U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-483/OL-91-01

Docket No. 50-483 License No. NPF-30

Licensee: Union Electric Company
Post Office Box 149
Mail Code 400
St. Louis, MO 63166

Facility Name: Callaway Nuclear Plant

Examination Administered At: Callaway Nuclear Plant

Examination Conducted: Week of January 28, 1991

Chief Examiner: Kris Shembarger 2/11/91
Kris Shembarger Date

Approved By: Thomas M Burdick 2/11/91
Thomas M. Burdick, Chief Date
Operator Licensing Section 2

Examination Summary

Examination administered during the week of January 28, 1991
(Report No. 50-483/OL-91-01(DRS))

Written and operating examinations were administered to five senior reactor operator and four reactor operator candidates.

Results: Five senior reactor operator and three reactor operator candidates passed the examinations. One reactor operator passed the operating examination, but failed the written examination.

REPORT DETAILS

1. Exit Meeting

a. On February 1, 1991, an exit meeting was held. The following personnel were present at the meeting:

D. Heinlein, Assistant Superintendent, Operations
E. B. Stewart, Operating Supervisor, Training
S. M. Halverson, Senior Training Supervisor, Simulator
S. V. Henderson II, Operating Supervisor, Training
P. J. McKenna, Training
D. W. Neterer, Senior Training Supervisor, Operations
K. Mills, Quality Assurance
J. D. Blosser, Manager, Callaway Plant
C. D. Naslund, Manager, Operations Support
M. S. Evans, Superintendent, Training
S. Johnson, NRC Examiner
B. Steinke, NRC Examiner
P. Isaksen, NRC Examiner
K. Shembarger, NRC Examiner

b. The following general observations were made by the examination team and were discussed with the utility:

(1) Pre-exam review at Region III was thorough, as evidenced by the low number of post-exam comments.

(2) Facility support during the operating examination was very good.

(3) A conflict arose during the week regarding facility observers during administration of the operating test. Arrangements had not been made with the Chief Examiner prior to the exam regarding observers and therefore observers were not allowed. The facility is encouraged to discuss all exam related issues with the Chief Examiner prior to exam administration.

(4) The Steam Tables provided by the facility to be used by the candidates during the written exam contained markings. The facility should ensure all materials provided to the candidates for use during the exam do not pose an exam compromise issue.

(5) The facility is encouraged to be more discrete during pre-exam JPM validation in the plant to prevent exam compromise.

(6) While implementing the EOPs during the operating examination, the SRO candidates stopped implementation of the EOPs prior to transitioning out of E-0 to classify the event and to request the BOP operator to monitor the CSF status trees. The facility was advised that it is acceptable to classify the event after scenario termination and to request an additional operator to monitor the CSF status trees.

(7) During the simulator operating examination, both an SRO and an RO candidate went behind the control board panels, leaving only one RO "at the controls".

(8) A copy of ECA-0.0, Attachment 2, "Locally Starting Emergency Diesel Generators" is not located locally by the Diesel Generators.

(9) Facility Lesson Plan No. 3 on the diesel generators states that the day tank high level alarm is received when the diesel generator is in operation. The alarm response procedure states that the alarm is automatically bypassed when the diesel generator is in operation.

(10) One candidate consistently reached over a Contaminated Area boundary and leaned on a railing by the spent fuel pool with a Contaminated Area sign on it during the walkthrough exam.

(11) Facility JPM's and Lesson Plans are not being maintained current with procedure revisions.

(12) The resemblance between the simulator and the control room is very good.

c. Generic Strength

The following generic strength was identified by the examiners and discussed with the facility:

(1) Performance of JPM's both inside the plant and on the simulator was very good, particularly in the area of review of procedure precautions and limitations.

d. Generic Weaknesses

The following generic weaknesses were identified by the examiners and discussed with the facility:

(1) Knowledge of radiation limits during emergency operations and guidelines for the selection of individuals to make emergency entries.

(2) Use of hand friskers

(3) Knowledge of Technical Specification requirements during a reactor startup and power escalation to 100% power with one of four power range NI channels out of service.

(4) Knowledge of the guidelines for AFW isolation to a ruptured steam generator as stated in the Standing Order describing the Callaway Emergency Operating Procedures Usage Policy.

(5) Lack of alarm response procedure usage during the simulator operating examination.

(6) During the simulator operating examination, candidates silenced annunciators without first identifying the window that alarmed.

2. Written Examination

a. The following generic weaknesses, common to both the RO and SRO exams, were identified (i.e. greater than or equal to 50% of RO/SRO candidates failed to identify the correct answer).

(1) Bases for maintaining VCT pressure greater than or equal to 15 psig prior to starting a reactor coolant pump.

(2) The plant conditions that would result in RHR pump vortexing with the plant in Mode 6.

b. The following generic weaknesses in the SRO candidates were identified (i.e. greater than or equal 50% of SRO candidates failed to identify the correct answer):

(1) Design of Hydrogen Control System for maintaining containment hydrogen concentrations.

(2) Worst case plant operating status prior to the occurrence of a Main Steam Line Break accident.

c. The following generic weaknesses in the RO candidates were identified (i.e. greater than or equal to 50% of RO candidates failed to identify the correct answer):

(1) The component in the Control Rod Drive System Bank Overlap Unit that maintains the correct overlap of control banks and moves either group 1 or 2.

(2) Identifying when DNB limit has been exceeded and determining the required action.

(3) Determine the reactivity added by a cooldown associated with a Main Steam Line Break.

ENCLOSURE 2

FACILITY COMMENTS AND NRC

RESOLUTION OF COMMENTS

QUESTION 023 on SRO/023 on RO:

The following plant conditions exists:

-Mode 3 following a reactor trip/Safety injection
-RCS temperature 500 degrees
-PZR pressure 1900 psig
-Steam Generator Pressure 600 psig
-Steam Generator Levels 50% wide range
-Containment pressure 3.2 psig

Which ONE of the following describes the status of the Safety Actuation Signal Logic assuming NO operator actions were taken?

a. A low pressurizer pressure and low steam pressure signal are active.

b. A low pressurizer pressure signal is present but can be blocked.

c. A low steam generator pressure signal is present but can be blocked.

d. A containment pressure signal is active but can't be blocked.

ANSWER 023 on SRO/023 on RO:

c. A low steam generator pressure signal is present but can be blocked.

REFERENCE 023 on SRO/023 on RO:

1. SNUPPS Funct. Diagram, 7250D64, Sheet 7, 8

2. Technical Specifications, Section 3.3, Table 3.3-4 E-0, Step 4.

Callaway Comment:

At a containment pressure of 1.5 psig, Environmental Allowance Modifier (EAM) will activate causing the S/G level Rx Trip setpoint to change from 14.8% to 20.2% narrow range. This EAM signal remains locked in until containment pressure drops below the reset point of less than 1.5 psig and must be reset by I&C.

Callaway Reference:

SNUPPS Funct. Diagram, 7250D64, Sheet 19
OTA-RL-RK108, Window 1082

Callaway Recommendation:

Accept answers c. and d. as correct.

NRC Resolution:

Comment accepted. The answer key was revised to reflect both c and d as correct answers.

QUESTION 038 on SRO/040 on RO:

Control of Pressurizer Heater Group A has been transferred to the Auxiliary Shutdown Panel and the control switch has been placed in the closed position.

Which ONE of the following conditions will trip the supply breaker (PG2101) for this group of heaters?

a. Pressurizer low level (17%)

b. Lockout on NB0106

c. NB01 Undervoltage

d. Safety Injection Signal

Answer 038 on SRO/040 on RO:

b. Lockout on NB0106

Reference 038 onSRO/040 on RO:

1. Facility Exam Bank Question No. 69 SBB-02PF-04C

2. LP B-4 T61.003B.6, PRZR Pressure/Level, P. 8 Obj.

Callaway Comment:

This question was taken directly from Section B of the RO/SRO Requalification Exam Bank. This exam bank was provided as a supplemental mailing to be used as a reference for writing the replacement exam. The operators are not required to memorize ALL trips for ALL breakers at Callaway. This question should be used only if the associated electrical drawing is supplied with the question.

Callaway Reference:

None

Callaway Recommendation:

Delete this question.

NRC Resolution:

Comment accepted. The question was deleted from both the RO and SRO examinations.

Question 092 on SRO:

Feedwater Flow to a Steam Generator that has a leaking or ruptured tube is not isolated in procedure E-3 (STEAM GENERATOR TUBE RUPTURE) or OTO-BB-00001 (STEAM GENERATOR TUBE LEAK) until level is greater than 4% in the narrow range. This is done to provide:

a. indication for inventory control.

b. a minimum radiological release to the environment.

c. a thermal layer to maintain Steam Generator pressure.

d. cooling initially which lowers Steam Generator pressure.

Answer 092 on SRO:

c. a thermal layer to maintain Steam Generator pressure.

Reference 092 on SRO:

1. LP D-5 T61.003D.6, E-3, OBJ.F P. 41

2. OTO-BB-00001, STEAM GENERATOR tube leak, STEP 6.6.6

Callaway Comment:

Westinghouse Background information and our lesson plans also include radiolo[illegible] release concerns along with the thermal stratification concern as rea[illegible] for filling the S/G above the tubes. This information is located in LP D-5 Pg 42.

Callaway Reference:

LP D-5, pg 42.

Callaway Recommendation:

Accept answers B. and C. as correct.

NRC Resolution:

Comment accepted. The answer key was revised to reflect both b. and c. as correct answers.

SIMULATION FACILITY REPORT

Facility Licensee: Callaway

Facility Licensee Docket No. 50-483

Operating Tests Administered On: January 29, 1991

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

ITEM	DESCRIPTION
Control Boards	The Control Boards lock-up due to dryness.
Seal Water High Temperature Alarm	The Seal Water High Temperature Alarm annunciates when charging is swapped back to the VCT.
Instrument Bus	With an Instrument Bus failure, the corresponding bistables will not trip.
"A" RCP Lift Oil Pump	When starting/stopping the "A" RCP Lift Oil Pump, the "B" and "C" RCP Lift Oil Pumps start/stop.

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION
REGION 3

FACILITY: Callaway

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 91/01/28

CANDIDATE: ____________

INSTRUCTIONS TO CANDIDATE:

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

NUMBER QUESTIONS	TOTAL POINTS	CANDIDATE'S POINTS	CANDIDATE'S OVERALL GRADE (%)
~~100~~ 99	~~100.00~~ 99.00		
KMS 2/4/91	KMS 2/4/91		

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

____
Candidate's Signature

MASTER COPY

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.
2. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done after you complete the examination.
3. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.
4. Use black ink or dark pencil only to facilitate legible reproductions.
5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet.
6. Fill in the date on the cover sheet of the examination (if necessary).
7. You may write your answers on the examination question page or on a separate sheet of paper. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.
8. If you write your answers on the examination question page and you need more space to answer a specific question, use a separate sheet of the paper provided and insert it directly after the specific question. DO NOT WRITE ON THE BACK SIDE OF THE EXAMINATION QUESTION PAGE.
9. Print your name in the upper right-hand corner of the first page of answer sheets whether you use the examination question pages or separate sheets of paper. Initial each of the following answer pages.
10. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.
11. If you are using separate sheets, number each answer and skip at least 3 lines between answers to allow space for grading.
12. Write "Last Page" on the last answer sheet.
13. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

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

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

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

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

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

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

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

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

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

DATA SHEET

REACTOR THEORY FORMULAS:

$$P = P_0 e^{t/\tau}$$

$$P = \frac{\Sigma \phi_{th} V}{3.12 \times 10^{10} \text{ fissions/sec}}$$

$$P_{th} = \frac{1}{1 + (B^2 L_{th}^2)} = e^{-(B^2 L_{th}^2)}$$

$$P_f = e^{-(B^2 L_f^2)}$$

$$p = e^{-[N][I_{eff}]/\xi\Sigma_s}$$

$$C_1 (1-K_{eff1}) = C_2 (1-K_{eff2})$$

$$m = \frac{1}{1-K} = \frac{C_{final}}{C_{initial}}$$

$$\alpha_T = \frac{1}{f}\frac{\Delta f}{\Delta t} + \frac{1}{p}\frac{\Delta p}{\Delta t} - B^2 \left(\frac{\Delta L_f^2}{\Delta t} + \frac{\Delta L_{th}^2}{\Delta t}\right)$$

$$K_{eff} = \epsilon P_f p P_{th} f \eta$$

$$P = P_0 10^{SUR(t)}$$

$$SUR = 26.06/\tau$$

$$\rho = \frac{l^*}{\tau} + \frac{\overline{\beta}_{eff}}{1 + \lambda\tau}$$

$$\rho = \frac{K - 1}{K}$$

$$\Delta\rho = \ln \frac{K_{final}}{K_{initial}}$$

$$\tau = \frac{\overline{\beta}_{eff} - \rho}{\lambda\rho}$$

$$\tau = \frac{l^*}{\rho}$$

$$P_1 = P_0 \frac{\overline{\beta}_{eff} - \rho_0}{\overline{\beta}_{eff} - \rho_1}$$

DATA_SHEET

THERMODYNAMICS_AND_FLUID_MECHANICS_FORMULAS:

$$\dot{Q} = \dot{m} \Delta h$$

$$\dot{Q} = \frac{2 \pi L \Delta T}{\frac{1}{K} + \frac{\ln R_2/R_1}{K_2} + \frac{\ln R_3/R_2}{K_3}}$$

$$\dot{Q} = U A (\Delta T_m)$$

$$\dot{Q} = \dot{m} c_p (\Delta T)$$

$$\dot{Q} = \alpha \delta A R^4$$

$$\eta = \frac{\dot{Q}_{in} - \dot{Q}_{out}}{\dot{Q}_{in}}$$

$$\eta = \frac{(h_{in} - h_{out})real}{(h_{in} - h_{out})ideal}$$

$$\eta_p = \frac{W_{actual}}{W_{supplied}}$$

$$\frac{P_1 V_1}{T_1} = \frac{P_2 V_2}{T_2}$$

$$\dot{m} = \rho A V$$

$$\rho_1 A_1 V_1 = \rho_2 A_2 V_2$$

$$\dot{m} = KA \sqrt{\Delta P_x \rho}$$

$$\dot{m}_{nc} = KA_Q \sqrt[3]{\dot{Q}} = KA \Delta T \sqrt{\Delta T} = KA \Delta p \sqrt{\Delta P}$$

$$\Delta T_m = \frac{\Delta T (in) - \Delta T (out)}{\ln \left(\frac{\Delta T (in)}{\Delta T (out)}\right)}$$

$$G = \frac{\Sigma_f \phi_{th}}{8.8 \times 10^9}$$

$$T_{cl} - T_{ps} = \frac{G r^2}{4k}$$

$$\dot{Q} = \frac{k A \Delta T}{\Delta x}$$

$$\dot{Q} = \frac{A \Delta T_{total}}{\frac{\Delta x_a}{K_a} + \frac{\Delta x_b}{K_b} + \ldots + \frac{\Delta x_n}{K_n}}$$

DATA SHEET

CENTRIFUGAL PUMP LAWS:

$$\frac{N_1}{N_2} = \frac{\dot{m}_1}{\dot{m}_2} \qquad \frac{(N_1)^2}{(N_2)^2} = \frac{H_1}{H_2} \qquad \frac{(N_1)^3}{(N_2)^3} = \frac{P_1}{P_2}$$

RADIATION AND CHEMISTRY FORMULAS:

$R/hr = 6CE/d^2$ $\qquad I_x = I_0 e^{-mx}$ $\qquad C_1V_1 = C_2V_2$

$G = \frac{\text{Dilution Rate}}{\text{Volume}}$ $\qquad I = I_0 \left(\frac{1}{10}\right)^n$ $\qquad C = C_0 e^{-Gt}$

$A = A_0 e^{-\lambda t}$ $\qquad A = \lambda N$

CONVERSIONS:

$1 \text{ gm/cm}^3 = 62.4 \text{ lbm/ft}^3$

1 gal = 8.345 lbm

$1 \text{ ft}^3 = 7.48 \text{ gal}$

1 gal = 3.78 liters

1 lbm = 454 grams

e = 2.72

$\pi = 3.14159$

1 KW = 738 ft-lbf/sec

1 KW = 3413 Btu/hr

1 HP = 550 ft-lbf/sec

1 HP = .746 KW

1 HP = 2545 Btu/hr

1 Btu = 778 ft-lbf

$1 \text{ MEV} = 1.54 \times 10^{-16} \text{ Btu}$

$h = 4.13 \times 10^{-21} \text{ M-sec}$

$1 \text{ W} = 3.12 \times 10^{10} \text{ fissions/sec}$

$g_c = 32.2 \text{ lbm-ft/lbf-sec}^2$

1 inch = 2.54 cm

$\sigma = 0.1714 \times 10^{-8} \text{ Btu/hr ft}^2 \text{ R}^4$

$\text{Density of water (20 C)} = 62.4 \text{ lbm/ft}^3$

$\text{Avogadro's Number} = 6.023 \times 10^{23}$

$\text{Heat of Vapor } (H_2O) = 970 \text{ Btu/lbm}$

Heat of Fusion (ICE) = 144 Btu/lbm

$1 \text{ AMU} = 1.66 \times 10^{-24} \text{ grams}$

Mass of Neutron = 1.008665 AMU

Mass of Proton = 1.007277 AMU

Mass of Electron = 0.000549 AMU

One atmosphere = 14.7 psia = 29.92 in. Hg

°F = 9/5 °C + 32

°C = 5/9 (°F - 32)

°R = °F + 460

°K = °C + 273

$c^2 = 931 \text{ MEV/AMU}$

$C = 3 \times 10^8 \text{ m/sec}$

DATA_SHEET

AVERAGE_THERMAL_CONDUCTIVITY_(K)

Material	K
Cork	0.025
Fiber Insulating Board	0.028
Maple or Oak Wood	0.096
Building Brick	0.4
Window Glass	0.45
Concrete	0.79
1% Carbon Steel	25.00
1% Chrome Steel	35.00
Aluminum	118.00
Copper	223.00
Silver	235.00
Water (20 psia, 200 degrees F)	0.392
Steam (1000 psia, 550 degrees F)	0.046
Uranium Dioxide	1.15
Helium	0.135
Zircaloy	10.0

MISCELLANEOUS_INFORMATION:

$E = mc^2$

$KE = 1/2\ mv^2$

$PE = mgh$

$V_f = V_0 + at$

Geometric Object	Area	Volume
Triangle	$A = 1/2\ bh$	//////////////////
Square	$A = S^2$	//////////////////
Rectangle	$A = L \times W$	//////////////////
Circle	$A = \pi r^2$	//////////////////
Rectangular Solid	$A = 2(L \times W + L \times H + W \times H)$	$V = L \times W \times H$
Right Circular Cylinder	$A = (2\ \pi r^2)h + 2(\pi r^2)$	$V = \pi r^2 h$
Sphere	$A = 4\ \pi r^2$	$V = 4/3\ (\pi r^2)$
Cube	////////////////////////////////	$V = S^3$

DATA SHEET

MISCELLANEOUS INFORMATION (continued):

				10 CFR 20 Appendix B			
				Table I		Table II	
Material	Half-Life	Gamma Energy MEV per Disintegration		Col I Air uc/ml	Col II Water uc/ml	Col I Air uc/ml	Col II Water uc/ml
Ar-41	1.84 h	1.3	Sub	2×10^{-6}	------	4×10^{-8}	------
Co-60	5.27 y	2.5	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	5×10^{-5}
I-131	8.04 d	0.36	S	9×10^{-9}	6×10^{-5}	1×10^{-10}	3×10^{-7}
Kr-85	10.72 y	0.04	Sub	1×10^{-5}	------	3×10^{-7}	------
Ni-65	2.52 h	0.59	S	9×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
Pu-239	2.41×10^{4} y	0.008	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
Sr-90	29 y	------	S	1×10^{-9}	1×10^{-5}	3×10^{-11}	3×10^{-7}
Xe-135	9.09 h	0.25	Sub	4×10^{-6}	------	1×10^{-7}	------
Any single radionuclide with $T_{1/2}$ > 2 hr which does not decay by alpha or spontaneous fission				3×10^{-9}	9×10^{-5}	1×10^{-10}	3×10^{-6}

Neutron Energy (MEV)	Neutrons per cm^2 equivalent to 1 rem	Average flux to deliver 100 mrem in 40 hours	
thermal	970×10^{6}	670	
0.02	400×10^{6}	280	(neutrons)
0.5	43×10^{6}	30	----------
10	24×10^{6}	17	cm^2 x sec

Linear Absorption Coefficients μ (cm^{-1})				
Energy (MEV)	Water	Concrete	Iron	Lead
0.5	0.090	0.21	0.63	1.7
1.0	0.067	0.15	0.44	0.77
1.5	0.057	0.13	0.40	0.57
2.0	0.048	0.11	0.33	0.51
2.5	0.042	0.097	0.31	0.49
3.0	0.038	0.088	0.30	0.47

TABLE 7-1 Rev. 000

REACTOR MAKEUP CONTROL SYSTEM NOMOGRAPHS

NOMOGRAPH CORRECTION FACTORS

7/26/83

Superintendent, Engineering Date

Plant Conditions			Correction Factor (K) (See Note)
Pressure (psig)	T (AVG) (F)	Pressurizer Level	
2235	547-570	Normal Operating	1.00
1600	500	No-Load	1.05
1200	450	No-Load	1.10
800	400	No-Load	1.16
400	350	No-Load	1.18
400	300	No-Load	1.20
400	300	Solid Water	1.35
400	200	No-Load	1.28
400	200	Solid Water	1.40
400	100	Solid Water	1.47

NOTE: CORRECTION FACTORS ARE APPLIED AS FOLLOWS:

(a) Boron Addition and Dilution Total Volume Nomographs

$$V_{(Corrected)} = K \times V_{(Nomograph)}$$

(b) Boron Addition and Dilution Rate Nomographs

$$\frac{dc}{dt} \text{ (Corrected)} = \frac{1}{K} \times \frac{dc}{dt} \text{ (Nomograph)}$$

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FIGURE 7-3 Rev. 001

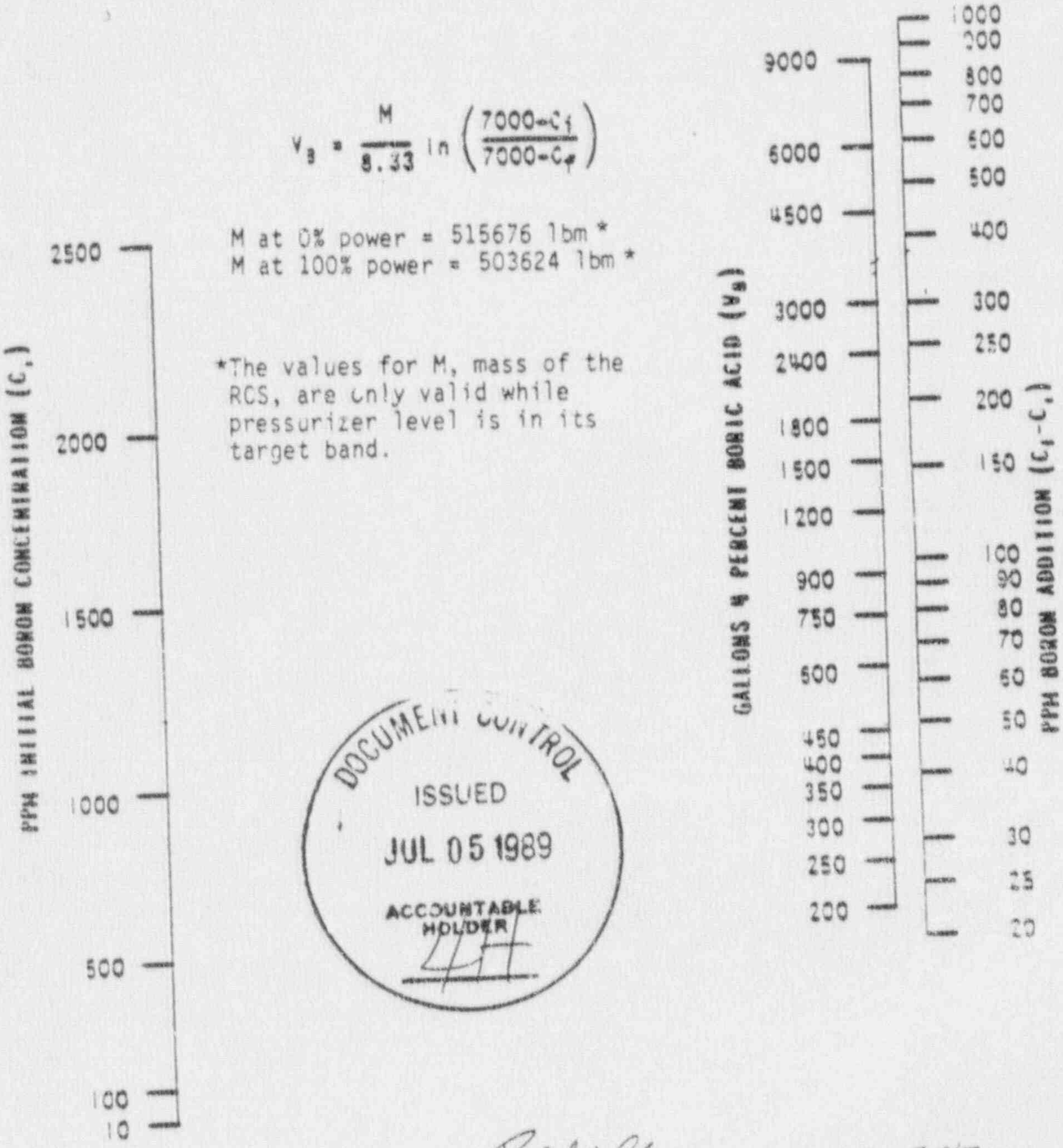
REACTOR MAKEUP CONTROL SYSTEM NOMOGRAPHS

BORON ADDITION

$$V_B = \frac{M}{8.33} \ln\left(\frac{7000-C_i}{7000-C_f}\right)$$

M at 0% power = 515676 lbm *
M at 100% power = 503624 lbm *

*The values for M, mass of the RCS, are only valid while pressurizer level is in its target band.



12-5-87
Superintendent, Engineering Date

NOTE: REFER TO TABLE 7-1 FOR CORRECTION FACTORS.

QUESTION: 001 (1.00)

The Workman's Protection Assurance Plan (APA-ZZ-00310) specifies that equipment under maintenance requiring operation for test be issued a:

a. Hold Off Tag (Orange).

b. Dispatcher's Hold Off Tag (White with Red Letters).

c. Caution Tag (Yellow).

d. Local Control Tag (Pink).

QUESTION: 002 (1.00)

According to rules of usage for procedures (APA-ZZ-00100 Procedure requirements), the word SHALL in a step denotes:

a. a recommendation imposed by UE management on its employees, contractors, and agents which should be met unless there is sufficient reason not to perform the activity.

b. permission to perform activities and is neither a requirement nor a recommendation.

c. a legally binding requirement to which UE management has committed (e.g. in the FSAR).

d. a requirement imposed by UE management on its employees, contractors, and agents which are above and in excess of the legally binding requirement of the appropriate regulatory body.

QUESTION: 003 (1.00)

Which ONE of the following is NOT defined as a deliberate action by Standing Order 90-010 for meeting T/S 3.0.3 shutdown requirements?

a. Mode 1 - reducing turbine load

b. Mode 2 - reducing NIS power

c. Mode 3 - reducing RCS temperature

d. Mode 4 - reducing RCS pressure

QUESTION: 004 (1.00)

An emergency entry may be made into a suspected or unknown hazardous atmosphere without a confined space entry permit (APA-ZZ-00802) providing:

a. chemistry has sampled the atmosphere within the last 24 hours.

b. a self-contained breathing apparatus is worn.

c. evolutions performed will not change the atmosphere.

d. the buddy system is used and two people enter.

QUESTION: 005 (1.00)

Which ONE of the following describes the ENTIRE scope of activities covered under the Callaway Plant ALARA program in accordance with APA-ZZ-01001, CALLAWAY PLANT ALARA PROGRAM:

a. Maintenance work performed in the Reactor Building during refueling periods, without Health Physics monitoring.

b. Maintenance and operations work in the Reactor Building during refueling periods, including work monitored by Health Physics.

c. All plant activities not monitored by Health Physics.

d. All plant activities, including those monitored by Health Physics.

QUESTION: 006 (1.00)

During a shift activity, an unplanned impairment of an Auxiliary Building pressure boundary door is required for 30 minutes. Prior to impairing the door, the individual should:

a. contact SS/OS for approval.

b. notify the QC Inspector.

c. hang a Fire Barrier Impairment Tag on the door.

d. place fire fighting equipment by the door.

QUESTION: 007 (1.00)

Access will be denied into the RCA to any Callaway worker without a completed Special Instruction Form CA-285 if his/her weekly exposure exceeds 80% of the:

a. 0.2 rem weekly limit.

b. 0.5 rem weekly limit.

c. 2.0 rem yearly limit.

d. 5.0 rem yearly limit.

QUESTION: 008 (1.00)

When the Emergency Implementing actions are taken per EIP-ZZ-00102, EMERGENCY IMPLEMENTING ACTIONS, plant personnel should be alerted by:

a. making a site-wide announcement over the Gaitronics.

b. sounding the Plant Emergency Alarm from the TSC and then making a plant-wide announcement over the Gaitronics.

c. sounding the Plant Emergency Alarm from the Control Room before making a plant-wide announcement from the Control Room or TSC, if manned.

d. dispatching a security officer to the siren control to initiate the siren ALERT tone and making an announcement over the Gaitronics.

QUESTION: 009 (1.00)

When a fire is reported to the control room, which ONE of the following actions should occur?

a. The person reporting the fire should be directed to use the nearest fire extinguisher and attempt to put out the fire.

b. The person reporting the fire should be directed to use the nearest hose station and attempt to put out the fire.

c. The person should be directed to stay at the location of the fire until assistance arrives.

d. The control room should supply the person with a location to meet the Fire Brigade Leader.

QUESTION: 010 (1.00)

Verification of position for valve EC-V020 (ESW Train A to SFP Upstream Isolation), an accessible LOCKED CLOSED valve, should be accomplished by which ONE of the following methods?

a. Attempting to move the handwheel in the close direction only.

b. Attempting to move the handwheel in the open direction and then reclose the valve.

c. Using the valve position indicator or stem position to verify condition.

d. Checking that the seal is intact and that the valve is on the Locked Valve Deviation List.

QUESTION: 011 (1.00)

A "Working File" copy of a diesel generator surveillance procedure is to be used by a Reactor Operator in the control room for several days. How often must this procedure be verified to be a "Controlled" copy in accordance with ODP-ZZ-00009, Operations Department - Control of Documents?

a. Every 8 hours

b. Every 12 hours

c. Daily

d. Weekly

QUESTION: 012 (1.00)

When working in a radiologically controlled area, a worker must leave if his/her PIC reading reaches what percent of full scale:

a. 25%

b. 50%

c. 75%

d. 100%

QUESTION: 013 (1.00)

Personnel Accountability is mandatory per EIP-ZZ-00102 (EMERGENCY IMPLEMENTING ACTIONS) if:

a. a radiological problem exists.

b. an Unusual Event has been declared for fire.

c. a Site Area Emergency or higher has been declared.

d. an Alert is declared for any radiological reason.

QUESTION: 014 (1.00)

Using the attached Table and figures, REACTOR MAKEUP CONTROL SYSTEM NOMOGRAPHS, which ONE of the following volumes is required to borate 50 ppm at 350 degrees, 400 psig with no-load pressurizer level? (current boron concentration is 1000 ppm)

a. 385 gallons

b. 410 gallons

c. 455 gallons

d. 537 gallons

QUESTION: 015 (1.00)

Emergency entries into the Reactor Building WITHOUT pre-entry surveys require which ONE of the following?

a. Shift Supervisor approval and SCBA's must be worn.

b. Emergency Duty Officer approval and SCBA's must be worn.

c. Health Physics Supervisor approval and a minimum of two personnel in respirators.

d. Plant Mangers' approval and a minimum of two personnel in respirators.

QUESTION: 016 (1.00)

When relieving the watch after 7 days absence from the control room, the watchstation logs should be reviewed, in accordance with ODP-ZZ-00003, Shift Relief And Turnover, for the previous:

a. day.

b. 3 days.

c. 5 days.

d. 7 days.

QUESTION: 017 (1.00)

The Control Room Command Function, during the absence of the Shift Supervisor, can be assumed by a(n):

a. STA, if licensed as a Senior Operator, in any mode.

b. Senior Reactor Operator (excluding the STA) in modes 1,2, and 3.

c. Reactor Operator in modes 4, 5 and 6.

d. Refueling Senior Reactor Operator in mode 6.

QUESTION: 018 (1.00)

During the performance of an NIS Power Range Heat Balance at 100% power, an operator uses a Feedwater Temperature 30 degrees LOWER than actual. Would the calculated value of power be HIGHER or LOWER than actual power and would an adjustment of the NIS Power Range Channels, based on this value, be CONSERVATIVE or NON-CONSERVATIVE with respect to protection setpoints.

a. higher/conservative

b. higher/non-conservative

c. lower/conservative

d. lower/non-conservative

QUESTION: 019 (1.00)

The unit is operating at 40% power steady state with all systems in their normal alignment. Reactor Coolant Pump B trips due to an electrical fault.

Which ONE of the following conditions describes the INITIAL response of S/G B level and steam flow after trip of the pump.

a. Level remains constant and steam flow decreases.

b. Level remains constant and steam flow increases.

c. Level changes rapidly and steam flow remains constant.

d. Level changes rapidly and steam flow decreases.

QUESTION: 020 (1.00)

With reactor power at 65%, penalty deviation outside the delta-I target band shall be accumulated on a time basis of:

a. One minute penalty for each minute outside of the target band.

b. One-half minute penalty for each minute outside of the target band.

c. One minute penalty for each one-half minute outside of the target band.

d. Zero penalty for time outside the target band.

QUESTION: 021 (1.00)

Control Rod M-12 in control bank D group 1 is being recovered after a blown stationary fuse was replaced. A procedure note in OTO-SF-00003, DROPPED CONTROL ROD, states an URGENT FAILURE alarm will be received when attempting to move the control rod.

Which ONE of the following describes why the alarm is present and why the control rod can be moved with this alarm present?

a. A regulation failure is present in power cabinet 2BD and rod M-12 is powered from cabinet 1BD.

b. A regulation failure is present in cabinets 1BD and 2BD but only inhibits auto rod movement.

c. A phase failure occurs when any lift coil is disconnected in a power cabinet but only inhibits auto rod movement.

d. A phase failure is present in power cabinet 1BD and 2BD but rods can be moved in individual bank position.

QUESTION: 022 (1.00)

The plant has experienced a large break LOCA. SI, CISB, and CTMT Spray have actuated. When should the Containment Spray Pump suction be transferred to the Containment Recirculation sump?

a. RWST level Lo-Lo 1 alarm and a lo-lo spray additive tank level.

b. RWST level Lo-Lo 1 alarm and containment pressure greater than 24 psig.

c. A lo-lo spray additive tank level alarm and containment pressure greater than 24 psig.

d. RWST empty alarm (ANN 47B).

QUESTION: 023 (1.00)

The following plant conditions exists:

-Mode 3 following a reactor trip/Safety injection	
-RCS temperature	500 degrees
-PZR pressure	1900 psig
-Steam Generator Pressure	600 psig
-Steam Generator Levels	50% wide range
-Containment pressure	3.2 psig

Which ONE of the following describes the status of the Safety Actuation Signal Logic assuming NO operator actions were taken?

a. A low pressurizer pressure and low steam pressure signal are active.

b. A low pressurizer pressure signal is present but can be blocked.

c. A low steam generator pressure signal is present but can be blocked.

d. A containment pressure signal is active but can't be blocked.

QUESTION: 024 (1.00)

Which ONE of the following conditions must be met prior to resetting the Motor Driven AFAS?

a. Indicated Steam Generator level in 3 of the 4 SG's above the lo-lo level setpoint.

b. Both Main Feedwater Pumps reset.

c. No undervoltage on NB01 or NB02.

d. AMSAC signal reset

QUESTION: 025 (1.00)

The Incore Thermocouples, by design, will operate satisfactorily in accident conditions and provide temperature indication up to:

a. 1300 degrees F.

b. 1650 degrees F.

c. 2300 degrees F.

d. 3300 degrees F.

QUESTION: 026 (1.00)

In an accident condition, the Containment Fan Coolers are all started in slow speed by the LOCA Sequencer. The reason for running in slow speed is:

a. to minimize the chance of fan motor overload with high containment pressure present.

b. to reduce the electrical load on the buses with all safeguard loads running.

c. to minimize the heat load on the Essential Service Water system.

d. to prevent over stressing the fan exhaust duct fusible links with high containment pressure present.

QUESTION: 027 (1.00)

The unit is at 100% power. Steam Generator D Level control system is in AUTOMATIC with the following instruments selected:

-Steam Flow Transmitter	(AB-FT-542)
-Feedwater Flow Transmitter	(AE-FT-540)
-Steam Generator Level Transmitter	(AE-LT-549)

Which ONE of the following describes the plant response to Steam Pressure Transmitter (AB-PT-544) failing LOW?

(ASSUME NO OPERATOR ACTION AND ALL PLANT CONTROLS IN AUTOMATIC)

a. Feedwater flow DECREASES until S/G D level stabilizes at a lower level or a low level trip setpoint is reached.

b. Steam Generator Feedwater Pump speed control system generates a HIGHER reference speed signal.

c. Indicated Steam Flow increases on S/G D and Feedwater pump RPM increases.

d. Feedwater Flow INCREASES until S/G D stabilizes at a HIGHER level or a high level trip setpoint is reached.

QUESTION: 028 (1.00)

The unit has tripped from 100% power due to a condenser problem and is removing heat through the Steam Generator Atmospherics. The unit reactor operator notes that Condensate Storage Tank (CST) level is at the minimum required Technical Specification volume.

Select the ONE statement below which describes the existing volume in the tank and the basis for it.

a. 281,000 gallons are available to maintain Hot Standby for 8 hours before cooling down to 350 degrees.

b. 281,000 gallons are available to maintain Hot Standby for 4 hours and then cool down to 350 degrees at 50 degrees per hour.

c. 394,000 gallons are available, which will allow 12 hours in Hot Standby and a 4 hour cooldown, at 50 degrees per hour to 350 degrees.

d. 394,000 gallons are available to maintain Hot Standby for 4 hours and then cool down to 350 degrees at 50 degrees per hour.

QUESTION: 029 (1.00)

Select the ONE statement which describes the Flow Control Valves for the Motor Driven Auxiliary Feedwater Pumps.

a. The valves are motor operated and throttle close automatically at high flow rates to prevent pump cavitation.

b. The valves are motor operated and throttle close automatically at high flow rates to limit containment pressure increase caused by a steam line rupture in containment.

c. The valves are air operated and throttle close automatically at high flow rates to prevent pump cavitation.

d. The valves are air operated and throttle close automatically to limit Auxiliary Feedwater flow to a faulted steam generator in the event of a feed line break.

QUESTION: 030 (1.00)

The Auxiliary Feedwater Pump Suction will automatically shift to Essential Service Water when:

a. the Condensate Storage Tank level decreases to 18%.

b. a low suction pressure is sensed on common suction header.

c. an AFAS signal is present and low suction pressure on 2 of 3 sensors.

d. an AFAS signal is present and Condensate Storage Tank level decreases to 18%.

QUESTION: 031 (1.00)

A Waste Gas release will be automatically terminated if radiation levels reach:

a. a high alarm (yellow) on GH-RE-10B.

b. a high alarm (yellow) on GH-RE-10A.

c. a high high alarm (red) on GE-RE-10A or GH-RE-10B.

d. a high high alarm (red) on GH-RE-23.

QUESTION: 032 (1.00)

Annunciator window 58D (RV FLG LEAKOFF TEMP HI) has just energized. The leakage from the reactor vessel flange will accumulate in which ONE of the following?

a. normal containment sumps

b. recirculation sump

c. pressurizer relief tank

d. reactor coolant drain tank

QUESTION: 033 (1.00)

The containment atmosphere radiation monitors GT-RE-31 AND GT-RE-32 sample:

a. containment utilizing the hydrogen control system and are isolated from the containment by a CIS-A actuation.

b. inside the containment isolation valves for the Hydrogen Control System and are not isolated by a CIS-A actuation.

c. between the containment isolation valves on the mini-purge exhaust line and can initiate a CIS-A actuation.

d. from the containment purge exhaust line outside containment and can initiate a CIS-A actuation.

QUESTION: 034 (1.00)

The following readings were taken from the Power Range NIS Detectors:

	N41	N42	N43	N44
Det. A (upper)	375.0	360.0	365.0	360.0
Det. B (lower)	325.0	345.0	330.0	360.0

All readings are in microamperes. Full power currents on all detectors is known to be 400.0 microamperes. Select the DETECTOR with the most limiting Quadrant Power Tilt Ratio (QPTR).

a. N41 upper

b. N42 upper

c. N43 lower

d. N44 lower

QUESTION: 035 (1.00)

Prior to starting a Reactor Coolant Pump, the Volume Control Tank (VCT) is pressurized to greater than or equal to 15 psig to insure:

a. Net Positive Suction Head for the charging pumps.

b. proper flow through the number 1 seal.

c. proper flow through the number 2 seal.

d. adequate hydrogen addition to the VCT.

QUESTION: 036 (1.00)

The Technical Specification COOLDOWN limit on the pressurizer is:

a. the same as the Reactor Coolant System because they are directly connected.

b. more restrictive than the limit for the Reactor Coolant System because the wall thickness of the pressurizer is larger.

c. less restrictive than the limit for the Reactor Coolant System because the wall thickness of the pressurizer is less.

d. less restrictive than the limit for the Reactor coolant System because there is little or no flow through the pressurizer.

QUESTION: 037 (1.00)

The unit has just experienced a Safety Injection and recovery is in progress. The RCS has been depressurized to recover pressurizer level and the following parameters exist:

-RCS Tave	540 degrees
-pressure	2200 psig
-pressurizer level	50%
-liquid temperature	610 degrees
-Vapor space temperature	615 degrees
-pressurizer heaters	ON

Using the provided steam tables, which ONE of the following describes the PRESENT state of the pressurizer.

a. The pressurizer is at equilibrium saturation conditions and normal pressure control is available with heaters and sprays.

b. The pressurizer liquid is subcooled and pressure is being maintained by Safety Injection flow compressing the vapor space.

c. Superheat conditions exist for the pressure but heaters and sprays will maintain pressure because of the lower liquid temperature.

d. The pressurizer liquid is subcooled with pressurizer heaters maintaining pressure.

~~QUESTION: 038 (1.00)~~

~~Control of Pressurizer Heater Backup Group A has been transferred to the Auxiliary Shutdown Panel and the control switch has been placed in the closed position.~~

~~Which ONE of the following conditions will trip the supply breaker (PG2101) for this group of heaters?~~

- ~~a. Pressurizer low level (17%)~~
- ~~b. Lockout on NB0106~~
- ~~c. NB01 Undervoltage~~
- ~~d. Safety Injection Signal~~

Deleted
2/4/91
KMS

QUESTION: 039 (1.00)

The following plant conditions exists:

-Mode 1 at 100% steady state	
-RCS pressure	2200 psig (slowly decreasing)
-pressurizer level	40% (decreasing)
-Tave	588 degrees F (stable)
-charging/letdown lineup	PDP running
-charging flow	98 gpm
-VCT level	30% (decreasing)
-containment humidity	increasing
-containment sump levels	increasing

The first IMMEDIATE action required for this situation per OTO-BB-00003 (Reactor Coolant System Excessive Leakage) is:

- a. operate the centrifugal charging pumps as necessary to maintain pressurizer level.
- b. isolate the letdown system by closing BG-LCV-459 and 460.
- c. swap the charging pump suction to the RWST to insure adequate makeup.
- d. trip the reactor because the leak is beyond the capacity of the running charging pump.

QUESTION: 040 (1.00)

The unit is in Mode 1 at 100% power. Five minutes ago pressurizer channel PT-456 failed low. The unit reactor operator is conducting a board walkdown and discovers that OT-Delta-T trip setpoint for loop 2 is indicating 30% lower than the other 3 loops.

Select the ONE statement which describes the OT-Delta-T Channel condition.

a. The channel has also failed but the bistables can't be tripped because it is the same protection set as the pressure channel.

b. The channel has also failed because the setpoint should have increased when PT-456 failed low for proper DNB protection.

c. The channel is responding correctly and no action is required.

d. The channel is responding correctly but must be declared inoperable with the bistables tripped within the required time interval.

QUESTION: 041 (1.00)

The Hydrogen Control System is designed to maintain containment hydrogen concentrations between:

a. 0% and 2%.

b. 0% and 4%.

c. 0% and 6%.

d. 0% and 8%.

QUESTION: 042 (1.00)

Containment Hydrogen is increasing following a major LOCA. The Hydrogen Purge System is being used to supplement the Electric Hydrogen Recombiners to lower the hydrogen concentration.

This potential release path from the containment would be isolated automatically if conditions degraded by:

a. a high radiation condition (red) sensed by either GG-RE-27/28 (emergency exhaust) or GT-RE-21A/B (unit vent) which would initiate a CIS-A.

b. a high radiation condition (red) sensed by either GG-RE-27/28 (emergency exhaust) or GT-RE-21A/B (unit vent) would initiate a FBVIS.

c. any signal initiating a CPIS actuation.

d. any signal initiating a CIS-A actuation.

QUESTION: 043 (1.00)

A complete loss of Condenser Vacuum occurs with the plant at:

-Mode 3
-pressure 2235 psig
-Tave 557 degrees F
-All four RCP's running

With NO operator actions, RCS Tave will stabilize at:

a. 550 degrees F.

b. 557 degrees F.

c. 561 degrees F.

d. 564 degrees F.

QUESTION: 044 (1.00)

The trip of a running Circulating Water Pump at 100% power will:

a. cause a turbine trip on low vacuum.

b. initiate a turbine runback to 75%.

c. have no effect because the other pumps have enough capacity.

d. cause reactor power to increase due to condenser efficiency decreasing.

QUESTION: 045 (1.00)

Starting Air Compressors A and B supply Emergency Diesel Generator A. Select the one statement which describes their power supply.

a. They are both powered from MCC PG19G.

b. One compressor is powered from MCC PG19G and one from MCC PG20G.

c. They are both powered from MCC PG20G.

d. One compressor is powered from NG03 and one from NG04.

QUESTION: 046 (1.00)

The plant is at 100% power with all systems in their normal lineups. Annunciator 14A "S/U XFMR LOCKOUT" alarms due to failure of the startup transformer (SUT). Select the ONE item below which occurs as a result of the SUT failure.

a. A load shed occurs on NB01 and NB02.

b. Both emergency diesels NE01 and NE02 start.

c. An automatic Reactor Trip and Turbine Trip actuates.

d. Both the normal and alternate feeder breakers to NB02 trip.

QUESTION: 047 (1.00)

Given the following conditions:

-Annunciator "JOCKEY PUMP MALFUNCTION" on Panel KC008
-Fire Water Accumulator level -2 1/2 inches below centerline
-normal valve lineup
-all Firewater Pumps available in auto
-Firewater pressure 160 psig
-Firewater storage tank levels 35 feet

Which ONE of the following actions should be taken if the low level in the accumulator is verified locally?

a. No action is required. The jockey pump will auto start after a 5 minute timer and fill the accumulator to 90%.

b. A local start, using the manual lever for the jockey pump, should be attempted to fill the accumulator to 90%.

c. The local control station manual start/stop pushbuttons should be used to operate the jockey pump and raise the accumulator level to the center line of the accumulator.

d. No action required. The jockey pump will auto start after a 15 minute timer and raise the level to the center line of the accumulator.

QUESTION: 048 (1.00)

Which ONE of the following describes the operation of the pressurizer safety valves?

a. Sequentially open from 2435 psig to 2485 psig to prevent a large LOCA due to all valves simultaneously opening and then failing to re-close.

b. Open at 2385 psig to limit pressurizer pressure to a value below the high pressure reactor trip setpoint.

c. Open at 2485 psig to prevent over-pressurization of the Reactor Coolant System due to complete loss of load without a reactor trip.

d. Sequentially open from 2335 psig to 2485 psig to provide adequate pressure control following an increase in RCS pressure.

QUESTION: 049 (1.00)

The unit is in Mode 1 at 100% power steady state conditions. A reactor operator trainee inadvertently changes the Pressure Master Controller setpoint to 2370 psig. Assume a step change in the setpoint and assume that pressurizer pressure control remains in automatic.

The immediate automatic responses of the system will be:

a. spray valves close, pressurizer heaters energize.

b. power operated relief valve PCV-455A opens, spray valves open, pressurizer heaters de-energize.

c. spray valves open, pressurizer heaters energize.

d. power operated relief valve PCV-455A and PCV-456A opens, spray valves open, pressurizer heaters energize.

QUESTION: 050 (1.00)

Which input signals are used to control the feedwater bypass flow control valves when in automatic?

a. Steam header pressure, feed pump discharge pressure, total steam flow.

b. Steam header pressure, Steam generator level, programmed steam generator level, steam flow, feed flow.

c. Steam generator level, programmed steam generator level, auctioneered high nuclear power.

d. Steam generator level, programmed steam generator level, total steam flow.

QUESTION: 051 (1.00)

Pressurizer level control is selected to the LT-459/460 position. A failure causes the following SEQUENTIAL plant events. (ASSUMING NO OPERATOR ACTIONS TAKEN)

-charging flow reduces to minimum
-pressurizer level begins to decrease
-letdown isolates and heaters turn off
-pressurizer level eventually increases to the high level reactor trip

Which ONE of the following failures occurred?

a. Reference pressurizer level failed to the no-load value.

b. Auctioneered Tave failed Hi due to a failed RTD.

c. Level channel 460 failed low.

d. Level channel 459 failed high.

QUESTION: 052 (1.00)

A unit startup is in progress with the following conditions:

-NIS Power	11%
-Tave	560 degrees F
-pressurizer pressure	2235 psig
-SG levels	50%
-turbine load	70 MWE
-containment pressure	1.0 psig
-containment temperature	90 degrees
-normal radiation readings	

Which ONE of the following describes the plant response to feedwater control bypass valve on Steam Generator A failing closed. Assume no operator action.

a. A reactor trip will occur immediately when Steam Generator level A reaches 14.8% narrow range.

b. A reactor trip will occur 122 seconds after the level in a Steam Generator reaches 20.2% narrow range.

c. A reactor trip will occur 232 seconds after reaching 14.8% Narrow range in Steam Generator A.

d. A reactor trip will occur immediately when Steam Generator A level decreases to 20.2%

QUESTION: 053 (1.00)

The following plant condition exist:

-power 100%
-normal system lineups with "A" CCP running
-Pressurizer level (slowly decreasing)
-CCW Surge tank "A" level (increasing)
-Radiation Monitor EG-RE-9 (increasing)

Based on the above information, which ONE of the following components could be a source of in-leakage to the CCW system under the existing conditions?

a. seal water heat exchanger

b. RHR Heat Exchanger A

c. RCP Thermal barrier heat exchanger

d. spent fuel pool cooling heat exchanger

QUESTION: 054 (1.00)

The following plant conditions exist:

-Mode 1	
-NIS power	6%
-Tave	559 degrees F
-Pressurizer pressure	2235 psig
-steam dumps	in pressure mode
-turbine rolling	500 rpm
-Steam Generator levels	50%
-feedwater controlling	on the bypass valves
-condenser vacuum	2.1 inches HgA

Which ONE of the following describes the response of the steam dump valves when AB-PT-507 (Steam Header Pressure) fails HIGH (ASSUME NO OPERATOR ACTION).

a. Group 1 strokes full open and remains there as the plant is cooled down.

b. Group 1 and 2 trip open and then reclose when terperature reaches 550 degrees.

c. All four groups stroke full open and then reclose when temperature reaches 550 degrees.

d. All four groups trip open and groups 2,3 and 4 close when temperature reaches 550 degrees.

QUESTION: 055 (1.00)

The following plant conditions exist:

-Cycle 5	
-Initial startup after refueling	
-Physic testing completed	(MTC +1 PCM/F)
-NIS power	10%
-turbine	80 MWE
-Pressurizer pressure	2235 psig
-Tave	559 degrees F
-Boron concentration	1500 ppm
-Control Bank D	200 steps

Which ONE of the following would describe the change in Moderator Temperature Coefficient (MTC) as the turbine load is ramped up to 580 MWE?

a. remains constant.

b. more positive due to Rod withdrawal.

c. more positive due to Xenon and samarium buildup.

d. less positive due to the required dilution.

QUESTION: 056 (1.00)

A plant cooldown is being performed in MODE 4 using both RHR trains. The following data has been recorded:

TIME	RCS TEMP	RCS PRESS
1000	330 degrees	355 psig
1030	306 degrees	360 psig
1100	282 degrees	360 psig
1130	260 degrees	358 psig
1200	225 degrees	355 psig

Which ONE of the following actions should be taken based on the above data?

a. Restore RCS temperature to 235 degrees within 30 minutes.

b. Restore the cooldown rate to less than 50 degrees per hour.

c. Continue cooldown at present rate to cooldown to MODE 5 within 30 hours.

d. Restore cooldown rate to within Technical Specifications limits, notify the NRC of Tech Spec violation on cooldown rate.

QUESTION: 057 (1.00)

The unit is being runback from 75% power due to the loss of a feedwater pump. The unit reactor operator notices that Control Bank D group 1 Rod M-12 position remained at 185 steps. Thirty seconds later, as the plant is stabilizing, Control Bank D group 2 Rod D-12 position is indicating 150 steps and the bank demand is 135 steps.

Which ONE of the following actions should be taken?

a. Trip the reactor and implement E-0.

b. Stabilize the plant and perform a Quadrant Power Tilt Ratio calculation.

c. Notify Reactor Engineering to perform an incore flux map to determine rod position.

d. Initiate immediate boration to withdraw rods out of the core.

QUESTION: 058 (1.00)

The unit is operating at full power when annunciator 79C (RPI DEV OR PR TILT) alarms. The unit reactor operator announces that Control Bank D rods D-4 and M-12 have dropped.

Which ONE of the following actions should be taken?

a. Place rod control in manual to prevent an uncontrolled restart.

b. Reduce load on the turbine immediately.

c. Match Tave with Tref using turbine load or boration/dilution.

d. Trip the reactor and go to E-0.

QUESTION: 059 (1.00)

Which ONE of the following describes the use of adverse containment values in the advent of a LOCA?

a. If containment temperature or radiation exceeds the stated value on the foldout page, adverse containment values are used for the duration of the event.

b. Once in adverse conditions because of temperature and radiation, normal values can be used, if temperature and radiation decrease less than the foldout page values.

c. Once in adverse conditions, a return to normal values can be made, if containment temperature was the only reason adverse conditions had been declared.

d. Once in adverse conditions, a return to normal values can be made, if containment radiation was the only reason adverse conditions had been declared.

QUESTION: 060 (1.00)

Following a large break LOCA with systems aligned for cold leg recirculation mode, the core decay heat is being removed primarily by:

a. heat transfer from the RCS to the Steam Generators with natural circulation.

b. heat transfer from the RCS to the Steam Generators with reflux boiling.

c. injection of water from the RWST.

d. injection of water from the Recirculation Sump.

QUESTION: 061 (1.00)

The plant is operating at steady state 75% power. Which ONE of the following conditions requires an IMMEDIATE manual reactor trip?

a. RCP A seal injection temperature of 115 degrees F and is increasing at 0.5 degree per minute.

b. RCP B frame vibration equals 4 mils and is increasing at .5 mils/hr.

c. RCP C #1 seal inlet temperature 240 degrees F with a loss of normal seal injection.

d. RCP D shaft vibration equals 17 mils and is increasing at 1.4 mils/hr.

QUESTION: 062 (1.00)

A reactor trip has just occurred and the following conditions exist:

-reactor trip breakers open
-reactor bypass breakers open
-NIS power decreasing
-control rod H-10 228 steps
-control rod H-6 228 steps

Which ONE of the following identifies the procedure flow path for this situation?

a. Immediate borate in accordance with OTO-ZZ-0003, Response to Loss of shutdown Margin.

b. Enter procedure E-0, Rx Trip/SI; at step 1 go to FR-S.1; at step 4 immediate borate according to attachment 1.

c. Enter procedure E-0, Rx Trip/SI; at step 4 go to ES-0.1, Rx Trip Response; at step 3 immediate borate.

d. Enter procedure FR-S.1, Response To Nuclear Power Generation, and immediate borate according to attachment 1.

QUESTION: 063 (1.00)

The unit is at 100% power steady state. The CCW System Train B is lined up to supply the service loop. Given the following indications:

-Annunciator 53B (CCW SRG TK B LEV HILO) alarming
-surge tank B level (decreasing rapidly)
-Annunciator 39A (LTDN HX TEMP HI DIVERT) alarming
-Annunciator 39B (LTDN HX DISCH TEMP HI) alarming

Which ONE of the following operator actions COULD NOT isolate this leak?

a. Isolate the radwaste building by shutting EG-HIS-69 AND 70.

b. Isolate Letdown heat exchanger by closing the Letdown orifice valves and BG-PCV-131 Letdown heat exchanger valve isolation.

c. Immediately open EG-HS-15 and close EG-HS-16 to shift the service header to train A.

d. Isolate the service header and trip the reactor after 2 minutes, if the service header can not be restored.

QUESTION: 064 (1.00)

Procedure FR-S.1 (Response To Nuclear Power Generation) step 4 initiates Immediate Boration of the RCS and directs opening the pressurizer PORVs and block valves if pressure is above 2335 psig.

Which ONE of the following describes the basis for opening the PORVs and block valves?

a. This will insure pressure remains below the setpoint of the pressurizer safety valves.

b. Opening the valves will lower pressure in an attempt to increase charging flow.

c. This insures lower pressure while a steam bubble exists and not later when the pressurizer might be water solid.

d. Opening the valves depressurizes the RCS to initiate a Safety Injection actuation for boron addition from all sources.

QUESTION: 065 (1.00)

Which ONE of the following symptoms would most clearly differentiate between a large LOCA and a large Main Steam Line break inside containment?

a. increasing containment radiation levels

b. increasing containment sump levels

c. increasing containment pressure

d. decreasing pressurizer pressure

QUESTION: 066 (1.00)

The unit has sustained a loss of off-site and onsite AC power. Proced ECA-0.0, LOSS OF ALL AC POWER, has been implemented.

Which ONE of the following describes the use of the PORVs in this p condition?

a. PORVs should be cycled as necessary to control pressure at psig with no pressurizer spray or auxiliary available.

b. PORVs should be opened to depressurize the RCS and minimize leakage through the RCP seals.

c. PORVs should be checked shut and monitored closely if pressure reaches 2335 psig and causes them to cycle.

d. Block valves should be shut to eliminate any mass loss out through the PORVs at any time during the event.

QUESTION: 067 (1.00)

The plant is in mode 2 with the following conditions:

-NIS power	7%
-Tave	557 degrees
-pressurizer pressure	2235 psig
-turbine	600 rpm

Intermediate Range Channel N35 de-energizes due to a loss of the 120 Volt AC Instrument Bus. Which ONE of the following identifies the implementing procedure flowpath?

a. Procedure E-O, REACTOR TRIP OR SAFETY INJECTION, will be entered after the reactor trip.

b. Implement OTO-SE-00002, INTERMEDIATE RANGE NUCLEAR FAILURE, and repair Channel N35 before exceeding 10% power to comply with Technical Specifications.

c. Implement OTN-NN-00001, 120 VITAL AC INSTRUMENT POWER-CLASS 1E, and re-energize the instrument bus on the backup power supply before exceeding 10% power.

d. Bypass the trips on Channel N35 in accordance with OTO-SE-00002, INTERMEDIATE RANGE NUCLEAR FAILURE, and continue the startup in accordance with OTG-ZZ-00003, PLANT STARTUP.

QUESTION: 068 (1.00)

Which ONE of the following statements describes the limit and basis for RCS Specific Activity of 1.0 microCurie per gram DOSE EQUIVALENT I-131 and less than or equal to 100/E microCuries per gram of gross radioactivity?

a. The limit is based on maintaining containment area radiation levels low for ALARA concerns when making at power containment entries.

b. The limit insures that the 2 hour limit at site boundary will not exceed a small fraction of 10 CFR 100 dose guidelines with a Steam Generator Tube leak of 1 gpm.

c. The limit insures that the dose received in the low population area will not exceed the 10 CFR 100 guidelines during the course of any event.

d. The activity level indicates 1% failed fuel in the core which is a design limit and limits the total number of operating hours to ensure FSAR Offsite dose assumptions are valid.

QUESTION: 069 (1.00)

The Technical Specification actions for exceeding the RCS Specific Activity Limit directs a RCS cooldown to less than 500 degrees within 6 hours.

Which ONE of the following describes the condition of the plant after the cooldown has been completed? (Assume Plant at 499 degrees)

a. The RCS would still be 50 degrees subcooled if depressurized at this time to the pressure of the steam generators.

b. The Steam Generator Atmospheric relief valve would lift, but not a safety valve, if a Steam Generator Tube rupture occurred after reaching 500 degrees.

c. The saturation pressure of the RCS is less than the setpoint of the Atmospheric relief and safety valves of the Steam Generator.

d. In this condition the steam generator delta-P limits cannot be exceeded and initiate a Steam Generator Tube Rupture.

QUESTION: 070 (1.00)

Following a loss of all A.C. Power, the operations crew implemented ECA-0.0, LOSS OF ALL AC POWER. When Diesel Generator NE01 has been successfully placed on bus NB01, the STA announces that core exit thermocouples are reading greater than 700 degrees.

Based on this information, which ONE of the following actions should be taken?

a. Continue on in ECA-0.0 until directed to exit to ECA-0.1, Loss of All AC Power Recovery Without SI Required, or ECA-0.2, Loss of All AC Power Recovery With SI Required.

b. Continue on in ECA-0.0 until completed and then go to FR-C.1, Response to Inadequate Core Cooling.

c. Immediately exit ECA-0.0 and go to FR-C.2, Response to Degraded Core Cooling.

d. Immediately exit ECA-0.0 and go to FR-C.1, Response to Inadequate Core Cooling.

QUESTION: 071 (1.00)

In accordance with OTO-ZZ-00001, CONTROL ROOM INACCESSIBILITY, evacuating the control room requires immediate classification as:

a. an Unusual Event.

b. an Alert.

c. a Site Area Emergency.

d. a General Emergency.

QUESTION: 072 (1.00)

The unit has experienced a fire in the control room and immediate actions of OTO-ZZ-00001, CONTROL ROOM INACCESSIBILITY, have been completed to the point of initiating control from the Auxiliary Shutdown Panel (attachment 5 of the procedure).

-NIS power	1 x 10E-8 (decreasing)
-Wide range T-Hot	562 degrees F
-Wide range T-Cold	552 degrees F
-steam generator pressure	1050 psig

Which ONE of the following describes the plant condition?

a. Hot Standby, Tave at 557 degrees controlling on the condenser steam dumps in Tave Mode with all four RCP's running.

b. Hot Standby, Tave at 557 degrees controlling on the Atmospheric Steam relief valves with RCP B and C running.

c. Hot Standby with Natural circulation, Tave control with Atmospherics Steam Relief Valves on B and D Steam Generators.

d. Hot standby with Natural Circulation, Tave at 557 degrees controlling on the condenser steam dumps in Tave Mode.

QUESTION: 073 (1.00)

In accordance with E-1, LOSS OF REACTOR OR SECONDARY COOLANT, the Reactor Coolant Pumps should be tripped if pressure decreases below 1400 psig.

The Trip Criteria:

a. minimizes a core uncovery in the event of a large LOCA.

b. minimizes the cooldown rate for a steam line break.

c. saves the RCPs for later use if RCS feed and bleed fails.

d. insures Peak Clad Temperatures remain below 2200 degrees in a small break LOCA.

QUESTION: 074 (1.00)

The following conditions exist:

-Reactor trip/SI actuated	
-NIS power	1 x 10E-9 amps
-Core Exit Thermocouples	680 degrees F
-pressurizer pressure	2185 psig
-Containment temperature	180 degrees F
-Containment pressure	0.1 psig
-Containment radiation	Normal

The table for required subcooling in CSF-1, CRITICAL SAFETY FUNCTION STATUS TREES, list the following values!

RCS PRESSURE	NORMAL CTMT INSTRUMENT ERROR	ADVERSE CTMT INSTRUMENT ERROR
1000-3000 (psig)	23 (DEG F)	43 (DEG F)

The RCS Coolant condition based on the above the parameters is:

a. subcooled enough to meet the required normal Containment value of 23 degrees.

b. subcooled enough to meet the required Adverse Containment value of 43 degrees.

c. superheated.

d. saturated.

QUESTION. 075 (1.00)

A Main Steam Line Break Accident and its associated cooldown are considered as a worst case accident when they occur at:

a. End of Life (EOL) with the plant in Hot Standby.

b. End of Life (EOL) at full power.

c. Beginning of Life (BOL) at full power.

d. Beginning of Life (BOL) in Hot Standby.

QUESTION: 076 (1.00)

The reactor is operating at 50% power when turbine load is increased 10% with NO control rod motion or change in boron concentration. An indication you would EXPECT to receive is a:

a. T REF/T AUCT HI annunciator.

b. T REF/T AUCT LO annunciator.

c. RPI ROD DEV annunciator.

d. RPI DEV PR TILT annunciator.

QUESTION: 077 (1.00)

Which ONE of the following describes the sequence of events at 45% power with decreasing vacuum in the condenser? (ASSUME NO OPERATOR ACTION)

a. At 8.4 inches Hga, the turbine will trip, causing a reactor trip. The plant will be in hot standby, with Tave controlled on the condenser dumps at 557 degrees.

b. Standby vacuum pump will start at 5 inches Hga. Turbine will trip at 8.4 inches Hga causing a reactor trip. The plant will be in hot standby, with Tave controlled by the Atmospheric steam relief valves.

c. The turbine will trip at 8.4 inches Hga. The condenser dumps and control rods will reduce Tave to 557 degrees.

d. The turbine will trip at 8.4 inches Hga. The steam generator safeties and Atmospherics will remove heat until control rods bring reactor power down.

QUESTION: 078 (1.00)

Which ONE of the following events would initiate automatic rod insertion?

a. Power Range Channel 42 lower detector failing low.

b. First Stage Turbine Impulse Pressure Channel PT 505 failing low.

c. Loop 2 Narrow Range T-cold RTD failing low.

d. Loop 2 Narrow Range T-hot RTD failing low.

QUESTION: 079 (1.00)

The basis for depressurizing all intact steam generators to atmospheric pressure in step [illegible] of FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, is to:

a. insure core exit thermocouple temperatures are reduced to less than 700 degrees F.

b. reduce S/G pressure to increase feedwater flow.

c. reduce RCS pressure for establishing low-head safety injection.

d. enhance natural circulation cooling of the reactor core.

QUESTION: 080 (1.00)

A major Steam Line Break has occurred blowing down Steam Generator A and cooling the RCS by 20[illegible] degrees F in a 15 minute period. Thermal Stresses on the reactor vessel wall will:

a. remain until the thermal gradient is reduced by a soak.

b. be removed immediately when the cooldown is stopped.

c. be removed quickly if a heat up is commenced.

d. off-set by the pressure stress present on the inner wall.

QUESTION: 081 (1.00)

Following an event that requires entry into FR-P.1 because of an Integrity ORANGE path condition, the operator must exit that procedure if:

a. the Integrity path turns GREEN while performing step 2 of FR-P.1.

b. a Containment path turns RED while performing FR-P.1.

c. an Inventory path turns ORANGE while performing FR-P.1.

d. a Subcriticality path turns YELLOW while performing FR-P.1.

QUESTION: 082 (1.00)

Procedure E-0, step 2, verifies that a turbine trip has occurred following a reactor trip. The reason for tripping the turbine is to:

a. prevent overheating the last stage of blades on the low pressure turbines.

b. reverse power the generator and prevent overspeeding the unit.

c. prevent uncontrolled cooldown of the RCS.

d. provide dryout protection for the steam generators, in the event of a loss of all feedwater accident.

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QUESTION: 083 (1.00)

The plant has just experienced a Reactor Trip with no SI from 30% powe Which ONE of the following describes the MOST COMPLETE response of the Feedwater System in this event?

When Tave decreases to 564 degrees,:

a. both Main Feedwater Pumps will trip and all four control valves will close.

b. both Main Feedwater Pumps will trip and all four control valves and bypasses close.

c. all four control valves and bypasses close.

d. all four feedwater control valves, bypass valves and isolation valves will close.

QUESTION: 084 (1.00)

Pressurizer PORV 456 has lifted and failed to fully reseat resulting in the following plant conditions: (assume the block valve has failed to close)

-Rx trip	
-pressurizer pressure	1985 psig
-pressurizer vapor space temperature	635 degrees
-Tave	557 degrees
-PRT level	75 %
-PRT pressure	35 psig

The tailpipe temperature indication for Pressurizer PORVS should read:

a. full scale high at 400 degrees F.

b. 280 degrees F.

c. 260 degrees F.

d. 220 degrees F.

QUESTION: 085 (1.00)

The unit has experienced a LOCA and the transition from procedure E-1 to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, has been made. Voiding in the RCS when depressurizing may be identified by:

a. rapidly increasing pressurizer level.

b. decreasing Safety Injection flow.

c. increasing RCS pressure.

d. rapid drop in subcooling.

QUESTION: 086 (1.00)

A pressurizer PORV valve seat is leaking to the PRT at a rate of 1 gpm and all other system components will function normally. According to Technical Specifications, this is:

a. identified leakage that requires shutdown.

b. unidentified leakage that requires shutdown.

c. identified leakage but does not require shutdown.

d. unidentified leakage but does not require shutdown.

QUESTION: 087 (1.00)

The unit is in Mode 6 with reduced RCS inventory. Which ONE statement describes the relationship of RHR pump vortexing versus flow and level?

Vortexing will occur:

a. at the loop low level alarm (BLI-0053 17 inches) with a minimum RHR flow rate of 1700 gpm.

b. at any RHR pump flow rate below the low level alarm (BLI-0053 17 inches).

c. with lower RHR flow rates at a lower reactor vessel level.

d. with lower RHR flow rates at a higher reactor vessel level.

QUESTION: 088 (1.00)

The plant is in Mode 5 for maintenance. RCS Level has been lowered in accordance with OTN-BB-00002 (RCS DRAINING) to 18 inches on BLI-0053 when the annunciator RHR LOOP 1 FLOW LO alarms. RHR Flow Indicators FI-618, 619 and pump discharge pressure are oscillating. The first IMMEDIATE action should be to:

a. increase charging flow to raise level.

b. stop the RHR pump.

c. dispatch an equipment operator to vent the running RHR pump.

d. start the standby RHR pump.

QUESTION: 089 (1.00)

The reactor is critical at 8% power, with the turbine rolling at 1800 rpm. Investigation reveals that IR Channel N36 has failed. (ASSUME NO REACTOR TRIP)

Which ONE of the following actions should be taken?

a. Restore the N36 Channel to operable status before increasing power above 10%.

b. Place the N36 Level Trip Bypass Switch in BYPASS position and continue power increase.

c. Place the N36 Level Trip Bypass Switch in BYPASS and then reduce power to less than 5%.

d. Reduce power to less than 5% and then place the N36 Level Trip Bypass Switch in BYPASS.

QUESTION: 090 (1.00)

The unit is in Mode 3 with the following conditions:

-Tave 557 degrees F
-Pressurizer pressure 2235 psig
-RCS Boron at 1000 ppm (ECP value)
-Source Range Counts 30 cps
-all rods inserted
-72 hours since shutdown

The Shutdown Banks are withdrawn in accordance with OTG-ZZ-00002, REACTOR STARTUP. When the Shutdown Banks are fully withdrawn, the Source Range counts are 110 cps on N31 and 65 cps on N32.

Which ONE of the following actions should be taken?

a. Manually shift the charging pump suction to the RWST.

b. Verify operability of Channels N31 and N32.

c. Immediately open the Reactor Trip Breakers.

d. Perform immediate boration in accordance with OTG-ZZ-00002.

QUESTION: 091 (1.00)

A Steam Generator Tube Leak as defined by OTO-BB-00001, STEAM GENERATOR TUBE LEAK, is a leak within the capacity of:

a. one centrifugal charging pump with letdown isolated.

b. two centrifugal charging pumps with normal letdown (75 gpm).

c. one centrifugal charging pump with normal letdown (75 gpm).

d. two centrifugal charging pumps with letdown isolated.

QUESTION: 092 (1.00)

Feedwater Flow to a Steam Generator that has a leaking or ruptured tube is not isolated in procedure E-3 (STEAM GENERATOR TUBE RUPTURE) or OTO-BB-00001 (STEAM GENERATOR TUBE LEAK) until level is greater than 4% in the narrow range. This is done to provide:

a. indication for inventory control.

b. a minimum radiological release to the environment.

c. a thermal layer to maintain Steam Generator pressure.

d. cooling initially which lowers Steam Generator pressure.

QUESTION: 093 (1.00)

When procedure E-3 (STEAM GENERATOR TUBE RUPTURE) is entered for a tube rupture occurring in B or C generator, steam supply to Turbine Driven Auxiliary Feedwater Pump is isolated:

a. locally with manual valves AB-V085 and 87 (Steam supply B and C).

b. from the control room with AB-HIS-5A and 6A (Steam supply B and C).

c. by a high (red) alarm on radiation monitor FC-RE-385 (TD AFW).

d. by an SI actuation but will reopen with an AFAS signal.

QUESTION: 094 (1.00)

The first step in the emergency procedures attachment for restarting a reactor coolant pump verifies that SI has been reset.

Which ONE of the following describes the consequences of omitting this condition?

a. The reactor coolant pump would have no number 1 seal leakoff flow path.

b. CCW would not be available to the any of the reactor coolant pump heat exchangers.

c. Starting any reactor coolant pump would cause an undervoltage condition and a shutdown sequencer on XBN02.

d. Starting reactor coolant pump C or D would cause a shutdown sequencer on XBN02 due to undervoltage.

QUESTION: 095 (1.00)

The following plant conditions exist:

-Reactor trip/ SI actuated	
-RCS Temperature	500 degrees F
-pressurizer pressure	2000 psig
-steam generator (A,B,C) pressure	450 psig
-steam generator level A,B,C	50% wide range
-steam generator D pressure	0 PSIG
-steam generator level D	5% wide range
-auxiliary feedwater flow	100,000 lbm/hr to each SG
-containment pressure	10 psig

Which ONE of the following actions should be taken with the auxiliary feedwater system? (ASSUME MSIVs ARE SHUT)

a. Reduce flow on all steam generators until 300,000 lbm/hr total flow is achieved.

b. Isolate auxiliary feedwater flow to D steam generator to minimize containment pressure and maintain 300,000 lbm/hr to the other 3 intact SGs.

c. Maintain 15,000 lbm/hr to the D steam generator to avoid dryout and 300,000 lbm/hr to the other 3 intact SGs.

d. Maintain 15,000 lbm/hr to the D steam generator to avoid dryout and reduce flow to 260,000 lbm/hr to the other 3 intact SGs to minimize cooldown of the RCS.

QUESTION: 096 (1.00)

In the event of a loss of Instrument air, the Atmospheric relief valve accumulators are sized to allow ______ hours of operation with an atmospheric relief cycle every ______ minutes and one turbine driven auxiliary feedwater valve cycle every ______ minutes.

a. 20, 10, 8

b. 20, 8, 10

c. 8, 20, 10

d. 8, 10, 20

QUESTION: 097 (1.00)

125 VDC Bus NK51 supply breaker has tripped due to a ground fault. The loss of this Red Train Power Supply to the Main Steam Isolation Valves will cause:

a. all four to drift shut.

b. A and C to drift shut.

c. a loss of slow close for A and C but fast close can still be accomplished with the Yellow train switch.

d. a loss of slow close for all four valves but fast close is still available with the Yellow train switch.

QUESTION: 098 (1.00)

Pressurizer level transmitter LT-459 has failed to the low end of the scale. The control functions have been transferred from LT-459 to the alternate channel LT-461. Which ONE of the following describes the status of the letdown system.

a. Letdown can be returned to service by first opening loop isolation BG-HIS-459 and then opening one of the orifice valves BG-HIS-8149AA,BA, or CA.

b. Letdown can be returned to service by opening loop isolation BG-HIS-459.

c. Excess letdown must be used until LT-459 is repaired.

d. Letdown can be returned to service by first opening one of the orifice valves BG-HIS-8149AA,BA, or CA and then opening loop isolation BG-HIS-459.

QUESTION: 099 (1.00)

The plant is in Mode 1 when an inadvertent safety injection occurs and all systems respond properly. Transition has been made from E-0 step 26 to ES-1.1, SI TERMINATION. SI has just been reset in accordance with step 1 when offsite power is lost.

Which ONE of the following describes the response of the on-site safety related electrical system?

a. Both emergency diesels start, the D/G output breakers shut, then the LOCA sequencer actuates.

b. An electrical load shed occurs, both D/G output breakers shut, then the S/D sequencer actuates.

c. Both emergency diesels start, the D/G output breakers shut, then the S/D sequencer actuates.

d. An electrical load shed occurs, both D/G output breakers shut, then the LOCA sequencer actuates.

QUESTION: 100 (1.00)

The plant is in Mode 3 following a loss of all A.C. Power. Electrical Bus NB02 has just been locally re-energized while the control room was adjusting steam generator levels in step 13 of ECA-0.0, LOSS OF ALL AC. The SRO leaves step 13 and goes immediately to step 24 as directed by an earlier caution statement. This action is required to:

a. preserve water in the Condensate Storage Tank.

b. maintain a secondary heat sink.

c. minimize RCS leakage through the RCP seals.

d. automatically load all ECCS pumps on the energized bus.

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

ANSWER: 001 (1.00)

d.

REFERENCE:

LP, Administrative Procedures, A-1, p22. OBJ. F3

[3.7/4.1]
194001K102 ..(KA's)

ANSWER: 002 (1.00)

c.

REFERENCE:

LP, Administrative Procedures, A-1, p17 OBJ. C2

[4.1/3.9]
194001A102 ..(KA's)

ANSWER: 003 (1.00)

d.

REFERENCE:

STANDING ORDER LOG, 90-010

[2.5/3.4]
194001A103 ..(KA's)

ANSWER: 004 (1.00)

b.

REFERENCE:

APA-ZZ-00802, CONFINED SPACE ENTRY PERMIT, P5.

$194001K114^{[3.3/3.6]}$ ..(KA's)

ANSWER: 005 (1.00)

d.

REFERENCE:

APA-ZZ-01001, Callaway Plant ALARA Program, P. 2

$194001K104^{[3.3/3.5]}$ ..(KA's)

ANSWER: 006 (1.00)

a.

REFERENCE:

APA-ZZ-00744, P8

$194001K116^{[3.5/4.2]}$ ..(KA's)

ANSWER: 007 (1.00)

b.

REFERENCE:

APA-ZZ-01000, Section 4.7.3.10.1, Rev 7.

194001K103 [2.8/3.4] ..(KA's)

ANSWER: 008 (1.00)

c.

REFERENCE:

EIP-ZZ-00102 P. 3

194001A116 [3.1/4.4] ..(KA's)

ANSWER: 009 (1.00)

d.

REFERENCE:

APA-ZZ-00702, P.2

194001K116 [3.5/4.2] ..(KA's)

ANSWER: 010 (1.00)

a.

REFERENCE:

ODP-ZZ-00004, P5

[3.6/3.7]
194001K101 ..(KA's)

ANSWER: 011 (1.00)

c.

REFERENCE:

LP, Operations Department Procedures, A-5, Pg. 27, OBJ. G2d

[3.3/3.4]
194001A101 ..(KA's)

ANSWER: 012 (1.00)

c.

REFERENCE:

LP A-3, HEALTH PHYSICS PROGRAM AND PROCEDURES, APA-ZZ-01000, REV.7, PAGE 28
PLANT MANUAL APA-ZZ-01000 REV. 7 PAGE 28, SECTION 4.7.1.5.3

[2.8/3.4]
194001K103 ..(KA's)

ANSWER: 013 (1.00)

c.

REFERENCE:

EIP-ZZ-00102, SECTION 4.5, pg. 3.

[3.1/4.4]

194001A116 ..(KA's)

ANSWER: 014 (1.00)

d.

REFERENCE:

Curve Book Section 7, Table 7-1
Curve Book Section 7, Figure 7.3, Boron Addition

[2.6/3.1]
194001A108 ..(KA's)

ANSWER: 015 (1.00)

b.

REFERENCE:

ODP-ZZ-00019, CONTAINMENT ACCESS AND INTEGRITY, P.8 SECTION 4.1.2.4

[3.1/3.4]
194001K105 ..(KA's)

ANSWER: 016 (1.00)

b.

REFERENCE:

ODP-ZZ-00003, SHIFT RELIEF AND TURNOVER, SECTION 4.2.3.2, 4.3.2.2

194001A103 [2.5/3.4] ..(KA's)

ANSWER: 017 (1.00)

b.

REFERENCE:

TECHNICAL SPECIFICATIONS TABLE 6.2-1.

194001A103 [2.5/3.4] ..(KA's)

ANSWER: 018 (1.00)

a.

REFERENCE:

OSP-SE-00004, NIS POWER RANGE HEAT BALANCE

015000A101 [3.5/3.8] ..(KA's)

ANSWER: 019 (1.00)

d.

REFERENCE:

OTN-BB-00003, REACTOR COOLANT PUMPS, SECTION 2.17

[3.2/3.5]
003000K504 ..(KA's)

ANSWER: 020 (1.00)

a.

REFERENCE:

Technical Specifications, pg. 3/4 2-2(a)
LP B-2 T61.003B.6, POWER DISTRIBUTION LIMITS, OBJ. O.

[3.1/3.8]
015000G011 ..(KA's)

ANSWER: 021 (1.00)

a.

REFERENCE:

LP-26 T61.0110.6, ROD CONTROL, P.19, OBJ. M
LP-26 ATTACHMENT SNP-CR-16 REV. 4, OBJ. H

[3.7/3.9]
001050A201 ..(KA's)

ANSWER: 022 (1.00)

b.

REFERENCE:

EOP, E-0 STEP 4
LP D-3 T61.003D.6, E-1, P. 43
Annunciator Response for Window 47D
[4.1/4.3]
026020K403 ..(KA's)

ANSWER: 023 (1.00)

c. or d. KMS
2/4/91

REFERENCE:

SNUPPS Funct. Diagram, 7250D64, Sheet 7, 8
Technical Specifications, Section 3.3, Table 3.3-4
E-0, Step 4

[4.2/4.4]

013000K101 ..(KA's)

ANSWER: 024 (1.00)

d.

REFERENCE:

LP-52 T61.0110.6, ESFAS, OBJ B.3 P.19

[4.3/4.4]
013000A402 ..(KA's)

ANSWER: 025 (1.00)

c.

REFERENCE:

LP-29 T61.0110.6, INCORE INSTRUMENTATION, P.10, OBJ. H

[3.1/3.3]
017020K403 ..(KA's)

ANSWER: 026 (1.00)

a.

REFERENCE:

LP-40 T61.0110.6, CONTAINMENT VENTILATION, P.17, OBJ. B.1

[2.6/3.0]
022000A203 ..(KA's)

ANSWER: 027 (1.00)

a.

REFERENCE:

OTO-AB-00003, STEAM GENERATOR PRESSURE CHANNEL FAILURE, SECTION 4.1
OTO-AB-00003, STEAM GENERATOR PRESSURE CHANNEL FAILURE, ATTACHMENT 4
[3.0/3.3]
059000A211 ..(KA's)

ANSWER: 028 (1.00)

b.

REFERENCE:

LP-25 T61.0110.6, AUXILIARY FEEDWATER SYSTEM, OBJ. E P.6
TECHNICAL SPECIFICATION 3.7.1.3 BASES

[2.7/3.8]
061000G006 ..(KA's)

ANSWER: 029 (1.00)

a.

REFERENCE:

LP-25 T61.0110.6, AUXILIARY FEEDWATER SYSTEM, OBJ, II.C.3 P. 18-19

[3.1/3.4]
061000K404 ..(KA's)

ANSWER: 030 (1.00)

c.

REFERENCE:

LP-25 T61.0110.6, AUXILIARY FEEDWATER SYSTEM, OBJ. II.C.5 P. 28

[3.9/4.2]
061000K401 ..(KA's)

ANSWER: 031 (1.00)

c.

REFERENCE:

LP-36 T61.0110, PROCESS RADIATION AND AREA RADIATION MONITORING SYSTEMS ATTACHMENT 1, P. 2 OF 34

[2.8/2.8]
071000A303 ..(KA's)

ANSWER: 032 (1.00)

d.

REFERENCE:

LP-16 T61.0110.6, RADWASTE SYSTEMS, P.19, OBJ. J

[2.7/2.9]
068000K107 ..(KA's)

ANSWER: 033 (1.00)

a.

REFERENCE:

LP-40 T61.0110.6, CONTAINMENT VENTILATION, OBJ. K P. 43
OTA-SP-RM011, P.3
SNUPPS DRWG M-22GS01(Q), CONTAINMENT HYDROGEN CONTROL

[3.5/3.9]
072000K102 ..(KA's)

ANSWER: 034 (1.00)

d.

REFERENCE:

OSP-SE-00003, SECTION 6.3
TECHNICAL SPECIFICATION 3.2.4
LP B-2 T61.003B.6, POWER DISTRIBUTION LIMITS, OBJ.Q

[3.5/3.7]
015000A104 ..(KA's)

ANSWER: 035 (1.00)

c.

REFERENCE:

LP A-6 T61.003A.6, PLANT HEATUP, OBJ.C.3 P.5

[2.5/2.8]
003000A205 ..(KA's)

ANSWER: 036 (1.00)

c.

REFERENCE:

LP A-11 T61.003A.6, PLANT COOLDOWN OBJ.C6 P.4

[3.4/3.9]
002000G010 ..(KA's)

ANSWER: 037 (1.00)

b.

REFERENCE:

LP-9 T61.0110.6, REACTOR COOLANT SYSTEM, OBJ.C.4 P. 23
STEAM TABLE

[3.5/4.0]
010000K501 ..(KA's)

~~ANSWER: 038 (1.00)~~

~~b.~~

Deleted
2/4/91
KMS

~~REFERENCE:~~

~~FACILITY EXAM BANK QUES. #69 SDB-02PF-04C~~
~~LP B-4 T61.003B.6, PRZR PRESSURE/LEVEL, P. 8 OBJ.A2~~

~~[3.6/3.4]~~
~~010000A402 ..(KA's)~~

ANSWER: 039 (1.00)

a.

REFERENCE:

OTO-BB-00003, RCS EXCESSIVE LEAKAGE, P. 3
IMMEDIATE ACTIONS

[3.8/3.9]
011000A203 ..(KA's)

ANSWER: 040 (1.00)

d.

REFERENCE:

LP B-4 T61.003B.6, I&C REVIEW, OBJ.C P.5
LP-27 T61.0110.6, REACTOR PROTECTION SYSTEM, OBJ. I.D
Tech Spec Table 2.2-1

[3.6/4.2]
012000G011 ..(KA's)

ANSWER: 041 (1.00)

b.

REFERENCE:

LP-40 T61.0110.6, CONTAINMENT VENTILATION, OBJ. H P. 35
OTN-GS-00001, CONTAINMENT HYDROGEN CONTROL SYSTEM, P.1 SECTION 2.

[2.6/3.1]
028000K601 ..(KA's)

ANSWER: 042 (1.00)

d.

REFERENCE:

LP-40 T61.0110.6, CONTAINMENT VENTILATION, OBJ.K P. 50

[3.5/3.9]
028000A202 ..(KA's)

ANSWER: 043 (1.00)

c.

REFERENCE:

LP-20 T61.0110.6, MAIN STEAM, OBJ. F P. 11

[3.2/3.3]
039000A105 ..(KA's)

ANSWER: 044 (1.00)

b.

REFERENCE:

LP-36 T61.0510 6, MAIN TURBINE CONTROLS AND CONTROL OIL, OBJ. 1 P.21

[3.4/3.5]
075020K301 ..(KA's)

ANSWER: 045 (1.00)

b.

REFERENCE:

LP-3 T61.016C.6, STANDBY DIESEL SYSTEM, OBJ. 2.2.1 P.13

[2.7/3.1]
064000K201 ..(KA's)

ANSWER: 046 (1.00)

d.

REFERENCE:

LP-2, SERVICE POWER, OBJ. F P.5

[3.5/3.6]
062000A305 ..(KA's)

ANSWER: 047 (1.00)

c.

REFERENCE:

LP-35, FIRE PROTECTION, OBJ. B,C P. 9-10

[2.9/3.3]
086000A301 ..(KA's)

ANSWER: 048 (1.00)

c.

REFERENCE:

TECHNICAL SPECIFICATION 3/4.4.2 and BASES

[3.0/3.5]
002000K612 ..(KA's)

ANSWER: 049 (1.00)

a.

REFERENCE:

LP B-4, I AND C SYSTEMS REVIEW, A.1 P. 6-7

[3.2/3.6]
010000K603 ..(KA's)

ANSWER: 050 (1.00)

c.

REFERENCE:

LP B-4, I AND C CONTROL SYSTEM REVIEW, OBJ.A.3 P.14

[3.6/3.8]
035010K401 ..(KA's)

ANSWER: 051 (1.00)

d.

REFERENCE:

CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, PRS-2, 2-1
LP B-4 T61.003B.6, I and C SYSTEMS REVIEW, OBJ. A.2 P.11-12

[3.4/3.6]
011000A210 ..(KA's)

ANSWER: 052 (1.00)

a.

REFERENCE:

LP-27 T61.0110.6, REACTOR PROTECTION, OBJ. D P.24
OTG-ZZ-00001, pg. 24a
OTG-ZZ-00005, pg. 9

[3.9/4.3]
012000K402 ..(KA's)

ANSWER: 053 (1.00)

c.

REFERENCE:

OTO-BB-00003, RCS EXCESSIVE LEAKAGE, P.2
CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, CCW-7, 1-1
LP-10 , COMPONENT COOLING WATER, OBJ. H P.2-3

[2.8/3.0]
008000K103 ..(KA's)

ANSWER: 054 (1.00)

c.

REFERENCE:

CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, MSS-13, 2-2
LP-20 MAIN STEAM, OBJ. K P.25-27

[3.1/3.2]
041020A102 ..(KA's)

ANSWER: 055 (1.00)

d.

REFERENCE:

CURVE BOOK, Figure 3-4a rev.4

[2.5/2.7]
045010K508 ..(KA's)

ANSWER: 056 (1.00)

b.

REFERENCE:

OTG-ZZ-00006, PLANT COOLDOWN HOT STANDBY TO COLD SHUTDOWN
PRECAUTIONS AND LIMITATIONS, P.2
TECHNICAL SPECIFICATION 3/4.4.9

[3.5/3.6]
005000A101 ..(KA's)

ANSWER: 057 (1.00)

a.

REFERENCE:

TO-SF-00004, MISALIGNMENT OF CONTROL RODS, IMMEDIATE ACTIONS

[3.4/3.6]
000005G010 ..(KA's)

ANSWER: 058 (1.00)

d.

REFERENCE:

OTO-SF-00003, DROPPED CONTROL ROD, IMMEDIATE ACTIONS

[3.9/3.8]
000003G010 ..(KA's)

ANSWER: 059 (1.00)

c.

REFERENCE:

LP D-3 T61.003d.6 890701, E-1 STEP 1 NOTE 4 OBJ.2

[4.0/4.1]
000011G012 ..(KA's)

ANSWER: 060 (1.00)

d.

REFERENCE:

LP D-3 R61.003d.6, E-1 ,PAGES 18,19

[4.2/4.2]
000011A111 ..(KA's)

ANSWFR: 061 (1.00)

c.

REFERENCE:

OTO-BB-00002, P. 4
LP B-1 T61.003B.6, OTO PROCEDURE REVIEW, OBJ. 04

[3.7/3.7]
000015A210 ..(KA's)

ANSWER: 062 (1.00)

b.

REFERENCE:

LP D-2 T61.003D.6, E-0, STEP 1, OBJ. A.3
LP D-7 T61.003D.6, FR-S.1, STEP 4

[3.9/4.4]
000024A202 ..(KA's)

ANSWER: 063 (1.00)

b.

REFERENCE:

OTO-EG-00001, IMMEDIATE ACTIONS, PAGE 2
LP B-1 T61.003B.6, CCW OTO-EG, OBJ. Y5

[2.8/3.1]
000026A206 ..(KA's)

ANSWER: 064 (1.00)

b.

REFERENCE:

LP D-7 T61.003D.6, FR-S.1, P. 27

[4.4/4.7]
000029K312 ..(KA's)

ANSWER: 065 (1.00)

a.

REFERENCE:

LP D-2 T61.003D.6, E-O, PAGE 50
CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, MSS-3, 2-1 EXPECTED RESPONSE
CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, RCS-6, 2-1 EXPECTED RESPONSE

[4.6/4.7]
000040A203 ..(KA's)

ANSWER: 066 (1.00)

c.

REFERENCE:

ECA-0.0 STEP 3
LP D-6 TT61.003D.6, ECA-0.0, OBJ.5 P. 98

[4.1/4.3]
000055G010 ..(KA's)

ANSWER: 067 (1.00)

a.

REFERENCE:

LP S-28 T61.0110.6, EXCORE NUCLEAR INSTRUMENTATION, OBJ. 1.E P. 59

[4.0/4.3]
000057A219 ..(KA's)

ANSWER: 068 (1.00)

b.

REFERENCE:

TECHNICAL SPECIFICATIONS 3/4.4.8 BASES

[2.8/3.4]
000076A202 ..(KA's)

ANSWER: 069 (1.00)

c.

REFERENCE:

TECHNICAL SPECIFICATION 3/4.4.8 BASES
STEAM TABLES

[2.8/3.4]
000076A202 ..(KA's)

ANSWER: 070 (1.00)

a.

REFERENCE:

LP D-6, T61.003D.6, ECA-0, OBJ.4.1 P. 21

[3.9/4.0]
000055G012 ..(KA's)

ANSWER: 071 (1.00)

b.

REFERENCE:

OTO-ZZ-00001, CONTROL INACCESSIBILITY, SECTION 5.1.1.1
LP B-1 T61.003B.6, OTO-ZZ-00001, CONTROL ROOM INACCESSIBILITY
OBJ. TT5a

[3.3/4.1]
000068G001 ..(KA's)

ANSWER: 072 (1.00)

c.

REFERENCE:

OTO-ZZ-00001, CONTROL ROOM INACCESSIBILITY, ATTACHMENT 5 P.1

[4.3/4.4]
000068A211 ..(KA's)

ANSWER: 073 (1.00)

d.

REFERENCE:

LP D-2 T61.003D.6, RCP TRIP CRITERIA. P.1 OBJ. B.2,5

[4.1/4.2]
000011K314 ..(KA's)

ANSWER: 074 (1.00)

c.

REFERENCE:

CSF-1, CRITICAL SAFETY FUNCTION STATUS TREE, P.2
LP D-6 T61.003D.6, CORE COOLING STATUS TREES, P.2
STEAM TABLES

[3.7/4.1]
000074K104 ..(KA's)

ANSWER: 075 (1.00)

a.

REFERENCE:

TECHNICAL SPECIFICATIONS, 3/4.1.1 BASES

[4.1/4.4]
000040K105 ..(KA's)

ANSWER: 076 (1.00)

a.

REFERENCE:

LP B-1 T61.003B.6, OTO-SF-00006, OBJ. RR1 P.137

[3.1/3.3]
000005G005 ..(KA's)

ANSWER: 077 (1.00)

d.

REFERENCE:

OTO-AD-00001, P.1
OTO-AC-00001, P.1

[3.9/4.1]
000051A202 ..(KA's)

ANSWER: 078 (1.00)

b.

REFERENCE:

OTO-SF-00001, CONTINUOUS CONTROL ROD INSERTION, 3.0
OTO-AD-00005, TURBINE IMPULSE PRESSURE CHANNEL FAILURE, ATTACHMENT 1.

[3.5/3.8]
000001K105 ..(KA's)

ANSWER: 079 (1.00)

c.

REFERENCE:

LP D-6 T61.003D.6, FR-C P. 41 OBJ. 4.1

[4.0/4.4]
000074K311 ..(KA's)

ANSWER: 080 (1.00)

a.

REFERENCE:

LP D-8 T61.003D-6, FR-P.1, P. 9 OBJ. 4.1

[4.1/4.4]
000040K101 ..(KA's)

ANSWER: 081 (1.00)

b.

REFERENCE:

LP D-1 T61.003D.6, ERG AND INTRO TO USERS GUIDE, OBJ. B.7 P. 26 & 27

[3.5/3.5]
000069G012 ..(KA's)

ANSWER: 082 (1.00)

c.

REFERENCE:

LP D-2 T61.003D.6, E-0, P. 14 OBJ. A.6

[3.7/4.0]
000007K103 ..(KA's)

ANSWER: 083 (1.00)

d.

REFERENCE:

LP-23 T61.0110.6, MAIN FEEDWATER SYSTEM, pg. 11-13.
SNUPPS DRWG 7250D64 SHT. 13 & 14.

[3.8/3.7]
000007A102 ..(KA's)

ANSWER: 084 (1.00)

b.

REFERENCE:

LP D-2 T61.003D.6, E-0, P.42 OBJ. A.6
STEAM TABLES

[3.9/3.9]
000008A203 ..(KA's)

ANSWER: 085 (1.00)

a.

REFERENCE:

LP D-3 T61.003D.6, ES-1.2, P.37 OBJ. 2

[3.4/3.6]
000009K301 ..(KA's)

ANSWER: 086 (1.00)

c.

REFERENCE:

TECHNICAL SPECIFICATIONS 3/4 4.6.2

[3.3/3.8]
000009A233 ..(KA's)

ANSWER: 087 (1.00)

c.

REFERENCE:

OTN-BB-00002, RCS DRAINING, ATTACHMENT 4
LP B-1, T61.003B.6, OTO-EJ-00001 OBJ. 2

[3.2/3.2]
000025K202 ..(KA's)

ANSWER: 088 (1.00)

b.

REFERENCE:

OTO-EJ-00001, LOSS OF RHR FLOW, IMMEDIATE ACTIONS
LP B-1 T61.003B.6, LOSS OF RHR FLOW, OBJ. 3

[3.9/3.9]
000025G010 ..(KA's)

ANSWER: 089 (1.00)

a.

REFERENCE:

LP B-1 T61.003B.6, OTO-SE-00002, OBJ. KK5 P.118
Tech Spec 3.3.1, Table 3.3-1, Item 5

[2.8/3.4]
000033G008 ..(KA's)

ANSWER: 090 (1.00)

b.

REFERENCE:

OTG-ZZ-00002, REACTOR STARTUP, PRECAUTIONS AND LIMITATIONS 2.8, 2.9
LP B-1 T61.003B.6, OTO-SE-00001, OBJ. JJ3 P.114
CURVE BOOK TABLE 2-2 , SUMMARY OF CONTROL ROD WORTH
ODP-ZZ-00020, Rev. 4, Attachment 1, pg. 1 of 4.
[3.6/3.9]
000032A202 ..(KA's)

ANSWER: 091 (1.00)

c.

REFERENCE:

OTO-BB-00001, STEAM GENERATOR TUBE LEAK, SECTION 1.0, 6.0
LP B-1 T61.003B.6, OTO-BB-00001, STEAM GENERATOR TUBE LEAK, OBJ. N1

[3.3/3.5]
000037G007 ..(KA's)

ANSWER: 092 (1.00)

c. or b. KMS
2/4/91

REFERENCE:

LP D-5 T61.003D.6, E-3, OBJ.F P. 41
OTO-BB-00001, STEAM GENERATOR TUBE LEAK, STEP 6.6.6

[4.2/4.4]
000037K307 ..(KA's)

ANSWER: 093 (1.00)

a.

REFERENCE:

LP D-5 T61.003D.6, E-3, STEP 3, OBJ. F.
LP B-1 T61.003B.6, OTO-BB-00001, P.42

[4.6/4.7]
000038A132 ..(KA's)

ANSWER: 094 (1.00)

d.

REFERENCE:

LP D-5 T61.003D.6, E-3, OBJ. F P.193

[4.2/4.5]
000038K306 ..(KA's)

ANSWER: 095 (1.00)

b.

REFERENCE:

LP D-4 T61.003D.6, E-2, STEP 4, OBJ. D

[4.5/4.4]
000054A101 ..(KA's)

ANSWER: 096 (1.00)

d.

REFERENCE:

LP-25 T61.0110.6, AUXILIARY FEEDWATER, OBJ. II.C.4 P.20-21

[2.9/3.4]
000065K303 ..(KA's)

ANSWER: 097 (1.00)

c.

REFERENCE:

LP-49 T61.0110.6, MSFIS, P.4 OBJ. C

[3.5/3.9]
000058A203 ..(KA's)

ANSWER: 098 (1.00)

a.

REFERENCE:

LP-11 T61.0110.6, CVCS, OBJ.C P.1-2, 1-4

[2.9/3.2]
000028A209 ..(KA's)

ANSWER: 099 (1.00)

b.

REFERENCE:

LP-51 T61.0110.6, LSELS, OBJ. D,E P. 4, 12.
FACILITY EXAM BANK, QUES. #56, CRK-01PD-02C

[3.8/3.9]
000056A247 ..(KA's)

ANSWER: 100 (1.00)

c.

REFERENCE:

LP D-6 T61.003D.6, ECA-0.0, P. 34 OBJ. E

[4.4/4.7]
eqb.3 ~ % *END
000056K302 ..(KA's)

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

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION
REGION 3

FACILITY: Callaway

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 91/01/28

CANDIDATE:

INSTRUCTIONS TO CANDIDATE:

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

NUMBER QUESTIONS	TOTAL POINTS	CANDIDATE'S POINTS	CANDIDATE'S OVERALL GRADE (%)
~~100~~ 99	~~100.00~~ 99.00		
KMS 2/4/91	KMS 2/4/91		

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

Candidate's Signature

MASTER COPY

DATA SHEET

REACTOR THEORY FORMULAS:

$$P = P_0 e^{t/\tau}$$

$$P = \frac{\Sigma \phi_{th} V}{3.12 \times 10^{10} \text{ fissions/sec}}$$

$$P_{th} = \frac{1}{1 + (B^2 L_{th}^2)} = e^{-(B^2 L_{th}^2)}$$

$$P_f = e^{-(B^2 L_f^2)}$$

$$p = e^{-[N][I_{eff}]/\xi\Sigma_s}$$

$$C_1 \; (1-K_{eff1}) = C_2 \; (1-K_{eff2})$$

$$m = \frac{1}{1-K} = \frac{C_{final}}{C_{initial}}$$

$$\alpha_T = \frac{1}{f}\frac{\Delta f}{\Delta t} + \frac{1}{p}\frac{\Delta p}{\Delta t} - B^2 \left(\frac{\Delta L_f^2}{\Delta t} + \frac{\Delta L_{th}^2}{\Delta t}\right)$$

$$K_{eff} = \epsilon \, P_f \, p \, P_{th} \, f \, \eta$$

$$P = P_0 10^{SUR(t)}$$

$$SUR = 26.06/\tau$$

$$\rho = \frac{l^*}{\tau} + \frac{\overline{\beta}_{eff}}{1 + \lambda\tau}$$

$$\rho = \frac{K - 1}{K}$$

$$\Delta\rho = \ln \frac{K_{final}}{K_{initial}}$$

$$\tau = \frac{\overline{\beta}_{eff} - \rho}{\lambda\rho}$$

$$\tau = \frac{l^*}{\rho}$$

$$P_1 = P_0 \frac{\overline{\beta}_{eff} - \rho_0}{\overline{\beta}_{eff} - \rho_1}$$

DATA_SHEET

THERMODYNAMICS_AND_FLUID_MECHANICS_FORMULAS:

$$\dot{Q} = \dot{m} \Delta h$$

$$\dot{Q} = \frac{2 \pi L \Delta T}{\frac{1}{K} + \frac{\ln R_2/R_1}{K_2} + \frac{\ln R_3/R_2}{K_3}}$$

$$\dot{Q} = U A (\Delta T_m)$$

$$\dot{Q} = \dot{m} c_p (\Delta T)$$

$$\dot{Q} = \alpha \delta A R^4$$

$$\eta = \frac{\dot{Q}_{in} - \dot{Q}_{out}}{\dot{Q}_{in}}$$

$$\eta = \frac{(h_{in} - h_{out})real}{(h_{in} - h_{out})ideal}$$

$$\eta_p = \frac{W_{actual}}{W_{supplied}}$$

$$\frac{P_1V_1}{T_1} = \frac{P_2V_2}{T_2}$$

$$\dot{m} = \rho AV$$

$$\rho_1A_1V_1 = \rho_2A_2V_2$$

$$\dot{m} = KA \sqrt{\Delta P_x \rho}$$

$$\dot{m}_{nc} = KA_Q \sqrt[3]{\dot{Q}} = KA \Delta T \sqrt{\Delta T} = KA \Delta \rho \sqrt{\Delta P}$$

$$\Delta T_m = \frac{\Delta T\ (in) - \Delta T\ (out)}{\ln \left(\frac{\Delta T\ (in)}{\Delta T\ (out)}\right)}$$

$$G = \frac{\Sigma_f \phi_{th}}{B.B \times 10^9}$$

$$T_{cl} - T_{ps} = \frac{Gr^2}{4k}$$

$$\dot{Q} = \frac{kA\Delta T}{\Delta x}$$

$$\dot{Q} = \frac{A \Delta T_{total}}{\frac{\Delta x_a}{K_a} + \frac{\Delta x_b}{K_b} + \ . \ . \ . + \frac{\Delta x_n}{K_n}}$$

DATA_SHEET

CENTRIFUGAL_PUMP_LAWS:

$\frac{N_1}{N_2} = \frac{\dot{m}_1}{\dot{m}_2}$

$\frac{(N_1)^2}{(N_2)^2} = \frac{H_1}{H_2}$

$\frac{(N_1)^3}{(N_2)^3} = \frac{P_1}{P_2}$

RADIATION_AND_CHEMISTRY_FORMULAS:

$R/hr = 6CE/d^2$

$I_x = I_0 e^{-mx}$

$C_1V_1 = C_2V_2$

$G = \frac{\text{Dilution Rate}}{\text{Volume}}$

$I = I_0 \left(\frac{1}{10}\right)^n$

$C = C_0 e^{-Gt}$

$A = A_0 e^{-\lambda t}$

$A = \lambda N$

CONVERSIONS:

$1\ gm/cm^3 = 62.4\ lbm/ft^3$

1 gal = 8.345 lbm

$1\ ft^3 = 7.48$ gal

1 gal = 3.78 liters

1 lbm = 454 grams

e = 2.72

π = 3.14159

1 KW = 738 ft-lbf/sec

1 KW = 3413 Btu/hr

1 HP = 550 ft-lbf/sec

1 HP = .746 KW

1 HP = 2545 Btu/hr

1 Btu = 778 ft-lbf

$1\ MEV = 1.54 \times 10^{-16}$ Btu

$h = 4.13 \times 10^{-21}$ M-sec

$1\ W = 3.12 \times 10^{10}$ fissions/sec

$g_c = 32.2\ lbm\text{-}ft/lbf\text{-}sec^2$

1 inch = 2.54 cm

$\sigma = 0.1714 \times 10^{-8}\ Btu/hr\ ft^2\ R^4$

Density of water (20 C) $62.4\ lbm/ft^3$

Avogadro's Number = 6.023×10^{23}

Heat of Vapor (H_2O) = 970 Btu/lbm

Heat of Fusion (ICE) = 144 Btu/lbm

1 AMU = 1.66×10^{-24} grams

Mass of Neutron = 1.008665 AMU

Mass of Proton = 1.007277 AMU

Mass of Electron = 0.000549 AMU

One atmosphere = 14.7 psia = 29.92 in. Hg

°F = 9/5 °C + 32

°C = 5/9 (°F - 32)

°R = °F + 460

°K = °C + 273

$c^2 = 931$ MEV/AMU

$C = 3 \times 10^8$ m/sec

DATA SHEET

AVERAGE THERMAL CONDUCTIVITY (K)

Material	K
Cork	0.025
Fiber Insulating Board	0.028
Maple or Oak Wood	0.096
Building Brick	0.4
Window Glass	0.45
Concrete	0.79
1% Carbon Steel	25.00
1% Chrome Steel	35.00
Aluminum	118.00
Copper	223.00
Silver	235.00
Water (20 psia, 200 degrees F)	0.392
Steam (1000 psia, 550 degrees F)	0.046
Uranium Dioxide	1.15
Helium	0.135
Zircaloy	10.0

MISCELLANEOUS INFORMATION:

$E = mc^2$

$KE = 1/2\ mv^2$

$PE = mgh$

$V_f = V_0 + at$

Geometric Object	Area	Volume
Triangle	$A = 1/2\ bh$	//////////////////
Square	$A = S^2$	//////////////////
Rectangle	$A = L \times W$	//////////////////
Circle	$A = \pi r^2$	//////////////////
Rectangular Solid	$A = 2(L \times W + L \times H + W \times H)$	$V = L \times W \times H$
Right Circular Cylinder	$A = (2\ \pi r^2)h + 2(\pi r^2)$	$V = \pi r^2 h$
Sphere	$A = 4\ \pi r^2$	$V = 4/3\ (\pi r^2)$
Cube	////////////////////////////////	$V = S^3$

DATA SHEET

MISCELLANEOUS INFORMATION (continued):

			10 CFR 20 Appendix B				
				Table I		Table II	
Material	Half-Life	Gamma Energy MEV per Disintegration		Col I Air uc/ml	Col II Water uc/ml	Col I Air uc/ml	Col II Water uc/ml
Ar-41	1.84 h	1.3	Sub	2×10^{-6}	-----	4×10^{-8}	------
Co-60	5.27 y	2.5	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	5×10^{-5}
I-131	8.04 d	0.36	S	9×10^{-9}	6×10^{-5}	1×10^{-10}	3×10^{-7}
Kr-85	10.72 y	0.04	Sub	1×10^{-5}	-----	3×10^{-7}	------
Ni-65	2.52 h	0.59	S	9×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
Pu-239	2.41×10^{4} y	0.008	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
Sr-90	29 y	-----	S	1×10^{-9}	1×10^{-5}	3×10^{-11}	3×10^{-7}
Xe-135	9.09 h	0.25	Sub	4×10^{-6}	-----	1×10^{-7}	------
Any single radionuclide with $T_{1/2}$ > 2 hr which does not decay by alpha or spontaneous fission				3×10^{-9}	9×10^{-5}	1×10^{-10}	3×10^{-6}

Neutron Energy (MEV)	Neutrons per cm^2 equivalent to 1 rem	Average flux to deliver 100 mrem in 40 hours
thermal	970×10^{6}	670
0.02	400×10^{6}	280 (neutrons)
0.5	43×10^{6}	30 ----------
10	24×10^{6}	17 cm^2 x sec

Linear Absorption Coefficients μ (cm^{-1})				
Energy (MEV)	Water	Concrete	Iron	Lead
0.5	0.090	0.21	0.63	1.7
1.0	0.067	0.15	0.44	0.77
1.5	0.057	0.13	0.40	0.57
2.0	0.048	0.11	0.33	0.51
2.5	0.042	0.097	0.31	0.49
3.0	0.038	0.088	0.30	0.47

TABLE 7-1 Rev. 000

REACTOR MAKEUP CONTROL SYSTEM NOMOGRAPHS

NOMOGRAPH CORRECTION FACTORS

7/26/83

Superintendent, Engineering Date

Plant Conditions			Correction Factor (K) (See Note)
Pressure (psig)	T (AVG) (F)	Pressurizer Level	
2235	547-570	Normal Operating	1.00
1600	500	No-Load	1.05
1200	450	No-Load	1.10
800	400	No-Load	1.16
400	350	No-Load	1.18
400	300	No-Load	1.20
400	300	Solid Water	1.35
400	200	No-Load	1.28
400	200	Solid Water	1.40
400	100	Solid Water	1.47

NOTE: CORRECTION FACTORS ARE APPLIED AS FOLLOWS:

(a) Boron Addition and Dilution Total Volume Nomographs

$$V_{(Corrected)} = K \times V_{(Nomograph)}$$

(b) Boron Addition and Dilution Rate Nomographs

$$\frac{dc}{dt}\text{ (Corrected)} = \frac{1}{K} \times \frac{dc}{dt}\text{ (Nomograph)}$$

FIGURE 7-3 Rev. 001

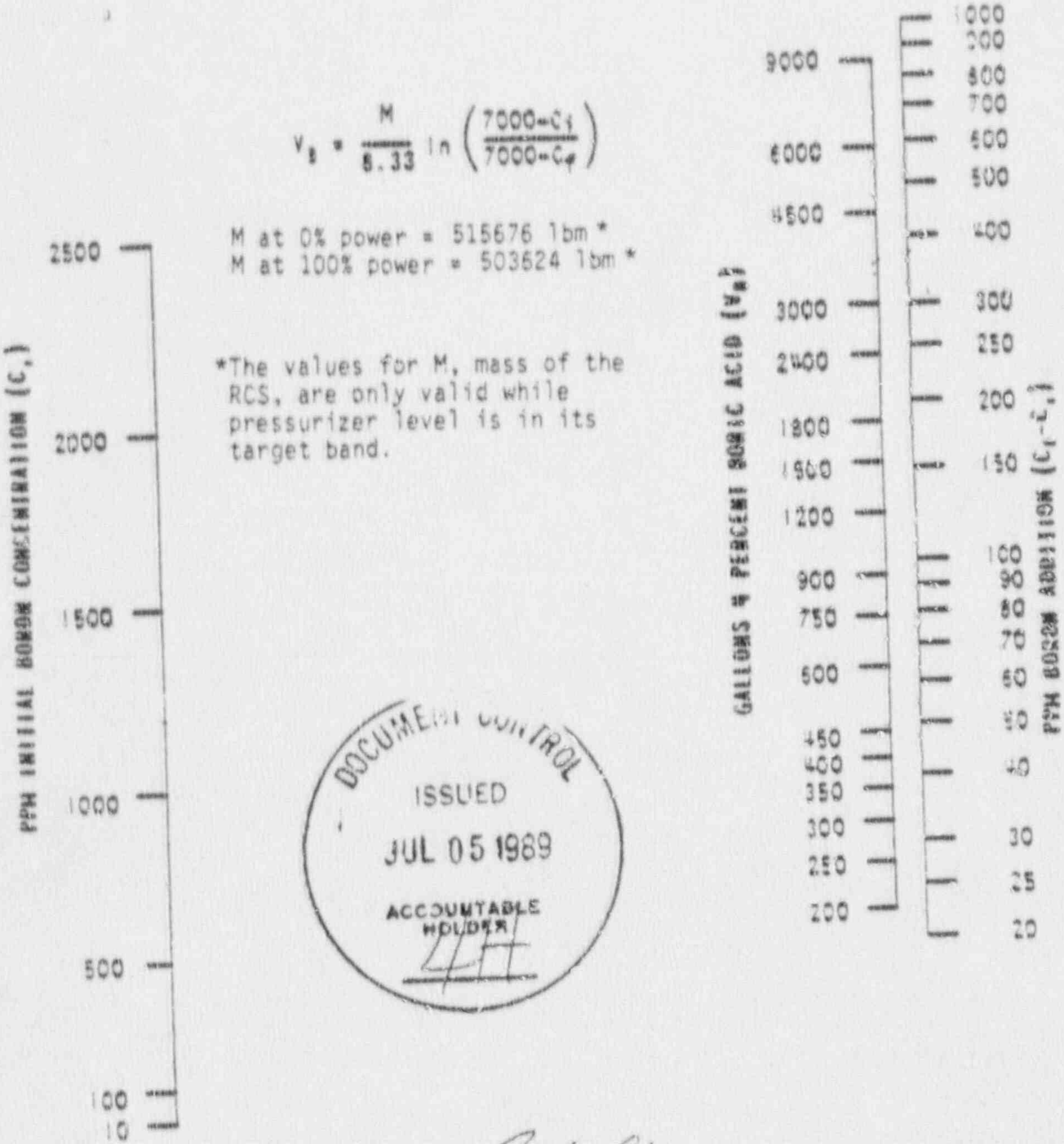
REACTOR MAKEUP CONTROL SYSTEM NOMOGRAPHS

BORON ADDITION

$$V_B = \frac{M}{8.33} \ln\left(\frac{7000-C_i}{7000-C_f}\right)$$

M at 0% power = 515676 lbm *
M at 100% power = 503624 lbm *

*The values for M, mass of the RCS, are only valid while pressurizer level is in its target band.



R Alfold 12-5-87
Superintendent, Engineering Date

NOTE: REFER TO TABLE 7-1 FOR CORRECTION FACTORS.

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.
2. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done after you complete the examination.
3. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.
4. Use black ink or dark pencil only to facilitate legible reproductions.
5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet.
6. Fill in the date on the cover sheet of the examination (if necessary).
7. You may write your answers on the examination question page or on a separate sheet of paper. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.
8. If you write your answers on the examination question page and you need more space to answer a specific question, use a separate sheet of the paper provided and insert it directly after the specific question. DO NOT WRITE ON THE BACK SIDE OF THE EXAMINATION QUESTION PAGE.
9. Print your name in the upper right-hand corner of the first page of answer sheets whether you use the examination question pages or separate sheets of paper. Initial each of the following answer pages.
10. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.
11. If you are using separate sheets, number each answer and skip at least 3 lines between answers to allow space for grading.
12. Write "Last Page" on the last answer sheet.
13. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

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

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

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

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

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

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

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

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

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

QUESTION: 001 (1.00)

The Workman's Protection Assurance Plan (APA-ZZ-003.0) specifies that equipment under maintenance requiring operation for test be issued a:

a. Hold Off Tag (Orange).

b. Dispatcher's Hold Off Tag (White with Red Letters).

c. Caution Tag (Yellow).

d. Local Control Tag (Pink).

QUESTION: 002 (1.00)

According to rules of usage for procedures (APA-ZZ-00100 Procedure requirements), the word SHALL in a step denotes:

a. a recommendation imposed by UE management on its employees, contractors, and agents which should be met unless there is sufficient reason not to perform the activity.

b. permission to perform activities and is neither a requirement nor a recommendation.

c. a legally binding requirement to which UE management has committed (e.g. in the FSAR).

d. a requirement imposed by UE management on its employees, contractors, and agents which are above and in excess of the legally binding requirement of the appropriate regulatory body.

QUESTION: 003 (1.00)

An emergency entry may be made into a suspected or unknown hazardous atmosphere without a confined space entry permit (APA-ZZ-00802) providing:

a. chemistry has sampled the atmosphere within the last 24 hours.

b. a self-contained breathing apparatus is worn.

c. evolutions performed will not change the atmosphere.

d. the buddy system is used and two people enter.

QUESTION: 004 (1.00)

Which ONE of the following describes the ENTIRE scope of activities covered under the Callaway Plant ALARA program in accordance with APA-ZZ-01001, CALLAWAY PLANT ALARA PROGRAM?

a. Maintenance work performed in the Reactor Building during refueling periods, without Health Physics monitoring.

b. Maintenance and operations work in the Reactor Building during refueling periods, including work monitored by Health Physics.

c. All plant activities not monitored by Health Physics.

d. All plant activities, including those monitored by Health Physics.

QUESTION: 005 (1.00)

During a shift activity, an unplanned impairment of an Auxiliary Building pressure boundary door is required for 30 minutes. Prior to impairing the door, the individual should:

a. contact SS/OS for approval.

b. notify the QC Inspector.

c. hang a Fire Barrier Impairment Tag on the door.

d. place fire fighting equipment by the door.

QUESTION: 006 (1.00)

Access will be denied into the RCA to any Callaway worker without a completed Special Instruction Form CA-285 if his/her weekly exposure exceeds 80% of the:

a. 0.2 rem weekly limit.

b. 0.5 rem weekly limit.

c. 2.0 rem yearly limit.

d. 5.0 rem yearly limit.

QUESTION: 007 (1.00)

When a fire is reported to the control room, which ONE of the following actions should occur?

a. The person reporting the fire should be directed to use the nearest fire extinguisher and attempt to put out the fire.

b. The person reporting the fire should be directed to use the nearest hose station and attempt to put out the fire.

c. The person should be directed to stay at the location of the fire until assistance arrives.

d. The control room should supply the person with a location to meet the Fire Brigade Leader.

QUESTION: 008 (1.00)

Verification of position for valve EC-V020 (ESW Train A to SFP Upstream Isolation), an accessible LOCKED CLOSED valve, should be accomplished by which ONE of the following methods?

a. Attempting to move the handwheel in the close direction only.

b. Attempting to move the handwheel in the open direction and then reclose the valve.

c. Using the valve position indicator or stem position to verify condition.

d. Checking that the seal is intact and that the valve is on the Locked Valve Deviation List.

QUESTION: 009 (1.00)

A "Working File" copy of a diesel generator surveillance procedure is to be used by a Reactor Operator in the control room for several days. How often must this procedure be verified to be a "Controlled" copy in accordance with ODP-ZZ-00009, Operations Department - Control of Documents?

a. Every 8 hours

b. Every 12 hours

c. Daily

d. Weekly

QUESTION: 010 (1.00)

When working in a radiologically controlled area, a worker must leave if his/her PIC reading reaches what percent of full scale:

a. 25%

b. 50%

c. 75%

d. 100%

QUESTION: 011 (1.00)

Using the attached Table and figures, REACTOR MAKEUP CONTROL SYSTEM NOMOGRAPHS, which ONE of the following volumes is required to borate 50 ppm at 350 degrees, 400 psig with no-load pressurizer level? (current boron concentration is 1000 ppm)

a. 385 gallons

b. 410 gallons

c. 455 gallons

d. 537 gallons

QUESTION: 012 (1.00)

When relieving the watch after 7 days absence from the control room, the watchstation logs should be reviewed, in accordance with ODP-ZZ-00003, Shift Relief And Turnover, for the previous:

a. day.

b. 3 days.

c. 5 days.

d. 7 days.

QUESTION: 013 (1.00)

The Control Room Command Function, during the absence of the Shift Supervisor, can be assumed by a(n):

a. STA, if licensed as a Senior Operator, in any mode.

b. Senior Reactor Operator (excluding the STA) in modes 1,2, and 3.

c. Reactor Operator in modes 4, 5 and 6.

d. Refueling Senior Reactor Operator in mode 6.

QUESTION: 014 (1.00)

During the performance of an NIS Power Range Heat Balance at 100% power, an operator uses a Feedwater Temperature 30 degrees LOWER than actual. Would the calculated value of power be HIGHER or LOWER than actual power and would an adjustment of the NIS Power Range Channels, based on this value, be CONSERVATIVE or NON-CONSERVATIVE with respect to protection setpoints.

a. higher/conservative

b. higher/non-conservative

c. lower/conservative

d. lower/non-conservative

QUESTION: 015 (1.00)

The unit is operating at 40% power steady state with all systems in their normal alignment. Reactor Coolant Pump B trips due to an electrical fault.

Which ONE of the following conditions describes the INITIAL response of S/G B level and steam flow after trip of the pump.

a. Level remains constant and steam flow decreases.

b. Level remains constant and steam flow increases.

c. Level changes rapidly and steam flow remains constant.

d. Level changes rapidly and steam flow decreases.

QUESTION: 016 (1.00)

With reactor power at 65%, penalty deviation outside the delta-I target band shall be accumulated on a time basis of:

a. One minute penalty for each minute outside of the target band.

b. One-half minute penalty for each minute outside of the target band.

c. One minute penalty for each one-half minute outside of the target band.

d. Zero penalty for time outside the target band.

QUESTION: 017 (1.00)

Which ONE of the following control interlocks will block only AUTO outward motion of the control rod banks?

a. C-1, HIGH FLUX IR

b. C-2, HIGH FLUX, PR

c. C-3, OT DELTA T

d. C-11, BANK D WITHDRAWAL STOP

QUESTION: 018 (1.00)

The CVCS is in operation with the Positive Displacement Pump running in automatic. Which ONE of the following describes the system response when a Safety Injection Signal is received?

The Positive Displacement Pump will:

a. continue in operation and will respond to changes in pressurizer level.

b. be load shed after the Undervoltage time delay expires.

c. be load shed and sequenced back on the LOCA Sequencer.

d. continue in operation at minimum speed, due to loss of instrument air to the controller.

QUESTION: 019 (1.00)

The unit is operating at 100% power, steady state, with all controls in auto, when the No. 1 SEAL LEAK OFF FLOW HIGH annunciator (72A) energizes. The operator discovers that the seal leakoff flow from the A reactor coolant pump is 9.5 gpm and seal leakoff temperature is 200 degrees. Which ONE of the following actions is required?

a. Trip the A pump immediately.

b. Trip the A pump within 2 minutes.

c. Close the No. 1 seal leakoff valve within five minutes and reduce power at 1% per minute to less than 48% before tripping the pump.

d. Close the No. 1 seal leakoff valve within five minutes and reduce power to allow tripping of the pump within 30 minutes.

QUESTION: 020 (1.00)

The Technical Specification action for insufficient shutdown margin requires immediate boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron. In accordance with OTO-ZZ-00003, Loss Of Shutdown Margin, the preferred method for accomplishing this requirement is to initiate boration flow:

a. with makeup control selected to borate and a 40 gpm flow rate.

b. through BG-HV-8104 Valve (emergency borate to the charging pumps suction).

c. through BG-V177 Valve (alternate immediate boration valve).

d. through BN-HIS-112D and BN-HIS-112E (RWST to charging pumps) suction.

QUESTION: 021 (1.00)

The Control Rod Drive System Bank Overlap Unit maintains the correct overlap of control banks and moves either group 1 or 2 utilizing input from which one of the following?

a. P/A Converter

b. Step Counters

c. Master Cycler

d. Slave Cyclers

QUESTION: 022 (1.00)

Control Rod M-12 in control bank D group 1 is being recovered after a blown stationary fuse was replaced. A procedure note in OTO-SF-00003, DROPPED CONTROL ROD, states an URGENT FAILURE alarm will be received when attempting to move the control rod.

Which ONE of the following describes why the alarm is present and why the control rod can be moved with this alarm present?

a. A regulation failure is present in power cabinet 2BD and rod M-12 is powered from cabinet 1BD.

b. A regulation failure is present in cabinets 1BD and 2BD but only inhibits auto rod movement.

c. A phase failure occurs when any lift coil is disconnected in a power cabinet but only inhibits auto rod movement.

d. A phase failure is present in power cabinet 1BD and 2BD but rods can be moved in individual bank position.

QUESTION: 023 (1.00)

The following plant conditions exists:

-Mode 3 following a reactor trip/Safety injection	
-RCS temperature	500 degrees
-PZR pressure	1900 psig
-Steam Generator Pressure	600 psig
-Steam Generator Levels	50% wide range
-Containment pressure	3.2 psig

Which ONE of the following describes the status of the Safety Actuation Signal Logic assuming NO operator actions were taken?

a. A low pressurizer pressure and low steam pressure signal are active.

b. A low pressurizer pressure signal is present but can be blocked.

c. A low steam generator pressure signal is present but can be blocked.

d. A containment pressure signal is active but can't be blocked.

QUESTION: 024 (1.00)

Which ONE of the following signals cause BOTH a safety injection and a main steamline isolation?

a. Low Steamline Pressure

b. High Steam Pressure Rate

c. High-1 containment Pressure

d. Low Pressurizer Pressure.

QUESTION: 025 (1.00)

Which ONE of the following conditions must be met prior to resetting the Motor Driven AFAS?

a. Indicated Steam Generator level in 3 of the 4 SG's above the lo-lo level setpoint.

b. Both Main Feedwater Pumps reset.

c. No undervoltage on NB01 or NB02.

d. AMSAC signal reset

QUESTION: 026 (1.00)

The Incore Thermocouples, by design, will operate satisfactorily in accident conditions and provide temperature indication up to:

a. 1300 degrees F.

b. 1650 degrees F.

c. 2300 degrees F.

d. 3300 degrees F.

QUESTION: 027 (1.00)

In an accident condition, the Containment Fan Coolers are all started in slow speed by the LOCA Sequencer. The reason for running in slow speed is:

a. to minimize the chance of fan motor overload with high containment pressure present.

b. to reduce the electrical load on the buses with all safeguard loads running.

c. to minimize the heat load on the Essential Service Water system.

d. to prevent over stressing the fan exhaust duct fusible links with high containment pressure present.

QUESTION: 028 (1.00)

Which ONE of the following Main Feedwater Pump trip signals will cause only ONE Main Feedwater pump to trip?

a. High feed pump discharge pressure

b. All condensate pumps trip

c. HI-HI Steam Generator Level in any SG

d. Exhaust Vacuum Low

QUESTION: 029 (1.00)

The unit is at 100% power, Steam Generator D Level control system is in AUTOMATIC with the following instruments selected:

-Steam Flow Transmitter	(AB-FT-542)
-Feedwater Flow Transmitter	(AE-FT-540)
-Steam Generator Level Transmitter	(AE-LT-549)

Which ONE of the following describes the plant response to Steam Pressure Transmitter (AB-PT-544) failing LOW?

(ASSUME NO OPERATOR ACTION AND ALL PLANT CONTROLS IN AUTOMATIC)

a. Feedwater flow DECREASES until S/G D level stabilizes at a lower level or a low level trip setpoint is reached.

b. Steam Generator Feedwater Pump speed control system generates a HIGHER reference speed signal.

c. Indicated Steam Flow increases on S/G D and Feedwater pump RPM increases.

d. Feedwater Flow INCREASES until S/G D stabilizes at a HIGHER level or a high level trip setpoint is reached.

QUESTION: 030 (1.00)

Select the ONE statement which describes the Flow Control Valves for the Motor Driven Auxiliary Feedwater Pumps.

a. The valves are motor operated and throttle close automatically at high flow rates to prevent pump cavitation.

b. The valves are motor operated and throttle close automatically at high flow rates to limit containment pressure increase caused by a steam line rupture in containment.

c. The valves are air operated and throttle close automatically at high flow rates to prevent pump cavitation.

d. The valves are air operated and throttle close automatically to limit Auxiliary Feedwater flow to a faulted steam generator in the event of a feed line break.

QUESTION: 031 (1.00)

The Auxiliary Feedwater Pump Suction will automatically shift to Essential Service Water when:

a. the Condensate Storage Tank level decreases to 18%.

b. a low suction pressure is sensed on common suction header.

c. an AFAS signal is present and low suction pressure on 2 of 3 sensors.

d. an AFAS signal is present and Condensate Storage Tank level decreases to 18%.

QUESTION: 032 (1.00)

A Waste Gas release will be automatically terminated if radiation levels reach:

a. a high alarm (yellow) on GH-RE-10B.

b. a high alarm (yellow) on GH-RE-10A.

c. a high high alarm (red) on GE-RE-10A or GH-RE-10B.

d. a high high alarm (red) on GH-RE-23.

QUESTION: 033 (1.00)

Annunciator window 58D (RV FLG LEAKOFF TEMP HI) has just energized. The leakage from the reactor vessel flange will accumulate in which ONE of the following?

a. normal containment sumps

b. recirculation sump

c. pressurizer relief tank

d. reactor coolant drain tank

QUESTION: 034 (1.00)

The containment atmosphere radiation monitors GT-RE-31 AND GT-RE-32 sample:

a. containment utilizing the hydrogen control system and are isolated from the containment by a CIS-A actuation.

b. inside the containment isolation valves for the Hydrogen Control System and are not isolated by a CIS-A actuation.

c. between the containment isolation valves on the mini-purge exhaust line and can initiate a CIS-A actuation.

d. from the containment purge exhaust line outside containment and can initiate a CIS-A actuation.

QUESTION: 035 (1.00)

The following readings were taken from the Power Range NIS Detectors:

	N41	N42	N43	N44
Det. A (upper)	375.0	360.0	365.0	360.0
Det. B (lower)	325.0	345.0	330.0	360.0

All readings are in microamperes. Full power currents on all detectors is known to be 400.0 microamperes. Select the DETECTOR with the most limiting Quadrant Power Tilt Ratio (QPTR).

a. N41 upper

b. N42 upper

c. N43 lower

d. N44 lower

QUESTION: 036 (1.00)

Prior to starting a Reactor Coolant Pump, the Volume Control Tank (VCT) is pressurized to greater than or equal to 15 psig to insure:

a. Net Positive Suction Head for the charging pumps.

b. proper flow through the number 1 seal.

c. proper flow through the number 2 seal.

d. adequate hydrogen addition to the VCT.

QUESTION: 037 (1.00)

When an SIS is initiated, the CCP Mini Flow Recirc Valves will:

a. Remain open to provide pump cooling under all conditions.

b. Cycle open on high BIT Flow and close on low BIT Flow.

c. Cycle closed on high BIT Flow and open on low BIT Flow.

d. Close to provide maximum flow through the BIT.

QUESTION: 038 (1.00)

The following plant conditions exists:

-all system in auto
-Mode 1 at 100%
-letdown flow 75 gpm
-seal injection flow 32 gpm
-charging flow 115 gpm
-pressurizer level 61% (constant)
-containment humidity (increasing)

Which ONE of the following statements describes the above conditions:

a. RCS leakage is 28 gpm and in excess of Technical Specifications allowable 1 gpm unidentified leakage.

b. RCS leakage is 40 gpm which is within the allowable 40 gpm (8 per pump) controlled leakage limit of Technical Specifications.

c. RCS leakage is 8 gpm and within the allowed 10 gpm identified limit of Technical Specifications.

d. RCS leakage is 20 gpm and in excess of the 1 gpm unidentified leakage limit in Technical Specifications.

QUESTION: 039 (1.00)

The unit has just experienced a Safety Injection and recovery is in progress. The RCS has been depressurized to recover pressurizer level and the following parameters exist:

-RCS Tave	540 degrees
-pressure	2200 psig
-pressurizer level	50%
-liquid temperature	610 degrees
-Vapor space temperature	615 degrees
-pressurizer heaters	ON

Using the provided steam tables, which ONE of the following describes the PRESENT state of the pressurizer.

a. The pressurizer is at equilibrium saturation conditions and normal pressure control is available with heaters and sprays.

b. The pressurizer liquid is subcooled and pressure is being maintained by Safety Injection flow compressing the vapor space.

c. Superheat conditions exist for the pressure but heaters and sprays will maintain pressure because of the lower liquid temperature.

d. The pressurizer liquid is subcooled with pressurizer heaters maintaining pressure.

~~QUESTION: 040 (1.00)~~

~~Control of Pressurizer Heater Backup Group A has been transferred to the Auxiliary Shutdown Panel and the control switch has been placed in the closed position.~~

~~Which ONE of the following conditions will trip the supply breaker (PG2101) for this group of heaters?~~

~~a. Pressurizer low level (17%)~~

~~b. Lockout on NB0106~~

~~c. NB01 Undervoltage~~

~~d. Safety Injection Signal~~

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2/4/91
KMS

QUESTION: 041 (1.00)

The unit is entering Mode 4 from Mode 3 with the following conditions:

-pressure is controlling at 400 psig
-all wide range cold leg temperatures 350 degrees
-Cold Overpressure Protection armed

Which ONE of the following describes the plant response to TE413 (loop 1 wide range Tc) failing low?

a. PORV 455A would open and depressurize the RCS.

b. Both PORV 455A and 456A would open and depressurize the RCS.

c. Neither valve would open because the auctioneered high temperature signal is used to determine the pressure setpoint for the system.

d. Neither valve would open because the pressure setpoint signal is only decreased to the 495 psig.

QUESTION: 042 (1.00)

The following plant conditions exists:

-Mode 1 at 100% steady state	
-RCS pressure	2200 psig (slowly decreasing)
-pressurizer level	40% (decreasing)
-Tave	588 degrees F (stable)
-charging/letdown lineup	PDP running
-charging flow	98 gpm
-VCT level	30% (decreasing)
-containment humidity	increasing
-containment sump levels	increasing

The first IMMEDIATE action required for this situation per OTO-BB-00003 (Reactor Coolant System Excessive Leakage) is:

a. operate the centrifugal charging pumps as necessary to maintain pressurizer level

b. isolate the letdown system by closing BG-LCV-459 and 460.

c. swap the charging pump suction to the RWST to insure adequate makeup.

d. trip the reactor because the leak is beyond the capacity of the running charging pump.

QUESTION: 043 (1.00)

A complete loss of Condenser Vacuum occurs with the plant at:

-Mode 3
-pressure 2235 psig
-Tave 557 degrees F
-All four RCP's running

With NO operator actions, RCS Tave will stabilize at:

a. 550 degrees F.

b. 557 degrees F.

c. 561 degrees F.

d. 564 degrees F.

QUESTION: 044 (1.00)

The plant is in Mode 1 at 100% power. Main Steam Isolation Valve Testing per OSP-AB-V0001 is in progress. Selector Switch AB-HS-71 on panel RL025 has been selected to Steam Generator D, Valve AB-HV-11, and the valve is currently indicating at the 10% closed condition when a Main Steam Isolation Signal is received.

The valve will:

a. fast close with the other MSIV's.

b. remain in its present position until a manual fast close is initiated.

c. fast close after the exercise test switch is released and the valve returns to full open.

d. fast close after selector switch AB-HS-71 is returned to normal.

QUESTION: 045 (1.00)

Which ONE of the following groups of signals will isolate service water from the Essential Service Water System?

a. Trip of a Service Water Pump or Circulating Water Pump.

b. High Condenser Pit level or Safety Injection.

c. Safety Injection or Loss of Offsite Power.

d. Auxiliary Feedwater Actuation Signal or trip of a Service Water Pump.

QUESTION: 046 (1.00)

The trip of a running Circulating Water Pump at 100% power will:

a. cause a turbine trip on low vacuum.

b. initiate a turbine runback to 75%.

c. have no effect because the other pumps have enough capacity.

d. cause reactor power to increase due to condenser efficiency decreasing.

QUESTION: 047 (1.00)

Given the following conditions for the service and instrument air system:

-all three compressors available
-normal valve and power lineup
-local sequence switch position for compressors in the A-B-C position
-Instrument Air Pressure decreases to 109 psig

Which one of the following describes the system indication that should be present at the main control board.

a. All three compressors running with white indicating lights on and isolation valve position KA-ZL-11 indicating closed.

b. Compressor A running loaded and compressors B/C in standby with isolation valve position KA-ZL-11 indicating closed.

c. All three compressors running with white indicating lights on and isolation valve position KA-ZL-11 indicating open.

d. Compressors A/B running with white indicating light on and compressor C in standby with isolation valve indication KA-ZL-11 open.

QUESTION: 048 (1.00)

After the Emergency Diesel Generator receives an emergency start signal and is running, protection is limited to:

a. overcurrent or reverse power.

b. loss of field or underfrequency.

c. voltage restrained overcurrent or neutral ground overcurrent relay.

d. engine shutdown relay or differential overcurrent relay.

QUESTION: 049 (1.00)

As load is increased on the Emergency Diesel Generator, procedure OTN-NE-00001 (STANDBY DIESEL GENERATION SYSTEM) specifies that VARS be maintained at a value of kw/2. This note is applicable under which condition(s) and is adjusted using which control system.

a. When the Emergency Diesel is the only power source and using the voltage control.

b. When the Emergency Diesel is the only power source and using the governor speed control system.

c. When paralleled with another generator and using the voltage control.

d. When paralleled with another generator and using the governor speed control.

QUESTION: 050 (1.00)

Starting Air Compressors A and B supply Emergency Diesel Generator A. Select the one statement which describes their power supply.

a. They are both powered from MCC PG19G.

b. One compressor is powered from MCC PG19G and one from MCC PG20G.

c. They are both powered from MCC PG20G.

d. One compressor is powered from NG03 and one from NG04.

QUESTION: 051 (1.00)

The plant is at 100% power with all systems in their normal lineups. Annunciator 14A "S/U XFMR LOCKOUT" alarms due to failure of the startup transformer (SUT). Select the ONE item below which occurs as a result of the SUT failure.

a. A load shed occurs on NB01 and NB02.

b. Both emergency diesels NE01 and NE02 start.

c. An automatic Reactor Trip and Turbine Trip actuates.

d. Both the normal and alternate feeder breakers to NB02 trip.

QUESTION: 052 (1.00)

Given the following conditions:

-Annunciator "JOCKEY PUMP MALFUNCTION" on Panel KC008
-Fire Water Accumulator level -2 1/2 inches below centerline
-normal valve lineup
-all Firewater Pumps available in auto
-Firewater pressure 160 psig
-Firewater storage tank levels 35 feet

Which ONE of the following actions should be taken if the low level in the accumulator is verified locally?

a. No action is required. The jockey pump will auto start after a 5 minute timer and fill the accumulator to 90%.

b. A local start, using the manual lever for the jockey pump, should be attempted to fill the accumulator to 90%.

c. The local control station manual start/stop pushbuttons should be used to operate the jockey pump and raise the accumulator level to the center line of the accumulator.

d. No action required. The jockey pump will auto start after a 15 minute timer and raise the level to the center line of the accumulator.

QUESTION: 053 (1.00)

Which input signals are used to control the feedwater bypass flow control valves when in automatic?

a. Steam header pressure, feed pump discharge pressure, total steam flow.

b. Steam header pressure, Steam generator level, programmed steam generator level, steam flow, feed flow.

c. Steam generator level, programmed steam generator level, auctioneered high nuclear power.

d. Steam generator level, programmed steam generator level, total steam flow.

QUESTION: 054 (1.00)

A Rod Position Indication (RPI) annunciator RPI ROD DEV (80C) is generated if:

a. the position of any shutdown bank rod is below 218 steps.

b. the position of any rod in control bank D is less than 161 steps.

c. deviation of greater than or equal to 12 steps between any rod and its bank demand (step counter).

d. any rod in a control bank compared to any other rod in that bank is equal to or greater than plus or minus 12 steps.

QUESTION: 055 (1.00)

Pressurizer level control is selected to the LT-459/460 position. A failure causes the following SEQUENTIAL plant events. (ASSUMING NO OPERATOR ACTIONS TAKEN)

-charging flow reduces to minimum
-pressurizer level begins to decrease
-letdown isolates and heaters turn off
-pressurizer level eventually increases to the high level reactor trip

Which ONE of the following failures occurred?

a. Reference pressurizer level failed to the no-load value.

b. Auctioneered Tave failed Hi due to a failed RTD.

c. Level channel 460 failed low.

d. Level channel 459 failed high.

QUESTION: 056 (1.00)

A unit startup is in progress with the following conditions:

-NIS Power	11%
-Tave	560 degrees F
-pressurizer pressure	2235 psig
-SG levels	50%
-turbine load	70 MWE
-containment pressure	1.0 psig
-containment temperature	90 degrees
-normal radiation readings	

Which ONE of the following describes the plant response to feedwater control bypass valve on Steam Generator A failing closed. Assume no operator action.

a. A reactor trip will occur immediately when Steam Generator level A reaches 14.8% narrow range.

b. A reactor trip will occur 122 seconds after the level in a Steam Generator reaches 20.2% narrow range.

c. A reactor trip will occur 232 seconds after reaching 14.8% Narrow range in Steam Generator A.

d. A reactor trip will occur immediately when Steam Generator A level decreases to 20.2%

QUESTION: 057 (1.00)

RHR Train A has been aligned to cooldown the reactor coolant system in accordance with OTG-ZZ-00006 (Plant Cooldown Hot Standby to Cold Shutdown) and OTN-EJ-00001 (RHR System). The desired RCS cooldown rate is established by adjusting the position of:

a. CCW Inlet Valve EG-HV-101 to RHR Heat Exchanger A.

b. CCW Outlet Valve EJ-V033 from RHR Heat Exchanger A.

c. RHR Heat Exchanger A Bypass Valve EJ-FCV-618.

d. RHR Heat Exchanger A Outlet Isolation Valve EJ-HCV-606.

QUESTION: 058 (1.00)

The following plant conditions exist:

-Reactor Trip/SI
-all ECCS pumps running
-RCS pressure 50 psig
-containment pressure 10 psig
-containment temperature 200 degrees
-RWST Level 50%

Which ONE of the following describes the response of the RHR system?

a. When RWST level reaches 36%, the pump suction will automatically switch to the recirculation sump because adequate level is available in the sump for the required RHR pump NPSH.

b. When RWST level reaches 36%, the pump suction will automatically switch to the recirculation sump because the RWST will no longer provide adequate NPSH for the RHR pump.

c. An annunciator will alarm at 36% level in the RWST and RHR pump suction will automatically switch to the recirculation sump at 11.11% to reserve water for the containment spray system.

d. When RWST level reaches 11.11%, the pump suction will automatically switch to the recirculation sump because adequate level is available in the sump for the required RHR pump NPSH.

QUESTION: 059 (1.00)

The following plant condition exist:

-power 100%
-normal system lineups with "A" CCP running
-Pressurizer level (slowly decreasing)
-CCW Surge tank "A" level (increasing)
-Radiation Monitor EG-RE-9 (increasing)

Based on the above information, which ONE of the following components could be a source of in-leakage to the CCW system under the existing conditions?

a. seal water heat exchanger

b. RHR Heat Exchanger A

c. RCP Thermal barrier heat exchanger

d. spent fuel pool cooling heat exchanger

QUESTION: 060 (1.00)

When a Safety injection occurs, the load sequencer starts the CCW pumps in which ONE of the following methods?

a. The pumps that were running will restart at five seconds on the timer.

b. CCW pump A, supplied from NB01, and CCW B, supplied from NB02, will start at five seconds regardless of which pumps were running.

c. CCW pumps A and B will start at five seconds on the timer and CCW Pumps C and D will start 10 seconds later.

d. CCW pumps A and B will start at five seconds on the timer and if a centrifugal charging pump is running, CCW pump C or D will start.

QUESTION: 061 (1.00)

The following plant conditions exist:

-Mode 1	
-NIS power	6%
-Tave	559 degrees F
-Pressurizer pressure	2235 psig
-steam dumps	in pressure mode
-turbine rolling	500 rpm
-Steam Generator levels	50%
-feedwater controlling	on the bypass valves
-condenser vacuum	2.1 inches HgA

Which ONE of the following describes the response of the steam dump valves when AB-PT-507 (Steam Header Pressure) fails HIGH (ASSUME NO OPERATOR ACTION).

a. Group 1 strokes full open and remains there as the plant is cooled down.

b. Group 1 and 2 trip open and then reclose when temperature reaches 550 degrees.

c. All four groups stroke full open and then reclose when temperature reaches 550 degrees.

d. All four groups trip open and groups 2,3 and 4 close when temperature reaches 550 degrees.

QUESTION: 062 (1.00)

The plant is ramping load up at 5% per minute when an OP Delta T rod block occurs. The following conditions exist at that time:

-NIS Power	90%
-Pressurizer pressure	2200 psig
-Tave	590 degrees F
-Control Bank D	200 steps
-Delta I	-10%

Which ONE of the following actions, taken over the next 10 minutes, would raise the OP Delta T setpoint?

a. increase pressure to 2235 psig.

b. decrease Tave to 585 degrees.

c. dilute the RCS by 10 ppm.

d. withdraw control bank D to 220 steps

QUESTION: 063 (1.00)

The following plant conditions exist:

-Cycle 5	
-Initial startup after refueling	
-Physic testing completed	(MTC +1 PCM/F)
-NIS power	10%
-turbine	80 MWE
-Pressurizer pressure	2235 psig
-Tave	559 degrees F
-Boron concentration	1500 ppm
-Control Bank D	200 steps

Which ONE of the following would describe the change in Moderator Temperature Coefficient (MTC) as the turbine load is ramped up to 580 MWE?

a. remains constant.

b. more positive due to Rod withdrawal.

c. more positive due to Xenon and samarium buildup.

d. less positive due to the required dilution.

QUESTION: 064 (1.00)

A plant cooldown is being performed in MODE 4 using both RHR trains. The following data has been recorded:

TIME	RCS TEMP	RCS PRESS
1000	330 degrees	355 psig
1030	306 degrees	360 psig
1100	282 degrees	360 psig
1130	260 degrees	358 psig
1200	225 degrees	355 psig

Which ONE of the following actions should be taken based on the above data?

a. Restore RCS temperature to 235 degrees within 30 minutes.

b. Restore the cooldown rate to less than 50 degrees per hour.

c. Continue cooldown at present rate to cooldown to MODE 5 within 30 hours.

d. Restore cooldown rate to within Technical Specifications limits, notify the NRC of Tech Spec violation on cooldown rate.

QUESTION: 065 (1.00)

The unit is being runback from 75% power due to the loss of a feedwater pump. The unit reactor operator notices that Control Bank D group 1 Rod M-12 position remained at 185 steps. Thirty seconds later, as the plant is stabilizing, Control Bank D group 2 Rod D-12 position is indicating 150 steps and the bank demand is 135 steps.

Which ONE of the following actions should be taken?

a. Trip the reactor and implement E-0.

b. Stabilize the plant and perform a Quadrant Power Tilt Ratio calculation.

c. Notify Reactor Engineering to perform an incore flux map to determine rod position.

d. Initiate immediate boration to withdraw rods out of the core.

QUESTION: 066 (1.00)

Following a large break LOCA with systems aligned for cold leg recirculation mode, the core decay heat is being removed primarily by:

a. heat transfer from the RCS to the Steam Generators with natural circulation.

b. heat transfer from the RCS to the Steam Generators with reflux boiling.

c. injection of water from the RWST.

d. injection of water from the Recirculation Sump.

QUESTION: 067 (1.00)

The plant is operating at steady state 75% power. Which ONE of the following conditions requires an IMMEDIATE manual reactor trip?

a. RCP A seal injection temperature of 115 degrees F and is increasing at 0.5 degree per minute.

b. RCP B frame vibration equals 4 mils and is increasing at .5 mils/hr.

c. RCP C #1 seal inlet temperature 240 degrees F with a loss of normal seal injection.

d. RCP D shaft vibration equals 17 mils and is increasing at 1.4 mils/hr.

QUESTION: 068 (1.00)

A reactor trip has just occurred and [illegible] ing conditions exist:

-reactor trip breakers open
-reactor bypass breakers open
-NIS power decreasing
-control rod H-10 228 steps
-control rod H-6 228 steps

Which ONE of the following identifies the procedure flow path for this situation?

a. Immediate borate in accordance with OTO-ZZ-0003, Response to Loss of shutdown Margin.

b. Enter procedure E-0, Rx Trip/SI; at step 1 go to FR-S.1; at step 4 immediate borate according to attachment 1.

c. Enter procedure E-0, Rx Trip/SI; at step 4 go to ES-0.1, Rx Trip Response; at step 3 immediate borate.

d. Enter procedure FR-S.1, Response To Nuclear Power Generation, and immediate borate according to attachment 1.

QUESTION: 069 (1.00)

The unit is at 100% power steady state. The CCW System Train B is lined up to supply the service loop. Given the following indications:

-Annunciator 53B (CCW SRG TK B LEV HILO) alarming
-surge tank B level (decreasing rapidly)
-Annunciator 39A (LTDN HX TEMP HI DIVERT) alarming
-Annunciator 39B (LTDN HX DISCH TEMP HI) alarming

Which ONE of the following operator actions COULD NOT isolate this leak?

a. Isolate the radwaste building by shutting EG-HIS-69 AND 70.

b. Isolate Letdown heat exchanger by closing the Letdown orifice valves and BG-PCV-131 Letdown heat exchanger valve isolation.

c. Immediately open EG-HS-15 and close EG-HS-16 to shift the service header to train A.

d. Isolate the service header and trip the reactor after 2 minutes, if the service header can not be restored.

QUESTION: 070 (1.00)

The unit is in process of ramping to full power at 10% per hour with the following conditions:

-NIS power	80%
-turbine	960 MWE
-Tave	580 degrees F
-pressurizer pressure	2205 psig

Which ONE of the following describes the plant condition with respect to Technical Specifications?

Pressure is:

a. less than the required DNB value but no action is required, if cause is due to a power ramp.

b. less than the required DNB value and the required action is to restore it to normal within the next 15 minutes or reduce Thermal Power by 5%.

c. above the required DNB value and no action is required.

d. less than the required DNB value and the required action is to restore it to normal within the next 2 hours or reduce Thermal Power to less than 5%.

QUESTION: 071 (1.00)

Which ONE of the following symptoms would most clearly differentiate between a large LOCA and a large Main Steam Line break inside containment?

a. increasing containment radiation levels

b. increasing containment sump levels

c. increasing containment pressure

d. decreasing pressurizer pressure

QUESTION: 072 (1.00)

The unit has sustained a loss of off-site and onsite AC power. Procedure ECA-0.0, LOSS OF ALL AC POWER, has been implemented.

Which ONE of the following describes the use of the PORVs in this plant condition?

a. PORVs should be cycled as necessary to control pressure at 2235 psig with no pressurizer spray or auxiliary available.

b. PORVs should be opened to depressurize the RCS and minimize leakage through the RCP seals.

c. PORVs should be checked shut and monitored closely if pressure reaches 2335 psig and causes them to cycle.

d. Block valves should be shut to eliminate any mass loss out through the PORVs at any time during the event.

QUESTION: 073 (1.00)

The plant is in mode 2 with the following conditions:

-NIS power	7%
-Tave	557 degrees
-pressurizer pressure	2235 psig
-turbine	600 rpm

Intermediate Range Channel N35 de-energizes due to a loss of the 120 Volt AC Instrument Bus. Which ONE of the following identifies the implementing procedure flowpath?

a. Procedure E-0, REACTOR TRIP OR SAFETY INJECTION, will be entered after the reactor trip.

b. Implement OTO-SE-00002, INTERMEDIATE RANGE NUCLEAR FAILURE, and repair Channel N35 before exceeding 10% power to comply with Technical Specifications.

c. Implement OTN-NN-00001, 120 VITAL AC INSTRUMENT POWER-CLASS 1E, and re-energize the instrument bus on the backup power supply before exceeding 10% power.

d. Bypass the trips on Channel N35 in accordance with OTO-SE-00002, INTERMEDIATE RANGE NUCLEAR FAILURE, and continue the startup in accordance with OTG-ZZ-00003, PLANT STARTUP.

QUESTION: 074 (1.00)

The Technical Specification actions for exceeding the RCS Specific Activity Limit directs a RCS cooldown to less than 500 degrees within 6 hours.

Which ONE of the following describes the condition of the plant after the cooldown has been completed? (Assume Plant at 499 degrees)

a. The RCS would still be 50 degrees subcooled if depressurized at this time to the pressure of the steam generators.

b. The Steam Generator Atmospheric relief valve would lift, but not a safety valve, if a Steam Generator Tube rupture occurred after reaching 500 degrees.

c. The saturation pressure of the RCS is less than the setpoint of the Atmospheric relief and safety valves of the Steam Generator.

d. In this condition the steam generator delta-P limits cannot be exceeded and initiate a Steam Generator Tube Rupture.

QUESTION: 075 (1.00)

Following a loss of all A.C. Power, the operations crew implemented ECA-0.0, LOSS OF ALL AC POWER. When Diesel Generator NE01 has been successfully placed on bus NB01, the STA announces that core exit thermocouples are reading greater than 700 degrees.

Based on this information, which ONE of the following actions should be taken?

a. Continue on in ECA-0.0 until directed to exit to ECA-0.1, Loss of All AC Power Recovery Without SI Required, or ECA-0.2, Loss of All AC Power Recovery With SI Required.

b. Continue on in ECA-0.0 until completed and then go to FR-C.1, Response to Inadequate Core Cooling.

c. Immediately exit ECA-0.0 and go to FR-C.2, Response to Degraded Core Cooling.

d. Immediately exit ECA-0.0 and go to FR-C.1, Response to Inadequate Core Cooling.

QUESTION: 076 (1.00)

The unit has experienced a fire in the control room and immediate actions of OTO-ZZ-00001, CONTROL ROOM INACCESSIBILITY, have been completed to the point of initiating control from the Auxiliary Shutdown Panel (attachment 5 of the procedure).

-NIS power	1 x 10E-8 (decreasing)
-Wide range T-Hot	562 degrees F
-Wide range T-Cold	552 degrees F
-steam generator pressure	1050 psig

Which ONE of the following describes the plant condition?

a. Hot Standby, Tave at 557 degrees controlling on the condenser steam dumps in Tave Mode with all four RCP's running.

b. Hot Standby, Tave at 557 degrees controlling on the Atmospheric Steam relief valves with RCP B and C running.

c. Hot Standby with Natural circulation, Tave control with Atmospherics Steam Relief Valves on B and D Steam Generators.

d. Hot standby with Natural Circulation, Tave at 557 degrees controlling on the condenser steam dumps in Tave Mode.

QUESTION: 077 (1.00)

Identify which step is NOT an IMMEDIATE operator action for ECA-0.0, LOSS OF ALL AC POWER.

a. Ensure Turbine Trip.

b. Ensure AFW Flow.

c. Check is RCS is isolated.

d. Try to restore Power to Either NB01 OR NB02.

QUESTION: 078 (1.00)

In accordance with E-1, LOSS OF REACTOR OR SECONDARY COOLANT, the Reactor Coolant Pumps should be tripped if pressure decreases below 1400 psig.

Which ONE of the following is the basis for tripping the pumps at this condition?

a. Prevent cavitation of the reactor coolant pumps.

b. Minimize mass loss out of the break.

c. Minimize heat input.

d. Reduce cooldown rate if a main steam line break was in progress.

QUESTION: 079 (1.00)

The following conditions exist:

-Reactor trip/SI actuated	
-NIS power	1 x 10E-9 amps
-Core Exit Thermocouples	680 degrees F
-pressurizer pressure	2185 psig
-Containment temperature	180 degrees F
-Containment pressure	0.1 psig
-Containment radiation	Normal

The table for required subcooling in CSF-1, CRITICAL SAFETY FUNCTION STATUS TREES, list the following values!

RCS PRESSURE	NORMAL CTMT INSTRUMENT ERROR	ADVERSE CTMT INSTRUMENT ERROR
1000-3000 (psig)	23 (DEG F)	43 (DEG F)

The RCS Coolant condition based on the above the parameters is:

a. subcooled enough to meet the required normal Containment value of 23 degrees.

b. subcooled enough to meet the required Adverse Containment value of 43 degrees.

c. superheated.

d. saturated.

QUESTION: 080 (1.00)

The following plant conditions exist:

-Mode 3	
-Burnup	19,000 MWD/MTU
-Boron	20 ppm
-Tave	557 degrees
-Pressurizer pressure	2235 psig
-Steam Generator Levels	50 %
-Containment Pressure	0.1 psig

Which ONE of the following values approximates (i.e. is closest to) the reactivity added by a 100 degree cooldown associated with a Main Steam Line Break? (Assume MTC is within required Technical Specifications)

a. 1600 pcm.

b. 2200 pcm.

c. 3600 pcm.

d. 5000 pcm.

QUESTION: 081 (1.00)

The reactor is operating at 50% power when turbine load is increased 10% with NO control rod motion or change in boron concentration. An indication you would EXPECT to receive is a:

a. T REF/T AUCT HI annunciator.

b. T REF/T AUCT LO annunciator.

c. RPI ROD DEV annunciator.

d. RPI DEV PR TILT annunciator.

QUESTION: 082 (1.00)

Which ONE of the following describes the sequence of events at 45% power with decreasing vacuum in the condenser? (ASSUME NO OPERATOR ACTION)

a. At 8.4 inches Hga, the turbine will trip, causing a reactor trip. The plant will be in hot standby, with Tave controlled on the condenser dumps at 557 degrees.

b. Standby vacuum pump will start at 5 inches Hga. Turbine will trip at 8.4 inches Hga causing a reactor trip. The plant will be in hot standby, with Tave controlled by the Atmospheric steam relief valves.

c. The turbine will trip at 8.4 inches Hga. The condenser dumps and control rods will reduce Tave to 557 degrees.

d. The turbine will trip at 8.4 inches Hga. The steam generator safeties and Atmospherics will remove heat until control rods bring reactor power down.

QUESTION: 083 (1.00)

Procedure E-0, step 2, verifies that a turbine trip has occurred following a reactor trip. The reason for tripping the turbine is to:

a. prevent overheating the last stage of blades on the low pressure turbines.

b. reverse power the generator and prevent overspeeding the unit.

c. prevent uncontrolled cooldown of the RCS.

d. provide dryout protection for the steam generators, in the event of a loss of all feedwater accident.

QUESTION: 084 (1.00)

The plant has just experienced a Reactor Trip with no SI from 30% power. Which ONE of the following describes the MOST COMPLETE response of the Main Feedwater System in this event?

When Tave decreases to 564 degrees,:

a. both Main Feedwater Pumps will trip and all four control valves will close.

b. both Main Feedwater Pumps will trip and all four control valves and bypasses close.

c. all four control valves and bypasses close.

d. all four feedwater control valves, bypass valves and isolation valves will close.

QUESTION: 085 (1.00)

Pressurizer PORV 456 has lifted and failed to fully reseat resulting in the following plant conditions: (assume the block valve has failed to close)

-Rx trip	
-pressurizer pressure	1985 psig
-pressurizer vapor space temperature	635 degrees
-Tave	557 degrees
-PRT level	75 %
-PRT pressure	35 psig

The tailpipe temperature indication for Pressurizer PORVS should read:

a. full scale high at 400 degrees F.

b. 280 degrees F.

c. 260 degrees F.

d. 220 degrees F.

QUESTION: 086 (1.00)

The unit has experienced a LOCA and the transition from procedure E-1 to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, has been made. Voiding in the RCS when depressurizing may be identified by:

a. rapidly increasing pressurizer level.

b. decreasing Safety Injection flow.

c. increasing RCS pressure.

d. rapid drop in subcooling.

QUESTION: 087 (1.00)

During the sequence of actions following a LOCA, adverse containment condition of high temperature will cause HIGHER:

a. pressurizer level indication.

b. core exit thermocouple indications.

c. RCS pressure indications.

d. RCS wide range RTD indications.

QUESTION: 088 (1.00)

In response to a LOCA event, the operating crew has implemented the emergency procedures and are currently in E-1 (LOSS OF REACTOR OR SECONDARY COOLANT). Step 1 of E-1 (checking if RCPs should be stopped) was completed with RCS pressure at 1500 psig. One minute later, while on step 3 (Checking SG levels), RCS pressure drops to 1200 psig. The reactor operator should:

a. continue on with step 3.

b. trip the RCPs within 30 minutes after #1 seal failure.

c. go to ES-0.0 because RCP tripping criteria is not on the foldout page.

d. verify one CCP or SI pump running and trip the RCPs.

QUESTION 89 (1.00)

The unit is in Mode 6 with reduced RCS inventory. Which ONE statement describes the relationship of RHR pump vortexing versus flow and level?

Vortexing will occur:

a. at the loop low level alarm (BLI-0053 17 inches) with a minimum RHR flow rate of 1700 gpm.

b. at any RHR pump flow rate below the low level alarm (BLI-0053 17 inches).

c. with lower RHR flow rates at a lower reactor vessel level.

d. with lower RHR flow rates at a higher reactor vessel level.

QUESTION: 090 (1.00)

The plant is in Mode 5 for maintenance. RCS Level has been lowered in accordance with OTN-BB-00002 (RCS DRAINING) to 18 inches on BLI-0053 when the annunciator RHR LOOP 1 FLOW LO alarms. RHR Flow Indicators FI-618, 619 and pump discharge pressure are oscillating. The first IMMEDIATE action should be to:

a. increase charging flow to raise level.

b. stop the RHR pump.

c. dispatch an equipment operator to vent the running RHR pump.

d. start the standby RHR pump.

QUESTION: 091 (1.00)

FR-S.1, RESPONSE TO NUCLEAR GENERATION, step 1 RNO directs the operator to trip the reactor using switch SB-HS-1 on RL003 and then try switch SB-HS-42 on RL006. These switches will initiate a reactor trip by:

a. switch SB-HS-1 actuating the Under Voltage trip on Reactor Trip and Bypass Breakers while switch SB-HS-42 actuates the shunt trip on all Reactor Trip Breakers.

b. switch SB-HS-1 actuating the Under Voltage trip on Reactor Trip Breakers A and B while switch SB-HS-42 actuates the shunt trip on all Reactor Trip Breakers.

c. actuating the Under Voltage coil on Reactor trip Breaker A and B which will actuate the shunt trip on the associated Bypass Trip Breakers.

d. actuating the Under Voltage coil on each Reactor Trip and Bypass Breaker which will in turn actuate the shunt trip on its respective breaker.

QUESTION: 092 (1.00)

The reactor is critical at 8% power, with the turbine rolling at 1800 rpm. Investigation reveals that IR Channel N36 has failed. (ASSUME NO REACTOR TRIP)

Which ONE of the following actions should be taken?

a. Restore the N36 Channel to operable status before increasing power above 10%.

b. Place the N36 Level Trip Bypass Switch in BYPASS position and continue power increase.

c. Place the N36 Level Trip Bypass Switch in BYPASS and then reduce power to less than 5%.

d. Reduce power to less than 5% and then place the N36 Level Trip Bypass Switch in BYPASS.

QUESTION: 093 (1.00)

The unit is in Mode 3 with the following conditions:

-Tave	557 degrees F
-Pressurizer pressure	2235 psig
-RCS Boron at	1000 ppm (ECP value)
-Source Range Counts	30 cps
-all rods inserted	
-72 hours since shutdown	

The Shutdown Banks are withdrawn in accordance with OTG-ZZ-00002, REACTOR STARTUP. When the Shutdown Banks are fully withdrawn, the Source Range counts are 110 cps on N31 and 65 cps on N32.

Which ONE of the following actions should be taken?

a. Manually shift the charging pump suction to the RWST.

b. Verify operability of Channels N31 and N32.

c. Immediately open the Reactor Trip Breakers.

d. Perform immediate boration in accordance with OTG-ZZ-00002.

QUESTION: 094 (1.00)

A Steam Generator Tube Leak as defined by OTO-BB-00001, STEAM GENERATOR TUBE LEAK, is a leak within the capacity of:

a. one centrifugal charging pump with letdown isolated.

b. two centrifugal charging pumps with normal letdown (75 gpm).

c. one centrifugal charging pump with normal letdown (75 gpm).

d. two centrifugal charging pumps with letdown isolated.

QUESTION: 095 (1.00)

When procedure E-3 (STEAM GENERATOR TUBE RUPTURE) is entered for a tube rupture occurring in B or C generator, steam supply to Turbine Driven Auxiliary Feedwater Pump is isolated:

a. locally with manual valves AB-V085 and 87 (Steam supply B and C).

b. from the control room with AB-HIS-5A and 6A (Steam supply B and C).

c. by a high (red) alarm on radiation monitor FC-RE-385 (TD AFW).

d. by an SI actuation but will reopen with an AFAS signal.

QUESTION: 096 (1.00)

The first step in the emergency procedures attachment for restarting a reactor coolant pump verifies that SI has been reset.

Which ONE of the following describes the consequences of omitting this condition?

a. The reactor coolant pump would have no number 1 seal leakoff flow path.

b. CCW would not be available to the any of the reactor coolant pump heat exchangers.

c. Starting any reactor coolant pump would cause an undervoltage condition and a shutdown sequencer on XBN02.

d. Starting reactor coolant pump C or D would cause a shutdown sequencer on XBN02 due to undervoltage.

QUESTION: 097 (1.00)

The following plant conditions exist:

-Reactor trip/ SI actuated	
-RCS Temperature	500 degrees F
-pressurizer pressure	2000 psig
-steam generator (A,B,C) pressure	450 psig
-steam generator level A,B,C	50% wide range
-steam generator D pressure	0 PSIG
-steam generator level D	5% wide range
-auxiliary feedwater flow	100,000 lbm/hr to each SG
-containment pressure	10 psig

Which ONE of the following actions should be taken with the auxiliary feedwater system? (ASSUME MSIVs ARE SHUT)

a. Reduce flow on all steam generators until 300,000 lbm/hr total flow is achieved.

b. Isolate auxiliary feedwater flow to D steam generator to minimize containment pressure and maintain 300,000 lbm/hr to the other 3 intact SGs.

c. Maintain 15,000 lbm/hr to the D steam generator to avoid dryout and 300,000 lbm/hr to the other 3 intact SGs.

d. Maintain 15,000 lbm/hr to the D steam generator to avoid dryout and reduce flow to 260,000 lbm/hr to the other 3 intact SGs to minimize cooldown of the RCS.

QUESTION: 098 (1.00)

Pressurizer level transmitter LT-459 has failed to the low end of the scale. The control functions have been transferred from LT-459 to the alternate channel LT-461. Which ONE of the following describes the status of the letdown system.

a. Letdown can be returned to service by first opening loop isolation BG-HIS-459 and then opening one of the orifice valves BG-HIS-8149AA,BA, or CA.

b. Letdown can be returned to service by opening loop isolation BG-HIS-459.

c. Excess letdown must be used until LT-459 is repaired.

d. Letdown can be returned to service by first opening one of the orifice valves BG-HIS-8149AA,BA, or CA and then opening loop isolation BG-HIS-459.

QUESTION: 099 (1.00)

The plant is in Mode 1 when an inadvertent safety injection occurs and all systems respond properly. Transition has been made from E-0 step 26 to ES-1.1, SI TERMINATION. SI has just been reset in accordance with step 1 when offsite power is lost.

Which ONE of the following describes the response of the on-site safety related electrical system?

a. Both emergency diesels start, the D/G output breakers shut, then the LOCA sequencer actuates.

b. An electrical load shed occurs, both D/G output breakers shut, then the S/D sequencer actuates.

c. Both emergency diesels start, the D/G output breakers shut, then the S/D sequencer actuates.

d. An electrical load shed occurs, both D/G output breakers shut, then the LOCA sequencer actuates.

QUESTION: 100 (1.00)

The plant is in Mode 3 following a loss of all A.C. Power. Electrical Bus NB02 has just been locally re-energized while the control room was adjusting steam generator levels in step 13 of ECA-0.0, LOSS OF ALL AC. The SRO leaves step 13 and goes immediately to step 24 as directed by an earlier caution statement. This action is required to:

a. preserve water in the Condensate Storage Tank.

b. maintain a secondary heat sink.

c. minimize RCS leakage through the RCP seals.

d. automatically load all ECCS pumps on the energized bus.

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

ANSWER: 001 (1.00)

d.

REFERENCE:

LP, Administrative Procedures, A-1, p22. OBJ. F3

[3.7/4.1]
194001K102 ..(KA's)

ANSWER: 002 (1.00)

c.

REFERENCE:

LP, Administrative Procedures, A-1, p17 OBJ. C2

[4.1/3.9]
194001A102 ..(KA's)

ANSWER: 003 (1.00)

b.

REFERENCE:

APA-ZZ-00802, CONFINED SPACE ENTRY PERMIT, P5.

[3.3/3.6]
194001K114 ..(KA's)

ANSWER: 004 (1.00)

d.

REFERENCE:

APA-ZZ-01001, Callaway Plant ALARA Program, P. 2

[3.3/3.5]
194001K104 ..(KA's)

ANSWER: 005 (1.00)

a.

REFERENCE:

APA-ZZ-00744, P8

[3.5/4.2]
194001K116 ..(KA's)

ANSWER: 006 (1.00)

b.

REFERENCE:

APA-ZZ-01000, Section 4.7.3.10.1, Rev 7.

[2.8/3.4]
194001K103 ..(KA's)

ANSWER: 007 (1.00)

d.

REFERENCE:

APA-ZZ-00702, P.2

[3.5/4.2]
194001K116 ..(KA's)

ANSWER: 008 (1.00)

a.

REFERENCE:

ODP-ZZ-00004, P5

[3.6/3.7]
194001K101 ..(KA's)

ANSWER: 009 (1.00)

c.

REFERENCE:

LP, Operations Department Procedures, A-5, Pg. 27, OBJ. G2d

[3.3/3.4]
194001A101 ..(KA's)

ANSWER: 010 (1.00)

c.

REFERENCE:

LP A-3, HEALTH PHYSICS PROGRAM AND PROCEDURES, APA-ZZ-01000, REV.7, PAGE 28
PLANT MANUAL APA-ZZ-01000 REV. 7 PAGE 28, SECTION 4.7.1.5.3
[2.8/3.4]
194001K103 ..(KA's)

ANSWER: 011 (1.00)

d.

REFERENCE:

Curve Book Section 7, Table 7-1
Curve Book Section 7, Figure 7.3, Boron Addition

[2.6/3.1]
194001A108 ..(KA's)

ANSWER: 012 (1.00)

b.

REFERENCE:

ODP-ZZ-00003, SHIFT RELIEF AND TURNOVER, SECTION 4.2.3.2, 4.3.2.2

[2.5/3.4]
194001A103 ..(KA's)

ANSWER: 013 (1.00)

b.

REFERENCE:

TECHNICAL SPECIFICATIONS TABLE 6.2-1.

[2.5/3.4]
194001A103 ..(KA's)

ANSWER: 014 (1.00)

a.

REFERENCE:

OSP-SE-00004, NIS POWER RANGE HEAT BALANCE

[3.5/3.8]
015000A101 ..(KA's)

ANSWER: 015 (1.00)

d.

REFERENCE:

OTN-BB-00003, REACTOR COOLANT PUMPS, SECTION 2.17

[3.2/3.5]
003000K504 ..(KA's)

ANSWER: 016 (1.00)

a.

REFERENCE:

Technical Specifications, pg. 3/4 2-2(a)
LP B-2 T61.003B.6, POWER DISTRIBUTION LIMITS, OBJ. O.

[3.1/3.8]
015000G011 ..(KA's)

ANSWER: 017 (1.00)

d.

REFERENCE:

LP-26 T61.0110.6, ROD CONTROL, OBJ. J P.23
OTO-SA-00001, Rev. 2, Table 2, pg. 1 of 1.

[3.7/3.8]
001000K407 ..(KA's)

ANSWER: 018 (1.00)

a.

REFERENCE:

LP-11 T61.0110.6, CVCS, OBJ. M P.9

[3.8/4.0]
004000K115 ..(KA's)

ANSWER: 019 (1.00)

d.

REFERENCE:

OTO-BB-00002, REACTOR COOLANT PUMP OFF-NORMAL, SECTION 2.1, 3.1, 6.3
LP B-1 T61.003B.6, OFF-NORMAL PROCEDURES. OBJ. 5.

[3.5/3.9]
003000A201 ..(KA's)

ANSWER: 020 (1.00)

b.

REFERENCE:

OTO-ZZ-00003, LOSS OF SHUTDOWN MARGIN, SECTION 5.0
LP-11 T61.0110.6, CVCS, OBJ. O. P. 2-18

[4.4/4.6]
004010K609 ..(KA's)

ANSWER: 021 (1.00)

c.

REFERENCE:

LP-26 T61.0110.6, ROD CONTROL, P. 16, 17
Lesson OBJ. D.3

[3.5/3.8]
001000K403 ..(KA's)

ANSWER: 022 (1.00)

a.

REFERENCE:

LP-26 T61.0110.6, ROD CONTROL, P.19, OBJ. M
LP-26 ATTACHMENT SNP-CR-16 REV. 4, OBJ. H

[3.7/3.9]
001050A201 ..(KA's)

ANSWER: 023 (1.00)

c. or d. kms
2/4/91

REFERENCE:

SNUPPS Funct. Diagram, 7250D64, Sheet 7, 8
Technical Specifications, Section 3.3, Table 3.3-4
E-0, Step 4

[4.2/4.4]

013000K101 ..(KA's)

ANSWER: 024 (1.00)

a.

REFERENCE:

LP D-2 T61.003D.6, E-0, STEP 4
LP-49 T61.0110.6, MSIV ISOLATION, OBJ. D P.5

013000K101 [4.2/4.4] ..(KA's)

ANSWER: 025 (1.00)

d.

REFERENCE:

LP-52 T61.0110.6, ESFAS, OBJ B.3 P.19

013000A402 [4.3/4.4] ..(KA's)

ANSWER: 026 (1.00)

c.

REFERENCE:

LP-29 T61.0110.6, INCORE INSTRUMENTATION, P.10, OBJ. H

017020K403 [3.1/3.3] ..(KA's)

ANSWER: 027 (1.00)

a.

REFERENCE:

LP-40 T61.0110.6, CONTAINMENT VENTILATION, P.17, OBJ. B.1

[2.6/3.0]
022000A203 ..(KA's)

ANSWER: 028 (1.00)

d.

REFERENCE:

LP-23 MAIN FEEDWATER, OBJ. D P.15

[3.1/3.2]
059000K416 ..(KA's)

ANSWER: 029 (1.00)

a.

REFERENCE:

OTO-AB-00003, STEAM GENERATOR PRESSURE CHANNEL FAILURE, SECTION 4.1
OTO-AB-00003, STEAM GENERATOR PRESSURE CHANNEL FAILURE, ATTACHMENT 4
[3.0/3.3]
059000A211 ..(KA's)

ANSWER: 030 (1.00)

a.

REFERENCE:

LP-25 T61.0110.6, AUXILIARY FEEDWATER SYSTEM, OBJ, II.C.3 P. 18-19

[3.1/3.4]
061000K404 ..(KA's)

ANSWER: 031 (1.00)

c.

REFERENCE:

LP-25 T61.0110.6, AUXILIARY FEEDWATER SYSTEM, OBJ. II.C.5 P. 28

[3.9/4.2]
061000K401 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

LP-36 T61.0110, PROCESS RADIATION AND AREA RADIATION MONITORING SYSTEMS ATTACHMENT 1, P. 2 OF 34

[2.8/2.8]
071000A303 ..(KA's)

ANSWER: 033 (1.00)

d.

REFERENCE:

LP-16 T61.0110.6, RADWASTE SYSTEMS, P.19, OBJ. J

[2.7/2.9]
068000K107 ..(KA's)

ANSWER: 034 (1.00)

a.

REFERENCE:

LP-40 T61.0110.6, CONTAINMENT VENTILATION, OBJ. K P. 43
OTA-SP-RM011, P.3
SNUPPS DRWG M-22GS01(Q), CONTAINMENT HYDROGEN CONTROL

[3.5/3.9]
072000K102 ..(KA's)

ANSWER: 035 (1.00)

d.

REFERENCE:

OSP-SE-00003, SECTION 6.3
TECHNICAL SPECIFICATION 3.2.4
LP B-2 T61.003B.6, POWER DISTRIBUTION LIMITS, OBJ.Q

[3.5/3.7]
015000A104 ..(KA's)

ANSWER: 036 (1.00)

c.

REFERENCE:

LP A-6 T61.003A.6, PLANT HEATUP, OBJ.C.3 P.5

[2.5/2.8]
003000A205 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

OTN-BG-00001, CVCS SYSTEM, SECTION 2.14
LP-11 T61.0110.6, CVCS, OBJ. M.

[2.7/3.1]
006030K401 ..(KA's)

ANSWER: 038 (1.00)

a.

REFERENCE:

TECHNICAL SPECIFICATIONS, 3.4.6.2

[3.6/4.1]
002000G005 ..(KA's)

ANSWER: 039 (1.00)

b.

REFERENCE:

LP-9 T61.0110.6, REACTOR COOLANT SYSTEM, OBJ.C.4 P. 23
STEAM TABLE

[3.5/4.0]
010000K501 ..(KA's)

~~ANSWER: 040 (1.00)~~

~~b.~~

Deleted
2/4/91
KMS

~~REFERENCE:~~

~~FACILITY EXAM BANK QUES. #69 SBB-02PF-04C~~
~~LP B-4 T61.003B.6, PRZR PRESSURE/LEVEL, P. 8 OBJ.A2~~

~~[3.6/3.4]~~
~~010000A402 ..(KA's)~~

ANSWER: 041 (1.00)

d.

REFERENCE:

LP A-18 T61.003A.6, COLD OVERPRESSURE PROTECTION, OBJ. B P. 3-4

[3.8/4.1]
010000K403 ..(KA's)

ANSWER: 042 (1.00)

a.

REFERENCE:

OTO-BB-00003, RCS EXCESSIVE LEAKAGE, P. 3
IMMEDIATE ACTIONS

011000A203 [3.8/3.9] ..(KA's)

ANSWER: 043 (1.00)

c.

REFERENCE:

LP-20 T61.0110.6, MAIN STEAM, OBJ. F P. 11

039000A105 [3.2/3.3] ..(KA's)

ANSWER: 044 (1.00)

a.

REFERENCE:

LP-49 T61.0110.6, MAIN STEAM ISOLATION, P.10

[3.1/3.5]
039000A302 ..(KA's)

ANSWER: 045 (1.00)

c.

REFERENCE:

LP-4 T61.0110.6, CIRC AND SERVICE WATER, OBJ. 8 P.60

[3.2/3.2]
075000A401 ..(KA's)

ANSWER: 046 (1.00)

b.

REFERENCE:

LP-38 T61.0110.6, MAIN TURBINE CONTROLS AND CONTROL OIL, OBJ. I P.21

[3.4/3.5]
075020K301 ..(KA's)

ANSWER: 047 (1.00)

a.

REFERENCE:

LP-14 T61.0110.6, SERVICE AND INSTRUMENT AIR, OBJ.D P.2, 4
LP-14 T61.0110.6, SERVICE AND INSTRUMENT AIR, OBJ.E ATTACHMENT AIR SYSTEMS

[2.7/2.7]
079000A401 ..(KA's)

ANSWER: 048 (1.00)

d.

REFERENCE:

LP-3 T61.016C.6, STANDBY DIESEL SYSTEMS, OBJ.J P. 46

[3.9/4.2]
064000K402 ..(KA's)

ANSWER: 049 (1.00)

c.

REFERENCE:

OTN-NE-00001, STANDBY DIESEL GENERATION SYSTEM. P.17

[2.7/2.9]
064000A202 ..(KA's)

ANSWER: 050 (1.00)

b.

REFERENCE:

LP-3 T61.016C.6, STANDBY DIESEL SYSTEM, OBJ. 2.2.1 P.13

[2.7/3.1]
064000K201 ..(KA's)

ANSWER: 051 (1.00)

d.

REFERENCE:

LP-2, SERVICE POWER, OBJ. F P.5

[3.5/3.6]
062000A305 ..(KA's)

ANSWER: 052 (1.00)

c.

REFERENCE:

LP-35, FIRE PROTECTION, OBJ. B,C P. 9-1

[2.9/3.3]
086000A301 ..(KA's)

ANSWER: 053 (1.00)

c.

REFERENCE:

LP B-4, I AND C CONTROL SYSTEM REVIEW, OBJ.A.3 P.14

[3.6/3.8]
035010K401 ..(KA's)

ANSWER: 054 (1.00)

d.

REFERENCE:

LP B-4 T61.003B.6, I and C SYSTEMS REVIEW, OBJ. C P. 46

[2.9/3.1]
014000G008 ..(KA's)

ANSWER: 055 (1.00)

d.

REFERENCE:

CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, PRS-2, 2-1
LP B-4 T61.003B.6, I and C SYSTEMS REVIEW, OBJ. A.2 P.11-12

[3.4/3.6]
011000A210 ..(KA's)

ANSWER: 056 (1.00)

a.

REFERENCE:

LP-27 T61.0110.6, REACTOR PROTECTION, OBJ. D P.24
OTG-ZZ-00001, pg. 24a
OTG-ZZ-00005, pg. 9

[3.9/4.3]
012000K402 ..(KA's)

ANSWER: 057 (1.00)

d.

REFERENCE:

LP-7 T61.0110.6, RESIDUAL HEAT REMOVAL, OBJ. C.1 P.15

[3.1/3.1]
005000K410 ..(KA's)

ANSWER: 058 (1.00)

a.

REFERENCE:

LP-7 T61.0110.6, RESIDUAL HEAT REMOVAL, OBJ. C.2A P.12

[3.2/3.5]
005000K402 ..(KA's)

ANSWER: 059 (1.00)

c.

REFERENCE:

OTO-BB-00003, RCS EXCESSIVE LEAKAGE, P.2
CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, CCW-7, 1-1
LP-10 , COMPONENT COOLING WATER, OBJ. H P.2-3

[2.8/3.0]
008000K103 ..(KA's)

ANSWER: 060 (1.00)

b.

REFERENCE:

LP-10, COMPONENT COOLING WATER, OBJ. C P.5

[3.6/3.7]
008030A304 ..(KA's)

ANSWER: 061 (1.00)

c.

REFERENCE:

CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, MSS-13, 2-2
LP-20 MAIN STEAM, OBJ. K P.25-27

[3.1/3.2]
041020A102 ..(KA's)

ANSWER: 062 (1.00)

b.

REFERENCE:

TECHNICAL SPECIFICATIONS, 2.2-1 TABLE NOTE 3

[3.1/3.2]
045010K421 ..(KA's)

ANSWER: 063 (1.00)

d.

REFERENCE:

CURVE BOOK, Figure 3-4a rev.4

[2.5/2.7]
045010K508 ..(KA's)

ANSWER: 064 (1.00)

b.

REFERENCE:

OTG-ZZ-00006, PLANT COOLDOWN HOT STANDBY TO COLD SHUTDOWN
PRECAUTIONS AND LIMITATIONS, P.2
TECHNICAL SPECIFICATION 3/4.4.9

[3.5/3.6]
005000A101 ..(KA's)

ANSWER: 065 (1.00)

a.

REFERENCE:

OTO-SF-00004, MISALIGNMENT OF CONTROL RODS, IMMEDIATE ACTIONS

[3.4/3.6]
000005G010 ..(KA's)

ANSWER: 066 (1.00)

d.

REFERENCE:

LP D-3 R61.003d.6, E-1 ,PAGES 18,19

[4.2/4.2]
000011A111 ..(KA's)

ANSWER: 067 (1.00)

c.

REFERENCE:

OTO-BB-00002, P. 4
LP B-1 T61.003B.6, OTO PROCEDURE REVIEW, OBJ. 04

[3.7/3.7]
000015A210 ..(KA's)

ANSWER: 068 (1.00)

b.

REFERENCE:

LP D-2 T61.003D.6, E-0, STEP 1, OBJ. A.3
LP D-7 T61.003D.6, FR-S.1, STEP 4

[3.9/4.4]
000024A202 ..(KA's)

ANSWER: 069 (1.00)

b.

REFERENCE:

OTO-EG-00001, IMMEDIATE ACTIONS, PAGE 2
LP B-1 T61.003B.6, CCW OTO-EG, OBJ. Y5

[2.8/3.1]
000026A206 ..(KA's)

ANSWER: 070 (1.00)

d.

REFERENCE:

TECHNICAL SPECIFICATION 3/4.2.5

[3.1/3.6]
000027G008 ..(KA's)

ANSWER: 071 (1.00)

a.

REFERENCE:

LP D-2 T61.003D.6, E-O, PAGE 50
CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, MSS-3, 2-1 EXPECTED RESPONSE
CALLAWAY SIMULATOR MALFUNCTION ABSTRACT, RCS-6, 2-1 EXPECTED RESPONSE

[4.6/4.7]
000040A203 ..(KA's)

ANSWER: 072 (1.00)

c.

REFERENCE:

ECA-0.0 STEP 3
LP D-6 TT61.003D.6, ECA-0.0, OBJ.5 P. 28

[4.1/4.3]
000055G010 ..(KA's)

ANSWER: 073 (1.00)

a.

REFERENCE:

LP S-28 T61.0110.6, EXCORE NUCLEAR INSTRUMENTATION, OBJ. 1.E P. 59

[4.0/4.0]
000057A219 ..(KA's)

ANSWER: 074 (1.00)

c.

REFERENCE:

TECHNICAL SPECIFICATION 3/4.4.8 BASES
STEAM TABLES

[2.8/3.4]
000076A202 ..(KA's)

ANSWER: 075 (1.00)

a.

REFERENCE:

LP D-6, T61.003D.6, ECA-0, OBJ.4.1 P. 21

[3.9/4.0]
000055G012 ..(KA's)

ANSWER: 076 (1.00)

c.

REFERENCE:

OTO-ZZ-00001, CONTROL ROOM INACCESSIBILITY, ATTACHMENT 5 P.1

[4.3/4.4]
000068A211 ..(KA's)

ANSWER: 077 (1.00)

d.

REFERENCE:

ECA-0.0, LOSS OF ALL AC POWER, IMMEDIATE ACTIONS

[4.1/4.3]
000055G010 ..(KA's)

ANSWER: 078 (1.00)

b.

REFERENCE:

LP D-2 T61.003D.6, RCP TRIP CRITERIA. P.3 OBJ. B

[4.1/4.2]
000011K314 ..(KA's)

ANSWER: 079 (1.00)

c.

REFERENCE:

CSF-1, CRITICAL SAFETY FUNCTION STATUS TREE, P.2
LP D-6 T61.003D.6, CORE COOLING STATUS TREES, P.2
STEAM TABLES
[3.7/4.1]
000074K104 ..(KA's)

ANSWER: 080 (1.00)

c.

REFERENCE:

TECHNICAL SPECIFICATION 3.1.1.3 LCO
CURVE BOOK FIGURE 3-9 REV.5

[4.1/4.4]
000040K105 ..(KA's)

ANSWER: 081 (1.00)

a.

REFERENCE:

LP B-1 T61.003B.6, OTO-SF-00006, OBJ. RR1 P.137

[3.1/3.3]
000005G005 ..(KA's)

ANSWER: 082 (1.00)

d.

REFERENCE:

OTO-AD-00001, P.1
OTO-AC-00001, P.1

[3.9/4.1]
000051A202 ..(KA's)

ANSWER: 083 (1.00)

c.

REFERENCE:

LP D-2 T61.003D.6, E-0, P. 14 OBJ. A.6

[3.7/4.0]
000007K103 ..(KA's)

ANSWER: 084 (1.00)

d.

REFERENCE:

LP-23 T61.0110.6, MAIN FEEDWATER SYSTEM, pg. 11-13.
SNUPPS DRWG 7250D64 SHT. 13 & 14.

[3.8/3.7]
000007A102 ..(KA's)

ANSWER: 085 (1.00)

b.

REFERENCE:

LP D-2 T61.003D.6, E-0, P.42 OBJ. A.6
STEAM TABLES

[3.9/3.9]
000008A203 ..(KA's)

ANSWER: 086 (1.00)

a.

REFERENCE:

LP D-3 T61.003D.6, ES-1.2, P.37 OBJ. 2

[3.4/3.6]
000009K301 ..(KA's)

ANSWER: 087 (1.00)

a.

REFERENCE:

LP D-2 T61.003D.6, E-0, P.10 OBJ. A.6

[3.7/3.9]
000011A204 ..(KA's)

ANSWER: 088 (1.00)

d.

REFERENCE:

LP D-2 T61.003D.6, RCP TRIP CRITERIA, P.5 OBJ. B
E-0, Foldout Page

[4.0/4.0]
000011A103 ..(KA's)

ANSWER: 089 (1.00)

c.

REFERENCE:

OTN-BB-00002, RCS DRAINING, ATTACHMENT 4
LP B-1, T61.003B.6, OTO-EJ-00001 OBJ. 2

[3.2/3.2]
000025K202 ..(KA's)

ANSWER: 090 (1.00)

b.

REFERENCE:

OTO-EJ-00001, LOSS OF RHR FLOW, IMMEDIATE ACTIONS
LP B-1 T61.003B.6, LOSS OF RHR FLOW, OBJ. 3

[3.9/3.9]
000025G010 ..(KA's)

ANSWER: 091 (1.00)

d.

REFERENCE:

LP-27 T61.0110.6, REACTOR PROTECTION, OBJ. A FIG.5
FR-S.1, RESPONSE TO NUCLEAR GENERATION, STEP 1

[2.9/3.1]
000029K206 ..(KA's)

ANSWER: 092 (1.00)

a.

REFERENCE:

LP B-1 T61.003B.6, OTO-SE-00002, OBJ. KK5 P.118
Tech Spec 3.3.1, Table 3.3-1, Item 5

[2.8/3.4]
000033G008 ..(KA's)

ANSWER: 093 (1.00)

b.

REFERENCE:

OTG-ZZ-00002, REACTOR STARTUP, PRECAUTIONS AND LIMITATIONS 2.8, 2.9
LP B-1 T61.003B.6, OTO-SE-00001, OBJ. JJ3 P.114
CURVE BOOK TABLE 2-2 , SUMMARY OF CONTROL ROD WORTH
ODP-ZZ-00020, Rev. 4, Attachment 1, pg. 1 of 4.
[3.6/3.9]
000032A202 ..(KA's)

ANSWER: 094 (1.00)

c.

REFERENCE:

OTO-BB-00001, STEAM GENERATOR TUBE LEAK, SECTION 1.0, 6.0
LP B-1 T61.003B.6, OTO-BB-00001, STEAM GENERATOR TUBE LEAK, OBJ. N1

[3.3/3.5]
000037G007 ..(KA's)

ANSWER: 095 (1.00)

a.

REFERENCE:

LP D-5 T61.003D.6, E-3, STEP 3, OBJ. F.
LP B-1 T61.003B.6, OTO-BB-00001, P.42

[4.6/4.7]
000038A132 ..(KA's)

ANSWER: 096 (1.00)

d.

REFERENCE:

LP D-5 T61.003D.6, E-3, OBJ. F P.193

[4.2/4.5]
000038K306 ..(KA's)

ANSWER: 097 (1.00)

b.

REFERENCE:

LP D-4 T61.003D.6, E-2, STEP 4, OBJ. D

[4.5/4.4]
000054A101 ..(KA's)

ANSWER: 098 (1.00)

a.

REFERENCE:

LP-11 T61.0110.6, CVCS, OBJ.C P.1-2, 1-4

[2.9/3.2]
000028A209 ..(KA's)

ANSWER: 099 (1.00)

b.

REFERENCE:

LP-51 T61.0110.6, LSELS, OBJ. D,E P. 4, 12.
FACILITY EXAM BANK, QUES. #56, CRK-01PD-02C

[3.8/3.9]
000056A247 ..(KA's)

ANSWER: 100 (1.00)

c.

REFERENCE:

LP D-6 T61.003D.6, ECA-0.0, P. 34 OBJ. E

[4.4/4.7]
eqb.3 - % *END
000056K302 ..(KA's)

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

OC/x 3DCB

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

FEB 12 1991

Docket No. 50-483

Union Electric Company
ATTN: Mr. Donald F. Schnell
Senior Vice President - Nuclear
Post Office Box 149 - Mail Code 400
St. Louis, MO 63166

Gentlemen:

SUBJECT: EXAMINATION REPORT

During the week of January 28, 1991, the NRC administered examinations to employees of your organization who had applied for licenses to operate your Callaway Nuclear Plant. At the conclusion of the examination, the examination questions and preliminary findings were discussed with those members of your staff identified in the enclosed report.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosures will be placed in the NRC Public Document Room.

Should you have any questions concerning this examination, please contact us.

Sincerely,

Monte P. Phillips, Chief
Operations Branch

Enclosures:
1. Examination Report
   No. 50-483/OL-91-01
2. Facility Comments and NRC
   Resolution of Comments
3. Examination(s) and
   Answer Key(s) (SRO/RO)
4. Simulation Facility Report

See Attached Distribution

~~9102200036~~

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IE42
1/1

Distribution

cc w/enclosures:
G. L. Randolph, General Manager,
  Nuclear Operations
J. V. Laux, Manager Quality
  Assurance
Tom P. Sharkey, Supervising
  Engineer, Site Licensing
DCD/DCB (RIDS)
OC/LFDCB
Resident Inspector, RIII
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Resident Inspector, Wolf Creek
K. Drey
Chris R. Rogers, P.E.
  Utility Division, Missouri
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Thomas Baxter, Esq.
R. A. Kucera, Deputy Director,
  Department of Natural Resources
D. Neterer, Plant Training Manager
M. J. Davis, Project Manager. NRR
C. Kvamme, Contract Exam Supervisor, INEL
R. M. Gallo, Branch Chief, OLB

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-483/OL-91-01

Docket No. 50-483 License No. NPF-30

Licensee: Union Electric Company
Post Office Box 149
Mail Code 400
St. Louis, MO 63166

Facility Name: Callaway Nuclear Plant

Examination Administered At: Callaway Nuclear Plant

Examination Conducted: Week of January 28, 1991

Chief Examiner: Kris Shemberger
Kris Shemberger
2/11/91
Date

Approved By: Thomas M Burdick
Thomas M. Burdick, Chief
Operator Licensing Section 2
2/11/91
Date

Examination Summary

Examination administered during the week of January 28, 1991
(Report No. 50-483/OL-91-01(DRS))
Written and operating examinations were administered to five senior reactor operator and four reactor operator candidates.
Results: Five senior reactor operator and three reactor operator candidates passed the examinations. One reactor operator passed the operating examination, but failed the written examination.

~~9102200447~~

REPORT DETAILS

1. Exit Meeting

a. On February 1, 1991, an exit meeting was held. The following personnel were present at the meeting:

D. Heinlein, Assistant Superintendent, Operations
E. B. Stewart, Operating Supervisor, Training
S. M. Halverson, Senior Training Supervisor, Simulator
S. V. Henderson II, Operating Supervisor, Training
P. J. McKenna, Training
D. W. Neterer, Senior Training Supervisor, Operations
K. Mills, Quality Assurance
J. D. Blosser, Manager, Callaway Plant
C. D. Naslund, Manager, Operations Support
M. S. Evans, Superintendent, Training
S. Johnson, NRC Examiner
B. Steinke, NRC Examiner
P. Isaksen, NRC Examiner
K. Shembarger, NRC Examiner

b. The following general observations were made by the examination team and were discussed with the utility:

(1) Pre-exam review at Region III was thorough, as evidenced by the low number of post-exam comments.

(2) Facility support during the operating examination was very good.

(3) A conflict arose during the week regarding facility observers during administration of the operating test. Arrangements had not been made with the Chief Examiner prior to the exam regarding observers and therefore observers were not allowed. The facility is encouraged to discuss all exam related issues with the Chief Examiner prior to exam administration.

(4) The Steam Tables provided by the facility to be used by the candidates during the written exam contained markings. The facility should ensure all materials provided to the candidates for use during the exam do not pose an exam compromise issue.

(5) The facility is encouraged to be more discrete during pre-exam JPM validation in the plant to prevent exam compromise.

(6) While implementing the EOPs during the operating examination, the SRO candidates stopped implementation of the EOPs prior to transitioning out of E-0 to classify the event and to request the BOP operator to monitor the CSF status trees. The facility was advised that it is acceptable to classify the event after scenario termination and to request an additional operator to monitor the CSF status trees.

(7) During the simulator operating examination, both an SRO and an RO candidate went behind the control board panels, leaving only one RO "at the controls".

(8) A copy of ECA-0.0, Attachment 2, "Locally Starting Emergency Diesel Generators" is not located locally by the Diesel Generators.

(9) Facility Lesson Plan No. 3 on the diesel generators states that the day tank high level alarm is received when the diesel generator is in operation. The alarm response procedure states that the alarm is automatically bypassed when the diesel generator is in operation.

(10) One candidate consistently reached over a Contaminated Area boundary and leaned on a railing by the spent fuel pool with a Contaminated Area sign on it during the walkthrough exam.

(11) Facility JPM's and Lesson Plans are not being maintained current with procedure revisions.

(12) The resemblance between the simulator and the control room is very good.

c. Generic Strength

The following generic strength was identified by the examiners and discussed with the facility:

(1) Performance of JPM's both inside the plant and on the simulator was very good, particularly in the area of review of procedure precautions and limitations.

d. Generic Weaknesses

The following generic weaknesses were identified by the examiners and discussed with the facility:

(1) Knowledge of radiation limits during emergency operations and guidelines for the selection of individuals to make emergency entries.

(2) Use of hand friskers

(3) Knowledge of Technical Specification requirements during a reactor startup and power escalation to 100% power with one of four power range NI channels out of service.

(4) Knowledge of the guidelines for AFW isolation to a ruptured steam generator as stated in the Standing Order describing the Callaway Emergency Operating Procedures Usage Policy.

(5) Lack of alarm response procedure usage during the simulator operating examination.

(6) During the simulator operating examination, candidates silenced annunciators without first identifying the window that alarmed.

2. Written Examination

a. The following generic weaknesses, common to both the RO and SRO exams, were identified (i.e. greater than or equal to 50% of RO/SRO candidates failed to identify the correct answer).

(1) Bases for maintaining VCT pressure greater than or equal to 15 psig prior to starting a reactor coolant pump.

(2) The plant conditions that would result in RHR pump vortexing with the plant in Mode 6.

b. The following generic weaknesses in the SRO candidates were identified (i.e. greater than or equal 50% of SRO candidates failed to identify the correct answer):

(1) Design of Hydrogen Control System for maintaining containment hydrogen concentrations.

(2) Worst case plant operating status prior to the occurrence of a Main Steam Line Break accident.

c. The following generic weaknesses in the RO candidates were identified (i.e. greater than or equal to 50% of RO candidates failed to identify the correct answer):

(1) The component in the Control Rod Drive System Bank Overlap Unit that maintains the correct overlap of control banks and moves either group 1 or 2.

(2) Identifying when DNB limit has been exceeded and determining the required action.

(3) Determine the reactivity added by a cooldown associated with a Main Steam Line Break.

ENCLOSURE 2

FACILITY COMMENTS AND NRC RESOLUTION OF COMMENTS

QUESTION 023 on SRO/023 on RO:

The following plant conditions exists:

-Mode 3 following a reactor trip/Safety injection	
-RCS temperature	500 degrees
-PZR pressure	1900 psig
-Steam Generator Pressure	600 psig
-Steam Generator Levels	50% wide range
-Containment pressure	3.2 psig

Which ONE of the following describes the status of the Safety Actuation Signal Logic assuming NO operator actions were taken?

a. A low pressurizer pressure and low steam pressure signal are active.

b. A low pressurizer pressure signal is present but can be blocked.

c. A low steam generator pressure signal is present but can be blocked.

d. A containment pressure signal is active but can't be blocked.

ANSWER 023 on SRO/023 on RO:

c. A low steam generator pressure signal is present but can be blocked.

REFERENCE 023 on SRO/023 on RO:

1. SNUPPS Funct. Diagram, 7250D64, Sheet 7, 8

2. Technical Specifications, Section 3.3, Table 3.3-4 E-0, Step 4.

Callaway Comment:

At a containment pressure of 1.5 psig, Environmental Allowance Modifier (EAM) will activate causing the S/G level Rx Trip setpoint to change from 14.8% to 20.2% narrow range. This EAM signal remains locked in until containment pressure drops below the reset point of less than 1.5 psig and must be reset by I&C.

Callaway Reference:

SNUPPS Funct. Diagram, 7250064, Sheet 19
OTA-RL-RK108, Window 108E

Callaway Recommendation:

Accept answers c. and d. as correct.

NRC Resolution:

Comment accepted. The answer key was revised to reflect both c and d as correct answers.

QUESTION 038 on SRO/040 on RO:

Control of Pressurizer Heater Group A has been transferred to the Auxiliary Shutdown Panel and the control switch has been placed in the closed position.

Which ONE of the following conditions will trip the supply breaker (PG2101) for this group of heaters?

a. Pressurizer low level (17%)

b. Lockout on NB0106

c. NB01 Undervoltage

d. Safety Injection Signal

Answer 038 on SRO/040 on RO:

b. Lockout on NB0106

Reference 038 onSRO/040 on RO:

1. Facility Exam Bank Question No. 69 SBB-02PF-04C

2. LP B-4 T61.003B.6, PRZR Pressure/Level, P. 8 Obj. A2

Callaway Comment:

This question was taken directly from Section B of the RO/SRO Requalification Exam Bank. This exam bank was provided as a supplemental mailing to be used as a reference for writing the replacement exam. The operators are not required to memorize ALL trips for ALL breakers at Callaway. This question should be used only if the associated electrical drawing is supplied with the question.

Callaway Reference:

None

Callaway Recommendation:

Delete this question.

NRC Resolution:

Comment accepted. The question was deleted from both the RO and SRO examinations.

Question 092 on SRO:

Feedwater Flow to a Steam Generator that has a leaking or ruptured tube is not isolated in procedure E-3 (STEAM GENERATOR TUBE RUPTURE) or OTO-BB-00001 (STEAM GENERATOR TUBE LEAK) until level is greater than 4% in the narrow range. This is done to provide:

a. indication for inventory control.

b. a minimum radiological release to the environment.

c. a thermal layer to maintain Steam Generator pressure.

d. cooling initially which lowers Steam Generator pressure.

Answer 092 on SRO:

c. a thermal layer to maintain Steam Generator pressure.

Reference 092 on SRO:

1. LP D-5 T61.003D.6, E-3, OBJ.F P. 41

2. OTO-BB-00001, STEAM GENERATOR tube leak, STEP 6.6.6

Callaway Comment:

Westinghouse Background information and our lesson plans also include radiological release concerns along with the thermal stratification concern as reasons for filling the S/G above the tubes. This information is located in LP D-5 Pg 42.

Callaway Reference:

LP D-5, pg 42.

Callaway Recommendation:

Accept answers B. and C. as correct.

NRC Resolution:

Comment accepted. The answer key was revised to reflect both b. and c. as correct answers.

SIMULATION FACILITY REPORT

Facility Licensee: Callaway

Facility Licensee Docket No. 50-483

Operating Tests Administered On: January 29, 1991

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

ITEM	DESCRIPTION
Control Boards	The Control Boards lock-up due to dryness.
Seal Water High Temperature Alarm	The Seal Water High Temperature Alarm annunciates when charging is swapped back to the VCT.
Instrument Bus	With an Instrument Bus failure, the corresponding bistables will not trip.
"A" RCP Lift Oil Pump	When starting/stopping the "A" RCP Lift Oil Pump, the "B" and "C" RCP Lift Oil Pumps start/stop.