Examination Report No. 50-528/OL-86-01

Docket No. 50-528/529

Licensee: Arizona Public Service Company
P. O. Box 21666
Phoenix, Arizona 85036

Facility Name: Palo Verde 1 and 2

Examination at: Wintersburg, Arizona

Examination conducted: March 11-21, 1986

Chief Examiner: John P O'Brien 4/14/86
J. P. O'Brien, Chief Examiner Date Signed

Examiner: Gary W. Johnston 4/22/86
G. Johnston, Examiner Date Signed

Approved by: J. O. Elin 4/16/86
J. O. Elin, Section Chief Date Signed

Summary:

Examinations were conducted from March 11 to March 21, 1986. The Written Exam was administered on March 11, 1986 to six SRO and eighteen RO candidates. Operating exams were conducted from March 12 to March 21, 1986 to five SRO and nineteen RO candidates. All candidates who took the written exams passed. All SRO candidates passed the operating exams, and all but one RO candidate passed the RO operating exam (he failed the plant walkthru portion).

8604300152 860424
PDR ADOCK 05000528
V PDR

DETAILS

1. Persons Examined

RO Candidates:

Eighteen candidates took the written examination. Nineteen candidates took the operating examination (Simulator and Oral).

SRO Candidates

Six candidates took the written examination. Five candidates took the operating examination (Simulator and Oral).

2. Examiners

J. O'Brien, RV (Chief Examiner)
G. Johnston, RV
J. Upton, PNL
J. Smith, PNL

3. Persons Attending the Exit Meetings

APS:

+*W. F. Fernow, Training Manager
*J. Allen, Manager of Operations
+*P. J. Wiley, Training Supervisor, Licensed Operators
+ W. Rudolph, Lead Nuclear Instructor
+ O. J. Zerinque, Technical Support Manager

NRC:

+ J. P. O'Brien, Chief Examiner
+ T. Meadows, Examiner
+*J. Upton, Examiner
+*J. D. Smith, Examiner
*G. Johnston, Examiner

+Present at Exit on March 14, 1986.
*Present at Exit on March 21, 1986.

4. Written Examination and Facility Review

Written examinations were administered as follows:

Eighteen Reactor Operator Exams - March 11, 1986
Six Senior Reactor Operator Exams - March 11, 1986

At the conclusion of the exam, the facility staff reviewed the exams. The comments made by the staff at the conclusion of the review are included in the attachment (1). The resolution of these comments are included in attachment (2). These comments were discussed with the staff, and appropriate changes were made to the exam keys prior to the grading of the exams.

General weaknesses were noted by the examiners in the accuracy of reference materials provided by the training staff.

5. Operating Examinations

Simulator and oral exams were conducted March 12, 1986 to March 21, 1986. The examiners noted a significant improvement in the operability and capabilities of the simulator.

General weaknesses were identified by the examiners in the area of Radiological procedures and release points, plant radiation monitor design, use and operation.

6. Instructor Certifications

In a NRC letter dated February 28, 1986 to E. E. Van Brunt of ANNP, the NRC authorized ANNP to administer certification examinations to two instructor candidates. This action was granted due to the lack of adequate examiner resources during this exam period. The Chief Examiner reviewed ANNP's documentation of their written and operating exams, and finds the exam comparable to a NRC SRO exam. Also, he agrees that both candidates successfully demonstrated SRO level of knowledge, and can be allowed to teach systems, integrated responses, transients, and simulator courses at the Palo Verde facility. To maintain Instructor Certifications the candidates are required to be enrolled in the NRC approved requalification program. The NRC reserves the right to examine these candidates when resources are available or possibly at a future requalification program audit.

7 Exit Meeting:

On March 14, 1986 and March 21, 1986, exit meetings were conducted with the licensee representatives listed in paragraph 3. Those candidates who clearly passed the operating exam were identified. General weaknesses noted in paragraphs 4 and 5 and procedures for Instructor Certification were also discussed.

Attachment 1

Following are facility comments on the NRC examinations administered on March 11, 1986.

Reactor Operator Examination.

Section 1 - No comment.

Section 2 -

2.05 - There are 2 correct answers listed in the provided selections. Answers b as well as d are correct. Ref. PGS-15D-10.

2.09 - All valves listed are motor operated valves. The loss of electrical power to these valves will cause all valves to fail as is. The reference is based on valve fail positions due to a loss of central power.
Therefore all answers should be AS - IS.

2.10 - Additional answers not in Key are:
1) Low Lube Oil Press.
2) Low disc press. shaft driven L.O. pump.
Ref: system description manual "Main Turbine" page PGS-3A-44.

Section 3

3.09 - b - Answer as stated is correct but may also see 4% of instrument range, as it is sometimes stated.

c - AMI will only occur if selection switch for neutron flux is in the "AVG" position.

Ref. - Sys. Descrip. RRS training Article NS 9C Rev. 1, page NS-9C-7.

3.10 - The steam flow signal used for the setpoint generation and the quick open signals are biased by PZR pressure. Some operators may include this information in the answer.
Ref. - Sys. Desc. "Steam Bypass Control System" NS-9B, page NS-9B-19.

3.11 - Detector (2) is located at the edge of the pool vs over the pool.
Ref. - Actual plant location.

3.12 - a - Correct answer is - SIAS, CIAS and MSIS. All are received on High Containment Pressure.

3.12 - b - Should be AFAS 1 vs AFAS 2. But AFAS 1 is locked out by high Steam Generator differential pressure. An acceptable answer should be:
AFAS 1 with a high delta P lockout, or
AFAS 1 is not listed as it will not occur due to the high delta P lockout.

Section 4 -

4.05 - Question needs to state if power is increasing or decreasing to be a valid question.
Also, valve does not fail closed but receives a closed signal at 15% power increasing.
Question should be invalidated.

4.13 - Answer should accept information indicating candidate is aware of panel location. Panel name and number designation are not required knowledge for RO's.

Section 5 -

5.02 - b - With conversion sheet provided the calculated answer is <u>9652</u> lbm/hr which is outside answer key acceptable range. Widen acceptable band on question.

5.06 - Another consideration is the change in Beta over core life. Beta gets smaller as the core ages causing increased reactor response for a given amount of positive reactivity.

5.13.C - Correct answer should be; The probability of a thermal fission is greater for Plutonium 239 than Uranium 235 due to Plutonium 239 having a larger fission cross-section.

Section 6 -

6.04 - a - True - Reference 41A012215 states that loss of power to PNA-D25 (since PKA-M41 is lost) causes train A bistables to trip (p. 27) and loss of PNC-D27 causes train C bistables to trip (p. 16 step 4.2.1). Therefore since the A and C bistables are tripped the AC matrix will cause a Rx trip.

b. Answer is correct as given.

c. Answer is correct as given. (CREFAS will actuate but CIAS will not.)

d. True. Same situation as in part a. This will cause the ESFAS "AC" Matrix to call for MSIS.

6.06 a2 - In the manual mode no setpoint is used. Instead the operator has direct control over pressure controller output via the manual adjustment switch Ref. - Sys Desc. NS-9G-7. PZR Press. Control System.

6.07 - Valves in a thru e are motor operated and as such will fail as is on loss of electrical power. Valve in part f is solenoid operated and will fail closed. The reference is based on valve fail positions due to loss of control power. Therefore answers a - e are fail as is.
f is closed.

6.11 - Answer is correct per reference material. However unit controllers were changed out from Foxboro 250 series to 270 series which have white backlit indication when in Auto or Manual mode. Expect response to be White.

6.12 - b - Remove words "and tagged out" from answer key. No requirement exists to tag out the power supplies.

Section 7 -

7.07 - b - When the high CPS alarm is received (approx. 2000 cps).
Present answer states (approx. 200 cps).
Ref. 41OP-12203.

7.08 - c - Answer is false based on the (about 500 pcm) - The procedure states (about 100 pcm) other than this area True is the correct answer.
Ref. 41OP-12203.

7.11 - Answer c is correct not d.
Ref. 41RO-1ZZ01 Rx trip pg. 27.

Section 8 -

8.10 - Question did not require that positions be specified as part of answer. Answer key reflects positions must be specified. Correct number should constitute correct answers.

ATTACHMENT 2, PART 1:

RESOLUTIONS TO FACILITY COMMENTS FOR
THE SENIOR REACTOR OPERATOR EXAM:

5.02.b - Corrected calculation using number given for conversion factor on the equation sheet. Acceptable band remained the same.

5.06 - Added the fourth consideration given by the licensee and redistributed points. Full credit now requires 4 considerations to be given.

5.13.c - Key already addresses this. "More likely to fission" means the same thing as a "larger cross section for fission". Graders will accept either verbiage.

6.04 - Correct a and d to read: TRUE. (Reference material provided was inaccurate).

6.06.a2 - KEY changed to read: "in manual operator controls direct output of controller output (reference material provided was inaccurate due to plant modification).

6.07 - Key corrected: a-e - as is
f - closed
(reference material provided inaccurate).

6.11.a - Facility has been change controller to a Foxboro Type 270 controller. Reference material describes Foxboro Type 250. Answer key corrected to accept either "white" or "red".

6.12.b - Deleted words "and tagged out" from key as per clearance procedure.

7.07.b - Typo read 200 vs. 2000 CPS. Key corrected.

7.08.c - No key change. 500 PCM is not the distractor to make the statement false.

7.11 - Key corrected.

8.10 - No change to key. Question specifically identified "other" positions to be considered.

ATTACHMENT 2, PART 2:

RESOLUTIONS TO FACILITY COMMENTS FOR
THE REACTOR OPERATOR EXAM

2.05	-	Answers (b.) and (d.) are mutually exclusive. (b.) is wrong because the swing charger is normally tied to bus-B. No change to the answer key.
2.09	-	Loss of electrical power to the signal system will result in the failed position listed in the reference. Loss of electrical power to the drive motors results in "as-is" positions. Hence, either the original answer key or "as-is" for each/every valve will be accepted.
2.10	-	These two items will be added to the list of trips in the Answer Key.
3.09b	-	Either "4% of instrument range" or "5% of power level" will be accepted.
3.09c	-	"AMI" will be accepted either with or without the qualifier.
3.10	-	Either "steam flowrate" or "steam flowrate biased by PZR pressure" will be accepted.
3.11	-	"edge of the pool", or "pool area", or "over the pool" will be accepted.
3.12a	-	The answer will be corrected to read "SIAS, CIAS, MSIS [+0.33 each]".
3.12b	-	The answer will be corrected to read "SIAS, CIAS, CSAS, MSIS and AFAS-1 with a high P lockout/or no AFAS-1 [+0.2 each]".
4.05	-	Question 4.05 will be deleted. The terminology in the reference material, "fail closed", is indeed misleading and should be corrected.
4.13	-	Agreed; inclusion of the panel name and especially number are not required for full credit.

MASTER KEY

OB

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

Facility: PALO VERDE 1 AND 2
Reactor Type: CE-PWR
Date Administered: MARCH 11, 1986
Examiner: J. O'BRIEN
Candidate: ______

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

Category Value	% of Total	Candidate's Score	% of Category Value		Category
25.0	25.0	______	______	5.	Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics
25.0	25.0	______	______	6.	Plant Systems Design, Control, and Instrumentation
25.0	25.0	______	______	7.	Procedures - Normal, Abnormal, Emergency, and Radiological Control
25.0	25.0	______	______	8.	Administrative Procedures, Conditions, and Limitations
100.0		______ Final Grade			Totals

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

____
Candidate's Signature

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

Facility: PALO VERDE 1 AND 2
Reactor Type: CE-PWR
Date Administered: MARCH 11, 1986
Examiner: J. O'BRIEN
Candidate: ____________

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

Category Value	% of Total	Candidate's Score	% of Category Value		Category
25.0	25.0	______	______	5.	Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics
25.0	25.0	______	______	6.	Plant Systems Design, Control, and Instrumentation
25.0	25.0	______	______	7.	Procedures - Normal, Abnormal, Emergency, and Radiological Control
25.0	25.0	______	______	8.	Administrative Procedures, Conditions, and Limitations
100.0		______ Final Grade			Totals

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

____
Candidate's Signature

EQUATION SHEET

$f = ma$

$w = mg$

$E = mC^2$

$KE = \frac{1}{2}mv^2$

$PE = mgh$

$W = v\Delta P$

$\Delta E = 931\Delta m$

$\dot{Q} = \dot{m}C_p\Delta T$

$\dot{Q} = UA\Delta T$

$Pwr = W_f\,\dot{m}$

$P = P_o\,10^{SUR(t)}$

$P = P_o\,e^{t/T}$

$SUR = 26.06/T$

$T = 1.44\ DT$

$SUR = 26\left(\frac{\lambda_{eff}\rho}{\bar{\beta} - \rho}\right)$

$T = (\ell^*/\rho) + [(\bar{\beta} - \rho)/\lambda_{eff}\rho]$

$T = \ell^*/(\rho - \bar{\beta})$

$T = (\bar{\beta} - \rho)/\lambda_{eff}\rho$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$\rho = [\ell^*/TK_{eff}] + [\bar{\beta}/(1 + \lambda_{eff}T)]$

$P = \Sigma\phi V/(3 \times 10^{10})$

$\Sigma = N\sigma$

$v = s/t$

$s = v_o t + \frac{1}{2}at^2$

$a = (v_f - v_o)/t$

$v_f = v_o + at$

$\omega = \theta/t$

$\text{Cycle efficiency} = \frac{\text{Net Work (out)}}{\text{Energy (in)}}$

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

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$t_{1/2}(eff) = \frac{(t_{1/2})(t_b)}{(t_{1/2} + t_b)}$

$I = I_o e^{-\Sigma x}$

$I = I_o e^{-\mu x}$

$I = I_o\,10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = 0.693/\mu$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$CR_1(1 - K_{eff})_1 = CR_2(1 - K_{eff})_2$

$M = 1/(1 - K_{eff}) = CR_1/CR_0$

$M = (1 - K_{eff})_0/(1 - K_{eff})_1$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 1 \times 10^{-5}$ seconds

$\lambda_{eff} = 0.1\ \text{seconds}^{-1}$

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

$R/hr = (0.5\ CE)/d^2$ (meters)

$R/hr = 6\ CE/d^2$ (feet)

WATER PARAMETERS

1 gal. = 8.345 lbm

1 gal. = 3.78 liters

1 ft^3 = 7.48 gal.

Density = 62.4 lbm/ft^3

Density = 1 gm/cm^3

Heat of vaporization = 970 Btu/lbm

Heat of fusion = 144 Btu/lbm

1 Atm = 14.7 psi = 29.9 in. Hg

1 ft. H_2O = 0.4335 lbf/in^2

MISCELLANEOUS CONVERSIONS

1 Curie = 3.7×10^{10} dps

1 kg = 2.21 lbm

1 hp = 2.54×10^3 BTU/hr

1 Mw = 3.41×10^6 Btu/hr

1 Btu = 778 ft-lbf

1 inch = 2.54 cm

$^oF = 9/5\,^oC + 32$

$^oC = 5/9\,(^oF - 32)$

Senior Operators Exam

SECTION 5

Theory of Nuclear Power Plant Operation,
Fluids, and Thermodynamics

5.01 (1.5)

The reactor is determined to be shutdown by 6% delta K/K with indication in the source range of 30 counts per second.

a) What is the Keff when the reactor is shutdown by 6% delta K/K? (0.5)

b) What would the count rate be if Keff is increased to 0.98? (0.5)

c) What would the count rate be if Keff is increased to 0.99? (0.5)

5.01 Answer:

$$\frac{\text{delta } K}{K} = 0.06$$

$$\frac{1 - K}{K} = 0.06$$

$$1 = K + 0.06(K)$$

$$1 = 1.06(K)$$

a) $KEff = 1/1.06 = 0.94$ (0.5)

$$\frac{1 - K1}{1 - K2} = \frac{CR2}{CR1} \quad \text{and} \quad \frac{0.06}{0.02} = \frac{CR2}{CR1}$$

b) $CR2 = (30)(3) = 90$ cps (0.5)

Conversly $\frac{CR3}{30} = \frac{0.06}{0.01}$

c) $CR3 = (30)(6) = 180$ cps (0.5)
(Rule of thumb, doubles)

Reference: Glastone and Sesonske, "Nuclear Engineering".
Palo Verde Reactor Theory notes p.8.6-8.11

5.02 (3.0)

The turbine of the Auxiliary Feedwater System (AFS) is rated at 1250 hp with steam supplied at a pressure of 1170 psi.

a. When the AFS turbine is being supplied with 1000 psia main steam, How much work can be supplied by the turbine? Assume thermodynamically ideal operation of the turbine. Express the answer in Btu/lbm. (2.0)

b. If the turbine was receiving 250 Btu/lbm, and was driving the AFS pump with a power of 950 hp, what is the steam flowrate through the turbine? (Express the answer in lbm/Hr.) (1.0)

5.02 Answer (3.0)

a. h1 = 1193 Btu/lbm (using CE Steam Tables) (0.5)

Isentropic expansion (0.5)

h2 = 905 Btu/lbm (0.5)

Delta h = 288 Btu/lbm (+/-10 Btu/lbm) (0.5)

b. 950 hp X ~~2545~~ 2.54x10³ Btu/hr / hp = ~~2.42~~ 2.413 M btu/hr (0.5)

(~~2.42~~ 2.413 M btu/hr)/(250 Btu/lbm) =

~~9,680~~ 9652 lbm/hr (+/-25 lbm/hr) (0.5)

reference: PGS-11,12,13 and Palo Verde HT&FF notes Section 2, pp.9-16

5.03 (3.0)

Refer to the FIGURE 5.2 that follows this page. The figure is of "Heat Flux" versus "Temperature Difference between the Clad wall and the Bulk Fluid" for an operating reactor. Note that there are two curves represented for two pressures: (Assume: P1 < P2).

a. What is the principle type of heat transfer that is occuring at pressure P1 and 1.0E4 BTU/Hr-ft^2 between the wall and the bulk fluid? (0.5)

b. What is the principle type of heat transfer that is occuring at Pressure P2 and 1.0E4 BTU/Hr-ft^2 between the wall and the bulk fluid? (0.5)

c. What pressure will yield a lower fuel centerline temperature at 1.0E4 BTU/Hr-ft^2? (0.5)

d. What pressure will yield a higher fuel centerline temperature at 3.0E5 BTU/Hr-ft^2? (0.5)

e) What type of heat transfer between the wall and the bulk fluid is occuring at Pressure P1 and 3.0E5 BTU/Hr-ft^2? (0.5)

f) Assuming bulk temperature well below saturation, Will decreasing the pressure affect the bulk fluid temperature at a heat flux of 3.0E5 BTU/Hr-ft? (0.5)

5.03 Answer: (0.5) for each.

a) Nucleat boiling.

b) Convection (other terms may be used).

c) Pressure 1.

d) Pressure 1.

e) Radiant heat transfer (steam blanketing).

f) The bulk fluid temperature remains constant (is independent of pressure below saturation.)

Reference: Palo Verde HT&FF notes pp.34-38.
General Physics Nuclear Technology, Section E, Pages 2-144, 2-151, 2-159, and 2-164.

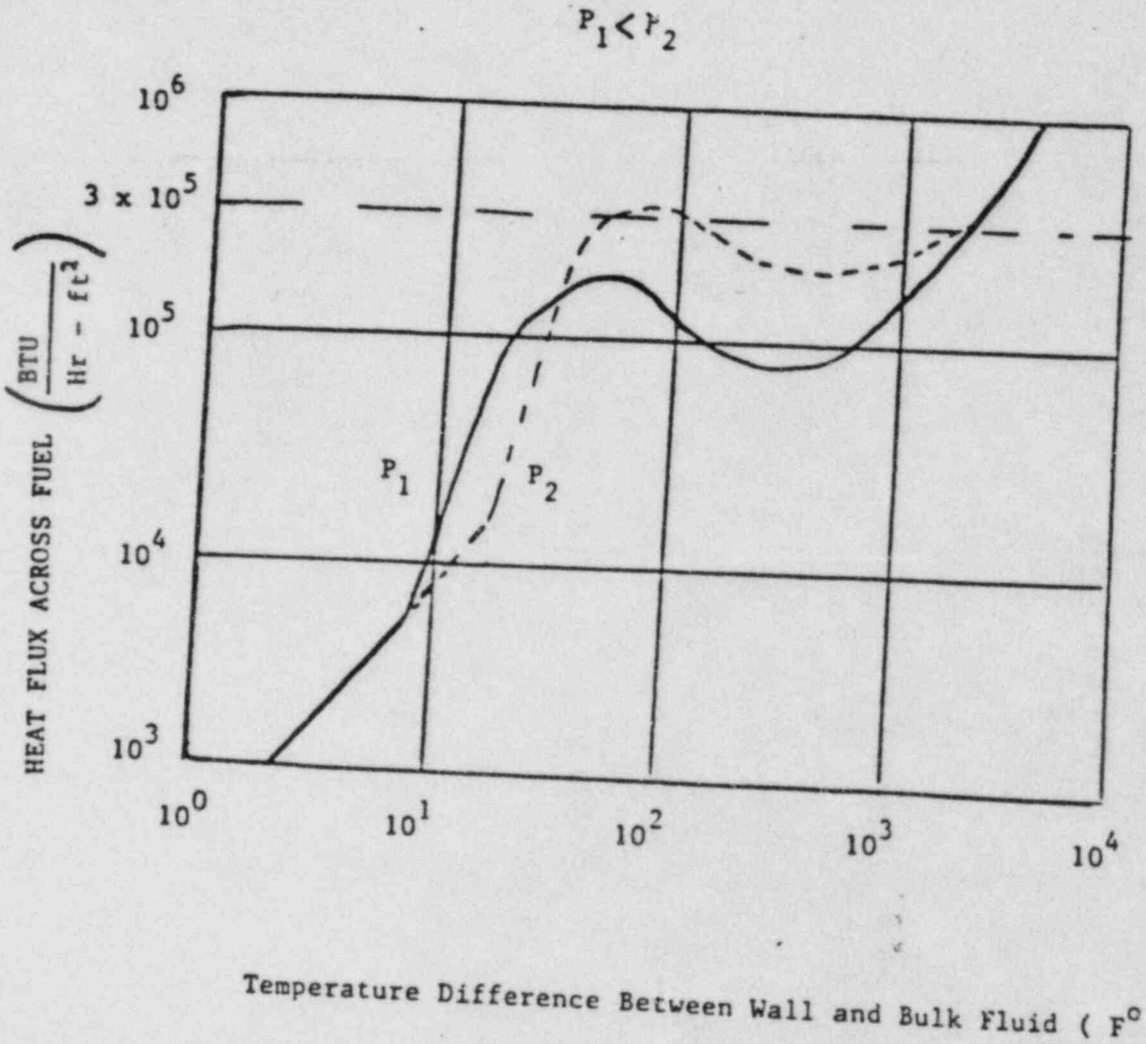

$P_1 < P_2$
10^6
3×10^5
10^5
10^4
10^3
HEAT FLUX ACROSS FUEL $\left(\frac{BTU}{Hr - ft^2}\right)$
P_1
P_2
10^0
10^1
10^2
10^3
10^4
Temperature Difference Between Wall and Bulk Fluid (F^o)

FIGURE 5.2

5.04 (1.0)

SELECT THE BEST ANSWER:
As a subcritical reactor nears criticality, the length of time to reach equilibrium count rate after an insertion of a given fixed amount of positive reactivity: (1.0)

a. decreases, primarily due to the increased population of delayed neutrons.

b. increases, primarily due to the increased population of delayed neutrons.

c. decreases, because the source neutrons are becoming less important in relation to the total neutron population.

d. increases, because of the larger number of neutron life cycles required to reach equilibrium.

5.04 Answer (1.0)

d.

Reference: Palo Verde - Reactor Theory review , Ch. 8

5.05 (2.0)

Answer the following TRUE or FALSE :

a. The production rate of indirect Xenon from Iodine is <u>faster</u> than the decay rate of Xenon to Cesium. (0.5)

b. Slowing the rate of a power decrease, lowers the height of the resultant Xenon peak. (0.5)

c. The resultant Xenon peak from a reactor Trip from 50% power is larger than a trip from 100% power. (0.5)

d. During an increase in power from equilibrium Xenon conditions, Xenon concentration initialy decreases. (0.5)

5.05 Answer (2.0)

a. True

b. True

c. False

d. True

Reference: Palo Verde Reactor Theory Notes Ch. 16.

5.06 (2.0)

Regarding a Main Steam line rupture:

a) Why is a rupture of a Main Steam line at End of Life (EOL) a much more limiting accident than at the Beginning of Life (BOL)? **(2.0)**

5.06 Answer:

a) The Moderator Temperature Coefficient is less negative at BOL <u>(0.67)</u>. Than at EOL. This difference in magnitude increases the severity of the addition of positive reactivity <u>(0.67)</u> due to the sudden cooling of the reactor coolant from the Main Steam line rupture <u>(0.67)</u>.

also consider β gets smaller as core ages increased Rx response for $+\Delta K$

Reference: Palo Verde Reactor Theory Notes Ch. 12
General Physics, Volume III, Chapter 3, Section 3.

5.07 (1.0)

A tank contains water to a level 40 ft above the bottom of of the tank. A Nitrogen cover gas is at 100 psia. The tank and its contents are at 70 degress F, and the density of the water is 62.4 lbm/cubic ft. The pressure at the bottom of the tank is: **(1.0)**

a. 117 psia

b. 132 psia

c. 208 psia

d. 308 psia.

5.07 Answer (1.0)

a. 117 psia (1.0)

Reference: Palo Verde Thermo Review Sect. 2 pp. 6-9
General Physics HTFF notes pp. 2-115 - 2-117

5.08 (1.0)

If the temperature of the tank in QUESTION 5.07 (container, water and gas) were raised and if no water or cover gas was allowed to enter or leave the tank, the pressure at the bottom of the tank would (1.0)

a. increase, because the gas volume has decreased and the temperature of the gas has increased.

b. increase, because of the pressure caused by the higher water level (water column effect), and the temperature of the gas has increased.

c. decrease, because the water density has decreased.

d. decrease, because the cover-gas density has decreased.

5.08 Answer (1.0)

a. (1.0)

Reference: See question 1.02

5.09 (1.0)

For the following indicate which is the most accurate statement:

a. The main condenser functions by: (0.5)

1. Removing the latent heat of vaporization at a constant temperature to allow the steam to condense.

2. Providing a low pressure volume that allows the steam to condense.

3. Cooling the steam to the point where it is at saturation temperature.

4. Lowering the temperature of the steam below the saturation temperature.

b. Condensate depression is: (0.5)

1. Maintained by adequate backpressure on the low pressure turbine exhaust.

2. Used to maintain constant temperature profile on the condenser tubesheets.

3. Used to maintain adequate Net Positive Suction Head for the condensate pumps.

4. Used to maintain adequate Net Positive Suction Head for the main feedwater pumps.

5.09 Answer:

a. 1. (0.5)

b. 3. (0.5)

Reference: General Physics "Heat Transfer and Fluid Flow"

5.10 (2.0)

At Beginning of Life (BOL) the plant nuclear instrumentation count rate will <u>increase</u> as plant temperature increases causing a Reactor Coolant System (RCS) coolant density decrease.

a) Assuming there is no rod movement or changes in RCS boron concentration. What <u>two</u> factors contribute to this phenomena? (2.0)

5.10 Answer:

a.

1. Neutrons travel farther, more leakage. (1.0)

2. Lower boron density, less boron absorption. (1.0)

Reference: General Physics, Volume II, Chapter 4, Sections B & D.

5.11 (1.0)

Of the following operations, which <u>one</u> will have a negative effect on the available Net Positive Suction Head (NPSH) for a given centrifugal pump: (1.0)

a. Throttling open the pump's suction valve.

b. Throttling open the pump's discharge valve.

c. Decreasing the pump's speed.

d. Decreasing the temperature of the fluid being pumped.

5.11 Answer (1.0)

b (1.0)

Reference: Palo Verve Thermo Review of L.O. pp.21-28

5.12 (3.0)

a. **Explain how the energy of nuclear fission is transfered to the reactor coolant.** (2.0)

b. **Explain why it is necessary to continue cooling the core following a shutdown from power generation.** (1.0)

5.12 Answer (3.0)

a. Most of the fission energy (93%) is in the form of fission product kinetic energy. The fission particles are slowed down in the fuel matrix by inelastic collisions, and this he: s up the fuel matrix (1.0). The fuel heats up the clad by conduction and/or convection (0.5); which in turn heats up the coolant (0.5)

b. the heat generated by fission product decay in the fuel matrix - decay heat - (0.5) is sufficent to damage the core if cooling is not continued (0.5)

Reference: General Physics Vol II, Ch 4.

5.13 (2.0)

a. **Describe what reaction produces Plutonium-239 in the core.** (1.0)

b. **Explain how and why having Plutonium-239 in the core affects the core's reactivity?** (0.5)

c. **As a fuel, what is the major difference between Plutonium-239's and Uranium-235's fission cross-section ?** (0.5)

5.13 Answer (2.0)

a. Pu-239 results from the neutron capture reaction with U-238 (1.0). (will also accept equation)

b. Pu-239 is fissionable, and therefore adds positive reactivity to the core. (0.5)

c. U-235 predominantly fissions from an absorbtion of a neutron of thermal energy. Where as Pu-239 is more likely to fission due to a interaction with neutron of epithermal or higher energy. (0.5) and Pu 239 has larger cross-section for fission

Reference: General Physics - Reactor Theory Notes Vol.II Ch.2-4

5.14 (1.5)

The following pairs of terms sound similar, but are in fact different. Explain for each, the difference between:

a. Fast and Prompt Neutrons. (0.5)

b. Slow and delayed Neutrons (0.5)

c. Activity and Reactivity (0.5)

5.14 Answer (1.5)

a. Fast neutrons refer to those at high energies. (0.25) Prompt neutrons are those released at the instant of the fission. (0.25)

b. Slow neutrons are those at low energy levels (0.25) Delayed neutrons are those which appear at some time after the fission event due to the decay of fission products. (0.25)

c. Activity refers to the rate of radioactive decay (0.25) reactivity expresses the deviation of a reactor core from the critical conditions (0.25)

Reference: General Physics - Reactor Theory Notes Vol.II, Ch.2

END OF SECTION 5

SECTION 6
PLANT SYSTEMS, DESIGN, CONTROL, AND INSTRUMENTATION

6.01 (3.0)

Operating and Bistable Trip Channel bypasses are used in the Reactor Protection System.

a. List two of the three Operating Bypasses and briefly describe at least one plant condition when you would expect to use each of these bypasses. (2.0)

b. How does placing one bistable trip channel in bypass for maintenance affect the Reactor Protection System 2 out of 4 logic ? (0.5)

c. What will automatically occur if an operator attempts to bypass the same parameter in more than one channel at the same time ? (0.5)

6.01 Answer

a. (1.0 each for any two of the following)

Low pressurizer pressure bypass: system tests at low power or low temperature or heatup or cooldown of the RCS.

High log power level bypass: during reactor startup or above 10-4% power.

DNBR/LPD trip bypass: at low power levels or below 1% power.

b. Converts RPS logic to 2 out of 3 logic (0.5)

c. the lowest priority channel will return to normal (interlock prevents two channels in bypass at the same time) (0.5)

Reference: NS 6-25 & 26, Reactor Protection System

6.02 (4.0)

Assume that the Plant is at 75 % power, control systems are in automatic, and the selector switches for Tave, and Nuclear power control channels selected to "average". Consider each of the following situations one at a time.

a. What will be the instrument response if a Tc RTD cable is cut ? (ie: fail high, fail low, or stay the same) (1.0)

b. With a Th RTD failed high list four of the five alarms or indications you would expect. (1.0)

c. How will the CEDMCS respond in "auto-sequential" if the single TLI which is selected for control fails low ? (1.0)

d. How does the instrument indication fail and what two alarm do you expect if one of the nuclear power control channel detectors shorts out ? (1.0)

6.02 Answer

(a) Fails high (open circuit) (1.0)

(b) (0.25 each for any four of the following) Tave-Tref Hi-Lo alarm, RCLoops Hi AWP, AMI, Channel deviation (on RRSTD), pressurizer level at or approaching maximum (1.0)

(c) CEAs will insert or an insert signal will be sent to CEAs (1.0)

2 of 3 for alarms

(d) ①SU and ①CONT CH TRBL Alarm, ②AMI and ③Channel Deviation LED on the RRSTD /(the indication fails high) (1.0)

Reference: NS-9C, 14-20, Reactor Regulating System
41AO-1ZZ34, RRS Malfunction

6.03 (3.0)

a. Why must the Emergency Diesel Generator (D/G) lube oil circulating pump be run for a minimum of 30 minutes after shutting down a hot D/G? (1.0)

b. List four (4) of the actuating signals that should automatically start the essential cooling water System. (2.0)

6.03 Answer (3.0)

a. to dissipate heat (0.5) from the turbocharger Bearings. (0.5)

b. (any 4, 0.5 each)

1. Loss of Offsite Power or (LOP)
2. CREFAS
3. SIAS
4. CRVIAS
5. AFAS-1
6. AFAS-2
7. CSAS

Reference: SD: Emergency Diesel Generator Sect.3.28, pp.10
SD: Essential Cooling Water Sys. Sect. 4.1.2 , pp.2

6.04 (2.0)

125 VDC IE bus PKC-M43 has just been denergized due to a fault. Alternate power is being lined up to the associated 125 VAC IE distribution pannel PNC-D27 while the DC bus is being examined to decide wheather to reenergize the bus or to shutdown the plant.

State wheather the following are true or false.

a. A loss of power on 125 VDC bus PKA-M41 before PNC-D27 is reenergized will SCRAM the plant. (0.5)

b. A loss of power on 125 VDC bus PKD-M44 before PNC-D27 is reenergized will cause a SIAS to occur. (0.5)

c. A loss of power on 125 VAC Distribution Pannel PNB-D26 after PNC-D27 is reenergized will cause CREFAS and CIAS actuation. (0.5)

d. A loss of power on 125 VAC Distribution Pannel PNA-D25 before PNC-D27 is reenergized will cause main steam and feedwater isolation. (0.5)

6.04 Answer

(0.5 each)

(a) ~~False~~ true as per provided material

(b) True

(c) False

(d) ~~False~~ TRUE same as A

~~(b) Reactor trips~~ (1.0)

Reference: PGS-15D, 125 VDC Class IE Power
PGS-15E, 125 VAC Class IE Power
41AO-1ZZ17, Loss of 125 VDC Class IE Electrical Power
41AO-1ZZ15, Loss of 125 VAC Class IE Electrical Power

6.05 (3.0)

Assume that the controlling pressure transmitter for the Pressurizer Pressure Control System has failed high while in service.

a. What two alar ., would you expect to receive in the control room ? (1.0)

b. What will the spray valve(s) and the heaters do ? (1.0)

c. If the operator does not take action how will the plant be affected ? (1.0)

6.05 Answer

(a) PZR TRBL and PZR PRESS HI-LO (1.0)

(b) Spray valve(s) open and all heaters will be off. (1.0)

(c) Plant pressure decreases until SIAS and reactor trip occur. (1.0)

Reference: NS-9G-13, Pressurizer Pressure Control System

6.06 (2.0)

The pressurizer pressure controller provides singular control of heaters and sprays.

a. How is the output of the pressurizer pressure master controller determine when in:

   1. AUTOMATIC (0.5)
   2. MANUAL (0.5)

b. As the controller output increases from 0% to 67% , What takes place to control pressure? (0.5)
(assume controller in AUTO)

c. As the controller output increases from 67% to 100%, What takes place to control pressure? (0.5)
(assume controller in AUTO)

6.06 Answer (2.0)

~~manual direct control of controller output~~

a.1 Difference between the setpoint and the measured pressure of the Pressurizer. (0.5)

in manual oper. controls direct output of controller output

a.2 Difference between the setpoint and the value set by the operator. (0.5)

b. The power supplied to the proportional heaters will go from full power to zero power. (0.5)

c. (the output of the spray valve controller will increase from 0% to 100%) The spray valve goes from full closed to full open. (0.5)

Reference: SD: NS-G9 PSZR Pressure Control Sys.

6.07 (2.0)

As a result of a loss of electrical power, what is the failed position for each of the following valves associated with the Safety Injection System. (OPEN, CLOSED, or AS-IS)

a. Low pressure header isolation valve(s) (0.33)

b. HPSI pump mini flow line isolation valve(s) (0.33)

c. Containment sump line isolation valve(s) (0.33)

d. LPSI pump suction isolation valve(s) (0.33)

e. Hot-leg injection isolation valve(s) (0.33)

f. SI Tank vent valve(s) (0.33)

6.07 Answer

(0.33 for each of the following)

(a) As-is licensee AS-IS vs ref. original answer is as per ref. mtl.

(b) Closed

(c) Open

(d) Closed

(e) As-is

(f) As-is CLOSED

Reference: NS-3A, Safety Injection and Shutdown Cooling System, page NS-3A-25

6.08 (2.0)

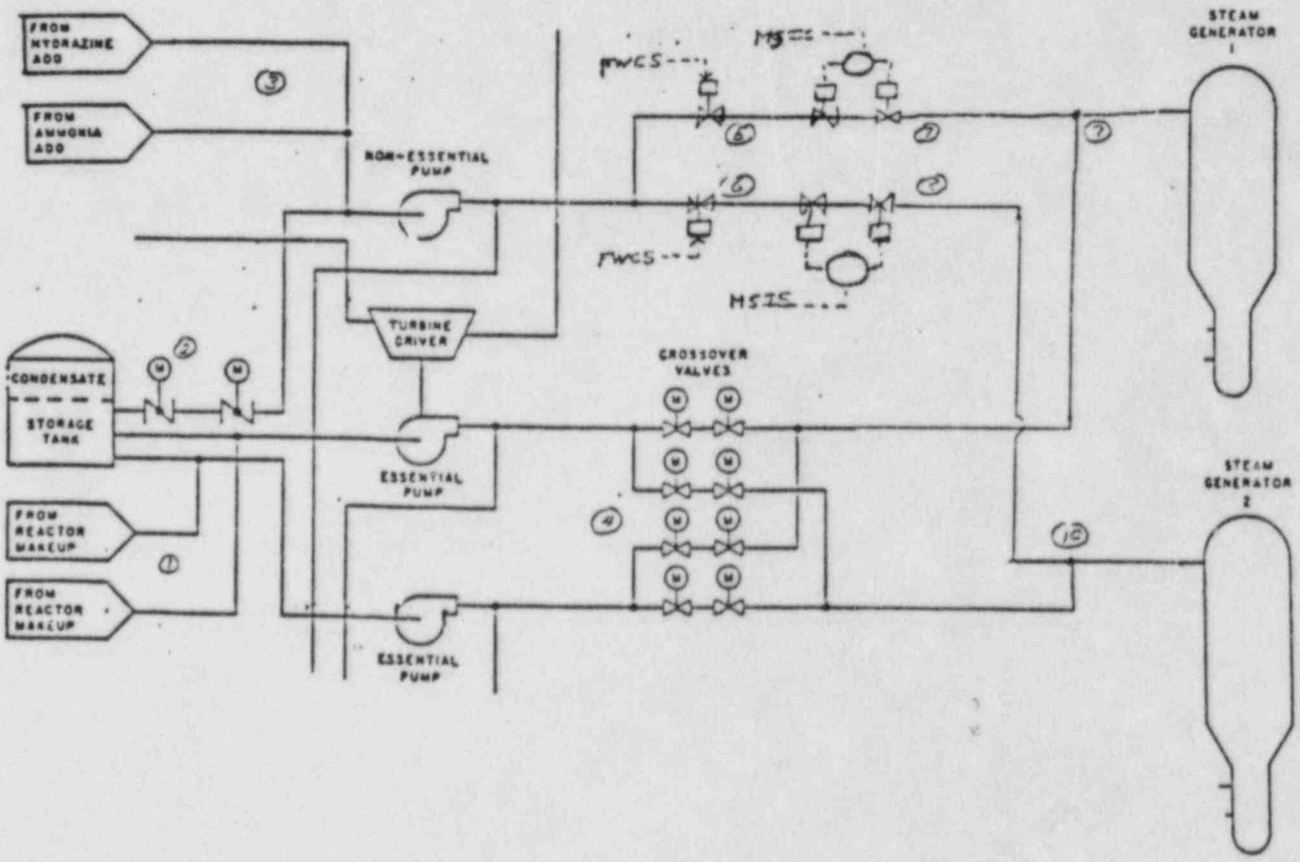
Draw a one-line diagram of the auxiliary feedwater system that contains the following components. Label the components on the diagram.

* Essential pump
* Turbine-driven essential pump
* Non-essential pump
* Line providing hydrizene addition
* Line providing ammonia addition
* Water sources for the three pumps
* The eight (8) cross-over valves
* Line from the main feedwater supply
* Valves contrclled by the FWCS
* MSIS valves
* Steam generators

6.08 Answer

See Figure 6.1 (2.0)

Reference: PGS-11, Auxiliary Feedwater System, pages 2 & 3



[+0.2 each connection 1-10]

Figure 6.1 (ANSWER)

-Section 6.0 Continued on Next Page-

6.09 (1.0)

Answer True or False to each of the following statements concerning control of each atmospheric dump valve.

a. A nitrogen gas accumulator is provided to ensure operability for at least eight hours following the loss of the Instrument and Service Air and Nitrogen Gas systems. (0.5)

b. One P/I converter is installed to convert the pneumatic signal of 3 to 30 psig to an electrical signal, which is used to control the position of the dump valve. (0.5)

6.09 Answer

(a) True (0.5)

(b) False (0.5)

Reference: PGS-1A, Main Steam System, pages 10 & 11

6.10 (1.0)

Listed below as "a" through "d" are components of the Fire Detection and Suppression System for the Emergency Diesel Generators. Also listed below as "1" through "4" are four possible results from the components exposed to a fire. Match the correct result with each listed component. Note that one result may occur for more than one component. Listing the letters "a" through "d" with the correct number(s) "1", "2", "3", or "4" is an adequate answer.

a. Sprinkler system fusible link (0.25)

b. Fusable link on the chain for the rollup fire door (0.25)

c. Ionization smoke detector in the DG local control room (0.25)

d. Ultra-violet flame detector in the DG engine room (0.25)

---

1. Activates water flow to the sprinkler heads in the DG engine room.

2. Activates water flow to the sprinkler heads heated in the silencer room.

3. Activates an alarm indication only.

4. Isolates the DG engine room from the DG local control room.

6.10 Answer

(a)	1	(0.25)
(b)	4	(0.25)
(c)	3	(0.25)
(d)	1	(0.25)

Reference: Diesel Generator and Auxiliaries Handbook, pages 7 & 8

6.11 (1.0)

A Foxboro control station is provided in the control room for positioning the condensate demineralizer bypass valve. Answer the following parts of this question which refer to the control of the Cndensate System.

a. When in the "Manual" mode, what color of indicating light is illuminated ? (0.25)

b. When in the "Auto" mode, what signal varies valve position ? (0.25)

c. When in the "Auto" mode the valve position will vary automatically only if three conditions are met. State two of these three conditions. (0.5)

ref mtl

licensee says they have these

6.11 Answer

FOXBORO 250

270 WHITE

(a) red (0.25)

(b) demineralizer differential pressure (0.25)

(c) (0.25 each for any two of the following)

Demineralizer inlet valve is fully open
Demineralizer outlet valve is fully open
Demineralizer differential pressure less than 35 psi

Reference: PGS-9A, Condensate System, pages 21, 22, & 30

6.12 (1.0)

Four Safety Injection Tanks (SIT) rapidly flood the core following an inadvertent depressurization of the Reactor Coolant System (RCS).

a. How are the tanks isolated from the RCS ? (0.5)

b. How is it ensured that the tanks will not become isolated from the RCS during normal at power operation. (0.5)

6.12 Answer (1.0)

a. Isolated from the RCS by means of a motor operated valve and two check valves. (0.5)

b. Motor Operated valve is locked open and power to the motor is removed ~~and tagged out~~. (0.5)
NA

Reference: SD: NS-3A Safety Injection

END OF SECTION 6
CONTINUE TO SECTION 7

SECTION 7

PROCEDURES: NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL

7.01 (3.0)

The plant is at 4 % reactor power and the secondary systems are being warmed up per 41OP-1ZZ04, Plant Start-up Mode 2 to Mode 1. All surveillances are complete and systems appear to be operating normally. Without any warning an MSIS occurs !

Respond to the following questions as though you were following up this MSIS per 41AO-1ZZ31, Inadvertent MSIS.

a. Besides the closed indications for MSIVs, FIVs, and S/G blowdown isolation valves there are three alarms and two direct indications of an MSIS. List four of these alarms and or indications and their locations. (1.0)

b. In order to evaluate wheather a MSIS was required or not the procedure directs the operator to look at three recorders which would indicate plant conditions just prior to the MSIS. List the three recorded plant conditions. (1.0)

c. Having determined that the MSIS should be reset the operator must reset the 4 Actuation Trip Path Channels. Briefly describe how and where this is done. (1.0)

6.01 Answer

(a) (0.25 each for any four of the following) (1.0)
1) MSIS A channel alarm on the main control board, 2) MSIS B channel alarm on the main control board, 3) MSIS alarm (5C03A) on the main control board, 4) Trip path indicators de-energized on the PPS remote operators modules on B05, 5) Trip path indicators de-energized on the indication pannel above the PPS cabinets (behind the main control board)

(b) (0.33 each) (1.0)
Steam generator pressure, Containment pressure, and Steam generator level

(c) Turn the Actuation Trip path Reset Key and at the same time push the MSIS pushbutton at the PPS cabinets. (1.0)

Reference: 41AO-1ZZ31, Inadvertent MSIS, pages 4, 5, & 6.

7.02 (3.0)

During a normal plant heat-up a bubble is being established in the pressurizer per procedure 41OP-1ZZ01, Cold Shutdown to Hot Standby. Heaters are on and the pressurizer will be vented to the Reactor Coolant Drain Tank (RDT) to vent the nitrogen bubble and to maintain pressure control.

a. One of the precautions for the procedure states that while a nitrogen bubble exists in the pressurizer the pressurizer temperature (not Pressure) should be used to determine the correct level instrument compensation curve (indicated vs actual level). Briefly state why this precaution is necessary. (1.0)

b. What initial RDT level and pressure indications do you expect to see during the venting ? (1.0)

c. What RDT level and pressure indications do you expect to see after all of the nitrogen gas has been vented from the pressurizer ? (1.0)

7.02 Answer

(a) The compensation is necessary due to the density change in water (and or steam) in the pressurizer which changes with temperature. and/or Density of water in pressurizer is controlled by pressure when a steam bubble is present, nitrogen pressure has a very small effect on density. (1.0)

(b) Initially pressure will rise and level will stay the same (constant temperature). (1.0)

(c) After nitrogen has been vented, pressure will rise slowly,and stablize, (and level will begin to increase, temperature will increase as seen on B03,ERFDADS).

Reference: 41OP-1ZZ01, Cold Shutdown to Hot Standby, pages 10 & 63
NS-9G-11, Pressurizer Pressure Control System

7.03 (2.0)

In the left-hand column below is a list (itemized numerically) of "symptoms" from the procedure "Shutdown Outside Control Room". The right-hand column is a list (itemized alphabetically) of "causes". Choose one (1) item from the cause list for each item in the symptom list. Use each item from the cause list for one (1) item from the symptom list. (0.4 each)

SYMPTOM	CAUSE
1. S/G level less than 0 % WR Tcold increasing above Tsat of S/G (Thot follows)	A. Condensible voids collecting in RCS flowpath
2. Tcold and S/G pressure increasing	B. Inadequate RCS inventory
3. RCS loop delta T increasing and abnormal PLCS response	C. Inadequate secondary steam flow
4. Thot increasing and low pressurizer level	D. Non-condensible voids collecting in RCS flowpath or physical blockage of RCS flowpaths
5. RCS loop delta T increasing and indication of no delta Press. across the RCP's.	E. Inadequate secondary water inventory

7.03 Answer

(0.4 each)

1. E
2. C
3. A
4. B
5. D

Reference: 41AO-1ZZ27, Shutdown Outside Control Room, page 58

7.04 (1.0)

According to 41AO-1ZZ13, Natural Circulation Cooldown, which of the following is not a proper indication of natual circulation ?

a. RCS cold leg temperature at saturation for S/G pressure and is controlable.

b. Core exit thermocouples indication are stable and trending down.

c. QSPDS indicates the RCS is becomming more subcooled.

d. Core delta temperature (Thot - Tcold) is more than 57 degrees F and is increasing.

7.04 Answer

D. (1.0)

Reference: 41AO-1ZZ13, Natural Circulation Cooldown, page 4

7.05 (1.0)

During natural circulation with the potential for void formation in the RCS. The Natural Circulation Cooldown procedure 41AO-1ZZ13 cautions against the use of the pressurizer level control system in the automatic mode. Why would operation of the pressurizer level control system in "automatic" in this situation be a problem ?

7.05 Answer (1.0)

If a void formed in the RCS, letdown will automatically increase in response to pressurizer level therby depleting RCS inventory.

Reference: 41AO-1ZZ13, Natural Circulation Cooldown, page 6

7.06 (3.0)

a. List the administrative control limits for whole body exposure. (1.0)

Weekly _____ mrem

Quarterly _____ rem

Yearly _____ rem

b. Which two individuals, by title, must approve a request for exceeding the normal quarterly administrative control limits ? (1.0)

c. What is the quarterly exposure limit for individuals who do not have their complete lifetime exposure history and who have received undocumented radiation exposure in the present quarter? (1.0)

_____ mrem

7.06 Answer

(a) (0.33 each) (1.0)

Weekly 300 mrem
Quarterly 1.0 rem
Yearly 4.0 rem

(b) Radiation Protection Supervisor and the Radiological Services Manager (1.0)

(c) 900 mrem (1.0)

Reference: 75AC-9ZZ01, Radiation Exposure and Access Control, page 25

7.07 (3.0)

A normal reactor start-up is being conducted in accordance with 41OP-1ZZ03, Reactor Start-up.

a. Verification of proper overlap between startup and log safety NI channels is required prior to turning off the high voltage to the startup channels. Over what range of each instrument should overlap occur ? (CPS and % power) (1.0)

b. When is the high voltage to the startup channels turned off ? (0.5)

c. Why is the high voltage turned off ? (0.5)

d. Below the point of adding heat what indication do you expect when the reactor becomes critical ? (1.0)

e. If the reactor is critical how can you tell you are at the point of adding heat ? (1.0)

7.07 Answer

(a) 20 to 2000 CPS = 2x10-8 to 2x10-6 % power (1.0)

(b) When the high CPS alarm is recieved (Approx. 2000 CPS) (0.5)

(c) To prolong the life of the startup detectors (0.5)

(d) Criticality is indicated when a control rod withdrawl results in a sustained positive startup rate which maintains after rod motion has stopped. (1.0)

(e) The reactor is adding heat when the effects of moderator feedback become evident and/or when reactivity addition results in a positive startup rate which slowly tapers off. (about 10-3 % power) (1.0)

Reference: 41OP-1ZZ03, Reactor Start-up, pages 5, 8, 13, and 15

7.08 (2.0)

Indicate whether each of the following statements are TRUE or FALSE as they pertain to the Reactor Start-up procedure.

a. If a required boron concentration change is greater than 50 ppm then pressurizer spray should be initiated to equalize pressurizer and RCS boron concentrations within 20 ppm. (0.5)

b. After criticality is achieved, reactor power is increased at a rate of 1 DPM to 10-4 % power to log data. (0.5)

c. If group 5 has been withdrawn to 135 inches and the reactor is not critical, insert group 5 a sufficient amount (about 500 pcm) to ensure the reactor remains subcritical, then dilute the RCS boron concentration to make up for rod insertion prior to continuing the startup. (0.5)

d. If the reactor is not critical upon reaching ECC + 500 pcm the regulating rod should be inserted to ECC - 500 and Reacto Engineering should be contacted. (0.5)

7.08 Answer

(0.5 each)

(a) False

(b) True

(c) True Procedure states 100 or 500 PCM

(d) False

Reference: 410P-1ZZ03, Reactor Startup, page 10, 13, 14, and 15

7.09 (4.0)

While the plant is operating at steady state power a minor steam generator tube leak occurs.

(a) List four of the six symptoms of the steam generator tube leak AO procedure. (1.0)

(b) Briefly, how you can determine the size of the leak ? (1.0)

(c) If it is not obvious how would you determine which steam generator is leaking. (1.0)

(d) What are two of the three reasons to keep the steam generator 100 psi below RCS pressure ? (1.0)

7.09 Answer

(a) (0.25 each for any four of the following) (1.0)
Main steam line high radiation
Condenser off-gas high radiation
S/G blowdown high radiation
Charging/letdown flow mismatch
Decreasing VCT level
Secondary side sample activity high

(b) With the plant stablized: Measure the VCT level change over a known time to calculate the leak rate. (1.0)

(c) The S/G with the highest activity is the leaker. (1.0)

(d) (0.5 each for any two of the following) (1.0)
Prevent boron dilution
Prevent S/G chemicals in the RCS
Ensure adequate RCS subcooling

Reference: 41AO-1ZZ08, Steam Generator Tube Leak, pages 4, 5. 7, and 9

7.10 (1.0)

Following a LOCA plant conditions appear under control, SI flow has been stopped, and SIAS has been reset.

What two conditions would require re-initiation of full SI flow ?

7.10 Answer

a. Subcooling is lost (0.5)
b. Pressurizer level is lost (0.5)

Reference: 41EP-1ZZ01, Emergency Operations, page 88

7.11 (1.0)

Which one of the following is not one of the four criteria that must be met before safety injection flow may be throttled ?

a. Reactor vessel level indicates void restricted to upper head

b. RCS subcooled more than 28 F

c. Core delta T is less than 10 F

d. Pressurizer level greater than 33 % and controllable.

7.11 Answer

c (1.0)

Reference: 41RO-1ZZ01, Reactor Trip, page 27

7.12 (1.0)

Match the following (numerical) emergency classifications with the appropriate (alphabetical) description. (An answer consisting of the numbers "1" through "4" and the correct associated letters is adequate.)

1. Unusual Event

2. Alert

3. Site Area Emergency

4. General Emergency

(a) Consists of events which are in progress or have occured which involve actual or likely major failures of plant functions needed for the protection of the public. Any releases are not expected to exceed Environmental Protection Agency Guideline exposure levels beyond the site boundary.

(b) This classification applies to events which are in progress or have occured which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

(c) Consists of events which are in progress or have occured which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed Environmental Protection Agency Protective Action Guideline exposure levels offsite for more than the immediate site area.

(d) This classification consists of events which are in progress or have occured which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the Environmental Protection Agency Protective Action Guideline exposure levels.

7.12 Answer

(0.25 each)

1. B
2. D
3. A
4. C

Reference: EP1P-02, Emergency Classification, page 5

END OF SECTION 7
CONTINUE TO SECTION 8

SECTION 8
ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS

8.01 (2.0)

In accordance with 10 CFR 20, "Standards for Protection Against Radiation":

a. What is a RADIATION AREA ? (1.0)

b. What is a HIGH RADIATION AREA ? (1.0)

8.01 Answer (2.0)

a. An area (accessible to personnel) where a major poriton of the body could receive (greater than):

5 Mrem in one hour (0.5)
or
100 Mrem in five (consecutive) days. (0.5)

b. An area (accessible to personnel) where a major portion of the body could receive (greater than):

100 Mrem in one hour. (1.0)

Reference:
10 CFR 20.203(b)(2) and (b)(3)

8.02 (3.0)

In accordance with 10 CFR 55, "Operators' Licenses":

a. As defined in 10 CFR 55, when is an individual deemed to be operating the controls of a nuclear facility ? (1.0)

b. What are the "controls" as defined in 10 CFR 55 ? (1.0)

c. According to the "Exemptions from License" provisions of 10 CFR 55 which individuals are allowed to operate the the reactor controls without a license ? (1.0)

8.02 Answer (3.0)

a. An individual is deemed to operate the controls of a nuclear facility if he directly manipulates the controls or directs another to manipulate the controls. (1.0)

b. "Controls" means apparatus and mechanisms the manipulation of which directly affect the reactivity or power level of the reactor. (1.0)

c. An individual may manipulate the controls as a part of his training to qualify for an operator license under the direction and in the presence of a licensed operator or senior operator. (1.0)

Reference:
10 CFR 55.4(d) & (f) and 55.9(b)

8.03 (3.0)

Refer to 10 CFR 50.72 and or 10 CFR 20.403 (ATTACHMENT). Which of the following events require immediate notification (within a period of one hour or sooner) to the NRC Operations Center via the Emergency Notification System ? (0.5 each)

a. Personnel exposure to an individuals hands of 200 Rems of radiation while performing steam generator tube repairs.

b. Personnel exposure to an individual's whole body of 30 Rem of radiation while working in a steam generator.

c. Declaration of an "Unusual Event" at PVNGS.

d. Taking the plant from mode 1 to mode 3 to comply with Technical Specification requirements.

e. Taking the plant from mode 3 to mode 5 to comply with Technical Specification requirements.

f. Discovery in mode 1 that two safety injection level instruments were improperly calibrated and that the levels have been above or below Technical Specification limits for two weeks.

8.03 Answer (3.0)

* B, C, D, and F require notification within one hour. (0.5 each)

** A and E do not require notification within one hour. (0.5 each)

Reference:
10 CFR 20.403(a) and 50.72(a)(1)i & 50.72(b)

[34 FR 7500, M 69, as amended at 38 FR 1271, Jan. 11 48 FR 33859, July 26, 1983]

§ 20.403 Notifications of incidents.

(a) *Immediate notification.* Each licensee shall immediately report any events involving byproduct, source, or special nuclear material possessed by the licensee that may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more or radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in Appendix B, Table II of this part; or

(3) A loss of one working week or more of the operation of any facilities affected; or

(4) Damage to property in excess of $200,000.

(b) *Twenty-four hour notification.* Each licensee shall within 24 hours of discovery of the event, report any event involving licensed material possessed by the licensee that may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 5 rems or more of radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms to 75 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in Appendix B, Table II of this part; or

(3) A loss of one day or more of the operation of any facilities affected; or

(4) Damage to property in excess of $2,000.

(c) Any report filed with the Commission pursuant to this section shall be prepared so that names of individuals who have received exposure to radiation will be stated in a separate part of the report.

(d) Reports made by licensees in response to the requirements of this section must be made as follows:

(1) Licensees that have an installed Emergency Notification System shall make the reports required by paragraphs (a) and (b) of this section to the NRC Operations Center in accordance with § 50.72 of this chapter.

(2) All other licensees shall make the reports required by paragraphs (a) and (b) of this section by telephone and by telegram, mailgram, or facsimile to the Administrator of the appropriate NRC Regional Office listed in Appendix D of this part.

§ 50.72 Immediate notification requirements for operating nuclear power reactors.

(a) *General requirements.*[1] (1) Each nuclear power reactor licensee licensed

[1] Other requirements for immediate notification of the NRC by licensed operating nuclear power reactors are contained elsewhere in this chapter, in particular, 20.205, 20.403, 50.36, and 73.71.

under § 50.21(b) or § 50.22 of this part shall notify the NRC Operations Center via the Emergency Notification System of:

(i) The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan;[2] or

(ii) Of those non-Emergency events specified in paragraph (b) of this section.

(2) If the Emergency Notification System is inoperative, the licensee shall make the required notifications via commercial telephone service, other dedicated telephone system, or any other method which will ensure that a report is made as soon as practical to the NRC Operations Center.[3]

(3) The licensee shall notify the NRC immediately after notification of the appropriate State or local agencies and not later than one hour after the time the licensee declares one of the Emergency Classes.

(4) When making a report under paragraph (a)(3) of this section, the licensee shall identify:

(i) The Emergency Class declared; or

(ii) Either paragraph (b)(1), "One-Hour Report," or paragraph (b)(2), "Four-Hour Report," as the paragraph of this section requiring notification of the Non-Emergency Event.

(b) *Non-emergency events—(1) One-hour reports.* If not reported as a declaration of an Emergency Class under paragraph (a) of this section, the licensee shall notify the NRC as soon as practical and in all cases within one hour of the occurrence of any of the following:

(i)(A) The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

(B) Any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part.[4]

(ii) Any event or condition during operation that results in the condition of the nuclear powerplant, including its principal safety barriers, being seriously degraded; or results in the nuclear power plant being:

(A) In an unanalyzed condition that significantly compromises plant safety;

(B) In a condition that is outside the design basis of the plant; or

(C) In a condition not covered by the plant's operating and emergency procedures.

(iii) Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.

(iv) Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal.

(v) Any event that results in a major loss of emergency assessment capability, offsite response capability, or communications capability (e.g., significant portion of control room indication, Emergency Notification System, or offsite notification system).

(vi) Any event that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

(2) *Four-hour reports.* If not reported under paragraphs (a) or (b)(1) of this section, the licensee shall notify the NRC as soon as practical and in all cases, within four hours of the occurrence of any of the following:

(i) Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety.

(ii) Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that results from and is part of the preplanned sequence during testing or

[2] These Emergency Classes are addressed in Appendix E of this part.

[3] Commercial telephone number of the NRC Operations Center is (202) 951-0550.

8.04 (1.0)

10 CFR 50.54(x) allows a licensee to take reasonable action that departs from a license condition or Technical Specification in an emergency when this action is immediatly needed to protect the public health and safety and no action consistent with license conditions and Technical Specifications that can provide adequate or equavilent protection is immediatly apparent.

Licensee action permitted by this paragraph shall be approved, as a minimum, by ________ prior to taking the action.

Which response best completes the above statement ?

a. the senior licensed operator present (RO)

b. the Plant Superintendent

c. any licensed control operator (RO)

d. any licensed Senior Reactor Operator (SRO)

8.04 Answer (1.0)

d

Reference:
10 CFR 50.54(x) and (y)

8.05 (2.0)

Answer each of the following TRUE or FALSE:

a. UNIDENTIFIED LEAKAGE shall be all leakage which does not constitute either IDENTIFIED LEAKAGE or reactor coolant pump controlled bleed-off flow. (0.5)

b. The local DNB ratio (DNBR), is defined as the ratio of the actual heat flux at a particular core location to that of the predicted DNB heat flux at same location. (0.5)

c. The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21 kw/ft. (0.5)

d. The Steam Generator Low Level trip in the RPS is is designed to protect the Reactor Coolant System from overpressure. (0.5)

8.05 Answer (2.0)

a. True

b. False

c. True

d. True

Reference: Tech. Spec's 1.0.2.1.1, 2.1.1.2, and 2.2.1

8.06 (1.0)

In accordance with PVNGS station tagging and clearance requirements, which one of the following statements is incorrect ?

a. More than one (red) danger tag may be attached to one device at the same time.

b. More than one (blue) men at work tag may be attached to one device at the same time.

c. A (yellow) caution tag and a (red) danger tag may be attached to one device at the same time.

d. A (blue) men at work tag may not hang on one device at the same time with a (red) danger tag.

8.06 Answer (1.0)

b, is the only incorrect statement (1.0)

Reference:
PVNGS Procedure 40AC-9ZZ15, Station Tagging and Clearance, page 19

8.07 (1.0)

a. What is the maximum time a RCP can be operated after a loss of nuclear cooling water, with seal injection flow established? (0.5)

b. What period of time may a RCP be operated without seal injection flow, assuming nuclear cooling water is still supplied? (0.5)

8.07 Answer (1.0)

a. Ten (10) minutes (0.5)

b. An indefinite period of time (0.5)

Reference: 41OP-1CH03 RCP Seal Injection System.

8.08 (3.0)

The PVNGS Technical Specification require that shutdown margin be greater than 6 % while in modes 1 through 4 and greater than 4 % while in mode 5.

a. What is the definition of shutdown margin ? (1.0)

b. What are the most restrictive accident conditions (ie. Technical Specification Bases) which require the 6 % SDM limit ? (1.0)

c. What are the most restrictive accident conditions (ie. Technical Specification Bases) which require the 4 % SDM limit ? (1.0)

8.08 Answer (3.0)

a. SDM is the amount of reactivity by which the reactor is, or would be, subcritical from its present condition assuming no change in part-length CEA position and all other CEAs are fully inserted except the single assembly of highest worth which is fully withdrawn. (1.0)

b. The 6 % SDM limit is based on controling the reactivity transient associated with an uncontrolled RCS cooldown caused by a steam line break at end of life and T(c) at no load operating temperature. (1.0)

c. The 4 % SDM limit is based on ensuring that reactivity transients resulting from a single CEA withdrawl event are minimal. (1.0)

Reference:
PVNGS Technical Specification, Definition 1.28, and Bases 3/4.1.1

8.09 (2.0)

Based on the PVNGS Technical Specifications, with regard to Minimum Temperature for Criticality, with the plant in mode 1:

a. What is the minimum operating loop temperature [i.e. LCO for T(c)] ? (1.0)

b. In the event T(c) decreased below its LCO what actions required by the Technical Specification Action statement(s) must be taken (including time limits) within one hour ? (1.0)

8.09 Answer (2.0)

a. 552 degrees F (1.0)

b. Restore T(c) to meet the LCO in 15 minutes (0.5)
or be in hot shutdown in the following 15 minutes. (0.5)

Reference:
PVNGS Technical Specifications, 3.1.1.4

8.10 (2.0)

Complete the minimum shift crew composition for the modes listed below. (Do not consider health physics, chemistry or I&C personnel.)

PERSONNEL	MODE 4 (1.0)	MODE 5 (1.0)
SROs	------	------
ROs	------	------
Others(AO's, STA's)	------	------

8.10 Answer (2.0)

PERSONNEL	MODE 4 (1.0)	MODE 5 (1.0)
SROs	2	1
ROs	2	1
Others(specify positions)	2(AO), 1(STA)	1(AO)

Reference:
PVNGS Technical Specifications, Table 6.2-1

8.11 (2.0)

During the absence of the Shift Supervisor from the control room for <u>modes</u> <u>1</u> through <u>6</u> , who may be designated to assume the control room command function ?

8.11 Answer (2.0)

* When in modes 1 through 4 an individual, other than the STA, with a SRO license may assume the Control Room command function. (1.0)

** While in modes 5 and 6 an individual with a RO or SRO license may assume the Control Room command function. (1.0)

Reference:
PVNGS Technical Specifications Table 6.2-1

8.12 (3.0)

The plant has been operating at 90 % power for about three weeks. The Non-Essential auxiliary feed pump (AFN-P01) was placed out of service the previous day due to a bad bearing. While testing the (G001) Diesel Generator, the Diesel Governor fails to control frequency and the Diesel must be shutdown for repairs. The associated Technical Specifications are attached for your reference and the time of various events are listed below.

1100 14 November	auxiliary feedpump placed out of service
1000 15 November	diesel generator placed out of service
1015 15 November	current time

a. What surveillances or verifications must be completed within the next two hours ? (1.0)

b. When must the plant be placed in hot standby if neither the diesel nor the auxiliary feed pump can be returned to service ? (1.0)

c. If the steam dr.ven auxiliary feedpump were found inoperable at 1130 15 November when would the plant have to be in hot standby ? (1.0)

8.12 Answer (3.0)

(a) Each independent offsite power supply must be checked operable. (.25)
The remaining diesel generator must be checked operable. (.25)
All required systems, subsystems, trains, components, and devices that depend on the remaining diesel generator for power must be checked for operability. (.25)
The steam driven auxiliary feed pump must be operable. (.25)

(b) 1700 17 November (1.0)

(c) 1730 15 November (1.0)

Reference:
PVNGS Technical Specifications, 3.7.1.2 and 3.8.1.1

END OF THE EXAM

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Testing the turbine-driven pump and both motor-driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine-driven pump for entry into MODE 3.

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that all manual valves in the suction lines from the primary AFW supply tank (condensate storage tank CTE-T01) to each essential AFW pump, and the manual discharge line valve of each AFW pump are locked, sealed or otherwise secured in the open position.

*Until the steam generators are no longer required for heat removal.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

c. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, by verifying on a STAGGERED TEST BASIS (by means of a flow test) that the normal flow path from the condensate storage tank to each of the steam generators through one of the essential auxiliary feedwater pumps delivers at least 750 gpm at 1270 psia or equivalent.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4 for the turbine-driven pump.

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits from the offsite transmission network to the switchyard and two physically independent circuits from the switchyard to the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. Separate day fuel tank with a minimum level of 2.75 feet (550 gallons of fuel), and

   2. A separate fuel storage system with a minimum level of 80% (71,500 gallons of fuel), and

   3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4 within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, 3, or 4*, the steam-driven auxiliary feed pump is OPERABLE.

   If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Until the steam generator is no longer required for heat removal.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availablity, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank,

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

FOR INFORMATION ONLY

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2. By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

d. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A.

Joe Upton

U.S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: Palo Verde 1 and 2 (50-528 & 529)

Reactor Type: PWR-CE

Date Administered: March 11, 1986

Examiner: Joe Upton, John Smith

Candidate: Answer Key

INSTRUCTIONS TO CANDIDATE:

Print your name on the line above marked "Candidate." The grade points available for each question are indicated within parentheses after each question. The passing grade is 70% in each of the four (4) categories and is 80% for the total grade. Use separate paper for your answers and write on only one (1) side of the paper, unless a specific question instructs you otherwise. Staple this question package to your answer sheets. The examination questions and answers will be picked up six (6) hours after the examination was started. Read the statement at the bottom of this page. When you have finished this examination, affirm the statement by signing your name.

Category Value	% of Total	Candidate's Score	% of Cat. Value	Category
25	25	______	______	1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
25	25	______	______	2. Plant Design Including Safety and Emergency Systems
25	25	______	______	3. Instruments and Controls
25	25	______	______	4. Procedures - Normal, Abnormal, Emergency and Radiological Control
100		______		TOTALS

Final Grade ______ %

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

____
Candidate's Signature

FURTHER INSTRUCTIONS TO CANDIDATE

1. At the end of the written examination package is a copy of a part of the Core Data Book for Unit 1 in Cycle 1. Use the tables and curves as appropriate.

2. At the end of the written examination package is a reference page containing equations, formulas, and constants. Use them as necessary.

3. Use the "Steam Tables" as necessary.

Points
Available

1.0 PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW (25.0)

QUESTIONS 1.01 through 1.06 are "multiple-choice" questions. For each question specify the letter designation for the phrase that provides the most correct statement.

QUESTION 1.01

A tank contains water to a level of 40 ft above the bottom of the tank. A nitrogen cover gas is at 100 psia. The tank and its contents are at 70°F and the density of the water is 62.4 lbm/ft^3. The pressure at the bottom of the tank is (1.0)

(a.) 117 psia.

(b.) 132 psia.

(c.) 208 psia.

(d.) 308 psia.

ANSWER 1.01

(a.) [+1.0]

Reference(s) 1.01

1. Palo Verde 1 and 2: Thermohydraulics Review for Licensed Operators, Section 2, pp. 6-9.

2. Generic: Academic Program for Nuclear Power Plant Personnel, Volume III, "Nuclear Power Plant Technology", 1973, General Physics Corporation, pp. 2-115 - 2-117.

3. St. Lucie 1 and 2: Thermodynamics and Heat Transfer - Module 4, "Thermodynamics", pp. 3-11.

Points
Available

QUESTION 1.02

If the temperature of the tank in QUESTION 1.01 (container, water and cover gas) were raised and if no water or cover gas was allowed to enter or leave the tank, the pressure at the bottom of the tank would (1.0)

(a.) increase because the gas volume has decreased and the temperature of the gas has increased.

(b.) increase because the pressure due to the water would rise and the temperature of the gas has increased.

(c.) decrease because the water density has decreased.

(d.) decrease because the cover-gas density has decreased.

ANSWER 1.02

(a.) [+1.0]

Reference(s) 1.02

1. Generic: Academic Program for Nuclear Power Plant Personnel, Volume III, "Nuclear Power Plant Technology", 1973, General Physics Corporation, pp. 2-6 - 2-10, 2.115 - 2.117.

2. St. Lucie 1 and 2: Thermodynamics and Heat Transfer - Module 4, "Thermodynamics", pp. 3-11, 38-40.

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.03

Delayed neutrons are emitted (1.0)

(a.) through the radioactive-decay chain of various fission-fragment nuclei and comprise less than 1% of the total number of neutrons released in the fission process.

(b.) through the radioactive-decay chain of various fission-fragment nuclei and appear, on the average, with more energy than the average energy associated with the neutrons produced immediately in the fission process.

(c.) by delayed-neutron precursor nuclei and appear, on the average, with less energy than the average energy associated with the neutrons produced immediately in the fission process.

(d.) through the radioactive-decay chain of various fission-fragment nuclei and comprise between 6 to 7% of the total number of neutrons released in the fission process.

ANSWER 1.03

(a.) [+1.0]

Reference(s) 1.03

1. Palo Verde 1 and 2: Palo Verde Licensed Operator Training, Reactor Theory, Pre-License Review Course NLC55, pp. 4.6, 5.2, and Figure 4.12.

2. St. Lucie 1 and 2: Reactor Physics Training Manual, "Reactor Physics", pp. 7.2-6 - 7.2-8.

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.04

Upon shutting down a reactor from full power by the instantaneous insertion of negative reactivity, it is impossible to obtain an arbitrarily small magnitude for the stable reactor period (the decay period several minutes after the insertion of the negative reactivity) by inserting a larger amount of negative reactivity because (1.0)

(a.) the decay rate associated with the reduction of the temperature of the fuel determines the long-term decay rate for the neutron flux.

(b.) most of the neutrons are "prompt" and hence the prompt neutron diffusion time determines the long-term decay rate for the neutron flux.

(c.) the delayed neutron emitters with the shortest half-life determine the long-term decay rate for the neutron flux.

(d.) the delayed neutron emitters with the longest half-life determine the long-term decay rate for the neutron flux.

ANSWER 1.04

(d.) [+1.0]

Reference(s) 1.04

1. Palo Verde 1 and 2: Palo Verde Licensed Operator Training, Reactor Theory, Pre-License Review Course NLC55, p. 11.5 and Figure 11.5.

2. GA: Syllabus & Triga Training Manual, "Reactor Kinetics", GA Technologies Inc., p. 6-1.

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.05

The nuclear reactors at Palo Verde are called "thermal reactors" because (1.0)

(a.) the reactors provide the thermal-energy input for the plant.

(b.) the thermal power produced by the reactors equals the power removed by the secondary-side of the plant (when in the normal full-power operation).

(c.) on the average, the neutrons produced by the fissioning process are at an energy level that corresponds to the temperature of the surrounding material.

(d.) on the average, the neutrons causing fission are at an energy level that corresponds to the temperature of the surrounding materials.

ANSWER 1.05

(d.) [+1.0]

Reference(s) 1.05

1. Palo Verde 1 and 2: Palo Verde Licensed Operator Training, Reactor Theory, Pre-License Review Course NLC55, pp. 3.3, 5.2, and 6.3.

2. St. Lucie 1 and 2: Reactor Physics Training Manual, "Reactor Physics", p. 7.1-2.

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.06

The isotope of plutonium, Pu^{239}, is found in the nuclear reactor core of Palo Verde Unit 1 because (1.0)

(a.) Pu^{239} is found in significant quantities (percentage wise) in pitchblend (the uranium material that is mined).

(b.) of the non-fission absorption of a thermal or epithermal neutron by U^{238} nuclei.

(c.) of the fissioning of U^{238} nuclei by fast neutrons.

(d.) of the non-fission absorption of a thermal neutron by U^{235} nuclei.

ANSWER 1.06

(b.) [+1.0]

Reference(s) 1.06

1. Palo Verde 1 and 2: Palo Verde Licensed Operator Training, Reactor Theory, Pre-License Review Course NLC55, pp. 4.3 - 4.6.

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.07

In the left-hand column below is a list of elements that are significant to the operation of nuclear-reactor power plants. In the right-hand column below is a list of element symbols (not all of which are correct element symbols). For each element in the left-hand column, choose its symbol from those in the right-hand column. (1.5)

Element	Symbol
1. Zirconium	Pl
2. Antimony	Fe
3. Beryllium	Pb
4. Sodium	N
5. Plutonium	O
6. Boron	Ni
7. Nitrogen	Pu
8. Bromine	Sm
9. Carbon	Na
10. Chlorine	U
11. Helium	Zr
12. Samarium	Xe
13. Iron	Sb
14. Lead	Be
15. Cobalt	C
	B
	Ar
	Br
	Cl
	He
	Cu
	H
	Co
	Nt

-Section 1.0 Continued on Next Page-

Points
Available

ANSWER 1.07

1. Zr
2. Sb
3. Be
4. Na
5. Pu
6. B
7. N
8. Br [+0.1 each]
9. C
10. Cl
11. He
12. H
13. Fe
14. Pb
15. Co

Reference(s) 1.07

1. Palo Verde 1 and 2: Palo Verde Licensed Operator Training, Reactor Theory, Pre-License Review Course NLC55, p. 1.2.

-Section 1.0 Continued on Next Page-

Points Available

QUESTION 1.08

The turbine of the Auxiliary Feedwater System (AFS) is rated at 1250 hp with steam supplied at a pressure of 1170 psig.

a. When the AFS turbine is being supplied with 1000 psia main steam, how much work can be supplied by the turbine? Assume thermodynamically ideal operation of the turbine. Express the answer in Btu/lbm. (2.0)

b. If the turbine was receiving 250 Btu/lbm and was driving the AFS pump with a power of 950 hp, what is the steam flowrate through the turbine? (1.0)

ANSWER 1.08

a. h_1 = 1193 Btu/lbm [+0.5]

isentropic expansion [+0.5]

h_2 = 905 Btu/lbm [+0.5]

Δh = 288 Btu/lbm [+0.5 for 280 - 296 Btu/lbm]

b. $950 \text{ hp} \times 2545 \frac{\text{Btu/hr}}{\text{hp}} = 2.42 \text{ M Btu/hr}$ [+0.5 for conversion]

$\frac{2.42 \text{ M Btu/hr}}{250 \text{ Btu/lbm}} = 9{,}680 \text{ lbm/hr}$ [+0.5 for 9,180 - 10,180 lbm/hr]

Reference(s) 1.08

1. Palo Verde 1 and 2: System Descriptions, "Auxiliary Feedwater System", Training Article PGS-11, pp. PSG-11-6, 12, 13.

2. Palo Verde 1 and 2: Thermohydraulics Review for Licensed Operators, Section 2, pp. 9-16.

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.09

What effect (INCREASE, DECREASE, REMAIN-THE-SAME) would each of the actions below have on the available Net Positive Suction Head (NPSH) to a centrifugal pump?

a. Raising the pump elevation to be closer to the surge tank that feeds it (Neglect friction losses in the pipe.) (0.5)

b. Decreasing the inlet pipe diameter (0.5)

c. Cooling the fluid upstream of the pump (0.5)

ANSWER 1.09

a. DECREASE [+0.5]

b. DECREASE [+0.5]

c. INCREASE [+0.5]

Reference(s) 1.09

1. Palo Verde 1 and 2: Thermohydraulics Review for Licensed Operators, Section 2, pp. 21, 28, 29.

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.10

Assume that Figure 1.10 (QUESTION) is a graphical illustration of the neutron population during one (1) neutron generation in nuclear-reactor core at Palo Verde Unit 1; not illustrated are the neutrons that "leak-out" of the core. Assume that the core is at MOL (middle-of-life). Complete the following parts of this QUESTION by "filling-in the blanks."

a. If the fraction of neutrons (include neutrons of all speeds) that "leak-out" in a neutron generation was 0.2, the core would be ______________________ (SUBCRITICAL, CRITICAL, SUPERCRITICAL). (0.5)

b. If CEAs were inserted further into this core, the primary affect on reactivity would be to ___________________ (INCREASE or DECREASE) the __________ (ϵ, p, f, or η) term. (1.0)

c. If the temperature of the core were increased, the "p" term would _____________ (INCREASE or DECREASE), the "f" term would __________ (INCREASE or DECREASE) and the total leakage of neutrons (fraction of neutrons per generation) would __________ (INCREASE or DECREASE). (1.5)

ANSWER 1.10

a. SUBCRITICAL [+0.5]

b. DECREASE [+0.5]
f [+0.5]

c. DECREASE [+0.5]
INCREASE [+0.5]
INCREASE [+0.5]

Reference(s) 1.10

1. Palo Verde 1 and 2: Palo Verde Licensed Operator Training, Reactor Theory, Pre-License Review Course NLC55, p. 6.2 - 7.2.

-Section 1.0 Continued on Next Page-

Points
Available

NEUTRON GENERATION IN A REACTOR OF INFINITE SIZE.

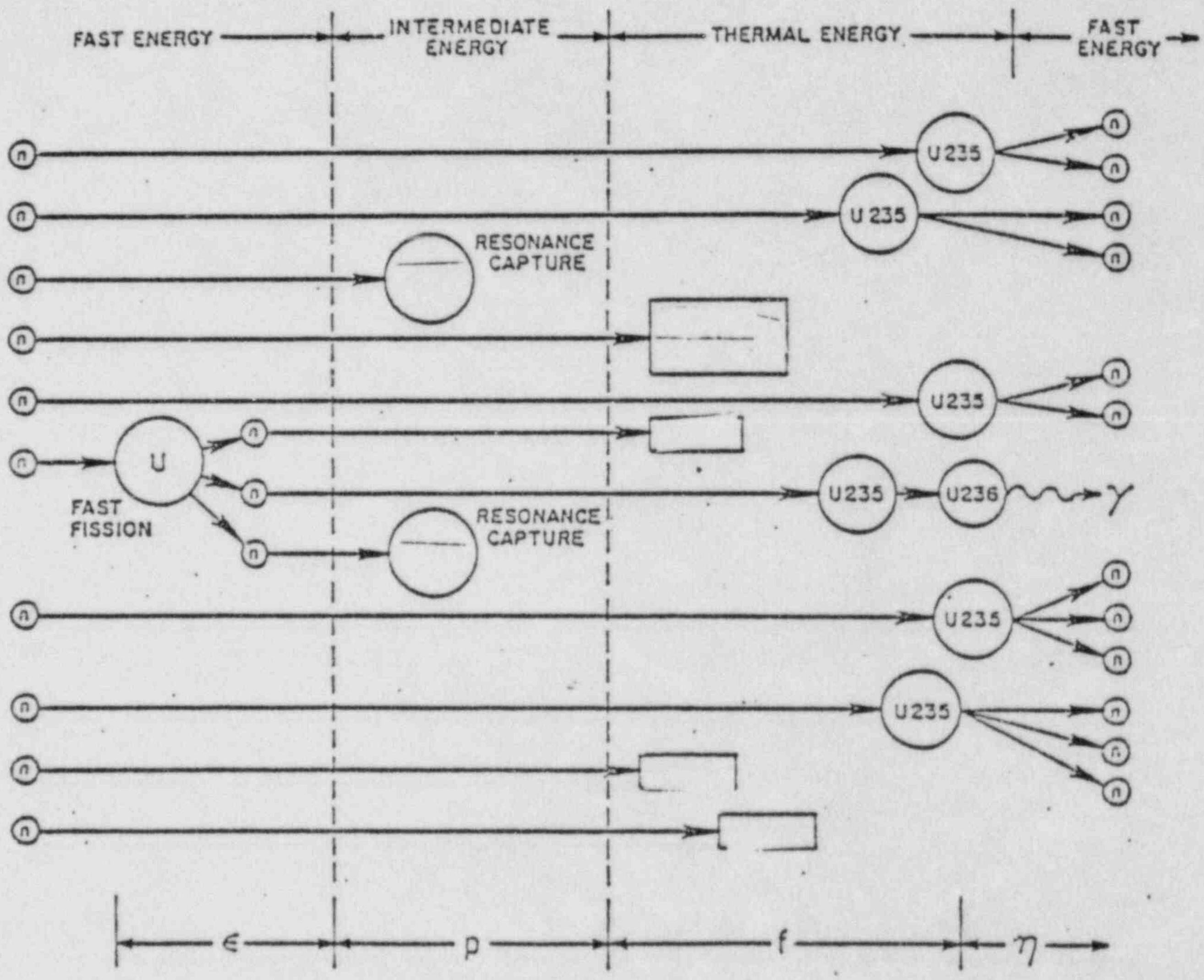


Figure 1.10 (QUESTION)

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.11

Answer the following parts of this QUESTION concerning the operation of the Palo Verde Unit 1 power plant with 208 EFPD.

a. If the plant had been operating at 100% of full power for 5 days, the magnitude of the xenon worth would be __________ pcm. (0.5)

b. If the plant had been operating at 100% of full power for 10 hours (the plant had been in HOT STANDBY for 15 days before FULL-POWER operation) the estimated magnitude of the xenon worth would be ________ pcm. (0.5)

c. If the plant tripped while in the conditon of part "a.", the magnitude of the peak xenon worth would be ________ pcm. (0.5)

d. If the plant tripped while in the condition of part "b.", the estimated magnitude of the peak xenon worth would be _____________ pcm. (0.5)

e. When, in part "c.", the power level instantaneously dropped; the "burnup" of Xe^{135} ____________________ (INCREASED, STAYED-THE-SAME or DECREASED) while the production of the Te^{135} and I^{135} __________ (INCREASED, STAYED-THE-SAME or DECREASED) and while the rate-of-decay of I^{135} ____________ (INCREASED, STAYED-THE-SAME or DECREASED). (1.5)

ANSWER 1.11

a. Using Curve 2.4.1, Table 2.4.2 or Curve 2.4.2, 2720 ± 20 pcm [+0.5]

b. 1400 ± 400 pcm [+0.5]

c. Using Table 1.1, 5470 ± 50 pcm [+0.5]

d. 3500 ± 600 pcm [+0.5]

e. DECREASED [+0.5]
DECREASED [+0.5]
STAYED-THE-SAME [+0.5]

-Section 1.0 Continued on Next Page-

Points
Available

Reference(s) 1.11

1. Palo Verde 1 and 2: Palo Verde Licensed Operator Training, Reactor Theory, Pre-License Review Course NLC55, p. 16.2 - 16.10.

2. Palo Verde 1 and 2: Core Data Book, Unit 1, Cycle 1, Revision 001, Table 1.1, Curve 2.4.1, Table 2.4.2, and Curve 2.4.2, pp. 1, 32, 33, 34.

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.12

The Unit 1 power plant is in the final stage of primary system heatup. The power level is 0%.

a. If the reactor was at BOC with 0 EFPD and 1500 ppm boron, specify the magnitude of the change in reactivity (in pcm) as the plant changes temperature from 550°F to 565°F? (0.5)

b. As the plant changes temperature from 550°F to 565°F, specify whether the reactivity of the core has INCREASED or DECREASED and provide an explanation for the direction of change of the reactivity. (2.0)

ANSWER 1.12

a. Using Curve 2.2.1 of the Core Data Book,

$\rho_{final} = 0$ pcm

$\rho_{initial} = -11$ pcm

$\Delta\rho = 11$ pcm. **[+0.5 for 11 ± 1 pcm]**

b. The reactivity has INCREASED **[+0.5]** as the temperature has increased.

This is due to the fact that as the temperature increased from 550 to 565°F, the moderator density decreased **[+0.5]**. With a lower density, there is less boron in the core, which provides an increase in reactivity and less moderating capability, which provides a decrease in reactivity **[+0.5]**. With a concentration of 1500 ppm of boron, the reactivity effect of the loss of boron is greater than that due to the decrease in moderating capability **[+0.5]**.

-Section 1.0 Continued on Next Page-

Points
Available

Reference(s) 1.12

1. Palo Verde 1 and 2: Palo Verde Licensed Operator Training, Reactor Theory, Pre-License Review Course NLC55, pp. 10.5 - 10.6.

2. Palo Verde 1 and 2: Core Data Book, Unit 1, Cycle 1, Revision 001, Curve 2.2.1, p. 8.

QUESTION 1.13

The Unit 1 nuclear reactor at BOC had just been taken critical with the CEAs in the manual-sequential (MS) mode. The boron concentration is 1250 ppm. The CEAs are at 90 inches on Group 4 and T_{avg} is 505°F. Determine the value of the boron concentration to which the primary coolant must be taken to reach ARO (all rods out). (1.5)

ANSWER 1.13

From Curve 2.5.6,

$\Delta\rho$ = 400 pcm. [+0.5, +0.25 for EOC value of 700]

From Table 2.3.4,

boron worth = 11.75 pcm/ppm [+0.5]

Hence,

$$\Delta \text{ boron worth} = \frac{400}{11.75} = 34.04 \text{ ppm}$$

new concentration = 1250 + 34 = 1284 ppm [+0.5]

Reference(s) 1.13

1. Palo Verde 1 and 2: Palo Verde Licensed Operator Training, Reactor Theory, Pre-License Review Course NLC55, p. 10.8 - 10.9.

2. Palo Verde 1 and 2: Core Data Book, Unit 1, Cycle 1, Revision 001, Curve 2.5.6, and Figure 2.3.4, pp. 23, 48.

-Section 1.0 Continued on Next Page-

Points
Available

QUESTION 1.14

Answer the following parts of this QUESTION by calculating the answer or by "filling-in the blanks."

a. The heat transfer in the Steam Generators involves a combination of types of heat transfer. Heat transfer from the bulk of the reactor coolant to the tube surface is by _________, through the tube is by _________, and from the tube surface to the bulk of the feedwater/steam is by _________________. (0.75)

b. Assume the following conditions for a Steam Generator:

- feedwater temperature = 400°F
- feedwater flowrate = 1.0×10^7 lbm/hr
- steam pressure = 1000 psia
- steam flowrate = 1.0×10^7 lbm/hr.

How much thermal energy in MW is being transferred to the secondary fluid in the Steam Generator? (Show your calculations.) (1.75)

ANSWER 1.14

a. convection [+0.25]
conduction [+0.25]
convection [+0.25]

-Section 1.0 Continued on Next Page-

Points
Available

ANSWER 1.14 (continued)

b. $\dot{Q} = \dot{m}\,(h_{steam} - h_{feedwater})$ [+0.5]

$$= 1.0 \times 10^{7}\ \frac{lbm}{hr}\left(1192\ \frac{Btu}{lbm} - 375.1\ \frac{Btu}{lbm}\right)$$

$$= 817 \times 10^{7}\ \frac{Btu}{hr} \qquad [+0.75]$$

$$= \frac{817 \times 10^{7}\ \frac{lbm}{hr}}{3.41 \times 10^{6}\ \frac{Btu}{hr\ MW}}$$

$$= 2396\ MW \qquad [+0.5]$$

Reference(s) 1.14

1. Generic: Academic Program for Nuclear Power Plant Personnel, Volume III, "Nuclear Power Plant Technology", 1973, General Physics Corporation, pp. 2-141 - 2-148.

-End of Section 1.0-

Points
Available

2.0 PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS (25.0)

QUESTIONS 2.01 through 2.05 are "multiple-choice" questions. For each question specify the letter designation for the phrase that provides the most correct statement.

QUESTION 2.01

The Circulating Water (CW) System functions to provide cooling water to the main condenser; it consists of four (4) CW pumps with (1.0)

(a.) each pump powered from a separate 13.8 kV bus, each pump with a motor-operated valve in its output line, and each pump feeding one (1) of four (4) independent tube bundles in each of the three (3) condenser sections.

(b.) each pump powered from a separate 13.8 kV bus, each pump with a motor-operated valve in its output line, and which combine their flows into two (2) flow paths with each path feeding one (1) of two (2) independent tube bundles in each of the three (3) condenser sections.

(c.) each pair of pumps powered from a separate 13.8 kV bus, each pump with a motor-operated valve in its output line, and which combine their flows into two (2) flow paths with each path feeding one (1) of two (2) independent tube bundles in each of the three (3) condenser sections.

(d.) each pair of pumps powered from a separate 13.3 kV bus, which combine their flows into three (3) flow paths with each path containing a motor-operated valve and feeding one (1) of the three (3) condenser sections.

ANSWER 2.01

(c.) [+1.0]

Reference(s) 2.01

1. Palo Verde 1 and 2: System Descriptions, "Circulating Water System", Training Article PGS-7A, pp. PGS-7A-4 - PGS-7A-8.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.02

Each Emergency Diesel Generator can be supplied diesel fuel oil to continuously operate at 100% load for up to (1.0)

(a.) 1 hour with the day tank and 7 days with the storage tank.

(b.) 6 hours with the day tank and 7 days with the storage tank.

(c.) 1 hour with the day tank and 2 days with the storage tank.

(d.) 6 hours with the day tank and 2 days with the storage tank.

ANSWER 2.02

(a.) [+1.0]

Reference(s) 2.02

1. Palo Verde 1 and 2: Diesel Generator and Auxiliaries Handbook, pp. 16-18.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.03

To protect the Steam Generators (S/Gs) from an overpressure condition, the main steam system contains twenty (20) safety valves that (1.0)

(a.) have a total relieving capacity of 100% of full steam flow and have sequential relief settings starting at 1255 psig and ending with 1315 psig.

(b.) must be combined with the atmospheric dump valves to have a combined total relieving capacity of 100% of full steam flow and have sequential relief settings starting at 1255 psig and ending with 1315 psig.

(c.) have a total relieving capacity of 100% of full steam flow and have sequential relief settings starting at 1070 psig and ending with 1130 psig.

(d.) must be combined with the atmospheric dump valves to have a combined total relieving capacity of 100% of full steam flow and have sequential relief settings starting at 1070 psig and ending with 1130 psig.

ANSWER 2.03

(a.) [+1.0]

Reference(s) 2.03

1. Palo Verde 1 and 2: System Description, "Main Steam System", Training Article PGS-1A, pp. PGS-1A-8 - PGS-1A-10.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.04

The four (4) motor-operated valves for the safety injection tanks (SITs) are (1.0)

(a.) closed during normal Mode 1 operation to isolate the SITs from the RCS but receive an open signal on a SIAS.

(b.) open during normal Mode 1 operation but receive an open signal on a SIAS.

(c.) interlocked with the Pressurizer pressure to automatically open if the RCS pressure decreases below 500 psig.

(d.) interlocked with the Pressurizer pressure to ensure that they are closed if the RCS pressure increases above 500 psig.

ANSWER 2.04

(b.) [+1.0]

Reference(s) 2.04

1. Palo Verde 1 and 2: System Descriptions, "Safety Injection and Shutdown Cooling System", Training Article NS-3A, Section 2.1.3, pp. NS-3A-13, NS-3A-14.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.05

The 125 VDC Class IE Power System consists of four 125 VDC control centers, A, B, C, D such that (1.0)

(a.) A is tied to the A battery charger and normally to the A-C backup battery charger and supplies DC loads including a 125 VDC to 480 VAC inverter for the shutdown-cooling isolation valves.

(b.) B is tied to the B battery charger and normally is NOT tied to the B-D backup battery charger and supplies DC loads including a 125 VDC to 120 VAC inverter for Class IE 120 VAC loads.

(c.) C is tied to the C battery charger and normally to the A-C backup batter charger and supplies DC loads including a 125 VDC to 120 VAC inverter for Class IE 120 VAC loads.

(d.) D is tied to the D battery charger and normally is NOT tied to the B-D backup battery charger and supplis DC loads including a 125 VDC to 480 VAC inverter for the shutdown-cooling isolation valves.

ANSWER 2.05

(d.) [+1.0]

Reference(s) 2.05

1. Palo Verde 1 and 2: System Descriptions, "125 VDC Class IE Power System", Training Article PGS-15D, pp. PGS-15D-7 through PGS-15D-10.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.06

Answer TRUE or FALSE.

A reactor coolant pump may be operated without seal injection water for an indefinite period of time as long as nuclear cooling water is still supplied. (0.5)

ANSWER 2.06

TRUE **[+0.5]**

Reference(s) 2.06

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Reactor Coolant Pump Operation", 41OP-1RC01, Section 3.17, p. 8.

QUESTION 2.07

Why must the Emergency Diesel Generator (D/G) lube oil circulating pump be run for a minimum of 30 minutes after shutting down a hot D/G? (1.0)

ANSWER 2.07

to dissipate heat **[+0.5]** from the turbocharger bearings **[+0.5]**

Reference(s) 2.07

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Emergency Diesel Generator", 41OP-1DG01, Section 3.28, p. 10.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.08

List four (4) of the actuating signals that should automatically start the Essential Cooling Water System. (2.0)

ANSWER 2.08

1. Loss of Offsite Power (LOP)
2. Control Room Essential Filtration Actuation Signal (CREFAS)
3. Safety Injection Actuation Signal (SIAS)
4. Control Room Ventilation Isolation Actuation Signal (CRVIAS)
5. Auxiliary Feedwater Actuation Signal 1 (AFAS-1)
6. Auxiliary Feedwater Actuation Signal 2 (AFAS-2)
7. Containment Spray Actuation Signal (CSAS)

[+0.5 each, +2.0 max]

Reference(s) 2.08

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Essential Cooling Water System", 41OP-1EW01, Section 4.1.2, p. 7.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.09

As a result of loss of electrical power, what is the failed position (OPEN, CLOSED, AS-IS) for each of the following actuator-operated valves associated with the Safety Injection System.

a. Low-pressure header isolation valve (0.5)

b. HPSI pump mini-flow line isolation valve (0.5)

c. Sump line isolation valve (0.5)

d. LPSI pump suction isolation valve (0.5)

e. Hot-leg injection isolation valve (0.5)

ANSWER 2.09

a. AS-IS [+0.5]

b. CLOSED [+0.5]

c. OPEN [+0.5]

d. CLOSED [+0.5]

e. AS-IS [+0.5]

Reference(s) 2.09

1. Palo Verde 1 and 2: System Descriptions, "Safety Injection and Shutdown Cooling System", Training Article NS-3A, Table NS-3-I, p. NS-3A-25.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.10

Other than an Overspeed Trip, list six (6) trips that will initiate closure of the main turbine steam admission valves. (3.0)

ANSWER 2.10

a. Main condenser low vacuum

b. Excessive thrust bearing wear

c. Reactor trip

d. Generator/reactor initiated trip

e. Master trip from control room

f. Manual trip using lever located at the turbine

g. Excessive vibration

h. High exhaust hood temperature

i. Moisture separator high-high level

j. Prolonged loss of stator coolant

k. Low hydraulic fluid pressure

l. Loss of EHC 125 VDC control power

[+0.5 each, +3.0 max].

Reference(s) 2.10

1. Palo Verde 1 and 2: System Descriptions, "Main Turbine System", Training Article PGS-3A, Section 4.3, p. PGS-3A-43.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.11

What is the function of the CVCS back pressure control valve? (1.0)

ANSWER 2.11

Maintains upstream pressure high enough to prevent letdown flow from flashing to steam in the letdown line. **[+1.0]**

Reference(s) 2.11

1. Palo Verde 1 and 2: System Descriptions, "CVCS-1", Training Article NS-2A, Section 1.3.1, p. NS-2A-7.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.12

Draw a one-line diagram of the auxiliary feedwater system that contains the following components. Label the components on the diagram. (2.5)

- essential pump
- turbine-driven essential pump
- non-essential pump
- line providing hydrozine addition
- line providing ammonia addition
- water sources for the three pumps
- the eight (8) cross over valves
- line from main feedwater supply
- valves controlled by the FWCS
- MSIS valves
- steam generators

ANSWER 2.12

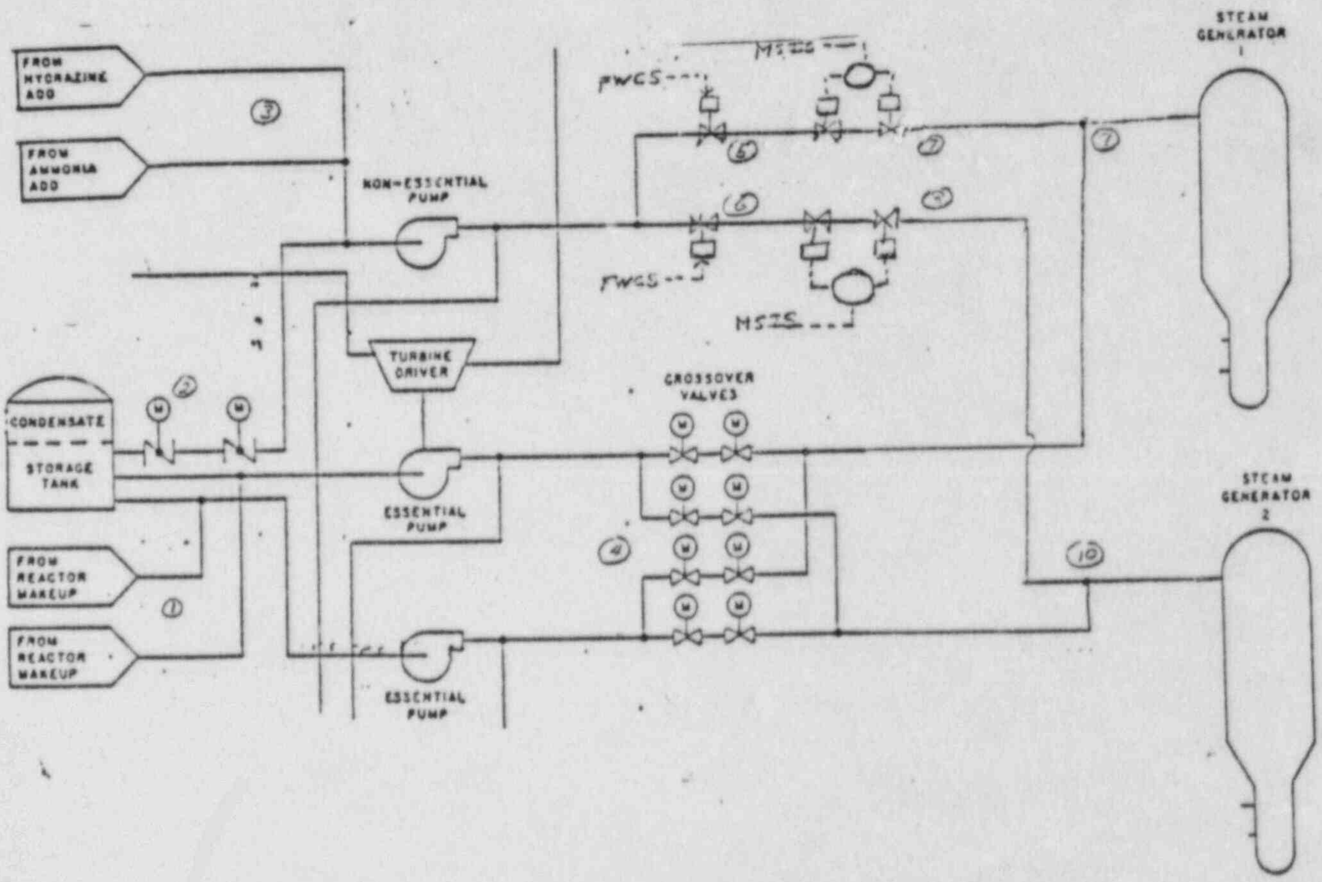
See Figure 2.12 (ANSWER) [+2.5]

Reference(s) 2.12

1. Palo Verde 1 and 2: System Descriptions, "Auxiliary Feedwater System", Training Article PGS-11, Figures PGS-11-2 and PGS-11-3.

-Section 2.0 Continued on Next Page-

Points
Available



[+0.25 each connection 1-10]

Figure 2.12 (ANSWER)

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.13

Answer the following parts of this QUESTION which pertains to the Essential Spray Pond (SP) and the Essential Spray Pond System (SPS).

a. According to the Safety Design Basis, up to what length of time can the SP and SPS be operated without water makeup? (Answer 1 hr, 1 day, 1 month or 1 year.) (0.5)

b. What are the normal (preferred) and the backup water supplies for the SPS? (0.5)

ANSWER 2.13

a. 1 month [+0.5]

b. Domestic Water System (preferred source) [+0.25]
station reservoir (non-preferred source) [+0.25]

Reference(s) 2.13

1. Palo Verde 1 and 2: System Descriptions, "Essential Spray Pond System", Training Article PGS-8A, Section 1.1, pp. PGS-8A-4 - PSG-8A-8.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.14

What operational consideration was used to design the minimum volumetric capacity of the CVCS volume control tank? (1.0)

ANSWER 2.14

sufficient capacity to accommodate the inventory change with no makeup system operation [+0.5] from a full to zero power decrease [+0.5]

Reference(s) 2.14

1. Palo Verde 1 and 2: System Descriptions, "CVCS-I", Training Article NS-2A, Section 1.2.b2, p. NS-2A-4.

QUESTION 2.15

Answer TRUE or FALSE to the following parts of this QUESTION which refers to the operation of the Main Turbine.

a. The Main Turbine shall not be operated above 2/3-rated-speed (1200 rpm) with a condenser pressure above a given maximum value. (0.5)

b. Upon loss of sealing steam, this maximum value for the condenser pressure (required for operation above 1200 rpm) is reduced; i.e., a better vacuum is required. (0.5)

ANSWER 2.15

a. TRUE [+0.5]

b. FALSE [+0.5]

Reference(s) 2.15

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Main Turbine", 41OP-1MT02, Section 3.2.3, p. 10.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.16

What is the purpose of the six (6) core stops that are welded to the inside wall of the reactor vessel? (1.0)

ANSWER 2.16

Limits the downward drop [+0.5] of the core support vessel [+0.5] in the event of a flange failure

Reference(s) 2.16

1. Palo Verde 1 and 2: System Descriptions, "Core Stops", Training Article NS-1C, Section 6.2.12, p. NS-1C-24.

QUESTION 2.17

What are the two (2) reasons for designing the nuclear fuel pellets to be slightly dished inward on both ends. (1.0)

ANSWER 2.17

1. Allow greater axial expansion at the center of the pellet during pellet heatup. [+0.5]

2. Reduce the effect of radiation induced swelling of stack height. [+0.5]

Reference(s) 2.17

1. Palo Verde 1 and 2: System Descriptions, "Fuel Rods", Training Article NS-1C, Section 2.2.3.2, p. NS-1C-9.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.18

Answer this QUESTION by "filling-in-the-blanks" in the following statement.

The minimum RCS flowrate of 111,400 gpm ensures ____________ ____________ capabilities in all modes of operation and the maximum flow rate of 129,225 gpm limits the ____________ ____________ generated on the core. (1.0)

ANSWER 2.18

heat removal [+0.5]

uplift forces [+0.5]

Reference(s) 2.18

1. Palo Verde 1 and 2: System Descriptions, "Reactor Coolant System", Training Article NS-1A, Section 1.2G, p. NS-1A-2.

-Section 2.0 Continued on Next Page-

Points
Available

QUESTION 2.19

Consider the situation in which a steam-line break has occured in Unit 1.

a. What will initially happen to T_{avg}? Why does T_{avg} change in this manner? (1.0)

b. What will initially happen to the Pressurizer level? (0.5)

ANSWER 2.19

a. T_{avg} will decrease [+0.5]. The increase in steam flow-rate will increase the heat rate between the primary and secondary [+0.5].

b. The Pressurizer level will drop (due to RCS shrinkage). [+0.5]

Reference(s) 2.19

1. Generic: Academic Program for Nuclear Power Plant Personnel, Volume III, General Physics Corporation, p. 3-37.

-End of Section 2.0-

Points
Available

3.0 INSTRUMENTS AND CONTROLS (25.0)

QUESTIONS 3.01 and 3.02 are "multiple-choice" questions. For each question specify the letter designation for the phrase that provides the most correct statement.

QUESTION 3.01

A reactor-vessel seal alarm would be received in the control room if (1.0)

(a.) the inner seal cannot hold the primary system pressure and the pressure between the two seals exceeds 1500 psig.

(b.) the outer seal cannot hold the primary system pressure and the pressure between the two seals drops below 500 psig.

(c.) the outer seal cannot hold the primary system pressure and the pressure between the seals exceeds 1000 psig.

(d.) both seals fail and the primary system pressure drops below 2100 psig.

ANSWER 3.01

(a.) **[+1.0]**

Reference(s) 3.01

1. Palo Verde 1 and 2: System Descriptions, "Reactor Coolant System", Training Article NS-1, Revision 1, August 1985, pp. NS-1A-5, NS-1A-23.

2. Palo Verde 1 and 2: Nuclear Generating Station Manual, "Excessive RCS Leakage", 41AO-1ZZ14, Appendix C, p. 2.

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.02

After the receipt of an MSIS and the closure of all four (4) MSIVs, the operator could control the temperature and pressure in the Steam Generators by (1.0)

(a.) changing the steam flowrate to the main feedwater pump turbine.

(b.) changing the steam flowrate to the auxiliary feedwater pump turbine.

(c.) taking manual control of the steam-bypass control valves.

(d.) opening or closing the valves from the main-steam lines to the upstream condensate drain line.

ANSWER 3.02

(b.) [+1.0]

Reference(s) 3.02

1. Palo Verde 1 and 2: System Descriptions, "Main Steam System", Training Article PGS-1A, pp. PGS-1A-5 - PGS-1A-6.

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.03

Answer TRUE or FALSE to each of the following statements concerning the control of each atmospheric dump valve.

a. A nitrogen gas accumulator is provided to ensure operability for at least 8 hours following the loss of the Instrument and Service Air and Nitrogen Gas Systems. (0.5)

b. One (1) solenoid-operated isolation valve is installed in the nitrogen gas supply line which can be opened from the control room or the remote-shutdown panel to unisolate the backup pneumatic supply (nitrogen gas) if instrument and service air is lost. (0.5)

c. One (1) P/I converter is installed to convert the pneumatic signal of 3 to 30 psig to an electrical signal, which is used to control the position of the dump valve. (0.5)

d. The two (2) safety related, solenoid-operated, three-way valves, installed in series in the air supply line to the actuator, must both be opened to vent the accumulator to the atmosphere. (0.5)

ANSWER 3.03

a. TRUE [+0.5]

b. FALSE [+0.5]

c. FALSE [+0.5]

d. FALSE [+0.5]

Reference(s) 3.03

1. Palo Verde 1 and 2: System Descriptions, "Main Steam System", Training Article PGS-1A, pp. PGS-1A-10 - PGS-1A-11.

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.04

Shown as Figure 3.04 (QUESTION) is the EW/NC X-TIE VALVES LOGIC DIAGRAM. Answer the following questions concerning logic diagrams in general and the control of the EW System in particular.

a. Region-1 on Figure 3.04 (QUESTION) represents what type of logic connection (AND, OR, NOR, NOT, NAND)? (0.5)

b. Region-2 on Figure 3.04 (QUESTION) represents what type of logic connection (AND, OR, NOR, NOT, NAND)? (0.5)

c. A typical logic wiring diagram is also shown on Figure 3.04 (QUESTION). Does this logic wiring diagram provide an alternate schematic for the logic represented by Region 1 or Region 2? (0.5)

d. If there were a LOP signal and a SIAS, what would be the condition of the cross tie (OPEN, CLOSED)? (0.5)

e. Describe the operator manipulation of the control-room switch to CLOSE UV-145 after it has been automatically OPENED. (0.5)

ANSWER 3.04

a. OR [+0.5]

b. AND [+0.5]

c. Region-2 [+0.5]

d. CLOSED [+0.5]

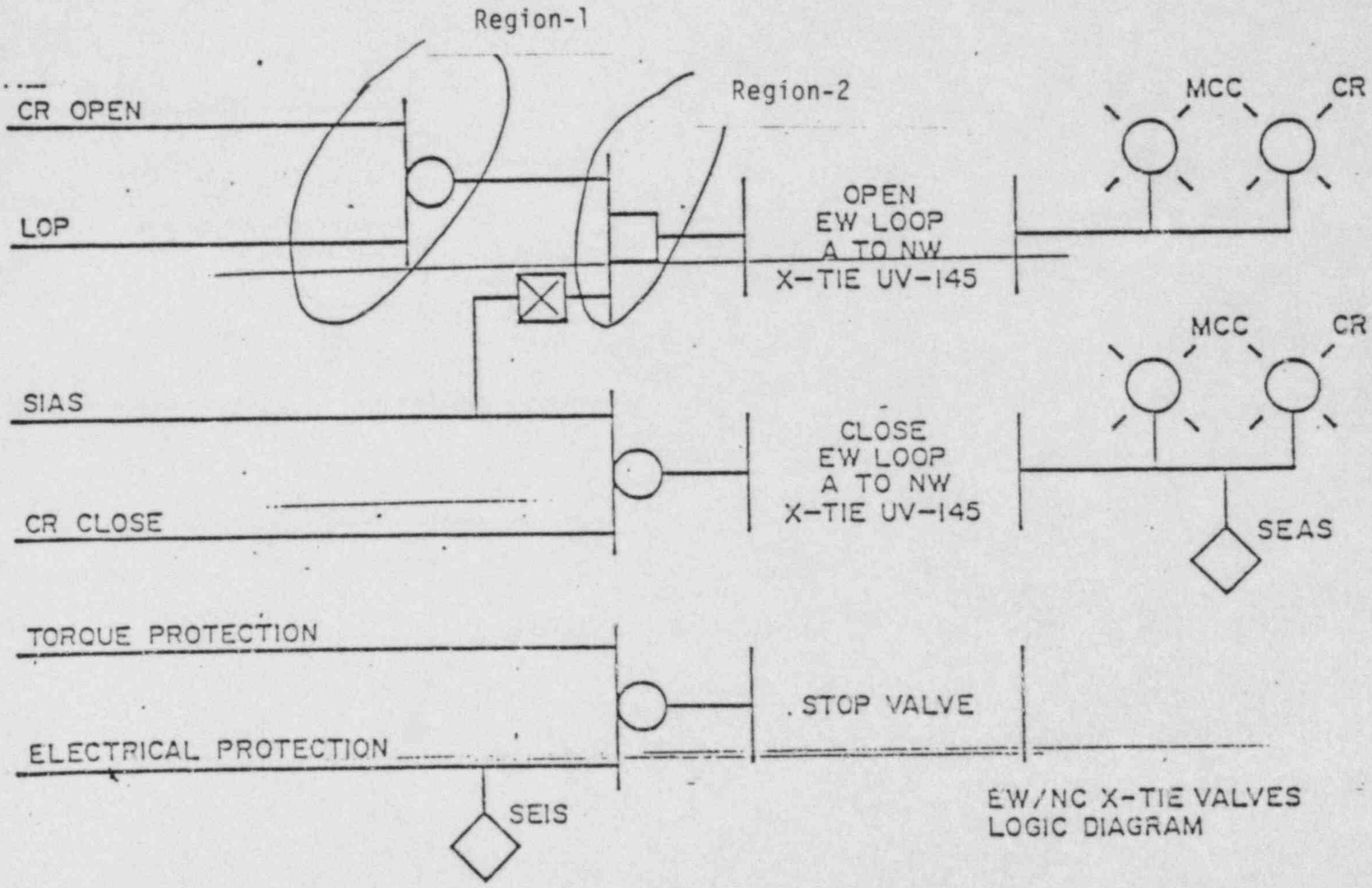
e. (Following automatic opening, the control-room operator can close the valve by) momentarily moving the control switch to the OPEN position and then to CLOSE. [+0.5]

Reference(s) 3.04

1. Palo Verde 1 and 2: System Descriptions, "Essential Cooling Water System", Training Article PGS-8B, pp. PGS-8B-17, PGS-8B-18, and Figure PGS-8B-9.

-Section 3.0 Continued on Next Page-

Points
Available



Typical logic wiring diagram for Part "c".

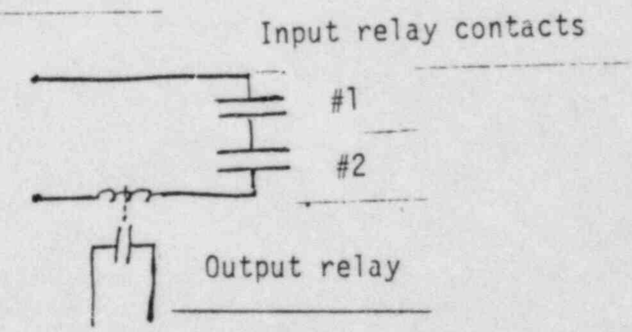


Figure 3.04 (QUESTION)

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.05

Listed below as "a." through "e." are components of the Fire Detection System for the Emergency Diesel Generator System. Listed below as "1." through "4." are four (4) possible actions of the components when they respond to the conditions of a fire. For each fire-control component, give the one (1) associated design action. (Any action item may be used for more than one fire-control component or not at all.) (2.5)

a. fixed-temperature detector above the day tank

b. sprinkler-system fusible link

c. fusible link on the chain for the rollup fire door

d. ionization smoke detector in the DG local-control room

e. ultraviolet flame detector in the DG engine room

---

1. activates water flow to the sprinkler heads located in the DG engine room

2. activates water flow to the sprinkler heads located in the silencer room

3. activates an alarm indication ONLY

4. isolates the DG engine room from the DG local-control room

ANSWER 3.05

a. 1. [+0.5]
b. 1. [+0.5]
c. 4. [+0.5]
d. 3. [+0.5]
e. 1. [+0.5]

Reference(s) 3.05

1. Palo Verde 1 and 2: Diesel Generator and Auxiliaries Handbook, pp. 7-8.

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.06

Refer to Figure 3.06 (QUESTION) which shows a typical level-measuring system for such closed tanks as the Pressurizer and the Steam Generator. Answer the following parts to this QUESTION by choosing the correct response, by "filling-in the blanks", or by completing the sentence.

a. The output of the D/P cell is $P_R - P_V$. This output is equal to the water density times _____. (h_1, h_2, h_3, h_4, or h_5) (0.5)

b. If the water level decreased, how would the output of the D/P cell ($P_R - P_V$) change? (INCREASE, DECREASE, or STAY-THE-SAME) (0.5)

c. If the reference leg broke and some of the water in the reference leg drained out, the output of the D/P cell ($P_R - P_V$) would ______________ (INCREASE, DECREASE, or STAY-THE-SAME) and the indicated level would __________ (INCREASE, DECREASE, or STAY-THE-SAME). (1.0)

d. If there were a large RCS or steam line break within containment that resulted in elevated temperatures, why would the accuracy of the level-measuring system be effected? (1.0)

ANSWER 3.06

a. h_2 [+0.5]

b. INCREASE [+0.5]

c. DECREASE [+0.5]
INCREASE [+0.5]

d. The ambient temperature of the reference leg would increase, which would decrease the density of the water in the reference leg, which would decrease P_R, which would increase the indicated tank level. [+1.0]

Reference(s) 3.06

1. Generic: C-E Training Document, "Controllers and Process Instrumentation", pp. 969(92W3)/ds-29 - 30.

-Section 3.0 Continued on Next Page-

Points
Available

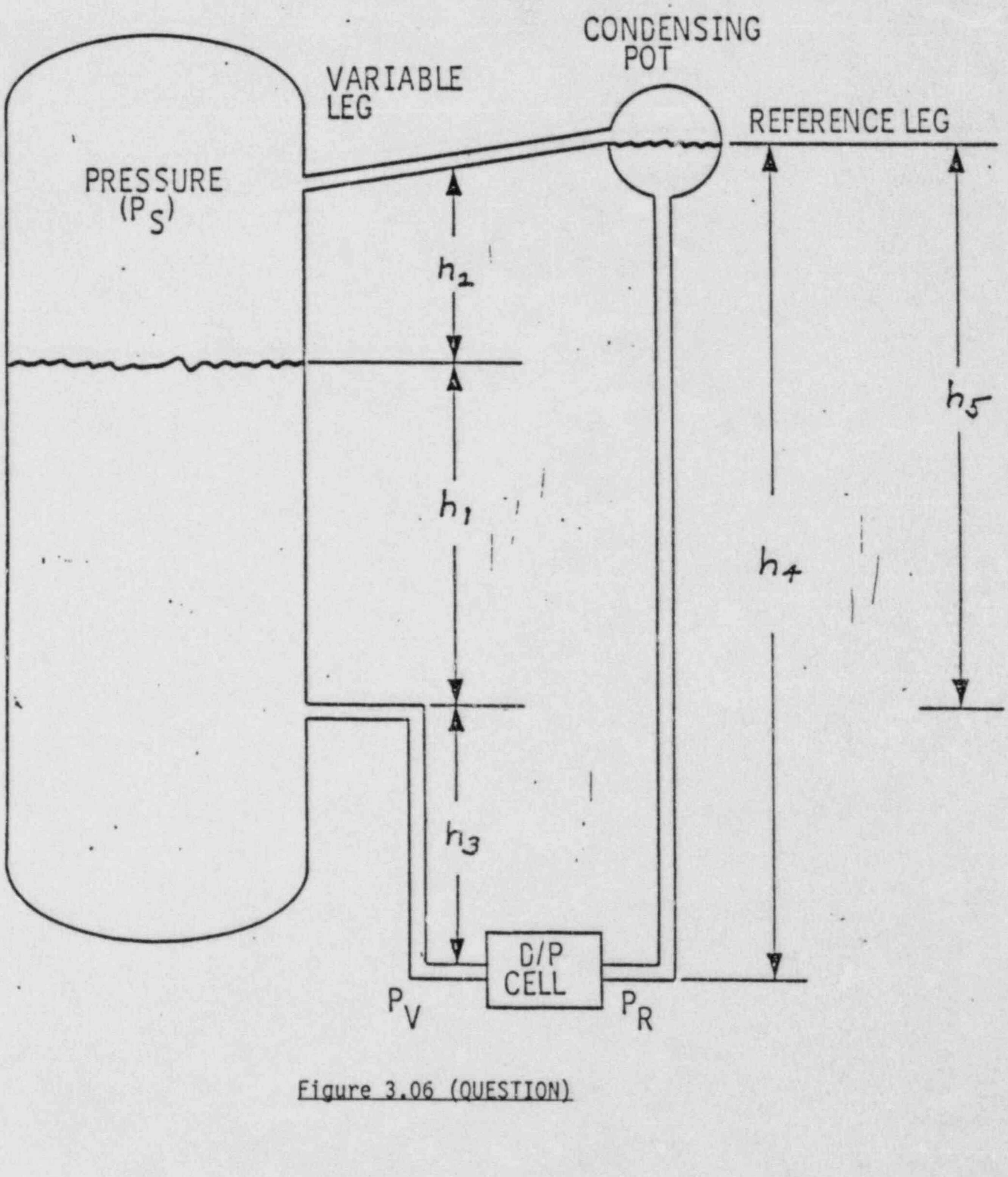


Figure 3.06 (QUESTION)

-Section 3.0 Continued on Next Page-

Points Available

QUESTION 3.07

Answer the following parts of this QUESTION which pertain to the control of the Condenser Air Removal System.

a. A two-position (START/STOP) control switch is provided in the control room for the A, B, and C vacuum pumps. List the three (3) results of momentarily placing the control switch for the A-pump in the START position. (Do not list any light indications or that the switch will return to the neutral position when released. Do list such actions as pump X receives a start signal, valve Y receives a close-permissive signal, etc.) (1.5)

b. A three-position (OPEN/CLOSE/AUTO) control switch is provided in the control room for the isolation valve in the suction line from the A section of the main condensor to the standby vacuum pump. When in the AUTO position, the valve will automatically OPEN if two (2) conditions are met. List these two (2) conditions. (1.0)

c. From the exhaust header are three (3) flowpaths to the atmosphere, There is one (1) radiation monitor, and one (1) filtering unit. If the radiation monitor is triggered on a high-radiation condition, describe the flowpaths by completing the table below. (1.5)

	Flowpath-1	Flowpath-2	Flowpath-3
Is the filter this flowpath? (YES or NO)	________	________	________
Isolation valve position? (OPEN, CLOSED, or STAY-THE-SAME)	________	________	________

-Section 3.0 Continued on Next Page-

Points
Available

ANSWER 3.07

a. • the vacuum pump A will receive a start signal

• an open-permissive is sent to the A pump's suction isolation valve

• associated seal-water recirculation pump will receive a start signal

[+0.5 each]

b. • a standby vacuum pump running signal is received

• the pressure in section A exceeds its setpoint (3.7 in. HgA)

[+0.5 each]

c.

flowpath-1	flowpath-2	flowpath-3
NO	YES	NO
CLOSED	OPEN	STAY-THE-SAME

will accept CLOSED for this flowpath, as it should be CLOSED when the others are in operation

[+1.5]

Reference(s) 3.07

1. Palo Verde 1 and 2: System Descriptions, "Condenser Air Removal System", Training Article PGS-9B, pp. PGS-9B-15 - PGS-9B-22.

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.08

Answer the following parts of this QUESTION which refer to the control of the Condensate System.

a. The Condenser hotwell makeup control valves LV-81 and LV-82 will automatically close upon receiving a signal indicating what condition? (0.5)

b. When in the AUTO mode, the condensate polishing demineralizer bypass valve position is varied in response to what signal? (0.5)

c. When in the AUTO mode, the condensate polishing demineralizer bypass valve position will vary automatically only if three (3) conditions are met. List these three (3) conditions. (1.0)

d. Specify the buses which normally power each of the condensate pumps (A, B, and C). (1.0)

ANSWER 3.08

a. Condensate storage tank level decreases to 28 ft actual level **[+0.5]**

b. demineralizer differential pressure **[+0.5]**

c.
- the deminalerizer input valve is fully OPEN **[+0.25]**
- the deminalerizer output valve is fully OPEN **[+0.25]**
- the differential pressure is less than 35 psi **[+0.5]**

d.
- A, B on S01 **[+0.5]**
- C on S02 **[+0.5]**

Reference(s) 3.08

1. Palo Verde 1 and 2: System Descriptions, "Condensate System", Training Article PGS-9A, pp. PGS-9A-21, PGS-9A-22, PGS-30.

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.09

The Reactor Regulating System generates a "ϕn signal" and a "ϕn Deviation" indication.

a. What two (2) signals are used to produce the "ϕn signal" and the "ϕn Deviation" indication? (0.5)

b. What is the acceptable relationship between the two (2) signals that would prevent the indication of "ϕn Deviation"? (0.5)

c. What signal would be sent to CEDMCS if there is a "ϕn Deviation"? (0.5)

ANSWER 3.09

a. excore neutron detectors in control channels 1 and 2 [+0.5]

b. difference in the signals must be within 5% [+0.5]

c. AMI [+0.5]

Reference(s) 3.09

1. Palo Verde 1 and 2: System Descriptions, "Reactor Regulating System", Training Article NS-9C, Revision 1, October 1985, p. NS-9C-7.

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.10

Answer the QUESTION by "filling-in-the blanks."

The Steam Bypass Control System (SBCS) controls the Steam Bypass Control valves by using the output of a PID controller which has two inputs. One signal input is the __________ which is compared to a setpoint which is a function of the __________ input signal. Actuation of a quick-opening sequence is based on the rate-of-change of the __________ input signal. (1.5)

ANSWER 3.10

steam-header pressure [+0.5]

steam flowrate [+0.5]

steam flowrate [+0.5]

Reference(s) 3.10

1. Palo Verde 1 and 2: System Descriptions, "Steam Bypass Control System", Training Article NS-9B, p. NS-9B-19.

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.11

The Fuel Building Essential Ventilation System (FBEVS) uses inputs from two (2) radiation monitors for ESF actuation. Specify the location for the detectors of these two (2) radiation monitors. (1.0)

ANSWER 3.11

1. exhaust duct [+0.5]

2. over the fuel pool [+0.5]

Reference(s) 3.11

1. Palo Verde 1 and 2: System Descriptions, "Engineered Safety Features Actuation System 1/2", Training Article NS-7B, pp. NS-7B-15, NS-7B-16.

-Section 3.0 Continued on Next Page-

Points
Available

QUESTION 3.12

If the following conditions existed at Unit 1, list the ESFAS actuation signals that should have been generated. (2.0)

a. Pressurizer pressure = 1650 psig (no bypass)
AND
containment pressure = 4 psig
AND
Steam Generator 1 & 2 pressures = 940 psia
AND
Steam Generator 1 & 2 level = 57%

b. Pressurizer pressure = 2200 psig (no bypass)
AND
containment pressure = 11 psig
AND
Steam Generator 1 pressure = 700 psia
AND
Steam Generator 2 pressure = 950
AND
Steam Generator 1 level = 20%
AND
Steam Denerator 2 level = 50%

ANSWER 3.12

CIAS, MSIS [+0.33 ...]

a. SIAS [~~+0.5~~]

b. SIAS
CSAS
CIAS
MSIS
~~AFAS-2~~ AFAS-1 with HI Cp lockout, ... AFAS-1

~~[+0.3 each, +1.5 max.]~~ [+0.2 each]

Reference(s) 3.12

1. Palo Verde 1 and 2: System Descriptions, "Engineered Safety Features Actuation System 2/4", Training Article NS-7A, pp. NS-7A-32 - NS-7A-35.

-End of Section 3.0-

Points
Available

4.0 PROCEDURES, NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL (25.0)

QUESTIONS 4.01 and 4.03 are "multiple-choice" questions. For each question specify the letter designation for the phrase that provides the most correct statement.

QUESTION 4.01

On-coming shift personnel shall review shift logs (1.0)

(a.) back to the last shift worked or the previous 3 days of logs, whichever is the shortest, prior to relieving the off-going shift and after relief back to the last shift worked.

(b.) of the last shift prior to relieving the off-going shift and after relief back to the last shift worked or the previous 3 days of logs whichever is the shortest.

(c.) of the last 3 days prior to relieving the off-going shift and after relief the Shift Turnover Comment Sheet.

(d.) back to the last shift worked prior to relieving the off-going shift and after relief the Shift Turnover Comment Sheet.

ANSWER 4.01

(a.) [+1.0]

Reference(s) 4.01

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Shift Turnover", 40AC-9ZZ16, Appendix D, p. 3.

Points
Available

QUESTION 4.02

A change in T_{avg} is taken into account in the calculation of estimated critical rod position by (1.0)

(a.) making a $\Delta\rho(T_{avg})$ calculation if T_{avg} (projected) and T_{avg} (previous) are both ≥ 565°F.

(b.) making a $\Delta\rho(T_{avg})$ calculation if T_{avg} (projected) and T_{avg} (previous) are both ≤ 565°F.

(c.) making a $\Delta\rho$(power) calculation if power (projected) and power (previous) are both ≥ 0%.

(d.) making a $\Delta\rho$(power) calculation if T_{avg} (projected) and T_{avg} (previous) are both ≤ 565°F.

ANSWER 4.02

(b.) [+1.0]

Reference(s) 4.02

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Calculation of Estimated Critical Condition", 72OP-9RX01, pp. 2-7.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.03

With Unit 1 in Mode 1 and one or more CEAs determined to be inoperable (1.0)

(a.) it is assumed that the shutdown margin is less than 6% $\Delta k/k$ if one or more inoperable CEAs are immovable/untrippable.

(b.) it is assumed that the shutdown margin is less than 6% $\Delta k/k$ if two or more inoperable CEAs are immovable/untrippable and the shutdown margin is greater than 6% $\Delta k/k$ if only one is immovable/untrippable.

(c.) boration of 40 gpm or greater of a solution of 4000 ppm or greater boron is immediately required if the inoperable CEAs are still trippable after 1 hour.

(d.) boration of 40 gpm or greater of a solution of 4000 ppm or greater boron is immediately required if one of the inoperable CEAs is the most reactive rod.

ANSWER 4.03

(a.) [+1.0]

Reference(s) 4.03

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Shutdown Margin", 72ST-1RX09, pp. 5-7.

Points
Available

QUESTION 4.04

Answer TRUE or FALSE.

Following a LOCA, the hydrogen monitors must be placed in service 1 hour after the event. (0.5)

ANSWER 4.04

FALSE [+0.5] in service within 30 minutes

Reference(s) 4.04

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Loss of Coolant Accident", 41RO-1ZZ07, Revision 1, Section 3.4, p. 6.

QUESTION 4.05

Answer TRUE or FALSE.

dropped

At approximately 15% power the steam generator downcomers will fail closed and all feed control will be with the economizer. (0.5)

ANSWER 4.05

TRUE [+0.5]

Reference(s) 4.05

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Plant Startup Mode 2 to Mode 1", 41OP-1ZZ04, Section 3.0, p. 23.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.06

Answer TRUE or FALSE.

The spray-pond pump motor, rated at 4160 volts, should be limited to a maximum of four (4) consecutive attempts-to-start with the motor initially at ambient temperature. (0.5)

ANSWER 4.06

FALSE [+0.5] only two (2) consecutive starts

Reference(s) 4.06

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Essential Spray Pond - Train A", 41OP-1SP01, Revision 3, p. 11.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.07

In the left-hand column below is a list (itemized numerically) of "symptoms" from the procedure "Shutdown Outside Control Room". The right-hand column is a list (itemized alphabetically) of "causes". Choose one (1) item from the cause list for each item in the symptom list. Use each item from the cause list for one (1) item from the symptom list. (2.5)

Symptom	Cause
1. S/G level less than 0% WR T_{cold} increasing above T_{sat} of S/G (T_{hot} follows)	A. Condensible voids collecting in RCS flowpath
2. T_{cold} and S/G pressure increasing	B. Inadequate RCS inventory
3. RCS Loop Delta T increasing and abnormal PLCS response	C. Inadequate secondary steam flow
4. T_{hot} increasing and low Pressurizer level	D. Non-condensible voids collecting in RCS flowpath or physical blockage of RCS flowpaths
5. RCS Loop Delta T increasing, no other indication at RCPs	E. Inadequate secondary water inventory

ANSWER 4.07

1. E
2. C
3. A
4. B
5. D

Reference(s) 4.07

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Shutdown Outside Control Room", 41AO-1ZZ27, Revision 1, p. 58.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.08

Specify when each of the tags listed below should be utilized; i.e., what does hanging that tag indicate?

a. red tag (1.0)

b. yellow tag (1.0)

c. blue tag (1.0)

ANSWER 4.08

a. Red Danger Tag (Appendix D)

A red danger tag shall be placed on equipment (mechanical or electrical) which, if the equipment was operated, would endanger personnel or damage equipment. [+1.0]

b. Yellow Caution Tag (Appendix E)

The yellow caution tag or sticker is utilized where no danger to personnel is involved. It shall state specific information which shall be understood and observed before operating the equipment. [+1.0]

c. Blue Men-at-Work Tag (Appendix F)

A blue tag is placed on system components to indicate that the equipment is to be operated only by the direction of the individual to whom the tag is issued, after that individual has obtained permission from the Responsible Supervisor in order to ensure operation of the equipment does not affect the system's operation. A blue men-at-work tag is to be used used only for trouble shooting or testing, maintenance work/repair is not allowed. [+1.0]

Reference(s) 4.08

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Station Tagging and Clearance", 40AC-9ZZ15, p. 6.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.09

According to Procedure No 41RO-1ZZ05, "Loss of Feedwater," there is a preferred order or priority for which Auxiliary Feedwater Pumps are utilized - assuming that there is no MSIS present. List the preferred order as indicated below. (1.5)

1. (pump to be utilized first)

2.

3. (last pump in priority)

ANSWER 4.09

1. startup auxiliary-feed pump

2. essential electric auxiliary-feed pump

3. essential steam-driven auxiliary-feed pump

[+0.5 each]

Reference(s) 4.09

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Loss of Feedwater", 41RO-1ZZ05, Revision 1, Section 4.1, Note, p. 9.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.10

List the four (4) criteria that must be met before safety injection (SI) flow may be throttled. (2.0)

ANSWER 4.10

1. RCS subcooled > 28°F

2. Rx vessel level indicates void restricted to upper head

3. Pzr level > 33% and controllable

4. one (1) S/G capable of maintaining heat removal

[+0.5 each]

Reference(s) 4.10

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Reactor Trip", 41RO-1ZZ01, Section 3.0, p. 27.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.11

Match the following (numerical) emergency classifications with the appropriate (alphabetical) description. (2.0)

1. Unusual Event — A. Consists of events which are in progress or have occurred which involve actual or likely major failures of plant functions needed for the protection of the public. Any releases are not expected to exceed Environmental Protection Agency Guideline exposure levels beyond the site boundary.

2. Alert — B. This classification applies to events which are in progress or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occur.

3. Site Area Emergency — C. Consists of events which are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed Environmental Protection Agency Protective Action Guideline exposure levels offsite for more than the immediate site area.

-Section 4.0 Continued on Next Page-

Points
Available

4. General Emergency D. This classification consists of events which are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the Environmental Protection Agency Protective Action Guideline exposure levels.

ANSWER 4.11

1. B

2. D

3. A

4. C

[+0.5 each]

Reference(s) 4.11

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Emergency Classifications", EP1P-02, Revision 4, Section 3.2, p. 5.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.12

If, during reactor startup and approach to criticality, the boron concentration were to be changed by more than 50 ppm, what action should the operator take with the Pressurizer system? (1.0)

ANSWER 4.12

Initialize Pressurizer spray to equalize RCS and pressurizer boron concentration. [+1.0]

Reference(s) 4.12

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Reactor Startup", 41OP-1ZZ03, Revision 3, Section 4.3.12 Caution, p. 10.

QUESTION 4.13

Procedure No. 41OP-1ZZ01, "Cold Shutdown to Hot Standby", directs you to reset both inadvertent-dilution alarms after the startup channel high voltage is turned off. Where are the controls to accomplish this located? (1.0)

ANSWER 4.13

Behind B04 in the Miscellaneous Equipment Cabinet, 1J-ZJN-C06 [+1.0]

Reference(s) 4.13

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Cold Shutdown to Hot Standby", 41OP-1ZZ01, Revision 2, p. 59.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.14

Explain why the differential temperature between the spray water temperature and the Pressurizer temperature should be maintained less than 200°F. (1.0)

ANSWER 4.14

to avoid unnecessary spray-nozzle fatigue [+1.0]

Reference(s) 4.14

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Cold Shutdown to Hot Standby", 41OP-1ZZ01, Revision 2, p. 49.

QUESTION 4.15

Give the administrative control limits for whole body exposure. (1.5)

Weekly ____________

Quarterly ____________

Yearly ____________

ANSWER 4.15

weekly - 300 mrem [+0.5]

quarterly - 1.0 rem [+0.5]

yearly - 4.0 rem [+0.5]

Reference(s) 4.15

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Radiation Exposure and Access Control", 75AC-9ZZ01, Revision 4, p. 25.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.16

Give the beta-gamma area contamination limits for the following

1. contamination area __________ (0.5)

2. high contamination area __________ (0.5)

ANSWER 4.16

1. $>$ 1000 dpm/100 cm^2 [+0.5]

2. $>$ 50,000 dpm/100 cm^2 [+0.5]

Reference(s) 4.16

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Radioactive Contamination Control", 75AC-9ZZ03, Revision 3, p. 11.

QUESTION 4.17

a. As specified by Tech-Specs., what is the minimum allowable value for T_{cold} when operating in Modes 1 and 2? Neglect any special test exceptions. (0.5)

b. How much time is allotted to correct T_{cold} before the reactor power level must be reduced? (0.5)

ANSWER 4.17

a. 552°F [+0.5]

b. 15 minutes [+0.5]

Reference(s) 4.17

1. Palo Verde 1 and 2: Technical Specifications, p. 3/4 1-6.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.18

According to the Unit 1 Tech-Specs., two (2) out of three (3) boron injection flow paths shall be OPERABLE when in Modes 1, 2, 3 and 4. List these three (3) flow paths. (1.5)

ANSWER 4.18

1. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System. **[+0.5]**

2. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System. **[+0.5]**

3. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System. **[+0.5]**

Reference(s) 4.18

1. Palo Verde 1 and 2: Technical Specifications, p. 3/4 1-8.

-Section 4.0 Continued on Next Page-

Points
Available

QUESTION 4.19

List five (5) pieces of information that would be found on a Radiation Exposure Permit (REP) at PVNGS. (1.5)

ANSWER 4.19

1. job description
2. radiological conditions at job location
3. required protective clothing
4. required dosimetry
5. required respiratory equipment
6. job classification
7. dates and times of validation
8. special job instructions

[+0.3 each, +1.5 max]

Reference(s) 4.19

1. Palo Verde 1 and 2: Palo Verde Nuclear Generating Station Manual, "Radiation Exposure and Access Control", 75AC-9ZZ01, Revision 4, p. 14.

-End of Section 4.0-

-End of Exam-

TABLE OF CONTENTS

FIGURE	REV.	TITLE	PAGE
TABLE 1.1	001	"TYPICAL VALUES" FOR QUICK REFERENCE	1
TABLE 2.1.1	001	POWER DEFECT VS POWER LEVEL: BOC. MOC. EOC	5
CURVE 2.1.1	001	POWER DEFECT VS POWER LEVEL: BOC. MOC. EOC	6
TABLE 2.2.1	001	ISOTHERMAL TEMP DEFECT VS T-AVG: 550-565 F. BOC	7
CURVE 2.2.1	001	ISOTHERMAL TEMP DEFECT VS T-AVG: 550-565 F. BOC	8
TABLE 2.2.2	001	ISOTHERMAL TEMP DEFECT VS T-AVG: 550-565 F. MOC	9
CURVE 2.2.2	001	ISOTHERMAL TEMP DEFECT VS T-AVG: 550-565 F. MOC	10
TABLE 2.2.3	001	ISOTHERMAL TEMP DEFECT VS T-AVG: 550-565 F. EOC	11
CURVE 2.2.3	001	ISOTHERMAL TEMP DEFECT VS T-AVG: 550-565 F. EOC	12
TABLE 2.3.1	000	BORON WORTH VS T-AVG. 505-615 F. 100% P. BOC	13
CURVE 2.3.1	000	BORON WORTH VS T-AVG. 505-615 F. 100% P. BOC	15
TABLE 2.3.2	000	BORON WORTH VS T-AVG. 505-615 F. 100% P. MOC	16
CURVE 2.3.2	000	BORON WORTH VS T-AVG. 505-615 F. 100% P. MOC	18
TABLE 2.3.3	000	BORON WORTH VS T-AVG. 505-615 F. 100% P. EOC	19
CURVE 2.3.3	000	BORON WORTH VS T-AVG. 505-615 F. 100% P. EOC	21
TABLE 2.3.4	000	BORON WORTH VS T-AVG. 70-565 F. 0% P. BOC	22
CURVE 2.3.4	000	BORON WORTH VS T-AVG. 70-565 F. 0% P. BOC	24
TABLE 2.3.5	000	BORON WORTH VS T-AVG. 70-565 F. 0% P. MOC	25
CURVE 2.3.5	000	BORON WORTH VS T-AVG. 70-565 F. 0% P. MOC	27
TABLE 2.3.6	000	BORON WORTH VS T-AVG. 70-565 F. 0% P. EOC	28
CURVE 2.3.6	000	BORON WORTH VS T-AVG. 70-565 F. 0% P. EOC	30
TABLE 2.4.1	000	EQ. XENON WORTH VS BURNUP. HFP	31
CURVE 2.4.1	000	EQ. XENON WORTH VS BURNUP. HFP	32
TABLE 2.4.2	000	EQ. XENON WORTH VS POWER LEVEL. BOC. MOC. EOC	33
CURVE 2.4.2	000	EQ. XENON WORTH VS POWER LEVEL. BOC. MOC. EOC	34
TABLE 2.5.1	000	REG GROUP 5 WORTH VS WITHDRAWAL. HFP. BOC. MOC. EOC	35
CURVE 2.5.1	000	REG GROUP 5 WORTH VS WITHDRAWAL. HFP. BOC. MOC. EOC	37
TABLE 2.5.2	000	REG CEA WORTH (OVERLAP) VS WITHDRAWAL. 50% P. BOC. MOC. EOC	38
CURVE 2.5.3	000	REG CEA WORTH (OVERLAP) VS WITHDRAWAL. 50% P. BOC	40
CURVE 2.5.4	000	REG CEA WORTH (OVERLAP) VS WITHDRAWAL. 50% P. MOC	41
CURVE 2.5.5	000	REG CEA WORTH (OVERLAP) VS WITHDRAWAL. 50% P. EOC	42
TABLE 2.5.6	001	REG CEA WORTH (OVERLAP) VS WITHDRAWAL. HZP. BOC. EOC	43
CURVE 2.5.6	001	REG CEA WORTH (OVERLAP) VS WITHDRAWAL. HZP. BOC. EOC	48
TABLE 2.6.1	000	PLCEA WORTH VS WITHDRAWAL. HFP. BOC	50
CURVE 2.6.1	000	PLCEA WORTH VS WITHDRAWAL. HFP. BOC	52
TABLE 2.6.2	000	PLCEA WORTH VS WITHDRAWAL. HFP. MOC	53
CURVE 2.6.2	000	PLCEA WORTH VS WITHDRAWAL. HFP. MOC	55
TABLE 2.6.3	000	PLCEA WORTH VS WITHDRAWAL. HFP. EOC	56
CURVE 2.6.3	000	PLCEA WORTH VS WITHDRAWAL. HFP. EOC	58
TABLE 2.6.4	000	PLCEA WORTH VS WITHDRAWAL. 50% P. BOC	59
CURVE 2.6.4	000	PLCEA WORTH VS WITHDRAWAL. 50% P. BOC	61
TABLE 2.6.5	000	PLCEA WORTH VS WITHDRAWAL. 50% P. MOC	62
CURVE 2.6.5	000	PLCEA WORTH VS WITHDRAWAL. 50% P. MOC	64
TABLE 2.6.6	000	PLCEA WORTH VS WITHDRAWAL. 50% P. EOC	65
CURVE 2.6.6	000	PLCEA WORTH VS WITHDRAWAL. 50% P. EOC	67
TABLE 2.6.7	000	PLCEA WORTH VS WITHDRAWAL. HZP	68
CURVE 2.6.7	000	PLCEA WORTH VS WITHDRAWAL. HZP	70
TABLE 2.7.1	000	EQUILIBRIUM SHAPE INDEX VS BURNUP AT 100% POWER	70A
CURVE 2.7.1	000	EQUILIBRIUM SHAPE INDEX VS BURNUP AT 100% POWER	70B
TABLE 2.8.1	000	DELTA ASI AT 80%. 50%. 20% POWER.0 TO 208 EFPD	70C
CURVE 2.8.1	000	DELTA ASI AT 80%. 50%. 20% POWER.0 TO 208 EFPD	70D
TABLE 2.9.1	000	EQUILIBRIUM IODINE VS POWER LEVEL.BOC THROUGH EOC	70E
CURVE 2.9.1	000	EQUILIBRIUM IODINE VS POWER LEVEL.BOC THROUGH EOC	70F

TABLE 1.1 (PAGE 1 OF 4)
"TYPICAL VALUES" FOR QUICK REFERENCE

CAUTION: These values are presented for a quick reference. The appropriate values from Section 2 or 3, or from the Operator Assistance Programs should be used for calculations affecting Reactor Operation.

1. Reactivity Effects of Temperature and Power

	BOL (960ppm)	EOL (24ppm)
Overall power defect, 0 ------> 100% power	-1380 PCM	-1950 PCM
Moderator Temp. Defect, 0 ----> 100%	- 150 PCM	- 700 PCM
Fuel Temp. (Doppler) Defect, 0 ----> 100%	-1230 PCM	- 820 PCM
Power Redistribution Effects, 0 ----> 100%	0 PCM	- 430 PCM
Reactivity defect from Hot Zero Power ----> 210F PCM	+1660 PCM	+2130 PCM
Reactivity defect from 210F----> 80F PCM	+1950 PCM	+2650 PCM
MTC, at 100% power PCM/F	- 5 PCM/F	-25 PCM
Fuel Temp. Coef., at 100% power PCM/F	- 1.3 PCM/F	- 1.5 PCM/F
Fuel Temp. at 100% power (effective)	1342 F	1084 F

The values above do not include effects of xenon change or PPM change.

2. Soluble Boron Worth

Hot full power	-11 PCM/PPM	-12 PCM/PPM
Hot zero power	-12 PCM/PPM	-13 PCM/PPM
210F, PPM=2000	-13 PCM/PPM	-14 PCM/PPM
210F, PPM=500	-15 PCM/PPM	-16 PCM/PPM
80F, PPM=2000	-13 PCM/PPM	-14 PCM/PPM

3. Xenon Worth

Equilibrium full power xenon worth	-2680 PCM	-2860 PCM
Peak xenon worth after trip from 100%	-5380 PCM	-5560 PCM
Time of xenon peak after trip	8.5 hours	8.5 hours

PLANT CONDITIONS
0 - 100% POWER
0-455.7 EFPD
COMMENTS: T1Z1Z1A

REFERENCE SOURCE OF DATA
V-CE-19010
85-NA-NFM-528

APPROVED FOR USE BY: Daniel Hixon 9-27-85
NUCLEAR SUPERVISOR DATE

TABLE 1.1 (PAGE 2 OF 4)
"TYPICAL VALUES" FOR QUICK REFERENCE

4. Control Rod Worths

	BOL*	EOL*
Total Rod Worth, hot full power	-18500 pcm	-19600 pcm
Total Rod Worth, hot zero power	-17270 pcm	

	ANYTIME IN CYCLE
Minimum Total Rod Worth**, hot zero power	-16400
, 210F	-11200
, 80F	-10500
Minimum Rod Worth* with Worst Rod Out,	
, hot zero power	- 9810
, 210F	- 5920
, 80F	- 5510

	BOL*	EOL*
Individual Rod Bank Worths, Hot full power		
SD Groups A & B (Reg.Grps 1-5 & P ARI)	-13900 pcm	-14800 pcm
PLCEA Group P (Reg. Grps 1-5 ARI)	- 510	550
Reg. Group 1 (Reg.Grps 2-5 ARI)	- 1400	- 1540
2 (Reg.Grps 3-5 ARI)	- 980	- 920
3 (Reg.Grps 4-5 ARI)	- 960	- 1100
4 (Reg. Grp 5 ARI)	- 480	- 440
5 (All other rods out)	- 300	- 320
PLCEA Group P (all other rods out)	- 240	- 270
Individual Rod Bank Worths, hot zero power		
SD Groups A & B	-13140	
PLCEA Group P (Reg. Grps. 1-5 ARI)	- 330	
Reg. Group 1	- 1270	
2	- 990	
3	- 850	
4	- 420	
5	- 260	
PLCEA Group P (all other rods out)	- 230	

*These rod worths were calculated at critical boron concentrations for each rod configuration; this method causes the total Rod Worth to be somewhat high. Do not use for Shutdown Margin calculations.

**These minimum Rod Worth values do not include the worth of the PLCEAs (approximately 200-400 pcm).

PLANT CONDITIONS
0 - 100% POWER
0-455.7 EFPD
COMMENTS: T1Z1Z1B

REFERENCE SOURCE OF DATA
V-CE-19010

TABLE 1.1 (PAGE 3 OF 4)
"TYPICAL VALUES" FOR QUICK REFERENCE

5. Reactivity Units and β-Effective

BOL (β =0.00718)
1 pcm = 0.001% $\Delta\rho$ = 0.139¢
1000 pcm = 1.000% $\Delta\rho$ = 139¢
718 pcm = 0.718% $\Delta\rho$ = 100¢

EOL (β =0.0052)
1 pcm = 0.001% $\Delta\rho$ = 0.192¢
1000 pcm = 1.000% $\Delta\rho$ = 192¢
520 pcm = 0.520% $\Delta\rho$ = 100¢

6. Startup Rate

$P = Po \cdot e^{(t/T)}$ $T = \frac{\overline{\beta eff} - \rho}{\lambda\rho}$

SUR=Startup Rate(decades/min.
λ = effective delayed neutro decay constant (1/sec)

$SUR = \frac{26.06}{T}$ $\rho = \frac{\overline{\beta eff}}{(1+\lambda T)}$ T = Reactor Period (seconds)

For BOL: $\overline{\beta eff}$ = 0.00718 λ = 0.07754 sec -1

SUR(DPM)	Period(sec)	PCM	%$\Delta\rho$	¢
1.0	26.1	237.7	0.2377	33.1
0.8	32.6	203.6	0.2036	28.4
0.6	43.4	164.4	0.1644	22.9
0.4	65.2	118.6	0.1186	16.5
0.2	130.3	64.7	0.0647	9.0
-0.2	130.3	-78.9	-0.0789	-11.0

7. Boration/Dilution

PPM	Gallons of RMW to Reduce PPM by 1PPM	Gallons of Refueling Water to Increase PPM by 1 PPM
100	836	19
500	153	21
1000	76	24

PLANT CONDITIONS
0 - 100% POWER
0-455.7 EFPD
COMMENTS: T1Z1Z1C

REFERENCE SOURCE OF DATA
V-CE-19010

TABLE 1.1 (PAGE 4 OF 4)
"TYPICAL VALUES" FOR QUICK REFERENCE

8. Decay Heat

Rate of Production of decay heat following shutdown from full power.

Time After Shutdown	% of Full Power
1 second	6
1 minute	4.5
30 minutes	2.0
1 hour	1.6
8 hours	1.0
24 hours	.7
48 hours	.6
1 month	.2
4 months	.02

9. Core Burnup Units

1 EFPD = 38.4 MWD/T

10. Approximate relationship of change in Power Operating Limit (POL) to changes in certain core parameters.

Δ(% POL) = 0.5 * Δ(% DNBR)
Δ(% POL) =-1.0 * Δ(% Radial Peaking Factor)
Δ(% POL) = 0.7 * Δ(% Core Flow)
Δ(% POL) = 0.03 * Δ(Primary Pressure, psia)
Δ(% POL) =-0.5 * Δ(Tin, F)

PLANT CONDITIONS
0 - 100% POWER
0-455.7 EFPD
COMMENTS: T1Z1Z1D

REFERENCE SOURCE OF DATA
V-CE-19010

TABLE 2.1.1

POWER DEFECT VS POWER LEVEL (BOC, MOC, EOC)

POWER LEVEL (%)	POWER DEFECT (PCM) BOC 0 EFPD	MOC 208.3 EFPD	EOC 455.7 EFPD
0	0	0	0
5	-85	-100	-90
10	-170	-196	-180
15	-252	-285	-270
20	-331	-365	-362
25	-408	-439	-452
30	-482	-509	-540
35	-555	-579	-631
40	-625	-653	-718
45	-695	-734	-805
50	-762	-816	-896
55	-829	-893	-990
60	-894	-966	-1084
65	-959	-1045	-1183
70	-1021	-1104	-1286
75	-1083	-1172	-1390
80	-1143	-1240	-1500
85	-1203	-1309	-1610
90	-1260	-1376	-1720
95	-1319	-1444	-1836
100	-1377	-1511	-1949

PLANT CONDITIONS

0 TO 100% POWER
0 TO 455.7 EFPD

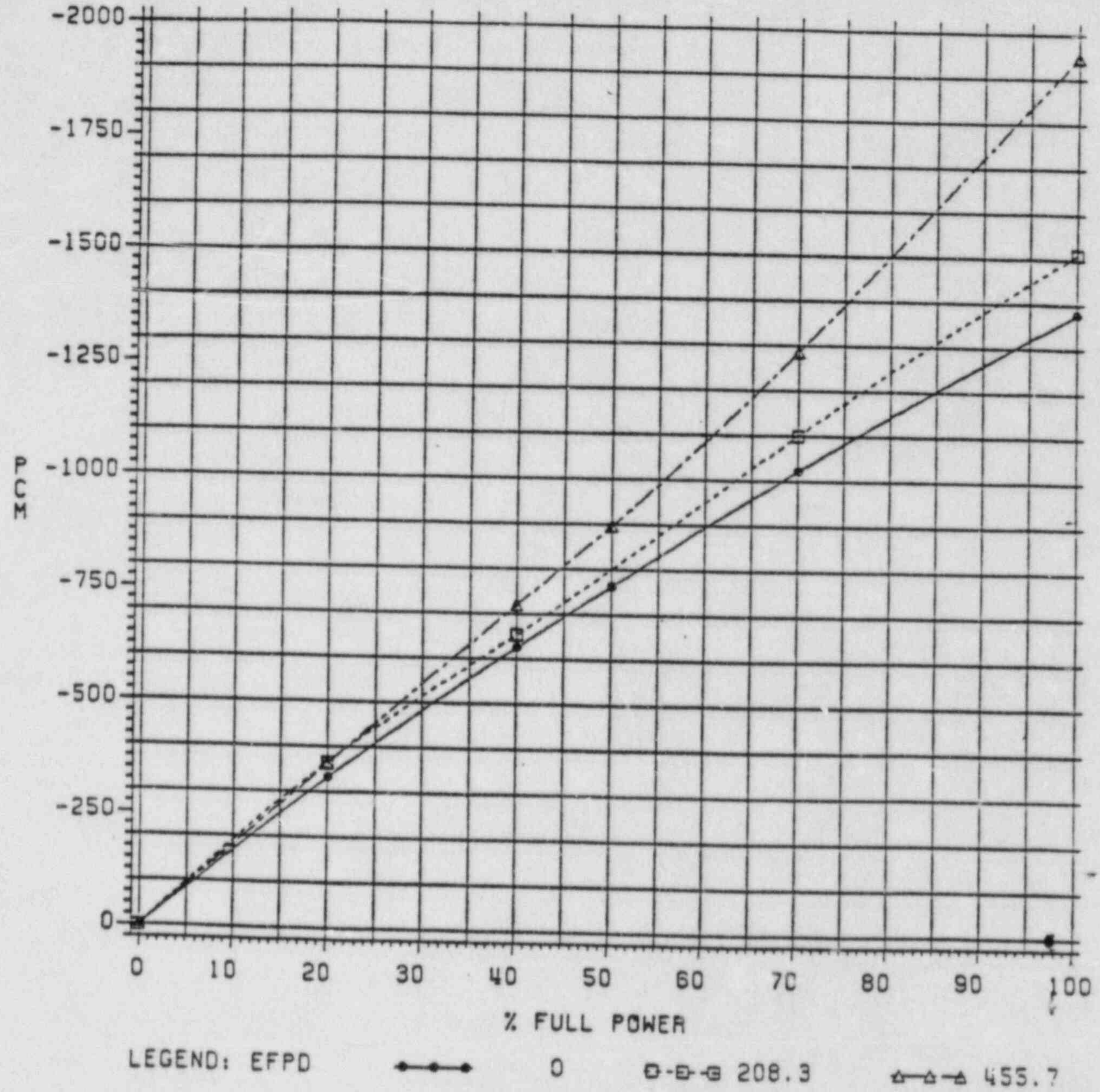
REFERENCE SOURCE OF DATA

V-CE-19010

COMMENTS: T2Z1Z1A, IDENTICAL TO TABLE 3.1.1

APPROVED FOR USE BY: David Hopper 2/27/84
NUCLEAR SUPERVISOR DATE

CURVE 2.1.1
POWER DEFECT VS POWER LEVEL (BOC, MOC, EOC)



PLANT CONDITIONS
0 TO 100% POWER
000 - 455.7 EFPD
COMMENTS: C2Z1Z1, IDENTICAL TO CURVE 3.1.1

REFERENCE SOURCE OF DATA
V-CE-19010

APPROVED FOR USE BY: David Hopper 2/27/84
NUCLEAR SUPERVISOR DATE

TABLE 2.2.1
ISOTHERMAL TEMPERATURE DEFECT VS T-AVG (550° - 565° F, 0% P, BOC)

	ISOTHERMAL TEMPERATURE DEFECT (PCM) BORON CONCENTRATION			
T-AVG (F)	600 PPM	1000 PPM	1500 PPM	2000 PPM
550	119.0	55.4	-11.3	-72.4
551	111.3	51.8	-10.5	-67.6
552	103.5	48.1	-9.8	-63.0
553	95.7	44.5	-9.1	-58.2
554	87.9	40.9	-8.3	-53.4
555	80.0	37.2	-7.6	-48.7
556	72.2	33.6	-6.8	-43.8
557	64.3	30.0	-6.0	-39.0
558	56.3	26.2	-5.3	-34.2
559	48.4	22.5	-4.5	-29.3
560	40.3	18.8	-3.3	-24.5
561	32.4	15.1	-3.0	-19.6
562	24.3	11.3	-2.3	-14.8
563	16.2	7.6	-1.5	-9.8
564	8.1	3.7	-0.8	-5.0
565	0.0	0.0	0.0	0.0

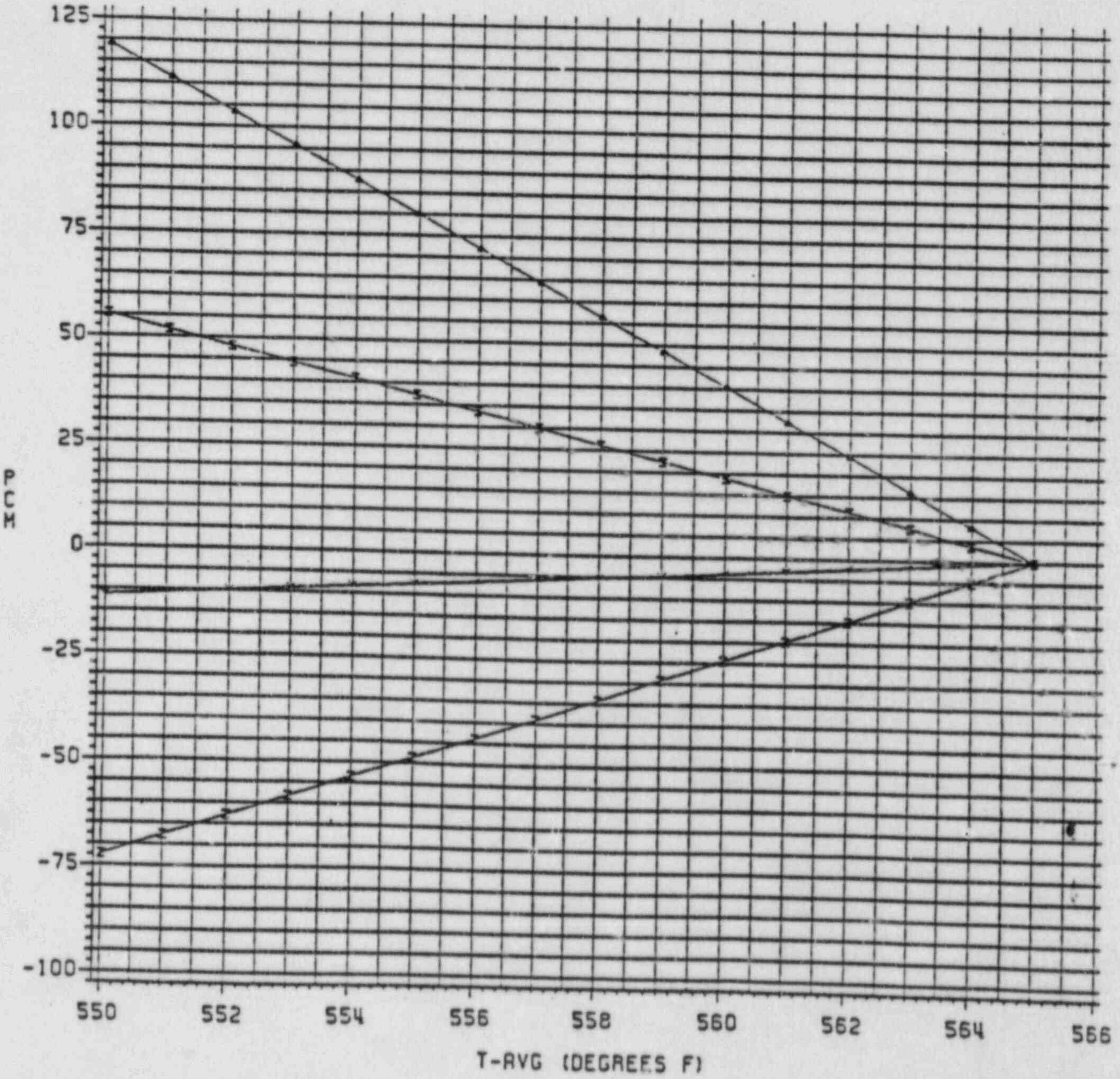
PLANT CONDITIONS
0% POWER
000 EFPD
COMMENTS: T2Z2Z1A, IDENTICAL TO TABLE 3.5.1

REFERENCE SOURCE OF DATA
CALC.FILE- NFMNA-85001

APPROVED FOR USE BY: David Hoover 9-27-85
NUCLEAR SUPERVISOR DATE

ITCBT

CURVE 2.2.1
ISOTHERMAL TEMPERATURE DEFECT VS T-AVG
(550° – 565° F, 0% P, BOC)



PLANT CONDITIONS
0% POWER
000 EFPD
COMMENTS: T2Z2Z1A. IDENTICAL TO CURVE 3.5.1

REFERENCE SOURCE OF DATA
CALC. FILE - NFMNA-85001

APPROVED FOR USE BY: Daniel Hassen (NUCLEAR SUPERVISOR) 9-27-85 (DATE)

ITC8

CONTROLLED BY USER

TABLE 2.2.2
ISOTHERMAL TEMPERATURE DEFECT VS T-AVG
(550° - 565° F, 0% P, MOC)

T-AVG (F)	ISOTHERMAL TEMPERATURE DEFECT (PCM) BORON CONCENTRATION 0 PPM	400 PPM	800 PPM	1200 PPM
550	202.2	110.7	60.0	11.1
551	189.2	103.7	56.4	10.8
552	176.0	96.6	52.7	10.3
553	162.8	89.5	49.0	9.8
554	149.6	82.4	45.3	9.3
555	136.3	75.1	41.4	8.7
556	123.0	67.9	37.5	8.1
557	109.6	60.6	33.6	7.5
558	96.0	53.2	29.5	6.7
559	82.5	45.8	25.5	5.9
560	68.9	38.2	21.4	5.0
561	55.3	30.7	17.2	4.1
562	41.5	23.1	13.0	3.1
563	27.8	15.5	8.7	2.2
564	13.9	7.7	4.4	1.1
565	0.0	0.0	0.0	0.0

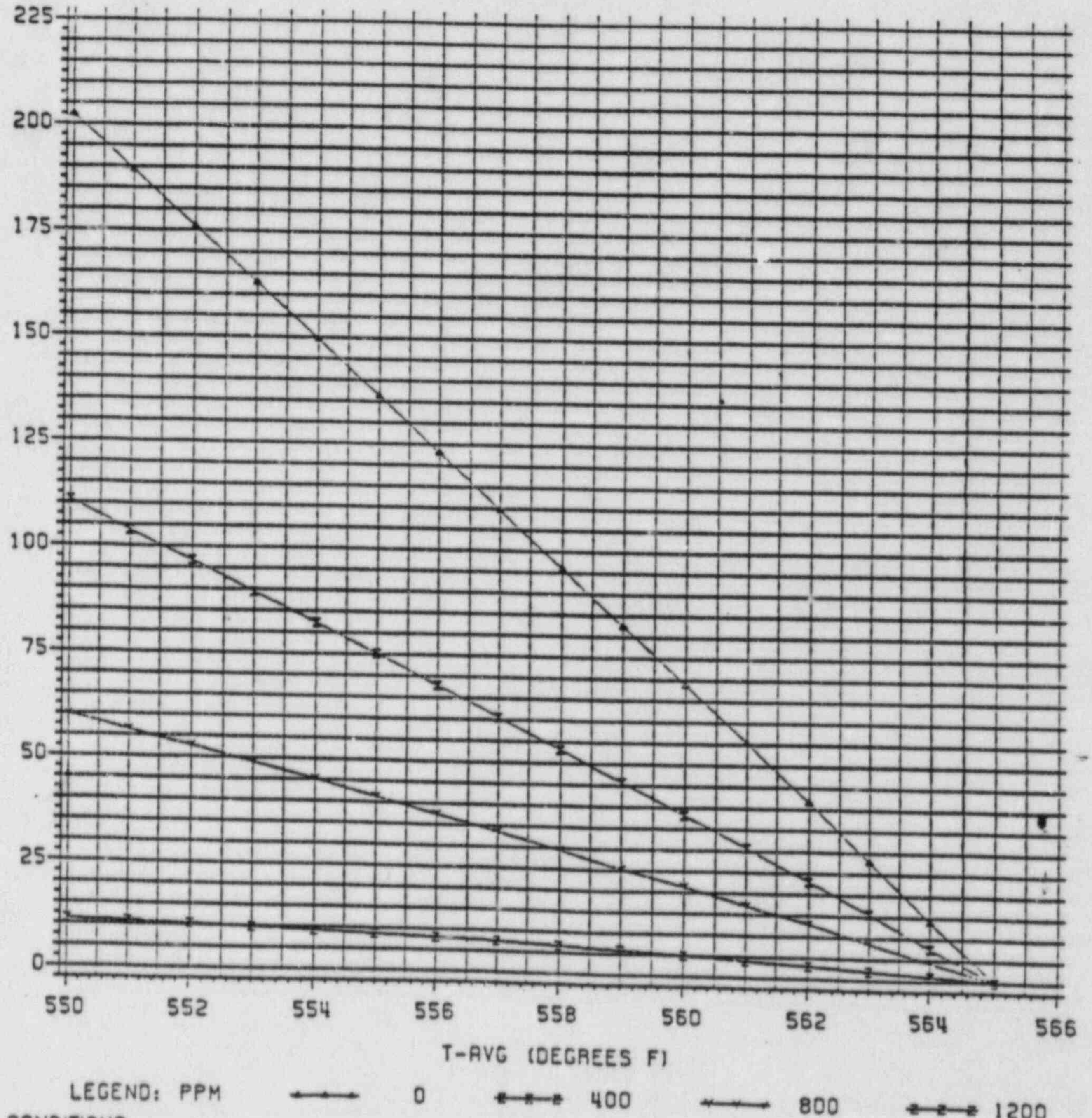
PLANT CONDITIONS
0% POWER
208 EFPD
COMMENTS: T2Z2Z2A, IDENTICAL TO TABLE 3.5.2

REFERENCE SOURCE OF DATA
CALC. FILE NFMNA-85001

APPROVED FOR USE BY: David Hansen 9-27-85
NUCLEAR SUPERVISOR DATE

ITC6T

CURVE 2.2.2
ISOTHERMAL TEMPERATURE DEFECT VS T-AVG
(550° - 565° F, 0% P, MOC)



PLANT CONDITIONS
0% POWER
208 EFPD
COMMENTS: C2Z2Z2, IDENTICAL TO CURVE 3.5.2

REFERENCE SOURCE OF DATA
CALC. FILE NFMNA-85001

APPROVED FOR USE BY: David Hopper 9-27-85
NUCLEAR SUPERVISOR DATE

ITC6

CONTROLLED BY USER

TABLE 2.2.3
ISOTHERMAL TEMPERATURE DEFECT VS T-AVG
(550° – 565° F, 0% P, EOC)

T-AVG (F)	ISOTHERMAL TEMPERATURE DEFECT (PCM) BORON CONCENTRATION 0 PPM	400 PPM	800 PPM	1200 PPM
550	240.7	144.5	68.1	3.7
551	225.3	135.4	64.0	3.8
552	209.6	126.1	59.7	3.8
553	194.0	116.8	55.5	3.8
554	178.3	107.4	51.2	3.8
555	162.5	97.9	46.8	3.6
556	146.6	88.5	42.4	3.5
557	130.7	78.9	38.0	3.3
558	114.6	69.2	33.4	3.0
559	98.5	59.6	28.8	2.8
560	82.3	49.8	24.1	2.4
561	66.0	40.0	19.4	2.0
562	49.6	30.0	14.6	1.5
563	33.2	20.1	9.8	1.1
564	16.6	10.0	4.9	0.5
565	0.0	0.0	0.0	0.0

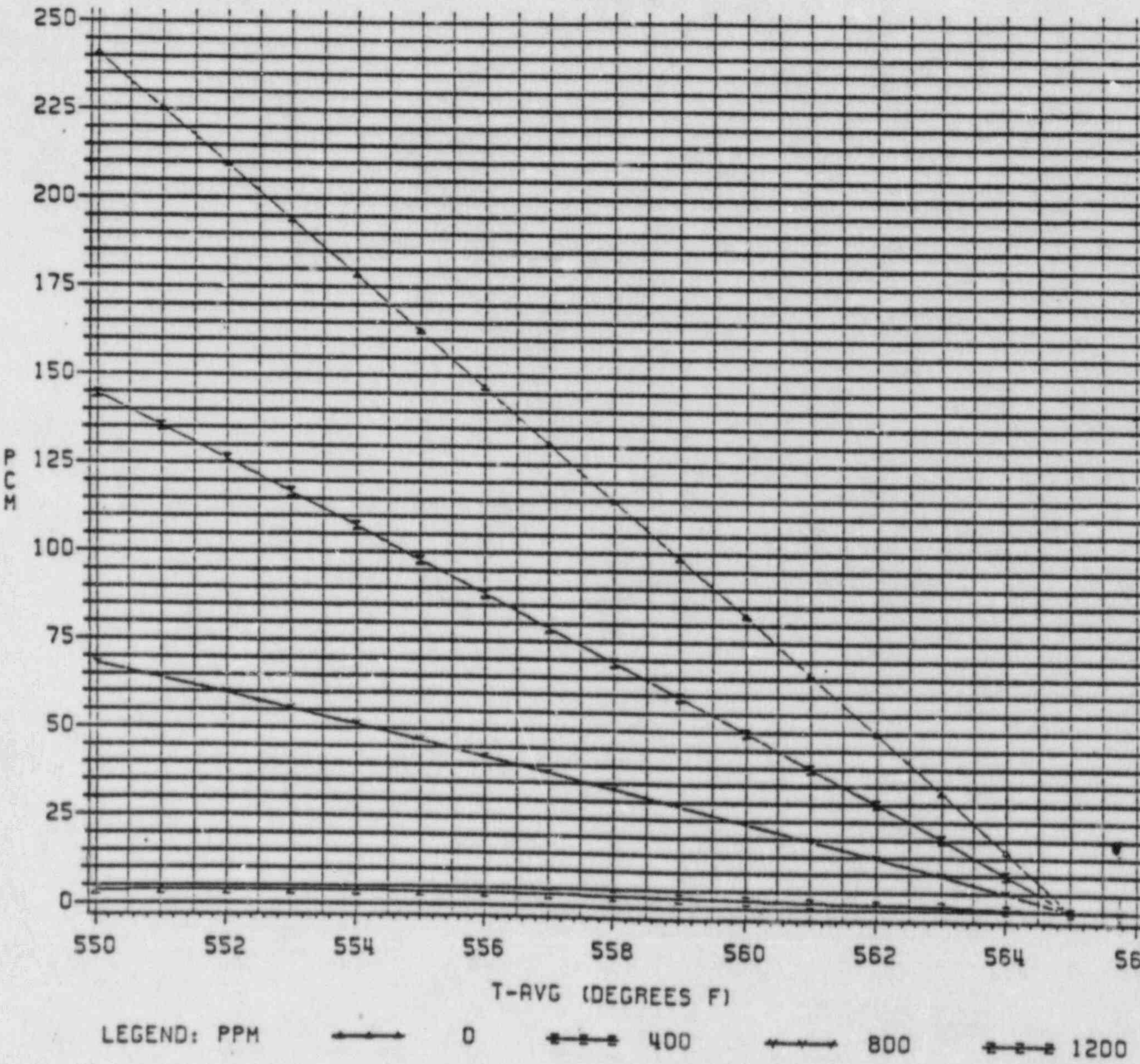
PLANT CONDITIONS
0% POWER
456 EFPD
COMMENTS: T2Z223A, IDENTICAL TO TABLE 3.5.3

REFERENCE SOURCE OF DATA
CALC FILE NFMNA-85001

APPROVED FOR USE BY: David Hopper 9-27-85
NUCLEAR SUPERVISOR DATE

ITC7T

CURVE 2.2.3
ISOTHERMAL TEMPERATURE DEFECT VS T-AVG
(550° - 565° F, 0% P, EOC)



PLANT CONDITIONS
0% POWER
456 EFPD
COMMENTS: C2Z2Z3, IDENTICAL TO CURVE 3.5.3

REFERENCE SOURCE OF DATA
CALC FILE NFMNA-85001

APPROVED FOR USE BY: David Harper 9-27-85
NUCLEAR SUPERVISOR DATE

ITC7

CONTROLLED BY USER

TABLE 2.3.1 (PAGE 1 OF 2)

BORON WORTH VS T–AVG

(505^0 – 615^0 *F*, 100% *P*, *BOC*)

	BORON WORTH (PCM/PPM) BORON CONCENTRATION				
T-AVG (F)	600 PPM	800 PPM	900 PPM	1000 PPM	1200 PP
505	-12.63	-12.44	-12.35	-12.26	-12.08
510	-12.57	-12.38	-12.29	-12.20	-12.02
515	-12.50	-12.31	-12.23	-12.13	-11.96
520	-12.44	-12.25	-12.26	-12.07	-11.90
525	-12.37	-12.19	-12.10	-12.01	-11.84
530	-12.31	-12.12	-12.04	-11.94	-11.78
535	-12.24	-12.05	-11.97	-11.88	-11.72
540	-12.17	-11.99	-11.91	-11.82	-11.65
545	-12.10	-11.92	-11.84	-11.75	-11.59
550	-12.02	-11.85	-11.77	-11.69	[illegible]
555	-11.94	-11.78	-11.70	-11.62	-11.46
560	-11.86	-11.70	-11.62	-11.55	-11.39
565	-11.78	-11.63	-11.54	-11.47	-11.32
570	-11.70	-11.55	-11.47	-11.40	-11.24
575	-11.62	-11.47	-11.38	-11.31	-11.16
580	-11.55	-11.38	-11.30	-11.23	-11.08
585	-11.44	-11.29	-11.21	-11.14	-10.99
590	-11.34	-11.20	-11.12	-11.05	-10.90
595	-11.24	-11.10	-11.02	-10.95	-10.81
600	-11.14	-11.00	-10.92	-10.85	-10.72

PLANT CONDITIONS

100% POWER
0 EFPD
COMMENTS: T2Z3Z1A, IDENTICAL TO TABLE 3.8.1

REFERENCE SOURCE OF DATA

V-CE-[illegible]10

APPROVED FOR USE BY: David Horner 2/27/84
NUCLEAR SUPERVISOR DATE

TABLE 2.3.3 (PAGE 2 OF 2)

BORON WORTH VS T-AVG

(505^0 – 615^0 *F*, 100% *P*, *EOC*)

T-AVG (F)	BORON WORTH (PCM/PPM) BORON CONCENTRATION 0 PPM	100 PPM	200 PPM	400 PPM
605	-11.79	-11.71	-11.64	-11.48
610	-11.66	-11.58	-11.51	-11.35
615	-11.52	-11.44	-11.37	-11.22

PLANT CONDITIONS

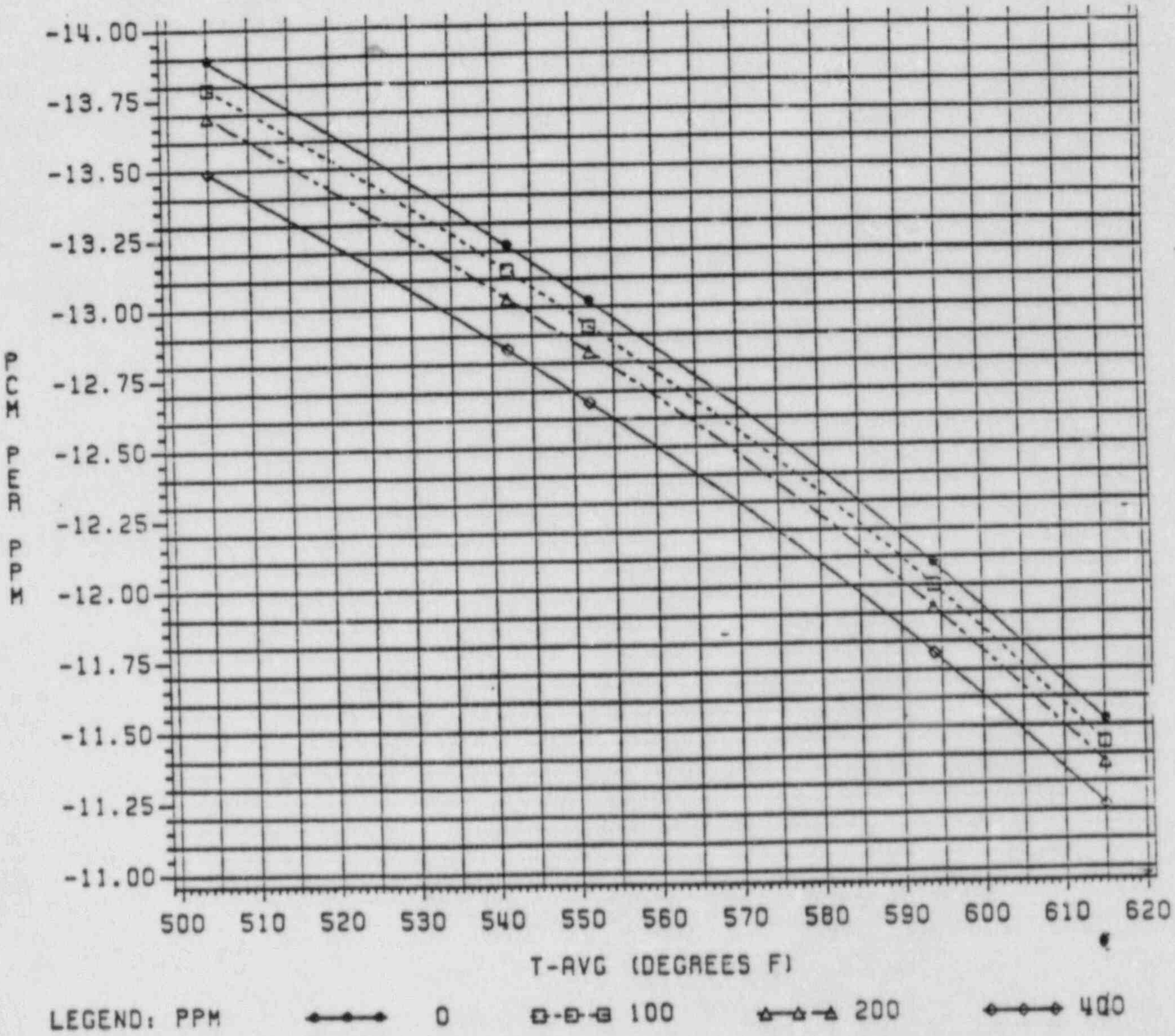
100% POWER
455.7 EFPD
COMMENTS: T2Z3Z3B, IDENTICAL TO TABLE 3.8.3

REFERENCE SOURCE OF DATA

V-CE-19010

CURVE 2.3.3

BORON WORTH VS T-AVG:
(505° - 615° F, 100% P, EOC)



PLANT CONDITIONS
100% POWER
455.7 EFPD
COMMENTS: C2Z3Z3, IDENTICAL TO CURVE 3.8.3

REFERENCE SOURCE OF DATA
V-CE-19010

APPROVED FOR USE BY: David Horn 2/27/84
NUCLEAR SUPERVISOR DATE

TABLE 2.3.4 (PAGE 1 OF 2)

BORON WORTH VS T-AVG

($70^0 - 565^0$ F, 0% P, BOC)

	BORON WORTH (PCM/PPM) BORON CONCENTRATION			
T-AVG (F)	600 PPM	1000 PPM	1500 PPM	2000 PPM
70	-15.02	-14.35	-13.68	-13.18
80	-14.99	-14.32	-13.65	-13.15
100	-14.92	-14.26	-13.59	-13.10
120	-14.85	-14.19	-13.53	-13.05
140	-14.78	-14.13	-13.47	-13.00
160	-14.71	-14.06	-13.41	-12.95
180	-14.63	-13.99	-13.35	-12.89
200	-14.55	-13.91	-13.28	-12.83
220	-14.46	-13.83	-13.21	-12.76
240	-14.37	-13.75	-13.13	-12.69
260	-14.26	-13.65	-13.04	-12.61
280	-14.15	-13.56	-12.95	-12.52
300	-14.04	-13.46	-12.86	-12.43
320	-13.92	-13.35	-12.76	-12.33
340	-13.79	-13.24	-12.65	-12.23
360	-13.66	-13.12	-12.54	-12.12
380	-13.52	-12.99	-12.43	-12.01
400	-13.38	-12.86	-12.30	-11.89
420	-13.22	-12.72	-12.17	-11.76
440	-13.06	-12.57	-12.03	-11.62

PLANT CONDITIONS

0% POWER
0 EFPD
COMMENTS: T2Z3Z4A, IDENTICAL TO TABLE 3.8.4

REFERENCE SOURCE OF DATA

V-CE-19010

APPROVED FOR USE BY: Daniel Hopper 2/27/84
NUCLEAR SUPERVISOR DATE

TABLE 2.3.4 (PAGE 2 OF 2)

BORON WORTH VS T-AVG

(70^0 – 565^0 *F*, 0% *P*, *BOC*)

	BORON WORTH (PCM/PPM) BORON CONCENTRATION			
T-AVG (F)	600 PPM	1000 PPM	1500 PPM	2000 PPM
460	-12.88	-12.41	-11.88	-11.48
480	-12.69	-12.23	-11.72	-11.33
500	-12.48	-12.04	-11.55	-11.16
520	-12.24	-11.83	-11.36	-10.98
540	-11.98	-11.60	-11.16	-10.79
560	-11.70	-11.35	-10.95	-10.58
565	-11.63	-11.29	-10.89	-10.53

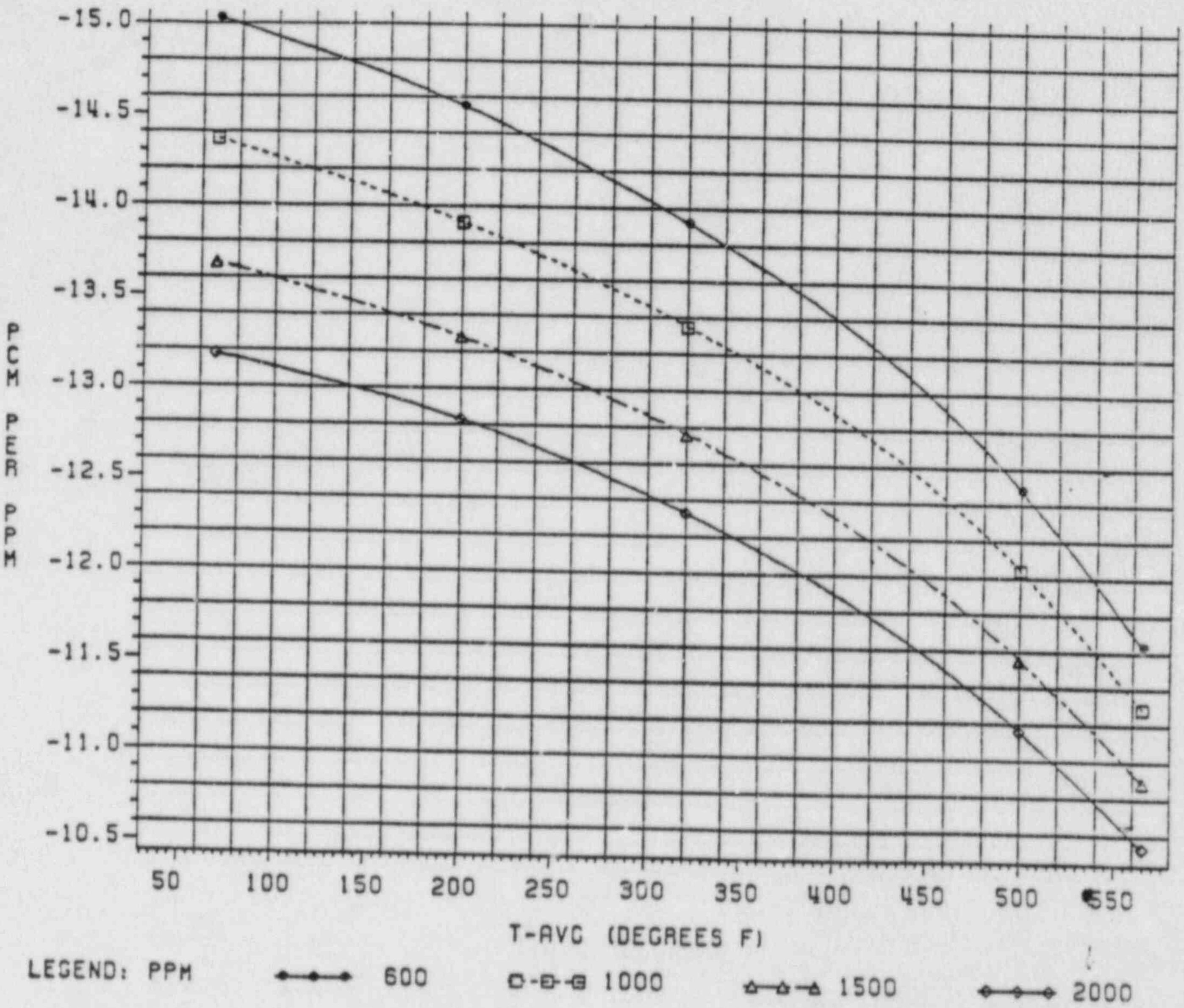
PLANT CONDITIONS

0% POWER
0 EFPD
COMMENTS: T2Z3Z4B, IDENTICAL TO TABLE 3.8.4

REFERENCE SOURCE OF DA

V-CE-19010

CURVE 2.3.4
BORON WORTH VS T-AVG: (70° - 565° F, 0% P, BOC)



PLANT CONDITIONS
0% POWER
000 EFPD
COMMENTS: C2Z3Z4, IDENTICAL TO CURVE 3.8.4.

REFERENCE SOURCE OF DATA
V-CE-19010

APPROVED FOR USE BY: David Horner 2/27/84
NUCLEAR SUPERVISOR DATE

TABLE 2.3.5 (PAGE 1 OF 2)
BORON WORTH VS T-AVG
(70° – 565° F, 0% P, MOC)

T-AVG (F)	BORON WORTH (PCM/PPM) BORON CONCENTRATION 0 PPM	400 PPM	800 PPM	1200 PPM
70	-16.29	-15.52	-14.85	-14.28
80	-16.23	-15.47	-14.81	-14.24
100	-16.11	-15.37	-14.72	-14.16
120	-15.99	-15.26	-14.63	-14.08
140	-15.87	-15.16	-14.54	-14.00
160	-15.75	-15.05	-14.44	-13.92
180	-15.62	-14.93	-14.34	-13.83
200	-15.48	-14.81	-14.23	-13.73
220	-15.33	-14.68	-14.12	-13.63
240	-15.17	-14.54	-14.00	-13.51
260	-15.01	-14.40	-13.86	-13.39
280	-14.83	-14.24	-13.72	-13.26
300	-14.65	-14.08	-13.57	-13.13
320	-14.46	-13.91	-13.42	-12.99
340	-14.26	-13.73	-13.26	-12.84
360	-14.06	-13.55	-13.09	-12.69
380	-13.85	-13.36	-12.92	-12.53
400	-13.63	-13.16	-12.74	-12.36
420	-13.40	-12.95	-12.55	-12.18
440	-13.16	-12.73	-12.35	-11.99

PLANT CONDITIONS

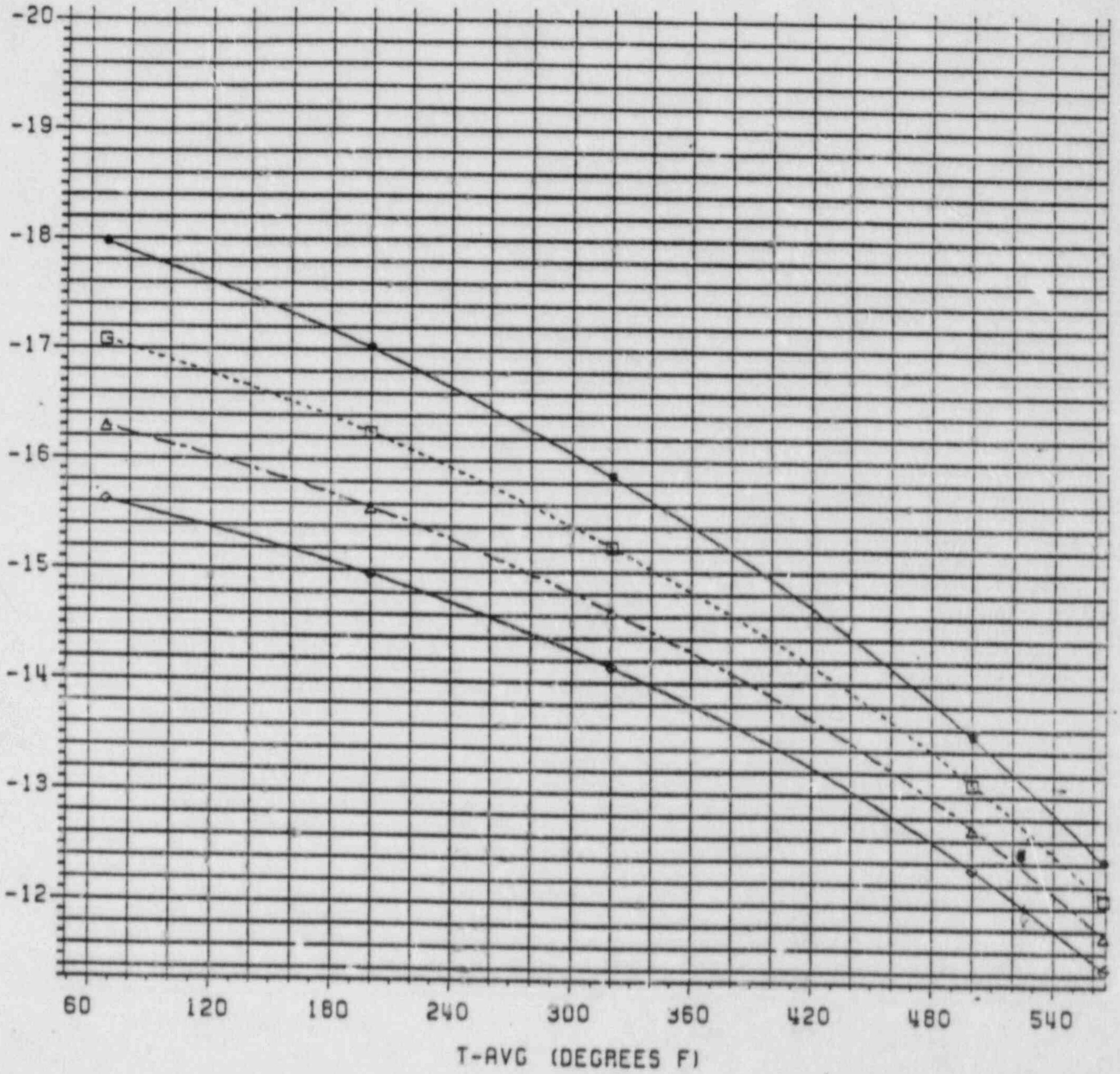
0% POWER
208.3 EFPD
COMMENTS: T2Z3Z5A, IDENTICAL TO TABLE 3.8.5

REFERENCE SOURCE OF DATA

V-CE-190

APPROVED FOR USE BY: David Hoppen 2/27/84
NUCLEAR SUPERVISOR DATE

CURVE 2.3.6
BORON WORTH VS T-AVG: (70° - 565° F, 0% P, EOC)



PLANT CONDITIONS
0% POWER
455.7 EFPD
COMMENTS: C2Z3Z6, IDENTICAL TO CURVE 3.8.6

REFERENCE SOURCE OF DATA
V-CE-19010

APPROVED FOR USE BY: Denis Hoppe 2/27/84
NUCLEAR SUPERVISOR DATE

TABLE 2.4.1
EQ. XENON WORTH VS BURNUP (HFP)

BURNUP (EFPD)	XENON WORTH (PCM)
0.0	0
1.3	-2680
13.0	-2800
26.0	-2780
52.1	-2750
78.1	-2740
104.2	-2720
130.2	-2730
156.3	-2730
182.3	-2720
208.3	-2720
234.4	-2740
260.4	-2750
286.5	-2770
312.5	-2790
338.5	-2800
364.6	-2820
390.6	-2840
416.7	-2860
429.7	-2860
442.7	-2860
455.7	-2860

PLANT CONDITIONS

REFERENCE SOURCE OF D

100% POWER

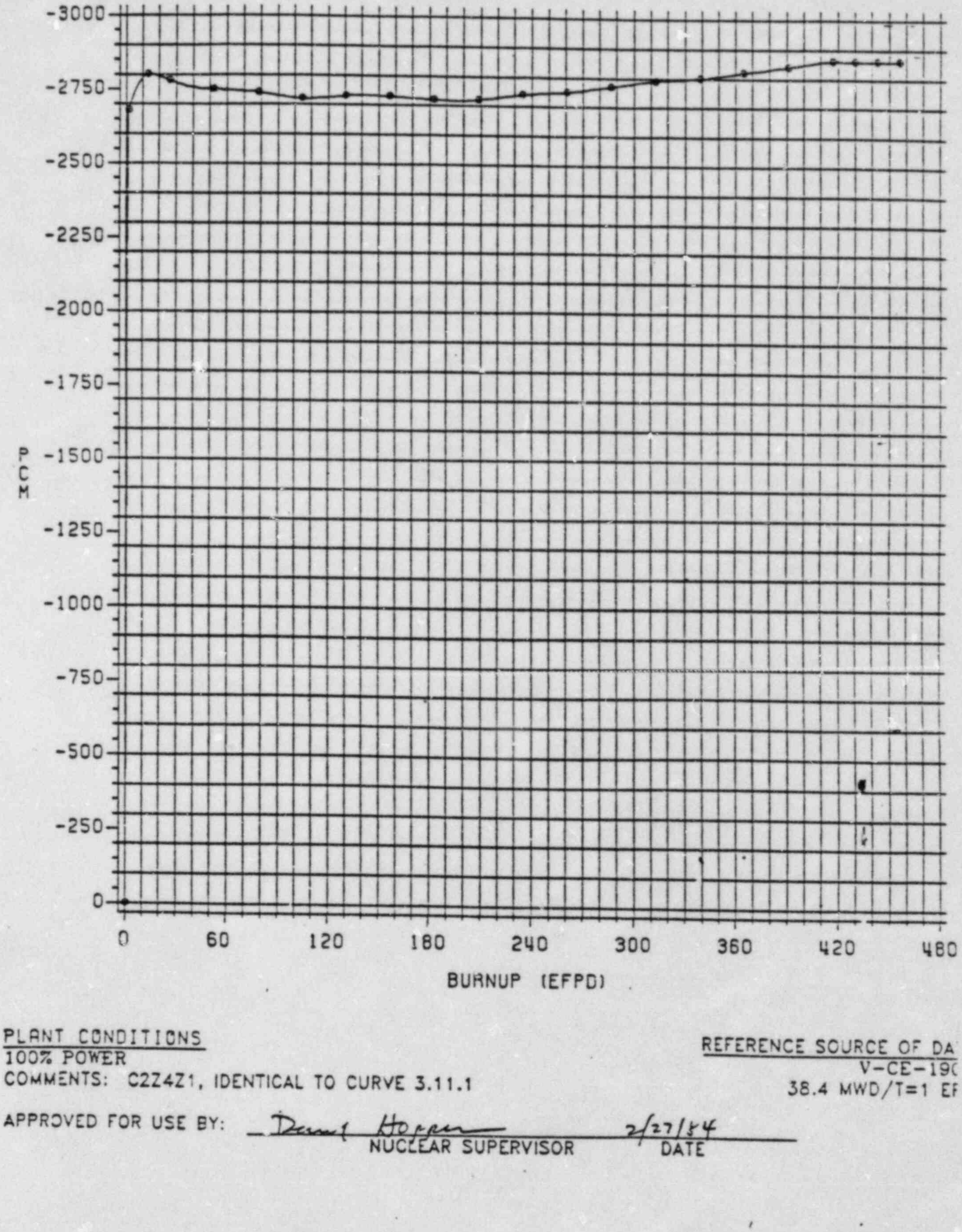
V-C 9

COMMENTS: T2Z4Z1A, IDENTICAL TO TABLE 3.11.1   38.4 MWD/T = . EF

APPROVED FOR USE BY: Dave Hopper   2/27/84
NUCLEAR SUPERVISOR   DATE

CURVE 2.4.1

EQUILIBRIUM XENON WORTH VS BURNUP (HFP



PLANT CONDITIONS
100% POWER
COMMENTS: C2Z4Z1, IDENTICAL TO CURVE 3.11.1

REFERENCE SOURCE OF DA
V-CE-19(
38.4 MWD/T=1 EF

APPROVED FOR USE BY: Dennis Hopper 2/27/84
NUCLEAR SUPERVISOR DATE

TABLE 2.4.2
EQUILIBRIUM XENON WORTH VS POWER LEVEL

POWER LEVEL (%)	EQUILIBRIUM XENON WORTH BOC 0 EFPD	MOC 208.3 EFPD	EOC 455.7 EFPD
0	0	0	0
5	-650	-665	-685
10	-1040	-1100	-1165
15	-1330	-1415	-1510
20	-1540	-1630	-1730
25	-1700	-1800	-1900
30	-1830	-1940	-2065
35	-1930	-2060	-2200
40	-2070	-2160	-2290
45	-2160	-2245	-2385
50	-2250	-2315	-2460
55	-2310	-2385	-2520
60	-2370	-2440	-2590
65	-2440	-2500	-2640
70	-2490	-2540	-2690
75	-2530	-2570	-2730
80	-2560	-2620	-2770
85	-2595	-2655	-2810
90	-2620	-2680	-2840
95	-2645	-2710	-2850
100	-2670	-2725	-2865

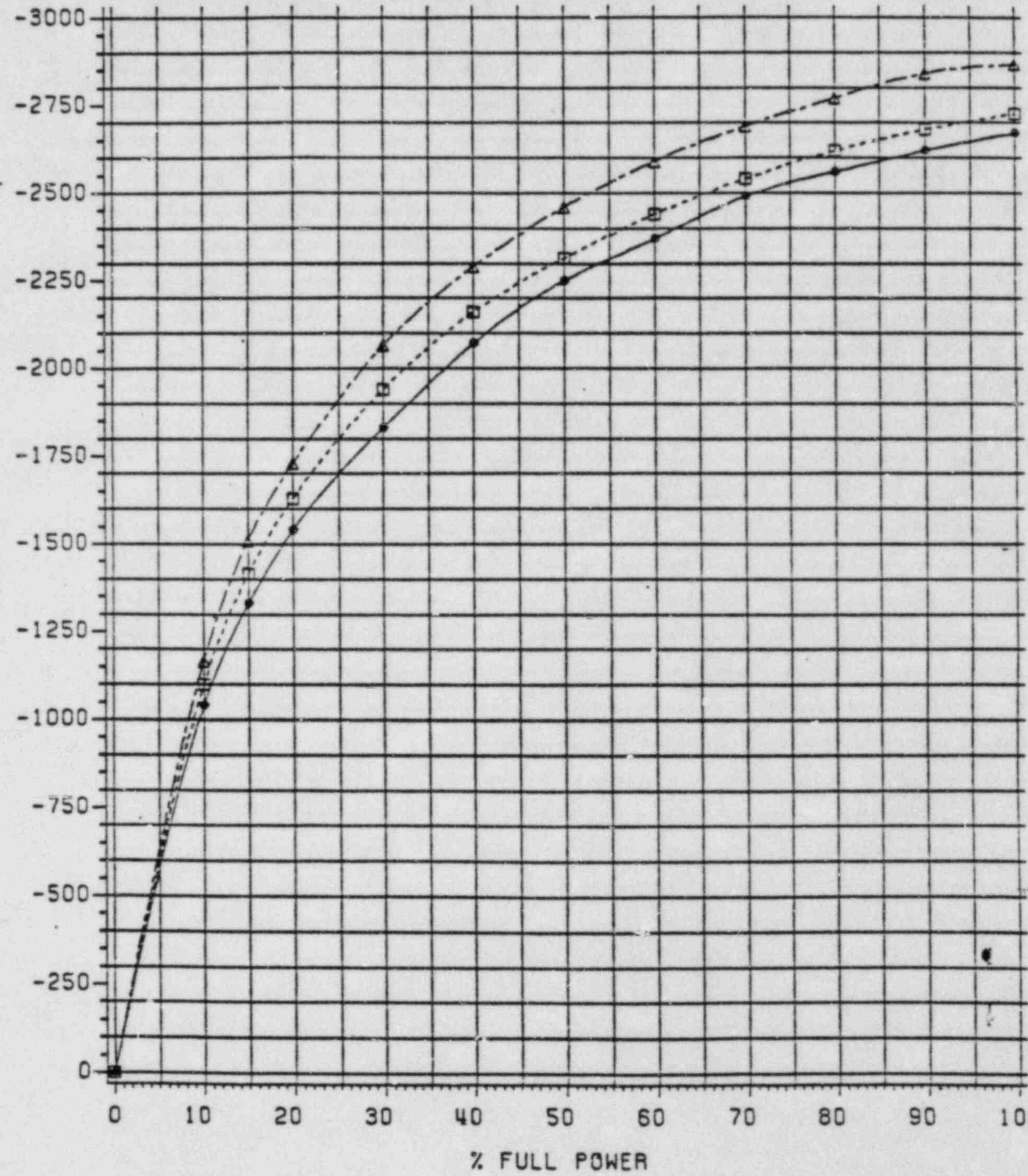
PLANT CONDITIONS
0-455.7 EFPD
COMMENTS: T2Z4Z2A, IDENTICAL TO TABLE 3.11.2

REFERENCE SOURCE OF DATA
V-CE-190

APPROVED FOR USE BY: Dennis Hooper 2/27/84
NUCLEAR SUPERVISOR DATE

CURVE 2.4.2

EQUILIBRIUM XENON WORTH VS POWER LEVEL: BOC, MOC, EOC



PLANT CONDITIONS

100% POWER

COMMENTS: C2Z4Z2, IDENTICAL TO CURVE 3.11.2

REFERENCE SOURCE OF DATA

V-CE-19010

APPROVED FOR USE BY: Daul Hoopes 2/27/84

NUCLEAR SUPERVISOR DATE

TABLE 2.5.1 (PAGE 1 OF 2)
REG GROUP 5 WORTH VS WITHDRAWAL (HFP)

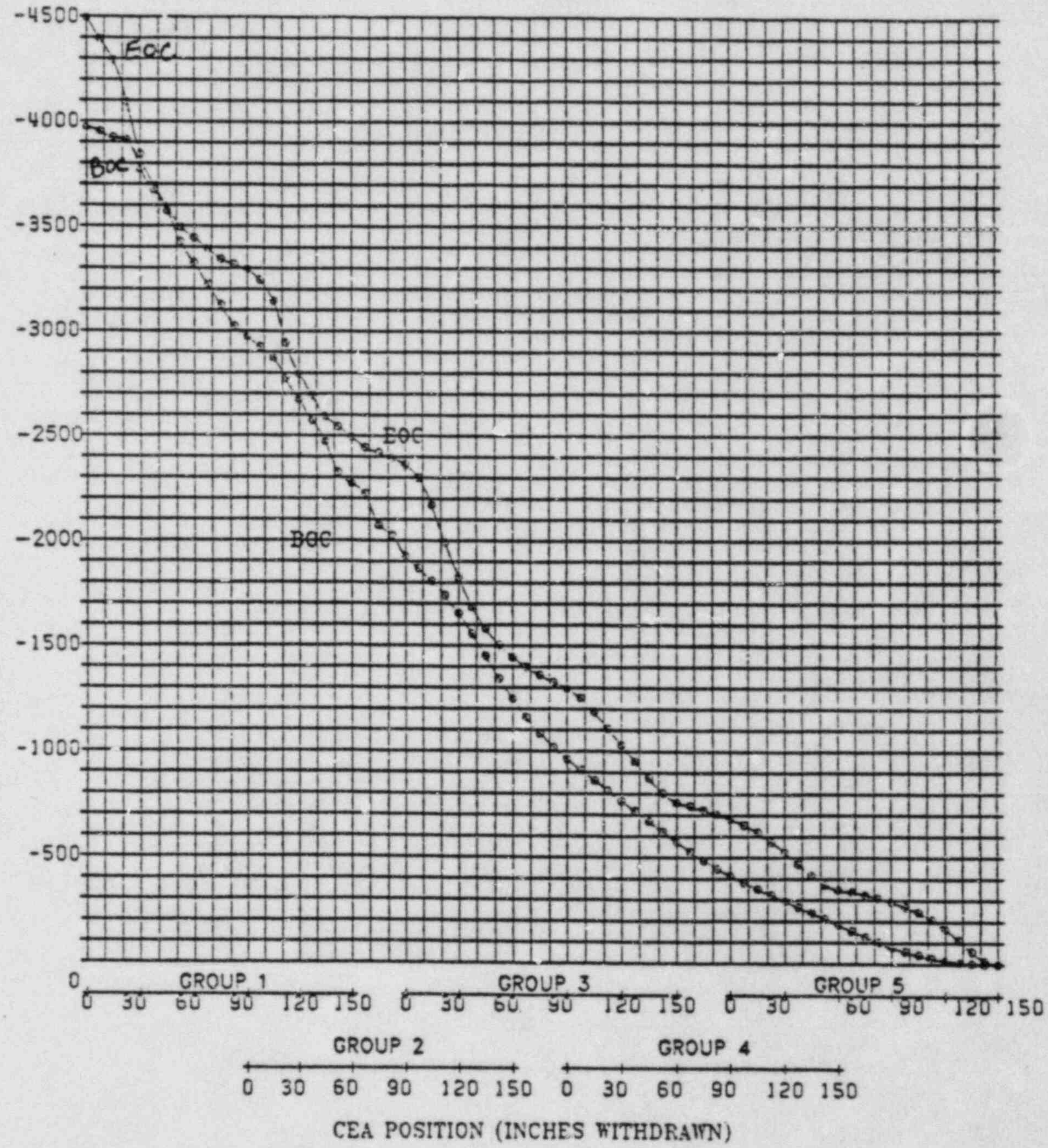
REG GROUP 5 POSITION (INCHES WITHDRAWN)	REG GROUP 5 WORTH (PCM) BOC 0 EFPD	MOC 208.3 EFPD	EOC 455.7 EFPD
0.0	-303.8	-295.1	-319.9
5.0	-302.1	-293.5	-318.3
10.0	-298.5	-290.1	-314.5
15.0	-293.0	-284.9	-308.9
20.0	-285.5	-278.0	-302.0
25.0	-276.3	-269.5	-294.4
30.0	-265.6	-259.9	-286.6
35.0	-253.8	-249.6	-278.9
40.0	-241.1	-238.7	-271.2
45.0	-227.7	-227.3	-263.5
50.0	-213.8	-215.7	-255.8
55.0	-199.7	-203.9	-247.9
60.0	-185.5	-192.0	-239.8
65.0	-171.2	-179.9	-231.4
70.0	-156.9	-167.7	-222.5
75.0	-142.7	-155.3	-213.3
80.0	-128.6	-142.8	-203.5
85.0	-114.8	-130.1	-193.2
90.0	-101.3	-117.3	-182.1
95.0	-88.1	-104.4	-170.1
100.0	-75.3	-91.5	-157.1

PLANT CONDITIONS
100% POWER
0-455.7 EFPD
ALL RODS OUT
COMMENTS: T2Z5Z1A, IDENTICAL TO TABLE 3.12.1

REFERENCE SOURCE OF DATA
V-CE-19010

APPROVED FOR USE BY: David Hooper 2/27/84
NUCLEAR SUPERVISOR DATE

CURVE 2.5.6
REG. CEA WORTH (OVERLAP) VS WITHDRAWAL
(HZP, BOC, EOC)



PLANT CONDITIONS

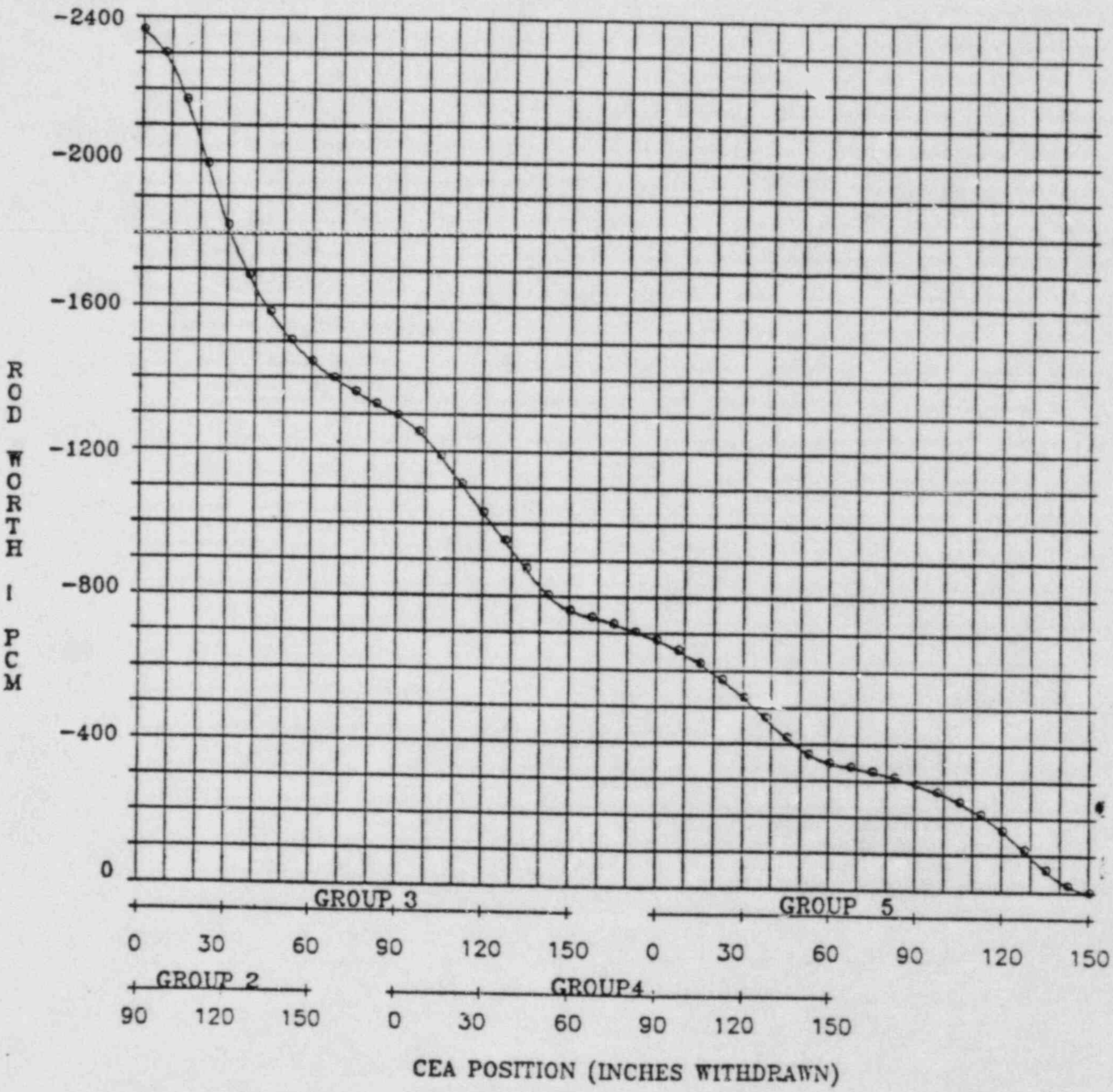
HOT ZERO POWER
BOC = 000 EFPD, EOC = 455.7 EFPD
COMMENTS: C2Z5Z6, IDENTICAL TO CURVE 3.12.6

REFERENCE SOURCE OF DATA
84-NFM-205 & V-CE-19010

APPROVED FOR USE BY: David Hogan 9-27-85
NUCLEAR SUPERVISOR DATE

CONTROLLED BY USER

CURVE 2.5.9
REG CEA WORTH (OVERLAP) VS WITHDRAWAL (HZP, EOC)



PLANT CONDITIONS
HOT ZERO POWER
455.7 EFPD
COMMENTS: C2Z5Z9, IDENTICAL TO CURVE 3.12.9

REFERENCE SOURCE OF DATA
V-CE-19010

APPROVED FOR USE BY: Dav-l Hoppes 2/27/84
NUCLEAR SUPERVISOR DATE

EQUATION FORMULA AND PARAMETER SHEET

Where $\dot{m}_1 = \dot{m}_2$

$(\text{density})_1(\text{velocity})_1(\text{area})_1 = (\text{density})_2(\text{velocity})_2(\text{area})_2$

$KE = \frac{mv^2}{2}$ $PE = mgh$ $PE_1+KE_1+P_1V_1 = PE_2+KE_2+P_2V_2$ where V = specific volume
P = Pressure

$Q = \dot{m}c_p(T_{out}-T_{in})$ $Q = UA\ (T_{ave}-T_{stm})$ $Q = \dot{m}(h_1-h_2)$

$P = P_o 10^{(SUR)(t)}$ $P = P_o e^{t/T}$ $SUR = \frac{26.06}{T}$ $T = \frac{(\beta - \rho)}{\rho}$ $\ell_m = \frac{(\beta - \rho)}{\rho\,\lambda_{eff}}$

$\lambda_{eff} = 0.08\ sec^{-1}$

$\text{delta } K = (K_{eff}-1)$ $CR_1(1-K_{eff1}) = CR_2(1-K_{eff2})$ $CR = S/(1-K_{eff})$

$M = \frac{(1-K_{eff1})}{(1-K_{eff2})}$ $SDM = \frac{(1-K_{eff}) \times 100\%}{K_{eff}}$

$\text{decay constant} = \frac{\ln(2)}{t_{1/2}} = \frac{0.693}{t_{1/2}}$ $A_1 = A_o e^{-(\text{decay constant}) \times (t)}$

<u>Water Parameters</u>

1 gallon = 8.345 lbs
1 gallon = 3.78 liters

1 ft^3 = 7.48 gallons

Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3

Heat of Vaporization = 970 Btu/lbm
Heat of Fusion = 144 Btu/lbm
1 Atm = 14.7 psia = 29.9 in Hg

1 ft H_2O = 0.4335 $lbf/in.^2$

<u>Miscellaneous Conversions</u>

1 Curie = 3.7×10^{10} dps
1 kg = 2.21 lbs

1 hp = 2.54×10^3 Btu/hr

1 MW = 3.41×10^6 Btu/hr
1 Btu = 778 ft-lbf

Degrees F = (1.8 x Degrees C) + 32
1 inch = 2.54 centimeters
g = 32.174 $ft\text{-}lbm/lbf\text{-}sec^2$
