fails closed and the RCPs will have to be tripped.

QUESTION: 086 (1.00)

Why is Emergency Operating Procedure ES-3.1, Post-SGTR Cooldown Using Backfill, the preferred method for Post-SGTR cooldown?

a. Radiological releases are minimized

b. Boron dilution is minimized

c. Adverse secondary side water chemistry concerns are eliminated

d. More rapid means of depressurizing the RCS is provided

QUESTION: 087 (1.00)

If a loss of all AC power occurs following an operator initiated reactor trip due to a SGTR, voiding in the main coolant system will be indicated by unexplained pressurizer:

a. Pressure decrease

b. Pressure increase

c. Level decrease

d. Level increase

QUESTION: 088 (1.00)

If the following indications or conditions are observed:

* 93% reactor power
* Auto makeup is in progress
* Unable to add primary water [PW] via normal flowpath
* PW hi/low flow alarm

Then reactor power and Tavg will be:

a. Decreasing because of the dilution in progress.

b. Decreasing because of the boration in progress.

c. Increasing because of the dilution in progress.

d. Increasing because of the boration in progress.

QUESTION: 089 (1.00)

If steam generator pressure decreases following a reactor trip, then:

a. Increase AUX feedwater flow to the steam generators.

b. Close the steam dump header isolation valves [MS-MOV-520 and MS-MOV-553].

c. Emergency borate if Tavg decreases to 515 degrees F.

d. Trip the operating RCPs and establish natural circulation cooldown.

QUESTION: 090 (1.00)

Emergency Operating Procedure E-2, Faulted Steam Generator Isolation, Step 3., Identify Faulted SG: Action/Expected Response is:

Check pressures in all SGs - ANY SG PRESSURE DECREASING UNCONTROLLABLY

What is the EMERGENCY OPERATING PROCEDURE meaning/definition of UNCONTROLLABLY?

a. Plant AUTOMATIC ACTIONS do not control the pressure decrease

b. Not under the control of the operator and incapable of being controlled by the operator using available equipment

c. Decreasing at a rate of 50 pounds/minute

d. Decreasing at a rate of 100 pounds/minute

QUESTION: 091 (1.00)

While operating at 98% power, a reactor trip and safety injection occur due to RCS pressure decreasing. You direct the operators in the performance of E-0, Reactor Trip Or Safety Injection, and at Step 23 you correctly go to E-1, Loss Of Reactor Or Secondary Coolant. E-1 Step 2 Action is "Check Secondary Integrity". E-0 Step 21 is "Check Secondary Integrity".

What is the BASIS for the E-1 action "Check Secondary Integrity" after just having performed the identical action step in E-0?

a. To identify any faulted SGs

b. To ensure proper S/G isolation

c. Check for a possible misdiagnosis or subsequent failure

d. E-1, Loss Of Reactor Or Secondary Coolant, is entered from other procedures as well as E-0, Reactor Trip Or Safety Injection

QUESTION: 092 (1.00)

Step 18 of E-0, Reactor Trip Or Safety Injection, reads:

Check RCS Average Temperature STABLE AT OR TRENDING TO 535 degrees F.

The DEFINITION of STABLE as used in this step is:

a. Within or outside normal control band but responding as expected to a controlled condition.

b. Temperature steady within 1.5 F of the specified temperature.

c. Temperature steady within 3 F of the specified temperature.

d. Within normal control band responding to automatic or manual controls.

QUESTION: 093 (1.00)

WHICH ONE (1) of the following is the reason all reactor coolant pumps are tripped according to FR-H.1, Response To Loss Of Secondary Heat Sink?

a. To get increased safety injection flow by decreasing RCS cold leg pressure

b. To conserve reactor coolant inventory by reducing seal leak off

c. To conserve steam generator secondary inventory by reducing heat input to the RCS

d. To minimize the possibility of a tube rupture as AFW is restored to the steam generator

QUESTION: 094 (1.00)

Per ES-0.1, Reactor Trip Response, operators must actuate SI then and go to E-0, Reactor Trip Or Safety Injection, Step 1, if pressurizer level DECREASES BELOW _____

a. 5% [14% for adverse containment]

b. 10% [24% for adverse containment]

c. 15% [34% for adverse containment]

d. 20% [44% for adverse containment]

QUESTION: 095 (1.00)

RCS cooldown is in progress in accordance with E-3, Steam Generator Tube Rupture. RCPs should be tripped if RCS pressure drops to:

a. 300 psig

b. 700 psig

c. 900 psig

d. 1100 psig

QUESTION: 096 (1.00)

Which of the following losses of Residual Heat Removal System flow will result in the fastest RCS heatup rate?

a. Refueling cavity full with the RHR piping and pumps intact

b. Refueling cavity full with the RHR piping or pumps not intact

c. Reactor head removed with RHR piping and pumps intact

d. Reactor head bolted in place

QUESTION: 097 (1.00)

A reactor startup is in progress with the reactor critical. The following conditions exist:

* P31 through P34 indicate 0% power
* Wide Range Channel 21 indicates 1.5 X 10 -10 amps
* Wide Range Channel 22 indicates 1.4 X 10 -11 amps
* Wide Range Channel 4 fails

What action[s] should be taken?

a. Proceed into the Power Range after tripping high SUR bistables for Channel 4

b. TRIP the reactor

c. FULLY INSERT all control rods and OPEN the reactor trip breakers

d. BORATE the reactor to the HOT STANDBY boron concentration and ENSURE positive reactivity is not added to the core

QUESTION: 098 (1.00)

The following conditions exist:

* Refueling is in progress
* Spent fuel handling is in progress
* Vent Stack Radiation Monitor [R-14A] radiation level is increasing

What action[s] should be taken?

a. EVACUATE containment and the spent fuel building

b. PLACE all fuel elements in a secure storage location and SECURE fuel movement until the source of activity is determined

c. PLACE all fuel elements in a secure storage location and ENSURE that a positive pressure is maintained in the spent fuel building

d. SHUTDOWN the spent fuel building supply and exhaust fans

QUESTION: 099 (1.00)

Given the following conditions:

* Pressurizer level channel 1 failed yesterday and was placed in the tripped condition in accordance with technical specification 3.3.1 action 6

* The I&C Department reports that pressurizer level channel 2 failed surveillance testing and can not be repaired for approximately 72 hours [time required to receive the required repair part]

What action[s] [including maximum time allowed] should be taken?

a. Place channel 2 in the tripped condition within 6 hours

b. Be in at least HOT SHUTDOWN within 6 hours

c. Be in at least HOT STANDBY within 7 hours

d. Be in at least COLD SHUTDOWN within 24 hours

QUESTION: 100 (1.00)

Given the following indications:

* 87% power
* SEMI-VITAL NORMAL SUPPLY LOW VOLTAGE alarm
* P-18-1A and P-18-1B CH PP U/V TRIP TIMER OPERATING alarm
* LEVEL TWO BUS 8 CH FAIL LEVEL THREE alarms
* LEVEL TWO BUS 9 CH FAIL LEVEL THREE alarms

What IMMEDIATE ACTION[s] should be performed?

a. E-0, Reactor Trip Or Safety Injection immediate actions and manually start the emergency diesel generators

b. Trip all reactor coolant pumps [RCPs] and manually start the emergency generators

c. ECA 0.0, Station Blackout immediate actions

d. E-0, Reactor Trip Or Safety Injection immediate actions and trip all reactor coolant pumps [RCPs]

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

ANSWER: 001 (1.00)

a.

REFERENCE:

ACP 1.0-4
194001K103 ..(KA's)

ANSWER: 002 (1.00)

b.

REFERENCE:

ACP 1.0-5, Containment Access, Page 2
194001K114 ..(KA's)

ANSWER: 003 (1.00)

d.

REFERENCE:

ACP 1.0-5, Containment Access, page 3
194001K114 ..(KA's)

ANSWER: 004 (1.00)

b.

REFERENCE:

ACP 1.0-6, Emergency Planning Responsibility And Station On-Call Organization And Responsibilities, page 2
194001A103 ..(KA's)

ANSWER: 005 (1.00)

a.

REFERENCE:

ACP 1.0-10, Refueling Operations, page 4
194001A103 ..(KA's)

ANSWER: 006 (1.00)

b.

REFERENCE:

ACP 1.0-12, Process Computer Data Base Changes, page 2
194001A115 ..(KA's)

ANSWER: 007 (1.00)

c.

REFERENCE:

ACP 1.0-15, Cable Vault Access, page 1
194001K116 ..(KA's)

ANSWER: 008 (1.00)

a.

REFERENCE:

ACP 1.0-31, PAB / Pipe Trench Floor Block Lifting Procedure, page 2
194001K103 ..(KA's)

ANSWER: 009 (1.00)

b.

REFERENCE:

ACP 1.0-57, Plant Labeling Procedure, page 6
194001K102 ..(KA's)

ANSWER: 010 (1.00)

a.

REFERENCE:

ACP 1.0-61, High Risk Testing, page 2
194001A102 ..(KA's)

ANSWER: 011 (1.00)

d.

REFERENCE:

ACP 1.0-62, Access An Work Control For PBX Room, page 3
194001A104 ..(KA's)

ANSWER: 012 (1.00)

d.

REFERENCE:

ACP 1.2-16.1, Plant Information Reports, pages 5 & 7
194001A106 ..(KA's)

ANSWER: 013 (1.00)

c.

REFERENCE:

ACP 1.2-14..1, Jumper, Lifted Lead, and Bypass Control (NOP-3.04), page 6
194001K102 ..(KA's)

ANSWER: 014 (1.00)

b.

REFERENCE:

ACP 1.2-14.2, Equipment Tagging, pages 9 & 10
194001K102 ..(KA's)

ANSWER: 015 (1.00)

d.

REFERENCE:

ODI #151, Double Valve Isolation, page 1
194001K101 ..(KA's)

ANSWER: 016 (1.00)

b.

REFERENCE:

EPIP 1.5-43, Personnel Radiation Exposure Control During Nuclear Emergencies, page 4
194001A116 ..(KA's)

ANSWER: 017 (1.00)

a.

REFERENCE:

EPIP 1.5-26, Manager of Control Operations, page 2
194001A116 ..(KA's)

ANSWER: 018 (1.00)

c.

REFERENCE:

Plant Information Book, Chapter 80, Rod Control System, para 3.2.1.4, page 26
001000K105 ..(KA's)

ANSWER: 019 (1.00)

d.

REFERENCE:

Plant Information Book, Chapter 80, Rod Control System, para 3.3.1, page 52
001000K203 ..(KA's)

ANSWER: 020 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 80, Rod Control System, para 3.2.6.2, page 41
001000K403 ..(KA's)

ANSWER: 021 (1.00)

c.

REFERENCE:

Connecticut Yankee Standard Technical Specifications, page B3/4 1-3
014000G006 ..(KA's)

ANSWER: 022 (1.00)

c.

REFERENCE:

Plant Information Book, Chapter 2.2, Reactor Coolant Pump, para 4.2.2, page 69
003000G007 ..(KA's)

ANSWER: 023 (1.00)

d.

REFERENCE:

Plant Information Book, Chapter 2.2, Reactor Coolant Pump, para 3.5.1.5, page 54
003000K604 ..(KA's)

ANSWER: 024 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 4, Chemical and Volume Control System, para 5.5, page 112
063000K302 ..(KA's)

ANSWER: 025 (1.00)

c.

REFERENCE:

Plant Information Book, Chapter 4, Chemical and Volume Control System, para 3.35.3 & 3.35.4, page 54
004010A305 ..(KA's)

ANSWER: 026 (1.00)

b.

REFERENCE:

Plant Information Book, Chapter 4, Chemical and Volume Control System, para 4.3.4, page 107
004010A403 ..(KA's)

ANSWER: 027 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 5, Emergency Core Cooling System, para 3.8.1.2, page 96, and Figure 7.14, page 154
013000K112 ..(KA's)

ANSWER: 028 (1.00)

b.

REFERENCE:

Plant Information Book, Chapter 5, Emergency Core Cooling System, para 3.8.2.2, page 101
013000A302 ..(KA's)

ANSWER: 029 (1.00)

c.

REFERENCE:

Connecticut Yankee Operator Training written examination dated 12/08/89, question 1.0005
015000K408 ..(KA's)

ANSWER: 030 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 77, Incore Instrumentation, para 4.2, page 55
017020A302 ..(KA's)

ANSWER: 031 (1.00)

c.

REFERENCE:

Plant Information Book, Chapter 56, Containment Ventilation, para 5.1, page 32
022000A204 ..(KA's)

ANSWER: 032 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 5, Emergency Core Cooling System, para 3.6.2.1, page 86
026000A101 ..(KA's)

ANSWER: 033 (1.00)

b.

REFERENCE:

Plant Information Book, Chapter 19, Feedwater System, para 5.3, page 29
059000A211 ..(KA's)

ANSWER: 034 (1.00)

b.

REFERENCE:

Plant Information Book, Chapter 14, Waste Gas System, para 4.1.4, page 47
071000K405 ..(KA's)

ANSWER: 035 (1.00)

d.

REFERENCE:

Plant Information Book, Chapter 82, Radiation Monitoring Syste ,
para 3.2.1, page 102
072000A401 ..(KA's)

ANSWER: 036 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 12, Liquid Waste System, para 5.1, page 67
068000A302 ..(KA's)

ANSWER: 037 (1.00)

d.

REFERENCE:

Plant Information Book, Chapter 2.4, Pressurizer and Pressure Relief System, para 4.4.3, page 50
002000G006 ..(KA's)

ANSWER: 038 (1.00)

b.

REFERENCE:

Plant Information Book, Chapter 2.4, Pressurizer and Pressurizer Relief, para 3.16.2, page 41
002000K403 ..(KA's)

ANSWER: 039 (1.00)

b.

REFERENCE:

Plant Information Book, Chapter 2.5, Pressurizer Pressure Control, para 3.7.2, page 31
010000K103 ..(KA's)

ANSWER: 040 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 2.5, Pressurizer Pressure Control, para 3.3.3, pages 17 & 18
010000G007 ..(KA's)

ANSWER: 041 (1.00)

c.

REFERENCE:

Plant Information Book, Chapter 2.6, Pressurizer Level Control, para 3.6.3, page 22
011000K301 ..(KA's)

ANSWER: 042 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 2.6, Pressurizer Level Control, para 3.3.3, page 16
011000K603 ..(KA's)

ANSWER: 043 (1.00)

c.

REFERENCE:

Plant Information Book, Chapter 5, Emergency Core Cooling System, para 3.2.1.3, pages 25 & 26
006000K201 ..(KA's)

ANSWER: 044 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 5, Emergency Core Cooling System, para 5.4, page 124
006000A208 ..(KA's)

ANSWER: 045 (1.00)

b.

REFERENCE:

Standard Technical Specification Bases
012000K501 ..(KA's)

ANSWER: 046 (1.00)

b.

REFERENCE:

Plant Information Book, Chapter 75, Reactor Protection System, para 3.3.7.2, page 83
012000K610 ..(KA's)

ANSWER: 047 (1.00)

c.

REFERENCE:

Plant Information Book, Chapter 56, Containment Ventilation, para 3.5.2.4, page 26
029000G009 ..(KA's)

ANSWER: 048 (1.00)

d.

REFERENCE:

Plant Information Book, Chapter 82, Radiation Monitoring System
073000K401 ..(KA's)

ANSWER: 049 (1.00)

c.

REFERENCE:

Standard Technical Specifications, page B3/4 9-2
034000G006 ..(KA's)

ANSWER: 050 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 64, 4 KV System, para 3.8..4, page 64
062000A305 ..(KA's)

ANSWER: 051 (1.00)

c.

REFERENCE:

Plant Information Book, Chapter 45, Spent Fuel Storage and Cooling, para 5.2, page 17
033000A202 ..(KA's)

ANSWER: 052 (1.00)

b.

REFERENCE:

Plant Information Book, Chapter 72, Emergency Diesel Generator, para 5.1, page 112
064000A406 ..(KA's)

ANSWER: 053 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 16, Main Steam System, para 3.6.3.4, page 37
039000A106 ..(KA's)

ANSWER: 054 (1.00)

a.

REFERENCE:

Plant Information Book, Chapter 6, Residual Heat Removal System, para 3.11.1 and 3.11..3, pages 29 & 30
005000G007 ..(KA's)

ANSWER: 055 (1.00)

d.

REFERENCE:

Plant Information Book, Chapter 44, Component Cooling Water System, para 5.1, page 54
008000K101 ..(KA's)

ANSWER: 056 (1.00)

c.

REFERENCE:

Plant Information Book. Chapter 17, High Pressure Steam Dump, para 2.3, page 5
041020K417 ..(KA's)

ANSWER: 057 (1.00)

b.

REFERENCE:

Plant Information Book, Chapter 35, Control Air, para 3.7.3.6, page 31
078000K402 ..(KA's)

ANSWER: 058 (1.00)

a.

REFERENCE:

CY-OP-LO-AOP-L13, Rod Control Malfunctions, Objective 16, page 40
000001K204 ..(KA's)

ANSWER: 059 (1.00)

c.

REFERENCE:

AOP 3.2-23, Malfunction of Rod Control System, 4.1.1, page 3
000003K304 ..(KA's)

ANSWER: 060 (1.00)

d.

REFERENCE:

AOP 3.2-23, Malfunction of Rod Control System, para 4.1.5, page 3
000003G001 ..(KA's)

ANSWER: 061 (1.00)

c.

REFERENCE:

Technical Specification 3.10.2 Action a.
000005G008 ..(KA's)

ANSWER: 062 (1.00)

d.

REFERENCE:

E-O Series Procedures Foldout
000011A103 ..(KA's)

ANSWER: 063 (1.00)

b.

REFERENCE:

E-O Series Continuous Action Page
000011A104 ..(KA's)

ANSWER: 064 (1.00)

c.

REFERENCE:

Technical Specification 3.4.7, Main Coolant System Specific Activity
000076A104 ..(KA's)

ANSWER: 065 (1.00)

b.

REFERENCE:

Technical Specification 3.4.8 Bases, Specific Activity, page B3/4 4-8
000076G004 ..(KA's)

ANSWER: 066 (1.00)

b.

REFERENCE:

E-1, Loss Of Reactor Or Secondary Coolant Basis
000040K304 ..(KA's)

ANSWER: 067 (1.00)

c.

REFERENCE:

E-2, Faulted Steam Generator Isolation Basis
000040K303 ..(KA's)

ANSWER: 068 (1.00)

c.

REFERENCE:

E-0 Reactor Scram Or Safety Injection Basis
000040K302 ..(KA's)

ANSWER: 069 (1.00)

b.

REFERENCE:

Emergency Operating Procedure E-1, Loss Of Reactor Or Secondary Coolant
000009A234 ..(KA's)

ANSWER: 070 (1.00)

c.

REFERENCE:

FR-S.1, Response To Nuclear Power Generation/ATWS
000029A101 ..(KA's)

ANSWER: 071 (1.00)

b.

REFERENCE:

ECA-0.0, Loss Of All AC Power, Step 3
000055A203 ..(KA's)

ANSWER: 072 (1.00)

a.

REFERENCE:

Functional Restoration Procedure, FR-C.1, Response To Inadequate Core Cooling Basis
000074K302 ..(KA's)

ANSWER: 073 (1.00)

a.

REFERENCE:

ECA-0.0, Loss Of All AC Power, Basis
000055K302 ..(KA's)

ANSWER: 074 (1.00)

d.

REFERENCE:

Emergency Operating Procedure, FR-C.1, Response To Inadequate Core Cooling
000074K103 ..(KA's)

ANSWER: 075 (1.00)

d.

REFERENCE:

Emergency Operating Procedure E-1, Loss Of Reactor Or Secondary Coolant, Foldout For E-1 Procedures
000069G011 ..(KA's)

ANSWER: 076 (1.00)

b.

REFERENCE:

Emergency Response Procedures Addendum, page 5
000024K301 ..(KA's)

ANSWER: 077 (1.00)

d.

REFERENCE:

AOP 3.2-10, Loss Of Component Cooling Water
000026G011 ..(KA's)

ANSWER: 078 (1.00)

b.

REFERENCE:

AOP 3.2-50, Plant Operations Outside the Control Room
000068K318 ..(KA's)

ANSWER: 079 (1.00)

a.

REFERENCE:

AOP 3.2-50, Plant Operations Outside the Control Room
000068A121 ..(KA's)

ANSWER: 080 (1.00)

a.

REFERENCE:

CY-OP-L0-AOP-L1402, RCP Trip At Power, Enabling Objective 10, page 19
000015K102 ..(KA's)

ANSWER: 081 (1.00)

d.

REFERENCE:

EPIP, Attachment 12.4, Reportable Liquid Or Gaseous Releases
000059G002 ..(KA's)

ANSWER: 082 (1.00)

A.

REFERENCE:

EOP 3.1-34, Complete Loss of Control Air Supply
000065G010 ..(KA's)

ANSWER: 083 (1.00)

d.

REFERENCE:

EOP 3.1-34, Complete Loss of Control Air Supply
000065A208 ..(KA's)

ANSWER: 084 (1.00)

b.

REFERENCE:

EOP 3.1-49, Partial Loss of DC, para 1.2, page 1
000058K302 ..(KA's)

ANSWER: 085 (1.00)

c.

REFERENCE:

EOP 3.1-49, Partial Loss of DC, para 1.8, page 2
000058A203 ..(KA's)

ANSWER: 086 (1.00)

a.

REFERENCE:

Emergency Operating Procedure E-3, Steam Generator Tube Rupture, Bases
000038K306 ..(KA's)

ANSWER: 087 (1.00)

d.

REFERENCE:

EOP 3.1-0, Emergency Response Procedures, Attachment 1, Emergency Response Procedures Addendum
000038K103 ..(KA's)

ANSWER: 088 (1.00)

b.

REFERENCE:

CY-OP-LO-AOP-L2201, Automatic Makeup System Malfunctions, page 13
000022G011 ..(KA's)

ANSWER: 089 (1.00)

b.

REFERENCE:

EOP 3.1-0, Emergency Response Procedures, Attachment 1, Emergency Response Procedures Addendum, page 5
000007A110 ..(KA's)

ANSWER: 090 (1.00)

b.

REFERENCE:

Emergency Operating Procedure E-2, Faulted Steam Generator Isolation Basis
000040A106 ..(KA's)

ANSWER: 091 (1.00)

c.

REFERENCE:

Emergency Operating Procedure E-1, Loss Of Reactor Or Secondary Coolant Basis
000009K321 ..(KA's)

ANSWER: 092 (1.00)

a.

REFERENCE:

Westinghouse EOP E-0 Basis
000007K301 ..(KA's)

ANSWER: 093 (1.00)

c.

REFERENCE:

Westinghouse EOP Basis FR-H.1, Page 8
000054K304 ..(KA's)

ANSWER: 094 (1.00)

b.

REFERENCE:

Foldout For E-0 Series Procedures
000007G011 ..(KA's)

ANSWER: 095 (1.00)

a.

REFERENCE:

E-3, Steam Generator Tube Rupture, Page 9
000038A217 ..(KA's)

ANSWER: 096 (1.00)

c.

REFERENCE:

AOP 3.2-12, Loss of Residual Heat Removal System, para 1.2, page 1
000025K101 ..(KA's)

ANSWER: 097 (1.00)

a.

REFERENCE:

AOP 3.2-14, Nuclear Instrumentation Malfunction, para 4.3.1, page 3
000032A204 ..(KA's)

ANSWER: 098 (1.00)

b.

REFERENCE:

AOP 3.2-3, High Air Activity Level In SFB Exhaust, para 4.1, page 1
000036G011 ..(KA's)

ANSWER: 099 (1.00)

c.

REFERENCE:

Technical Specifications 3.0.3 and 3.3.1
000028G003 ..(KA's)

ANSWER: 100 (1.00)

d.

REFERENCE:

EOP 3.1-46, Total Loss Of Semi Vital Power, para 4.1 & 4.2, page 3
000056G010 ..(KA's)

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

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

001	a	b	c	d	______
002	a	b	c	d	______
003	a	b	c	d	______
004	a	b	c	d	______
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	______
023	a	b	c	d	______
024	a	b	c	d	______
025	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.

026	a	b	c	d	______
027	a	b	c	d	______
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	______
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	a	b	c	d	______
048	a	b	c	d	______
049	a	b	c	d	______
050	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.

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	____
058	a	b	c	d	____
059	a	b	c	d	____
060	a	b	c	d	____
061	a	b	c	d	____
062	a	b	c	d	____
063	a	b	c	d	____
064	a	b	c		____
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	____

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

076	a	b	c	d	______
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	______
082	a	b	c	d	______
083	a	b	c	d	______
084	a	b	c	d	______
085	a	b	c	d	______
086	a	b	c	d	______
087	a	b	c	d	______
088	a	b	c	d	______
089	a	b	c	d	______
090	a	b	c	d	______
091	a	b	c	d	______
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	______
098	a	b	c	d	______
099	a	b	c	d	______
100	a	b	c	d	______

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

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

A N S W E R   K E Y

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

ANSWER KEY

026	b
~~027~~	~~a~~ delete SHB 6/19/90
028	b
029	c
030	a
031	c
032	a
033	b
034	b
035	d
036	a
037	d
038	b
039	b
040	a
041	c
042	a
043	c
044	a
045	b
046	b
047	c
048	d
049	c
050	a

ANSWER KEY

051	c
052	b
053	a
054	a
055	d
056	c
057	b
058	a
059	c
060	d
061	c
062	d
063	b
064	c
065	b
~~066~~	~~b~~ delete JHB 6/19/90
067	c
068	c
069	b
070	c
071	b
072	a
073	a
074	d
075	d

A N S W E R K E Y

076	b
077	d
078	b
079	~~a~~ b typo error SHB 6/19/90
080	a
081	d
082	A
083	d
084	b
085	c
086	a
087	d
088	b
089	b
090	b
091	c
092	a
093	c
094	b
095	a
096	c
097	a
098	b
099	c
100	d

ANSWER KEY

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

QUESTION	VALUE	REFERENCE
001	1.00	9000001
002	1.00	9000002
003	1.00	9000003
004	1.00	9000004
005	1.00	9000005
006	1.00	9000006
007	1.00	9000007
008	1.00	9000008
009	1.00	9000009
010	1.00	9000010
011	1.00	9000011
012	1.00	9000012
013	1.00	9000013
014	1.00	9000014
015	1.00	9000015
016	1.00	9000016
017	1.00	9000017
018	1.00	9000018
019	1.00	9000019
020	1.00	9000020
021	1.00	9000021
022	1.00	9000022
023	1.00	9000023
024	1.00	9000024
025	1.00	9000025
026	1.00	9000026
027	1.00	9000027
028	1.00	9000028
029	1.00	9000029
030	1.00	9000030
031	1.00	9000031
032	1.00	9000032
033	1.00	9000033
034	1.00	9000034
035	1.00	9000035
036	1.00	9000036
037	1.00	9000037
038	1.00	9000038
039	1.00	9000039
040	1.00	9000040
041	1.00	9000041
042	1.00	9000042
043	1.00	9000043
044	1.00	9000044
045	1.00	9000045
046	1.00	9000046
047	1.00	9000047
048	1.00	9000048
049	1.00	9000049
050	1.00	9000050
051	1.00	9000051
052	1.00	9000052
053	1.00	9000053
054	1.00	9000054

QUESTION	VALUE	REFERENCE
055	1.00	9000055
056	1.00	9000056
057	1.00	9000057
058	1.00	9000058
059	1.00	9000059
060	1.00	9000060
061	1.00	9000061
062	1.00	9000062
063	1.00	9000063
064	1.00	9000064
065	1.00	9000065
066	1.00	9000066
067	1.00	9000067
068	1.00	9000068
069	1.00	9000069
070	1.00	9000070
071	1.00	9000071
072	1.00	9000072
073	1.00	9000073
074	1.00	9000074
075	1.00	9000075
076	1.00	9000076
077	1.00	9000077
078	1.00	9000078
079	1.00	9000079
080	1.00	9000080
081	1.00	9000081
082	1.00	9000082
083	1.00	9000083
084	1.00	9000084
085	1.00	9000085
086	1.00	9000086
087	1.00	9000087
088	1.00	9000088
089	1.00	9000089
090	1.00	9000090
091	1.00	9000091
092	1.00	9000092
093	1.00	9000093
094	1.00	9000094
095	1.00	9000095
096	1.00	9000096
097	1.00	9000097
098	1.00	9000098
099	1.00	9000099
100	1.00	9000100

	100.00	

	100.00	

ATTACHMENT 2

NRC Response to Facility Comments

Question No.: No. 27

Licensee Comment: Exam question infers tie between the initiation relay and the low voltage interlock. The interlock is actually labelled low voltage accident interlock on prints and more commonly is called "Level one" undervoltage. Candidate spent 45 minutes trying to find a tie on prints where none existed. Candidate requested clarification but examiner did not provide any. Haddam Neck recommends question be deleted from the exam.

NRC Response: Disagree with licensee comment requesting deletion of question. Candidate incorrectly interpreted the stem of the question.

Question No.: No. 38

Licensee Comment: The pressurizer vent valves are only used in accident conditions with DSEO approval. Question design was from a single failure design standpoint and not from an operational/procedural standpoint. Haddam Neck recommends question be deleted or credit be given for D.

NRC Response: Disagree with licensee comment. Certain design questions are considered within the knowledge level expected of an SRO.

Question No.: No. 66

Licensee Comment: Question doesn't constrain Reactor Coolant Pump operation. Since CY strips 2 RCPs on Bus Transfer and if one assumes the remaining two are operating - this eliminates B as correct answer. This leaves C as the best answer remaining. Haddam Neck recommends question be deleted from the exam or credit be given for either answer.

NRC Response: Agree with licensee comment. Credit will be given for answer C.

Question No.: No. 79

Licensee Comment: Question doesn't imply distinct functions of the three operators or timeliness in which they must be done. Answers A or B are both correct. Haddam Neck recommends question be deleted from the exam or credit be given for either answer.

NRC Response: Answer key is incorrect. The correct answer to Question No. 79 is B. Credit will be given.

ATTACHMENT 3

REQUALIFICATION EXAMINATION TEST ITEMS

1.0 SIMULATOR EXAMINATION

SCENARIO NOs.

AE S009
AE S016
AE S017
AE S023

2.1 WRITTEN EXAMINATION - PART A

STATIC SIMULATOR SCENARIO - RO MINOR - RO:900019

QUESTION NUMBER	QUESTION VALUE
002682	1.2
002686	2.0
002687	1.7
002689	2.0
002694	2.1
002693	1.0
003460	1.6
002696	2.0
002701	2.3
003458	1.1
003459	1.4
002695	1.4
002683	1.9
002685	2.0
002688	1.3

STATIC SIMULATOR SCENARIO - SRO MINOR - RO:900020

QUESTION NUMBER	QUESTION VALUE
002683	1.7
002684	2.2
002685	1.9
002688	1.2
002690	2.2
002694	2.0
002694	1.3
002696	1.9
002700	1.9
002702	1.6
002703	1.9
003459	1.3

STATIC SIMULATOR SCENARIO - SRO MINOR (CON'T) - RO:900020

002682	1.1
002686	1.9
002693	0.9

STATIC SIMULATOR SCENARIO - RO MAJOR - RO:900017

QUESTION NUMBER	QUESTION VALUE
002552	1.4
002554	1.2
002557	1.5
002558	1.9
002560	1.6
002561	1.2
002565	2.0
002677	1.9
002551	1.6
002556	1.1
002566	2.3
002678	1.6
002553	2.2
002559	1.9
002562	1.6

STATIC SIMULATOR SCENARIO - SRO MAJOR - RO:900018

QUESTION NUMBER	QUESTION VALUE
002553	2.0
002554	1.2
002551	1.6
002559	1.9
002562	1.6
002563	1.6
002566	2.2
002676	2.1
002678	1.6
002681	2.1
002564	1.4
002677	1.9
002557	1.5
002561	1.2
002556	1.1

2.2 WRITTEN EXAMINATION - PART B - REACTOR OPERATOR

QUESTION NUMBER	QUESTION VALUE
003207	1.9
003409	1.8
002955	1.9
003427	2.2
002729	2.8
003162	2.1
003429	2.0
003175	2.2
003164	2.0
003348	1.9
003311	2.9
003370	1.9
002908	2.7
003155	1.9
004339	1.8
003404	3.0
003310	2.9
003385	2.4
002736	2.3
002949	2.6
002738	2.6
002746	2.2

2.3 WRITTEN EXAMINATION - PART B - SENIOR REACTOR OPERATOR

QUESTION NUMBER	QUESTION VALUE
003207	2.3
003409	1.6
002955	1.7
003427	1.9
002729	2.5
003162	1.9
003429	1.8
003359	2.7
003250	2.1
003368	1.7
003311	2.6
003370	1.7
003293	2.6
003376	2.4
003266	1.6
004356	2.2

2.3 WRITTEN EXAMINATION - PART B - SENIOR REACTOR OPERATOR (CONT'D.)

QUESTION NUMBER	QUESTION VALUE
003404	2.7
003310	2.6
002747	2.9
002949	2.3
003183	2.1
003218	2.0
003385	2.1

3.0 JOB PERFORMANCE MEASURES (JPM)

JPM NUMBER	JPM TASK	LOCATION
1	TRIP REACTOR/TRIP TURBINE OUTSIDE C/R	IN-PLANT
2	MANUAL START EDG & ENERGIZE BUS 9	IN-PLANT
3	MAINTAIN SECONDARY HEAT SINK-LOCAL S/G LEVEL CONTROL	IN-PLANT
5	LOCAL COOLDOWN OUTSIDE THE CONTROL ROOM	IN-PLANT
8	LOCAL-MANUAL OPERATION OF AFW PUMPS	IN-PLANT
26	EMERGENCY BORATION OF THE RCS	CONTROL ROOM
27	OPERATE RHR STOP VALVES AND DISCONNECT SWITCHES	CONTROL ROOM
30	PERFORM A DILUTION OF THE RCS	CONTROL ROOM
46	RESET SI WITH HIGH CONTAINMENT PRESSURE	CONTROL ROOM
52	POWER RANGE CHANNEL CALIBRATION	CONTROL ROOM
64	OPERATE CAR FANS DURING LOCA-ONE DIESEL	CONTROL ROOM
65	MANUALLY INITIATE CONTAINMENT SPRAY	CONTROL ROOM
92	MANUALLY INITIATE SI & PERFORM E-0 IMMEDIATE OPERATOR ACTIONS	CONTROL ROOM
127	RESET AUTO AUXILIARY FEED	CONTROL ROOM
128	LOCALLY DEENERGIZE 4160 BUSES	IN-PLANT
134	MONITOR THE INADEQUATE CORE COOLING SYSTEM LOCALLY	CONTROL ROOM
135	LOCAL OPERATION OF THE METERING PUMP	IN-PLANT