Enclosure (1)

REPORT DETAILS

Examination Report No.: 50-528/OL-89-01

Facility Licensee: Palo Verde Nuclear Generating Station

Facility Docket No.: 50-528, 50-529, 50-530

Facility License No.: NPF-41, NPF-51, NPF-74

Examinations administered at Palo Verde Nuclear Generating Station, Wintersburg, Arizona.

Chief Examiner:	Michael J. Royack Licensing Examiner	9/21/89 Date Signed
Examiner:	David Pereira Licensing Examiner	9/21/89 Date Signed
Examiner:	T B Sundsmo Todd Sundsmo Licensing Examiner	9/21/89 Date Signed
Approved:	Lew Miller Chief, Operations Section	9/26/89 Date Signed

Examinations on August 22 through 24, 1989, and August 29, 1989.

Written and operating examinations were administered to eight (8) Senior Reactor Operator and one (1) Senior Reactor Operator Instant Candidates. Seven (7) of the nine (9) candidates passed these examinations. One (1) of the nine (9) candidates failed the written examination, and one (1) of the nine (9) candidates failed both the written and operating examinations.

8910180053 890929
PDR ADOCK 05000528
V PDC

REPORT DETAILS

1. Examiners

   Michael J. Royack, RV, Chief Examiner
   David B. Pereira, RV, Examiner
   Todd Sundsmo, RV, Examiner

2. Persons Attending the Exit Meeting

   M. J. Royack, RV, Chief Examiner
   T. Sundsmo, RV, Examiner
   J. Haynes, Vice President Nuclear
   W. Marsh, Plant Director
   J. Bailey, Unit 3 Assistant Plant Manager
   J. Scott, Unit 1 Operations Manager
   D. Ensign, Unit 2 Operations Manager
   D. Gouge, Unit 3 Operations Manager
   J. Allen, Relief Plant Manager
   W. Fernow, Training Manager
   D. Craig, Operations Training Manager
   D. Brown, Simulator Project Manager
   T. Bradish, Compliance Supervisor
   L. Clyde, STA Supervisor
   B. Ballard Sr., Director QA
   T. Cannon, Lead Simulator Instructor
   J. Stavely Jr., Lead Simulator Engineer

3. Written Examination and Facility Review

   The written examinations were administered on August 22, 1989 at a licensee contracted examination facility. Seven (7) of the nine (9) operators who took the written examination passed.

   The written examination was prevalidated on August 14, 1989 by a combined NRC operator licensing examiner team and licensee training and operations team. The NRC members who reviewed the examination with the facility representatives were Messrs. M. Royack, D. Pereira, and T. Sundsmo. The facility representatives who conducted the pre-review of the examination are documented in Attachment A, "Written Examination Security Agreements", which also serves as documentation of examination security.

   All comments and concerns presented by the licensee review personnel were addressed by the NRC examination team. All revisions or corrections made to the examination were agreed upon by both the licensee and NRC review teams. After the written examination review was completed all of the written examination material was collected by the NRC examination team to insure examination security.

At the conclusion of the written examination a review of the examination was conducted by a licensee representative and the Chief Examiner to verify that all of the licensee pre-review comments were incorporated. The candidate's examinations were then graded in accordance with the corrected key. After the review was complete, one question, 6.04, was deleted from the examination due to the unexpected large number of incorrect responses..

4. Operating Examinations

The operating portion of the examinations were administered on August 23, 24 and 29, 1989. Eight (8) of the nine (9) candidates passed the operating portion of the examination.

As documented in Enclosure 4, Simulation Facility Report, the facility's site specific simulator continued to display fidelity and operating problems. Fidelity and operating problems were experienced on the simulator during the administration of the examination even though the simulator scenarios were prevalidated with the licensee examination team on August 14 and 15, 1989. Attachment B, Operating Examination Security Agreements, documents the security measures required for the prevalidation of the operating examination.

The examiners identified a generic weakness concerning candidates' failure to reference plant operating procedures during the simulator portion of the examination. The failure to use procedures to perform (or verify) system lineups during both normal and abnormal evolutions appeared to be a generic weakness in operator performance and training. This weakness was identified specifically in the areas of reactor coolant system dilution, boration, and emergency boration. The operators performed these evaluations from memory and without reference to operating procedures. There were instances when the tasks were "completed," but were not completed in compliance with the steps designated in the appropriate operating procedure.

The operator's reluctance to reference normal operating procedures, and the subsequent management defense of this poor practice at the Exit Interview, is a significant weakness, and a safety concern.

10 CFR 50 Appendix B, Criterion V, Instructions, Procedures and Drawings clearly requires that safety related activities be conducted in accordance with procedures. The detail, frequency of revision, and predictable fallibility of memory, strongly indicate that applicable procedures should be referenced prior to and during the performance of such activities.

A second weakness was identified by the examiners during the administration of the walkthrough section of the examination. This weakness was in the area of identification and operation of electrical equipment in the plant.

The two specific areas here were:

(1) Identifying safety related DC power equipment and its operation, and

(2) Knowledge of the proper method of locally tripping electrical breakers, specifically the reactor trip breakers.

The examiners followed up on LER 1-88-016, Reactor Trip Following Earlier Than Anticipated Criticality. The operators questioned were able to reference correct procedures and steps to be taken in the event of an "earlier than anticipated criticality".

5. Exit Meeting

On August 30, 1989 NRC examiners met with representatives of the licensee staff to discuss the examination.

In response to NRC examiner comments regarding the use of plant operating procedures, a licensee representative stated that the operators were not required to use or reference plant procedures when performing normal or frequent plant evolutions and that it was common for the operators to perform evolutions from memory. However, the representative stated that the depth of the problem would be reviewed by the staff.

The NRC examiners were informed by the licensee staff that the generic problem of electrical system knowledge and operation had been identified by earlier internal audits. The licensee staff member indicated that the problem was being addressed by scheduling a block of time in the requalification and training cycles for electrical systems.

50-528/OL-89-01

ATTACHMENT A

WRITTEN EXAMINATION SECURITY AGREEMENTS

ATTACHMENT 2 (continued)

Enclosure 4

REQUIREMENTS FOR FACILITY REVIEW OF WRITTEN EXAMINATIONS

1. At the option of the Chief Examiner, the facility may review the written examination up to two weeks prior to its administration. This review may take place at the facility or in the Regional office. The Chief Examiner will coordinate the details of the review with the facility. An NRC examiner will always be present during the review.

Whenever this option of examination review is utilized, the facility reviewers will sign the following statement prior to being allowed access to the examination. The examination or written notes will not by retained by the facility.

a. Pre-Examination Security Agreement

I JEFF MAHAN agree that I will not knowingly divulge any information concerning the replacement (or initial) examination scheduled for AUG. 22, 1989 to any unauthorized persons. I understand that I am not to participate in any instruction involving those reactor operator or senior reactor operator applicants scheduled to be administered the above replacement (or initial) examination from now until after the examination has been administered. I understand that violation of this security agreement could result in the examination being voided.

Jeff Mahan 8/14/89
Signature/Date

In addition, the facility staff reviewers will sign the following statement after the written examination has been administered.

b. Post-Examination Security Agreement

I JEFF MAHAN did not, to the best of knowledge, divulge any information concerning the written examination administered on AUG 22, 1989 to any unauthorized persons. I did not participate in providing any instruction to those reactor operator and senior reactor operator applicants who were administered the examination from the time that I was allowed access to the examination.

Jeff Mahan 8/23/89
Signature/Date

ATTACHMENT 2 (continued)

Enclosure 4

REQUIREMENTS FOR FACILITY REVIEW OF WRITTEN EXAMINATIONS

1. At the option of the Chief Examiner, the facility may review the written examination up to two weeks prior to its administration. This review may take place at the facility or in the Regional office. The Chief Examiner will coordinate the details of the review with the facility. An NRC examiner will always be present during the review.

   Whenever this option of examination review is utilized, the facility reviewers will sign the following statement prior to being allowed access to the examination. The examination or written notes will not by retained by the facility.

   a. Pre-Examination Security Agreement

      I Thomas Cannon agree that I will not knowingly divulge any information concerning the replacement (or initial) examination scheduled for Aug 22, 1989 to any unauthorized persons. I understand that I am not to participate in any instruction involving those reactor operator or senior reactor operator applicants scheduled to be administered the above replacement (or initial) examination from now until after the examination has been administered. I understand that violation of this security agreement could result in the examination being voided.

      8-14-89
      Signature/Date

   In addition, the facility staff reviewers will sign the following statement after the written examination has been administered.

   b. Post-Examination Security Agreement

      I Thomas Cannon did not, to the best of knowledge, divulge any information concerning the written examination administered on Aug 22, 1989 to any unauthorized persons. I did not participate in providing any instruction to those reactor operator and senior reactor operator applicants who were administered the examination from the time that I was allowed access to the examination.

      8-22-89
      Signature/Date

50-528/OL-89-01

ATTACHMENT B

OPERATING EXAMINATION SECURITY AGREEMENTS

Pre-Examination Security Agreement

I, JACK BAILEY agree that I will not knowingly divulge any information concerning the operating tests scheduled for AUGUST 23-30, 1989 to any unauthorized persons. I understand that I am not to participate in any instruction involving those reactor operator or senior reactor operator applicants scheduled to be administered the above operating test from now until after the examination has been administered.

Jack A. Bailey 8/15/89
Signature/Date

Post-Examination Security Agreement

I JACK BAILEY did not, to the best of my knowledge, divulge any information concerning the operating tests administered on Aug 23-30 to any unauthorized persons. I did not participate in providing any instruction to those reactor operator and senior reactor operator applicants who were administered the operating test from the time I was allowed access to the operating test.

Jack A. Bailey 8/30/89
Signature/Date

Pre-Examination Security Agreement

I Thomas Cannon agree that I will not knowingly divulge any information concerning the operating tests scheduled for 8-23 thru 8-30-1989 to any unauthorized persons. I understand that I am not to participate in any instruction involving those reactor operator or senior reactor operator applicants scheduled to be administered the above operating test from now until after the examination has been administered.

8-14-89
Signature/Date

Post-Examination Security Agreement

I Thomas Cannon did not, to the best of my knowledge, divulge any information concerning the operating tests administered on 8-23 thru 8-29-89 to any unauthorized persons. I did not participate in providing any instruction to those reactor operator and senior reactor operator applicants who were administered the operating test from the time I was allowed access to the operating test.

8-29-89
Signature/Date

Pre-Examination Security Agreement

I Mark A. McGhee agree that I will not knowingly divulge any information concerning the operating tests scheduled for 8-23-89 To 8-30-89 to any unauthorized persons. I understand that I am not to participate in any instruction involving those reactor operator or senior reactor operator applicants scheduled to be administered the above operating test from now until after the examination has been administered.

Mark A. McGhee
Signature/Date

Post-Examination Security Agreement

I Mark A. McGhee did not, to the best of my knowledge, divulge any information concerning the operating tests administered on 8-23-89 To 8-30-89 to any unauthorized persons. I did not participate in providing any instruction to those reactor operator and senior reactor operator applicants who were administered the operating test from the time I was allowed access to the operating test.

Mark A. McGhee 8-30-89
Signature/Date

Pre-Examination Security Agreement

I JAMES M. STAVELY, JR. agree that I will not knowingly divulge any information concerning the operating tests scheduled for 8/23 - 8/30/89 to any unauthorized persons. I understand that I am not to participate in any instruction involving those reactor operator or senior reactor operator applicants scheduled to be administered the above operating test from now until after the examination has been administered.

James M. Stavely, Jr. 8/10/89
Signature/Date

Post-Examination Security Agreement

I JAMES M. STAVELY JR. did not, to the best of my knowledge, divulge any information concerning the operating tests administered on 8/23 - 8/30 to any unauthorized persons. I did not participate in providing any instruction to those reactor operator and senior reactor operator applicants who were administered the operating test from the time I was allowed access to the operating test.

James M. Stavely 8/29/89
Signature/Date

Enclosure (4)

PALO VERDE NUCLEAR GENERATING STATION
AUGUST 1989 REPLACEMENT EXAMINATION
SIMULATION FACILITY FIDELITY REPORT

PALO VERDE NUCLEAR GENERATING STATION

AUGUST 1989 REPLACEMENT EXAMINATION

SIMULATION FACILITY FIDELITY REPORT

The simulator scenarios prepared by the NRC examiners for the examination were prevalidated by licensee representatives and the NRC examination team. The NRC prepared scenarios were developed using the facility supplied initial condition (IC) list and the simulator malfunction list. The IC list and simulator malfunction list indicated a low level of simulator capability as indicated below.

The IC and simulator malfunction lists supplied for the preparation of the examination were limiting in that they did not provide a full spectrum of plant initial conditions or failures.

The ICs provided included a list of twelve (12) ICs. The ICs varied from 2E-8% to 100% power. These initial conditions are only modeled for beginning of core cycle (BOC) and only for Unit 1 core. The ICs from 2E-8% to 40% power did not have known xenon conditions indicated. This is sparse coverage for a unit with three reactors which are operated at different stages of core life.

The simulator malfunction list limited the equipment and instrument failures to selected items.. Examples of the limitations are:

Only being able to fail an NI pre-amp on safety channels A or B, not A, B, C, and D.

Only being able to fail a safety channel NI middle detector on channels C and D, not A, B, C, and D.

Only being able to fail a safety channel NI upper detector on channel D, not A, B, C, and D.

Only being able to simulate a water space leak on one safety injection tank.

Only being able to simulate a gas space leak on one safety injection tank.

Only being able to fail one of the LPSI pumps on an SIAS. Failure of either HPSI or the other LPSI not included in malfunctions.

Similar cases of the inability to fail specific instrumentation or components were encountered during the examination prevalidation. Numerous malfunctions listed on the simulator malfunction list required extended overrides and simulator operator input for the malfunction to perform properly.

During the administration of the simulator portion of the operating examination several simulator anomalies occurred. The anomalies which occurred during the examination did not occur in the prevalidation of the examination, nor could the same anomaly be repeated after the scenario was run.

Specific events or simulator malfunctions are as follows:

a. Computer trend recorder failure during scenario run.

b. Reactor power cutback system would not manually go into manual select, apparent hardware problem.

c. Reactor coolant pump (RCP) seal controller failed to the zero position without simulator operator or board operator actions. The RCP seal bypass flow should have been sufficient to cool the seal, however the seal temperatures did increase uncharacteristically.

d. Auxiliary feed water valves to steam generator 2, HV-33 and 37, did not open or respond as required by automatic signals.

e. Feed flow to steam generator 2 was lost even though the auxiliary feed water pump was lined up to feed the steam generator.

f. Steam generator downcomer controller, SGN-FIK-1123, operated erratically, jumping from zero to mid-scale.

g. Motor driven auxiliary feed water pump operating without indication of amps.

h. Low auxiliary feed water system pressure for no apparent reason. Steam generator levels were maintained despite the lower than required pressure indication.

i. Main feed water pumps failed to trip with plant operating at greater than 80% with only one (1) condensate pump in operation. This event was specifically reviewed and enhanced during the pre-validation review.

j. Main feed water pump minimum flow valve did not open as required when main feed water pump turbine was reset.

k. A high T cold of 595^{O}F, and increasing, occurred after a trip of a main feed water pump and subsequent reactor power cut back for no apparent reason. Event was not able to be repeated.

l. Following a condensate pump trip a low level in a condenser hot well occurred. This condition was not credible since the hot wells are cross connected. The low condenser level did not occur during the prevalidation of the scenario, therefore, this appears to be a modeling problem.

m. The non essential feed water pump indicated approximately 60 amps at a low flow when the amp reading should have been approximately 100 amps.

n. A steam generator repressurized after blowing down from a blow down line rupture. This is a known simulator problem previously identified by the facility.

o. Loss of process radiation monitoring system due to computer "lock up" during steam generator blow down scenario. Failure did not occur during prevalidation.

p. Flow could not be established through the feed water block valve bypasses to the steam generators. The bypass valves are the same size as the block valves therefore flow should have been indicated. This condition caused operator confusion as to the cause of the no feed water flow problem. Once the block valves were opened flow was established.

As Given Key
&
Graded To Key

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR OPERATOR LICENSE EXAMINATION
REGION V

*****KEY*****

FACILITY: Palo Verde Nuclear Generatig Station

REACTOR TYPE: CE - PWR

DATE ADMINISTERED: August 22, 1989

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
24.00	~~24.00~~ 24.2 R			4.	REACTOR PRINCIPLES (7%) THERMODYNAMICS (7%) AND COMPONENTS (10%) [FUNDAMENTALS EXAM)
33.00	~~33.00~~ 33.3 R			5.	EMERGENCY AND ABNORMAL PLANT EVOLUTIONS (33%)
~~43.00~~ 42.00 R	~~43.00~~ 42.4 R			6.	PLANT SYSTEMS (30%) AND PLANT-WIDE GENERIC RESPONSIBILITIES (13%)
~~100.00~~ 99.0 R	*****KEY***** FINAL GRADE		%		TOTALS

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

************KEY************
Candidate's Signature

Reviewed and accepted

Michael J. Royack

Thomas C. Cannon

ATTACHMENT 2 (continued)

Enclosure 3

PROCEDURES FOR THE ADMINISTRATION OF WRITTEN EXAMINATIONS

1. Check identification badges.

2. Pass out examinations and all handouts. Remind applicants not to review examination until instructed to do so.

READ THE FOLLOWING INSTRUCTIONS VERBATIM:

During the administration of this examination the following rules apply:

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

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

READ THE FOLLOWING INSTRUCTIONS

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

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

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

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

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

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

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

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

ATTACHMENT 2 (Continued)
Enclosure 3 (Continued)

9. If you are using separate sheets, number each answer as to category and number (i.e. 1.04, 6.10) and skip at least 3 lines between answers to allow space for grading.

10. Write "End of Category ____" at the end of your answers to a category.

11. Start each category on a new page.

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

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

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

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

16. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK. NOTE: partial credit will NOT be given on multiple choice questions. 5/17/8 Interac

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

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

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

20. To pass the examination, you must achieve an overall grade of 80% or greater and at least 70% in each category.

21. There is a time limit of (6) hours for completion of the examination. (or some other time if less than the full examination is taken.)

22. When you are done and have turned in your examination, leave the examination area (DEFINE THE AREA). If you are found in this area while the examination is still in progress, your license may be denied or revoked.

Table 1. Saturated Steam: Temperature Table—*Continued*

Temp. Fahr. t	Abs Press. Lb per Sq In. p	Specific Volume: Sat. Liquid v_f	Specific Volume: Evap v_{fg}	Specific Volume: Sat. Vapor v_g	Enthalpy: Sat. Liquid h_f	Enthalpy: Evap h_{fg}	Enthalpy: Sat. Vapor h_g	Entropy: Sat. Liquid s_f	Entropy: Evap s_{fg}	Entropy: Sat Vapor s_g	Temp Fahr t
460.0	466.87	0.01961	0.97463	0.99424	441.5	763.2	1204.8	0.6405	0.8299	1.4704	460.0
464.0	485.56	0.01969	0.93588	0.95557	446.1	758.6	1204.7	0.6454	0.8213	1.4667	464.0
468.0	504.83	0.01976	0.89885	0.91862	450.7	754.0	1204.6	0.6502	0.8127	1.4629	468.0
472.0	524.67	0.01984	0.86345	0.88329	455.2	749.3	1204.5	0.6551	0.8042	1.4592	472.0
476.0	545.11	0.01992	0.82958	0.84950	459.9	744.5	1204.3	0.6599	0.7956	1.4555	476.0
480.0	566.15	0.02000	0.79716	0.81717	464.5	739.6	1204.1	0.6648	0.7871	1.4518	480.0
484.0	587.81	0.02009	0.76613	0.78622	469.1	734.7	1203.8	0.6696	0.7785	1.4481	484.0
488.0	610.10	0.02017	0.73641	0.75658	473.8	729.7	1203.5	0.6745	0.7700	1.4444	488.0
492.0	633.03	0.02026	0.70794	0.72820	478.5	724.6	1203.1	0.6793	0.7614	1.4407	492.0
496.0	656.61	0.02034	0.68065	0.70100	483.2	719.5	1202.7	0.6842	0.7528	1.4370	496.0
500.0	680.86	0.02043	0.65448	0.67492	487.9	714.3	1202.2	0.6890	0.7443	1.4333	500.0
504.0	705.78	0.02053	0.62938	0.64991	492.7	709.0	1201.7	0.6939	0.7357	1.4296	504.0
508.0	731.40	0.02062	0.60530	0.62592	497.5	703.7	1201.1	0.6987	0.7271	1.4258	508.0
512.0	757.72	0.02072	0.58218	0.60289	502.3	698.2	1200.5	0.7036	0.7185	1.4221	512.0
516.0	784.76	0.02081	0.55997	0.58079	507.1	692.7	1199.8	0.7085	0.7099	1.4183	516.0
520.0	812.53	0.02091	0.53864	0.55956	512.0	687.0	1199.0	0.7133	0.7013	1.4146	520.0
524.0	841.04	0.02102	0.51814	0.53916	516.9	681.3	1198.2	0.7182	0.6926	1.4108	524.0
528.0	870.31	0.02112	0.49843	0.51955	521.8	675.5	1197.3	0.7231	0.6839	1.4070	528.0
532.0	900.34	0.02123	0.47947	0.50070	526.8	669.6	1196.4	0.7280	0.6752	1.4032	532.0
536.0	931.17	0.02134	0.46123	0.48257	531.7	663.6	1195.4	0.7329	0.6665	1.3993	536.0
540.0	962.79	0.02146	0.44367	0.46513	536.8	657.5	1194.3	0.7378	0.6577	1.3954	540.0
544.0	995.22	0.02157	0.42677	0.44834	541.8	651.3	1193.1	0.7427	0.6489	1.3915	544.0
548.0	1028.49	0.02169	0.41048	0.43217	546.9	645.0	1191.9	0.7476	0.6400	1.3876	548.0
552.0	1062.59	0.02182	0.39479	0.41660	552.0	638.5	1190.6	0.7525	0.6311	1.3837	552.0
556.0	1097.55	0.02194	0.37966	0.40160	557.2	632.0	1189.2	0.7575	0.6222	1.3797	556.0
560.0	1133.38	0.02207	0.36507	0.38714	562.4	625.3	1187.7	0.7625	0.6132	1.3757	560.0
564.0	1170.10	0.02221	0.35099	0.37320	567.6	618.5	1186.1	0.7674	0.6041	1.3716	564.0
568.0	1207.72	0.02235	0.33741	0.35975	572.9	611.5	1184.5	0.7725	0.5950	1.3675	568.0
572.0	1246.26	0.02249	0.32429	0.34678	578.3	604.5	1182.7	0.7775	0.5859	1.3634	572.0
576.0	1285.74	0.02264	0.31162	0.33426	583.7	597.2	1180.9	0.7825	0.5766	1.3592	576.0
580.0	1326.17	0.02279	0.29937	0.32216	589.1	589.9	1179.0	0.7876	0.5673	1.3550	580.0
584.0	1367.7	0.02295	0.28753	0.31048	594.6	582.4	1176.9	0.7927	0.5580	1.3507	584.0
588.0	1410.0	0.02311	0.27608	0.29919	600.1	574.7	1174.8	0.7978	0.5485	1.3464	588.0
592.0	1453.3	0.02328	0.26499	0.28827	605.7	566.8	1172.6	0.8030	0.5390	1.3420	592.0
596.0	1497.8	0.02345	0.25425	0.27770	611.4	558.8	1170.2	0.8082	0.5293	1.3375	596.0
600.0	1543.2	0.02364	0.24384	0.26747	617.1	550.6	1167.7	0.8134	0.5196	1.3330	600.0
604.0	1589.7	0.02382	0.23374	0.25757	622.9	542.2	1165.1	0.8187	0.5097	1.3284	604.0
608.0	1637.3	0.02402	0.22394	0.24796	628.8	533.6	1162.4	0.8240	0.4997	1.3238	608.0
612.0	1686.1	0.02422	0.21442	0.23865	634.8	524.7	1159.5	0.8294	0.4896	1.3190	612.0
616.0	1735.9	0.02444	0.20516	0.22960	640.8	515.6	1156.4	0.8348	0.4794	1.3141	616.0
620.0	1786.9	0.02466	0.19615	0.22081	646.9	506.3	1153.2	0.8403	0.4689	1.3092	620.0
624.0	1839.0	0.02489	0.18737	0.21226	653.1	496.6	1149.8	0.8458	0.4583	1.3041	624.0
628.0	1892.4	0.02514	0.17880	0.20394	659.5	486.7	1146.1	0.8514	0.4474	1.2988	628.0
632.0	1947.0	0.02539	0.17044	0.19583	665.9	476.4	1142.2	0.8571	0.4364	1.2934	632.0
636.0	2002.8	0.02566	0.16226	0.18792	672.4	465.7	1138.1	0.8628	0.4251	1.2879	636.0
640.0	2059.9	0.02595	0.15427	0.18021	679.1	454.6	1133.7	0.8686	0.4134	1.2821	640.0
644.0	2118.3	0.02625	0.14644	0.17269	685.9	443.1	1129.0	0.8746	0.4015	1.2761	644.0
648.0	2178.1	0.02657	0.13876	0.16534	692.9	431.1	1124.0	0.8806	0.3893	1.2699	648.0
652.0	2239.2	0.02691	0.13124	0.15816	700.0	418.7	1118.7	0.8868	0.3767	1.2634	652.0
656.0	2301.7	0.02728	0.12387	0.15115	707.4	405.7	1113.1	0.8931	0.3637	1.2567	656.0
660.0	2365.7	0.02768	0.11663	0.14431	714.9	392.1	1107.0	0.8995	0.3502	1.2498	660.0
664.0	2431.1	0.02811	0.10947	0.13757	722.9	377.7	1100.6	0.9064	0.3361	1.2425	664.0
668.0	2498.1	0.02858	0.10229	0.13087	731.5	362.1	1093.5	0.9137	0.3210	1.2347	668.0
672.0	2566.6	0.02911	0.09514	0.12424	740.2	345.7	1085.9	0.9212	0.3054	1.2266	672.0
676.0	2636.8	0.02970	0.08799	0.11769	749.2	328.5	1077.6	0.9287	0.2892	1.2179	676.0
680.0	2708.6	0.03037	0.08080	0.11117	758.5	310.1	1068.5	0.9365	0.2720	1.2086	680.0
684.0	2782.1	0.03114	0.07349	0.10463	768.2	290.2	1058.4	0.9447	0.2537	1.1984	684.0
688.0	2857.4	0.03204	0.06595	0.09799	778.8	268.2	1047.0	0.9535	0.2337	1.1872	688.0
692.0	2934.5	0.03313	0.05797	0.09110	790.5	243.1	1033.6	0.9634	0.2110	1.1744	692.0
696.0	3013.4	0.03455	0.04916	0.08371	804.4	212.8	1017.2	0.9749	0.1841	1.1591	696.0
700.0	3094.3	0.03662	0.03857	0.07519	822.4	172.7	995.2	0.9901	0.1490	1.1390	700.0
702.0	3135.5	0.03824	0.03173	0.06997	835.0	144.7	979.7	1.0006	0.1246	1.1252	702.0
704.0	3177.2	0.04108	0.02192	0.06300	854.2	102.0	956.2	1.0169	0.0876	1.1046	704.0
705.0	3198.3	0.04427	0.01304	0.05730	873.0	61.4	934.4	1.0329	0.0527	1.0856	705.0
705.47*	3208.2	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	705.47*

*Critical temperature

Table 2: Saturated Steam: Pressure Table

Abs Press. Lb/Sq In. p	Temp Fahr t	Specific Volume: Sat. Liquid v_f	Specific Volume: Evap v_{fg}	Specific Volume: Sat. Vapor v_g	Enthalpy: Sat. Liquid h_f	Enthalpy: Evap h_{fg}	Enthalpy: Sat. Vapor h_g	Entropy: Sat. Liquid s_f	Entropy: Evap s_{fg}	Entropy: Sat. Vapor s_g	Abs Press. Lb/Sq In. p
0.08865	32.018	0.016022	3302.4	3302.4	0.0003	1075.5	1075.5	0.0000	2.1872	2.1872	0.08865
0.25	59.323	0.016032	1235.5	1235.5	27.382	1060.1	1087.4	0.0542	2.0425	2.0967	0.25
0.50	79.586	0.016071	641.5	641.5	47.623	1048.6	1096.3	0.0925	1.9446	2.0370	0.50
1.0	101.74	0.016136	333.59	333.60	69.73	1036.1	1105.8	0.1326	1.8455	1.9781	1.0
5.0	162.24	0.016407	73.515	73.532	130.20	1000.9	1131.1	0.2349	1.6094	1.8443	5.0
10.0	193.21	0.016592	38.404	38.420	161.26	982.1	1143.3	0.2836	1.5043	1.7879	10.0
14.696	212.00	0.016719	26.782	26.799	180.17	970.3	1150.5	0.3121	1.4447	1.7568	14.696
15.0	213.03	0.016726	26.274	26.290	181.21	969.7	1150.9	0.3137	1.4415	1.7552	15.0
20.0	227.96	0.016834	20.070	20.087	196.27	960.1	1156.3	0.3358	1.3962	1.7320	20.0
30.0	250.34	0.017009	13.7266	13.7436	218.9	945.2	1164.1	0.3682	1.3313	1.6995	30.0
40.0	267.25	0.017151	10.4794	10.4965	236.1	933.6	1169.8	0.3921	1.2844	1.6765	40.0
50.0	281.02	0.017274	8.4967	8.5140	250.2	923.9	1174.1	0.4112	1.2474	1.6586	50.0
60.0	292.71	0.017383	7.1562	7.1736	262.2	915.4	1177.6	0.4273	1.2167	1.6440	60.0
70.0	302.93	0.017482	6.1875	6.2050	272.7	907.8	1180.6	0.4411	1.1905	1.6316	70.0
80.0	312.04	0.017573	5.4536	5.4711	282.1	900.9	1183.1	0.4534	1.1675	1.6208	80.0
90.0	320.28	0.017659	4.8779	4.8953	290.7	894.6	1185.3	0.4643	1.1470	1.6113	90.0
100.0	327.82	0.017740	4.4133	4.4310	298.5	888.6	1187.2	0.4743	1.1284	1.6027	100.0
110.0	334.79	0.01782	4.0306	4.0484	305.8	883.1	1188.9	0.4834	1.1115	1.5950	110.0
120.0	341.27	0.01789	3.7097	3.7275	312.6	877.8	1190.4	0.4919	1.0960	1.5879	120.0
130.0	347.33	0.01796	3.4364	3.4544	319.0	872.8	1191.7	0.4998	1.0815	1.5813	130.0
140.0	353.04	0.01803	3.2010	3.2190	325.0	868.0	1193.0	0.5071	1.0681	1.5752	140.0
150.0	358.43	0.01809	2.9958	3.0139	330.6	863.4	1194.1	0.5141	1.0554	1.5695	150.0
160.0	363.55	0.01815	2.8155	2.8336	336.1	859.0	1195.1	0.5206	1.0435	1.5641	160.0
170.0	368.42	0.01821	2.6556	2.6738	341.2	854.8	1196.0	0.5269	1.0322	1.5591	170.0
180.0	373.08	0.01827	2.5129	2.5312	346.2	850.7	1196.9	0.5328	1.0215	1.5543	180.0
190.0	377.53	0.01833	2.3847	2.4030	350.9	846.7	1197.6	0.5384	1.0113	1.5498	190.0
200.0	381.80	0.01839	2.2689	2.2873	355.5	842.8	1198.3	0.5438	1.0016	1.5454	200.0
210.0	385.91	0.01844	2.16373	2.18217	359.9	839.1	1199.0	0.5490	0.9923	1.5413	210.0
220.0	389.88	0.01850	2.06779	2.08629	364.2	835.4	1199.6	0.5540	0.9834	1.5374	220.0
230.0	393.70	0.01855	1.97991	1.99846	368.3	831.8	1200.1	0.5588	0.9748	1.5336	230.0
240.0	397.39	0.01860	1.89909	1.91769	372.3	828.4	1200.6	0.5634	0.9665	1.5299	240.0
250.0	400.97	0.01865	1.82452	1.84317	376.1	825.0	1201.1	0.5679	0.9585	1.5264	250.0
260.0	404.44	0.01870	1.75548	1.77418	379.9	821.6	1201.5	0.5722	0.9508	1.5230	260.0
270.0	407.80	0.01875	1.69137	1.71013	383.6	818.3	1201.9	0.5764	0.9433	1.5197	270.0
280.0	411.07	0.01880	1.63169	1.65049	387.1	815.1	1202.3	0.5805	0.9361	1.5166	280.0
290.0	414.25	0.01885	1.57597	1.59482	390.6	812.0	1202.6	0.5844	0.9291	1.5135	290.0
300.0	417.35	0.01889	1.52384	1.54274	394.0	808.9	1202.9	0.5882	0.9223	1.5105	300.0
350.0	431.73	0.01912	1.30642	1.32554	409.8	794.2	1204.0	0.6059	0.8909	1.4968	350.0
400.0	444.60	0.01934	1.14162	1.16095	424.2	780.4	1204.6	0.6217	0.8630	1.4847	400.0
450.0	456.28	0.01954	1.01224	1.03179	437.3	767.5	1204.8	0.6360	0.8378	1.4738	450.0
500.0	467.01	0.01975	0.90787	0.92762	449.5	755.1	1204.7	0.6490	0.8148	1.4639	500.0
550.0	476.94	0.01994	0.82183	0.84177	460.9	743.3	1204.3	0.6611	0.7936	1.4547	550.0
600.0	486.20	0.02013	0.74962	0.76975	471.7	732.0	1203.7	0.6723	0.7738	1.4461	600.0
650.0	494.89	0.02032	0.68811	0.70843	481.9	720.9	1202.8	0.6828	0.7552	1.4381	650.0
700.0	503.08	0.02050	0.63505	0.65556	491.6	710.2	1201.8	0.6928	0.7377	1.4304	700.0
750.0	510.84	0.02069	0.58880	0.60949	500.9	699.8	1200.7	0.7022	0.7210	1.4232	750.0
800.0	518.21	0.02087	0.54809	0.56896	509.8	689.6	1199.4	0.7111	0.7051	1.4163	800.0
850.0	525.24	0.02105	0.51197	0.53302	518.4	679.5	1198.0	0.7197	0.6899	1.4096	850.0
900.0	531.95	0.02123	0.47968	0.50091	526.7	669.7	1196.4	0.7279	0.6753	1.4032	900.0
950.0	538.39	0.02141	0.45064	0.47205	534.7	660.0	1194.7	0.7358	0.6612	1.3970	950.0
1000.0	544.58	0.02159	0.42436	0.44596	542.6	650.4	1192.9	0.7434	0.6476	1.3910	1000.0
1050.0	550.53	0.02177	0.40047	0.42224	550.1	640.9	1191.0	0.7507	0.6344	1.3851	1050.0
1100.0	556.28	0.02195	0.37863	0.40058	557.5	631.5	1189.1	0.7578	0.6216	1.3794	1100.0
1150.0	561.82	0.02214	0.35859	0.38073	564.8	622.2	1187.0	0.7647	0.6091	1.3738	1150.0
1200.0	567.19	0.02232	0.34013	0.36245	571.9	613.0	1184.8	0.7714	0.5969	1.3683	1200.0
1250.0	572.38	0.02250	0.32306	0.34556	578.8	603.8	1182.6	0.7780	0.5850	1.3630	1250.0
1300.0	577.42	0.02269	0.30722	0.32991	585.6	594.6	1180.2	0.7843	0.5733	1.3577	1300.0
1350.0	582.32	0.02288	0.29250	0.31537	592.3	585.4	1177.8	0.7906	0.5620	1.3525	1350.0
1400.0	587.07	0.02307	0.27871	0.30178	598.8	576.5	1175.3	0.7966	0.5507	1.3474	1400.0
1450.0	591.70	0.02327	0.26584	0.28911	605.3	567.4	1172.8	0.8026	0.5397	1.3423	1450.0
1500.0	596.20	0.02346	0.25372	0.27719	611.7	558.4	1170.1	0.8085	0.5288	1.3373	1500.0
1550.0	600.59	0.02366	0.24235	0.26601	618.0	549.4	1167.4	0.8142	0.5182	1.3324	1550.0
1600.0	604.87	0.02387	0.23159	0.25545	624.2	540.3	1164.5	0.8199	0.5076	1.3274	1600.0
1650.0	609.05	0.02407	0.22143	0.24551	630.4	531.3	1161.6	0.8254	0.4971	1.3225	1650.0
1700.0	613.13	0.02428	0.21178	0.23607	636.5	522.2	1158.6	0.8309	0.4867	1.3176	1700.0
1750.0	617.12	0.02450	0.20263	0.22713	642.5	513.1	1155.6	0.8363	0.4765	1.3128	1750.0
1800.0	621.02	0.02472	0.19390	0.21861	648.5	503.8	1152.3	0.8417	0.4662	1.3079	1800.0
1850.0	624.83	0.02495	0.18558	0.21052	654.5	494.6	1149.0	0.8470	0.4561	1.3030	1850.0
1900.0	628.56	0.02517	0.17761	0.20278	660.4	485.2	1145.6	0.8522	0.4459	1.2981	1900.0
1950.0	632.22	0.02541	0.16999	0.19540	666.3	475.8	1142.0	0.8574	0.4358	1.2931	1950.0
2000.0	635.80	0.02565	0.16266	0.18831	672.1	466.2	1138.3	0.8625	0.4256	1.2881	2000.0
2100.0	642.76	0.02615	0.14885	0.17501	683.8	446.7	1130.5	0.8727	0.4053	1.2780	2100.0
2200.0	649.45	0.02669	0.13603	0.16272	695.5	426.7	1122.2	0.8828	0.3848	1.2676	2200.0
2300.0	655.89	0.02727	0.12406	0.15133	707.2	406.0	1113.2	0.8929	0.3640	1.2569	2300.0
2400.0	662.11	0.02790	0.11287	0.14076	719.0	384.8	1103.7	0.9031	0.3430	1.2460	2400.0
2500.0	668.11	0.02859	0.10209	0.13068	731.7	361.6	1093.3	0.9139	0.3206	1.2345	2500.0
2600.0	673.91	0.02938	0.09172	0.12110	744.5	337.6	1082.0	0.9247	0.2977	1.2225	2600.0
2700.0	679.53	0.03029	0.08165	0.11194	757.3	312.3	1069.7	0.9356	0.2741	1.2097	2700.0
2800.0	684.96	0.03134	0.07171	0.10305	770.7	285.1	1055.8	0.9468	0.2491	1.1958	2800.0
2900.0	690.22	0.03262	0.06158	0.09420	785.1	254.7	1039.8	0.9588	0.2215	1.1803	2900.0
3000.0	695.33	0.03428	0.05073	0.08500	801.8	218.4	1020.3	0.9728	0.1891	1.1619	3000.0
3100.0	700.28	0.03681	0.03771	0.07452	824.0	169.3	993.3	0.9914	0.1460	1.1373	3100.0
3200.0	705.08	0.04472	0.01191	0.05663	875.5	56.1	931.6	1.0351	0.0482	1.0832	3200.0
3208.2*	705.47	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	3208.2*

*Critical pressure

EQUATION SHEET

$f = ma$

$w = mg$

$E = mC^2$

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

$PE = mgh$

$W = v\Delta P$

$\Delta E = 931\Delta m$

$v = s/t$

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

$a = (v_f - v_o)/t$

$v_f = v_o + at$

$\omega = \theta/t$

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

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

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

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

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

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

$Pwr = W_f \dot{m}$

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

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

$SUR = 26.06/T$

$T = 1.44\ DT$

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

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

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

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

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

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

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

$\Sigma = N\sigma$

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

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

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

$TVL = 1.3/\mu$

$HVL = 0.693/\mu$

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

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

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

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

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

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

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

$\lambda_{eff} = 0.1$ 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

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

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

CATEGORY IV
REACTOR THEORY (7%), THERMODYNAMICS (7%)
COMPONENTS (10%)

*QUESTION
4.01 (1.0)

MULTIPLE CHOICE (Select the correct answer)

A startup is to be performed 10 hours after a trip from 100% power, and the Estimated Critical Position (ECP) has already been calculated.

Which of the following events would cause the ACTUAL critical control rod position to be LOWER than the calculated ECP, assuming a negative MTC?

a. The condenser steam dump pressure setpoint is increased 25 psig.

b. Boron concentration is increased from 1140 ppm to 1150 ppm.

c. The startup is delayed until 20 hours after the trip.

d. All steam generator levels are decreased 5% just prior to startup.

*ANSWER
C

*REFERENCE
PVNGS Reactor Theory, page 7-26.
KA 192008K101 3.4/3.5

*QUESTION
4.02 (1.0).

MULTIPLE CHOICE (Select the correct answer)

A reactor is subcritical with a Keff of 0.95 and a source range count rate of 200 counts per second (CPS). Control rods are subsequently withdrawn to add reactivity and raise Keff to 0.967.

Which of the following is the steady state count rate after this reactivity addition?

a. 300 CPS

b. 600 CPS

c. 6,000 CPS

d. 12,000 CPS

*ANSWER
A

S / (1 - 0.95) = 200 (given information); S = 10

10 / (1 - 0.967) = X X = 300 CPS

*REFERENCE
PVNGS Reactor Theory, 11.5 (page 11-25).
KA 192008K103 3.9/4.0

*QUESTION
4.03 (1.0)

MULTIPLE CHOICE (Select the correct answer)

A natural circulation cooldown is in progress using 41AO-1ZZ13, Natural Circulation Cooldown. Average RCS temperature is 500 F and RCS pressure is 2220 psig. The Reactor Operator has just finished adding enough boron for a 300 ppm boration to comply with Technical Specifications (3.1.1.1).

Which of the following events could cause a dilution of the RCS?

a. Isolating the SG steam bleedoff.

b. Degassing the RCS.

c. Opening the Pressurizer Main spray valves.

d. Void formation in the Reactor Vessel Head.

*ANSWER
D .

*REFERENCE
Procedure 41AO-1ZZ13, page 10 of 47 (Vol 9, tab 21).
KA 192007K105 3.2

*QUESTION
4.04 (1.0)

MULTIPLE CHOICE (Select the correct answer)

A natural circulation cooldown is in progress in accordance with 41AO-1ZZ13, Natural Circulation Cooldown. The Reactor Operator has just finished adding enough boron for a 300 ppm boration to comply with Technical Specifications (3.1.1.1).

Which of the following time periods identifies how long will it take to achieve proper boron mixing?

a. 5 to 20 minutes.

b. 20 to 60 minutes.

c. 1 to 2 hours.

d. 2 to 4 hours.

*ANSWER
B

*REFERENCE
Procedure 41AO-1ZZ13, page 10 of 47 (Vol 9, tab 21).
KA 192007K105 3.2

*QUESTION
4.05 (1.0)

MULTIPLE CHOICE (Select the correct answer)

Technical Specification 3.1.3.6, Regulating CEA Insertion Limits, requires that CEAs be withdrawn above specified limits. If the Long Term and Short Term Insertion Limits are exceeded for greater than 4 hours per 24 hour period, power increases must be limited to less than 5% per hour.

Which of the following reasons describes why the rate of power increase is limited?

a. To assure that Shutdown Margin will be maintained at the higher power level.

b. To limit the effects of radial Xenon redistribution to within FSAR assumptions.

c. To assure that the effects of a CEA ejection accident are limited to acceptable levels.

d. To protect the fuel cladding from Pellet-Clad Interaction in high flux areas.

*ANSWER
B

*REFERENCE
PVNGS Technical Specification Bases, page B 3/4 1-6.
KA 192005K115 3.9

*QUESTION
4.06 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The reactor is at Beginning of Cycle (BOC) and has been operating steady state at 75% for the last two days. Power is increased to 100% in one hour by primarily using CEA withdrawal.

Which of the following statements describes how Xenon will affect the core flux shape over the next two (2) hours?

a. Xenon buildup in the upper core regions will force the flux peak lower in the core.

b. Xenon buildup in the lower core regions will force the flux peak higher in the core.

c. Xenon burnout in the upper core regions will force the flux peak higher in the core.

d. Xenon burnout in the lower core regions will force the flux peak lower in the core.

*ANSWER
D

*REFERENCE
PVNGS Reactor Theory NLA09-03, 10.2.5 (page 10-42).
KA 192006K108 3.4

*QUESTION
4.07 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The Power Defect is the sum of the Moderator Defect and the Fuel Defect. The value of each component is determined by using the Moderator Temperature Coefficient (MTC) and Fuel Temperature Coefficient (FTC) multiplied times a change in temperature.

Which of the following statements identifies the LARGEST component of the Power Defect over core life during power operations (Mode 1)?

a. The Moderator Defect is largest throughout core life.

b. The Fuel Defect is largest throughout core life.

c. Each component contributes approximately an equal amount, actual power level determines the largest component.

d. The Moderator Defect at BOC, and the Fuel defect at EOC.

*ANSWER
B

*REFERENCE
PVNGS Reactor Theory, NLA09-03, 8.7.4 (page 8-56).
KA 192004K108 3.1

*QUESTION
4.08 (1.0)

MULTIPLE CHOICE (Select the correct answer)

SHUTDOWN MARGIN is the amount of reactivity that the reactor is subcritical, or could be made subcritical, assuming certain conditions.

Which of the following assumptions IS part of the SHUTDOWN MARGIN definition?

a. No changes in any part-length CEA position.

b. The reactor is Xenon free.

c. RCS cooldown to less than 210 F.

d. All CEAs are fully inserted.

*ANSWER
A

*REFERENCE
PVNGS Technical Specification Definitions, page 1-6.
KA 192002K110 3.6

*QUESTION
4.09 (1.0)

MULTIPLE CHOICE (Select the correct answer)

Brittle fracture is a failure mechanism of concern for low alloy steel pressure vessels. Several operational limitations help reduce the possibility of brittle fracture occurring.

Which of the following operational limits effects brittle fracture potential?

a. RCS Pressure Safety Limit.

b. RCS cooldown rate limit.

c. Minimum temperature for criticality limit.

d. RCS chemistry limits.

*ANSWER
B

*REFERENCE
PVNGS Technical Specification Bases 3/4.4.8, page B 3/4 4-6.
KA 193010K104 3.7

*QUESTION
4.10 (1.0)

MULTIPLE CHOICE (Select the correct answer)

Technical Specifications limits AZIMUTHAL POWER TILT to within specified limits when the Reactor is operated above 20% rated thermal power.

Which of the following statements describes the Technical Specification Basis for this limit?

a. It provides additional assurance that the core was properly loaded.

b. It assures a 95% probability and 95% confidence level that that the margin to DNB will not be exceeded.

c. It maintains the accuracy of radial peaking factors used for core power distribution calculations.

d. It assures that each azimuthal section of the core will not exceed 2200 F following a LOCA.

*ANSWER
C

*REFERENCE
PVNGS Technical Specification Bases 3/4.2.3.
KA 193009K107 3.3

*QUESTION
4.11 (1.0)

MULTIPLE CHOICE (Select the correct answer)

PVNGS Technical Specifications limits the Linear Heat Rate to 13.5 kW/ft (LCO 3.2.1).

Which of the following is the Technical Specification basis for this limit?

a. It prevents fuel centerline melting after a LOCA.

b. It limits peak cladding temperature to 2200 F after a LOCA.

c. It limits (95% assurance) DNB for any anticipated operational occurrence.

d. It prevents Critical Heat Flux after a LOCA.

*ANSWER
B

*REFERENCE
PVNGS Technical Specification Bases 3/4.2.1.
KA 193009K105 3.5

*QUESTION
4.12 (1.0)

MULTIPLE CHOICE (Select the correct answer)

Departure from Nucleate Boiling (DNB) can cause rapid increases in fuel centerline and cladding temperatures by greatly reducing the heat transfer coefficient.

Which of the following statements best describes the cause of DNB?

a. High heat flux produces boiling bubbles that begin to form a film on the fuel rod before they can be swept away.

b. High heat flux raises the bulk coolant temperature to greater than saturation and the coolant instantly flashes into a steam blanket.

c. Excessive local heat flux causes a rapid increase in the Heat Transfer Coefficient which results in fuel centerline melting.

d. High heat flux causes Nucleate Boiling transition to Bulk Boiling which increases Peak Cladding Temperature to greater than 2200 F.

*ANSWER
A

*REFERENCE
PVNGS Thermodynamics and Heat Transfer, page 7-5.
KA 193008K104 3.1/3.3

*QUESTION
4.13 (1.0)

MULTIPLE CHOICE (Select the correct answer)

PVNGS has experienced a load rejection and loss of offsite power. A subcooled natural circulation cooling has been established using AFW and Steam Generator atmospheric dump valves. RCS hot leg temperatures are decreasing at a 10 F/hr cooldown rate.

Which of the following actions would INCREASE the amount of RCS natural circulation flow?

a. Increasing RCS pressure.

b. Lowering SG water level.

c. Increasing steam dump flow.

d. Waiting one hour for decay heat to decrease.

*ANSWER
C

*REFERENCE
Lesson NLA06-02-RC-009, page 13, 9.1.4 (objective E06).
KA 193008K123 4.1

*QUESTION
4.14 (1.0)

MULTIPLE CHOICE (Select the correct answer)

PVNGS has experienced a load rejection and loss of offsite power. A subcooled natural circulation cooling has been established using AFW and Steam Generator atmospheric dump valves. RCS hot leg temperatures are decreasing at a 10 F/hr cooldown rate.

Which of the following conditions could severely reduce steady state natural circulation flow?

a. Lowered S/G water level uncovering 20% of the S/G U-tube surface area.

b. Formation of a steam bubble in the upper regions of the reactor head.

c. Rapidly raising S/G water level.

d. Gas or vapor voiding inside the S/G U-tubes.

*ANSWER
D

*REFERENCE
Lesson NLA06-02-RC-009, page 15, 9.1.4 (objective E04).
KA 193007K104 3.0

*QUESTION
4.15 (1.0)

MULTIPLE CHOICE (Select the correct answer)

A subcooled liquid (350 F, 350 psig) is pumped through a piping system at 3000 gpm. A sudden valve closure (5 second stroke time) at the end of a long piping run causes water hammer that damages several piping snubbers and ruptures the piping.

Which of the following parameter changes would REDUCE the amount of water hammer? [ASSUME all other parameters remain constant.]

a. Decreasing the system temperature.

b. Decreasing the system pressure.

c. Decreasing the system flow rate.

d. Decreasing the valve stroke time.

*ANSWER
C

*REFERENCE
Fluids NLA04-03-003-H001 (Vol 20, tab 3, p 3-4, #9).
KA 193006K104 3.6

*QUESTION
4.16 (1.0)

MULTIPLE CHOICE. (Select the correct answer)

An operator drops his Self Indicating Dosemeter (SID) while working with a maintenance team in a radiation area. Visual inspection reveals no obvious damage to the SID, but it now reads off scale high.

Which of the following actions should the operator take?

a. The operator should try to find a Radiation Protection Technician and inform him of the situation.

b. The operator should leave the radiation area and report the dropped SID to Radiological Protection.

c. The operator should conclude his work within his assigned stay time, and ensure that his TLD is read after he leaves the RCA.

d. The operator should inform his co-workers, leave the RCA, and inform Radiological Protection.

*ANSWER
D

*REFERENCE
PVNGS Radiological Work Practices Training, page C05.
KA 191002K119 3.1/3.3

*QUESTION
4.17 (1.0)

MULTIPLE CHOICE (Select the correct answer)

Prior to closing any alternating current (A.C.) circuit breaker, the two power sources must be in phase to prevent component damage.

Which of the following statements describes how this requirement is assured when closing the Main Generator output breakers, with the syncroscope selector switch in the COMP/MAN position.

a. The output breakers are interlocked with the syncroscope to prevent closing unless both lines are in phase.

b. The output breakers are always closed onto a deenergized bus so that paralleling is not necessary.

c. The operator must properly parallel across the breaker, no electrical interlocks are provided.

d. All 230 KV breakers are the break-before-make design which allows for any breaker closing sequence.

*ANSWER
C

*REFERENCE
Procedure 410P-1MB01, Main Generation and Excitation, 3.32 (vol 3, tab 24, page 9).
KA 191008K107 3.0/3.3

*QUESTION
4.18 (1.0)

MULTIPLE CHOICE (Select the correct answer)

During power operations, one 4.16 KV ESF bus has been deenergized because of a spurious Normal Supply Breaker trip. The Emergency Diesel Generator failed to start. The operator is ready to reenergize the ESF bus by reclosing the Normal Supply Breaker.

Which of the following statements describes the syncroscope needle motion just prior to closing the feeder breaker onto a deenergized bus?

a. It will be stationary, no rotation will be observed.

b. It will rotate slowly in the FAST (clockwise) direction.

c. It will rotate slowly in the SLOW (counter clockwise) direction.

d. It will oscillate (about 15 degrees) at the 12 O'clock position.

*ANSWER
A

*REFERENCE
Training Article PB, 4.16 KVAC Class 1E Power System (vol 4, tab 10).
KA 191008K107 3.0/3.3

*QUESTION
4.19 (1.0)

MULTIPLE CHOICE (Select the correct answer)

During power operations, Normal Supply Breaker (PBA-S03) one 4.16 KV ESF bus has been deenergized because of a spurious trip. The Emergency Diesel Generator failed to start. The operator is ready to reenergize the ESF bus by reclosing the Normal Supply Breaker. The operator places the Control Room switch to close WITHOUT turning on the syncroscope.

Which of the following statements describes the breaker response?

a. It will close, but the syncroscope circuit will delay actual breaker closure until both lines are in phase.

b. It will close, because closure interlocks are bypassed when the oncoming bus is deenergized.

c. It will NOT close, because the two lines may be out of phase and breaker closure could damage the breaker.

d. It will NOT close, because the syncroscope selector switch is interlocked with the breaker closure circuitry.

*ANSWER
D

*REFERENCE
Training Article PB, 3.1.1 (Vol 4, tab 10, p 4).
KA 191008K107 3.0/3.3

*QUESTION
4.20 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The VCT is provided with two level indicators, one with a DRY reference leg (LT226) and one with a WET reference leg (LT227)

A problem with the Aux Building HVAC causes both reference leg temperatures to increase by the same amount.

Which of the following correctly describes the effect on the indicated VCT level?

a. LT226 indication is the same as actual level, and
LT227 indication is higher than actual level.

b. LT226 indication is lower than actual level, and
LT227 indication is higher than actual level.

c. LT226 indication is higher than actual level, and
LT227 indication is higher than actual level.

d. LT226 indication is lower than actual level, and
LT227 indication is the same as actual level.

*ANSWER
A

*REFERENCE
RC-PLCS Training Article 4.2.1 (Vol 5, tab 14, p 10).
KA 191002K108 3.1

D wright: clarified that ΔT in reference leg caused by Aux Building did not effect Temp/Level inside VCT. [re-read the problem].

*QUESTION
4.21 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The normal supply breaker for bus PBA-S03 has just tripped on fault and its lockout relay is energized.

Which of the following statements describes how the Reactor Operator can identify the faulted condition with the lockout relay energized?

a. Both the tripped and closed indicating lights are energized.

b. Both the tripped and closed indicating lights are NOT energized.

c. This indication is NOT available in the Control Room and can only be determined locally.

d. The tripped indicating light is brighter.

*ANSWER
d

*REFERENCE
Training Article PB, section 3.2.1.2 (Vol 4, tab 10, p 5).
KA 191008K111 3.3

*QUESTION
4.22 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The B Emergency Diesel Generator is operating at 95% load in parallel with the Normal Supply 4.16 KV Class (PBB-S04) power supply for surveillance testing. The operator places the GOVERNOR Raise/Lower Control Switch to the LOWER position.

Which of the following statements describes how the Emergency Diesel Generator (EDG) will respond?

a. Generator kilowatts will decrease until either the operator releases the switch or the EDG trips.

b. Generator frequency, RPMs, and kilowatts will decrease until either the operator releases the switch, or the EDG trips.

c. Generator voltage will decrease until the 4.16 KV Vital Bus supply breakers open on Under Voltage.

d. Generator frequency and RPMs will decrease until either the operator releases the switch or the EDG trips.

*ANSWER
A.

*REFERENCE
Training Article DG & DF, 4.3.2.6.1.2 (vol 2, tab 5, page 34).
KA 064000K101 4.1/4.4

*QUESTION
4.23 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The B Emergency Diesel Generator is operating at 95% load in parallel with the Normal Supply 4.16 KV Class (PBB-S04) power supply for surveillance testing. The operator places the VOLTAGE Raise/Lower Control Switch to the LOWER position for three (3) seconds.

Leading or (Lagging mode

Which of the following statements describes how the Emergency Diesel Generator (EDG) will respond?

Give 100 kvar lagging

a. Generator Voltage will decrease.

b. Generator Kilovars will decrease.

c. Generator Kilowatts will decrease.

d. EDG Frequency and RPMs will decrease.

*ANSWER
B

*REFERENCE
Training Article DG & DF, 4.3.2.6.1.2 (vol 2, tab 5, page 34).
KA 064000K101 4.1/4.4

*QUESTION
4.24 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The Pressurizer Level Controller is a Porportional-Integral-Derivative (PID) type of controller, with the Derivative portion of the control function tuned out. A spurious charging pump trip initially causes a steadily increasing (LINEAR) level error magnitude.

Which of the following statements describes how the PID controller output responds?

a. Output will quickly change to a minimum value, and then remain steady.

b. Output will decrease steadily (linear) to a minimum value.

c. Output will decrease at a increasing rate (faster than linear) to a minimum value.

d. Output will decrease at a decreasing rate (slower than linear) to a minimum value.

END OF CATEGORY IV

*ANSWER
C

*REFERENCE
RC-PLCS Training Article 3.1.5 (Vol 5, tab 14, p 6).
KA 191003K101 3.2

CATEGORY V
EMERGENCY PLANT EVOLUTIONS (33%)

*QUESTION
5.01 (1.0)

MULTIPLE CHOICE (Select the correct answer)

Reactor and Plant Startup are in progress. Reactor power is 25% and Group 5 is withdrawn to 132 inches. The Reactor Operator notices that one of the Group 5 CEAs position indicates 120 inches. When the Shift Supervisor verifies this condition, both operators notice that one Shutdown Bank CEA is still fully inserted.

Which of the following courses of action should the Shift Supervisor take when the Shutdown Bank CEA is found fully inserted?

a. Stabilize the plant, reduce turbine load to match reactor power, and verify actual rod positions.

b. Place the CEDM control system in Standby, stabilize the plant, reduce reactor power based on the amount of Time After Deviation.

c. Commence reactor shutdown, be in Hot Standby within 6 hours, and verify that Shutdown Margin requirements are met.

d. Immediately initiate a manual trip of the reactor, and proceed to 41EP-1ZZ01, Emergency Operations.

*ANSWER
D

*REFERENCE
PVNGS 41AO-1ZZ11, Dropped or Slipped CEA, page 5 (vol 9, tab 17).
KA 000005A203 4.4

*QUESTION
5.02 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The reactor is operating at 50% power. A dropped CEA has just been recovered per 41AO-1ZZ11, Dropped or Slipped CEA. Reactor power must now be held constant for 1 hour.

Which of the following reasons describes why reactor power is held constant for 1 hour?

a. This will allow any induced Xenon transient enough time to stabilize.

b. This allows the Reactor Engineer time to assess the power distribution.

c. This ensures COLSS has accurate reactor power inputs from incore detectors.

d. This assures that Thermal Margins will not be exceeded upon returning to higher power levels.

*ANSWER
B

*REFERENCE
PVNGS 41AO-1ZZ11, Dropped or Slipped CEA, page 8 (vol 9, tab 17).
KA 000005K106 3.8

*QUESTION
5.03 (1.0)

MULTIPLE CHOICE (Select the correct answer)

During a loss of all AC power procedure 41RO-1ZZ09, Blackout, requires that Steam Generator pressure be maintained less than 1250 psia. One basis for this pressure limit is to reduce the possibility of a Steam Generator Safety Valve failing to reseat.

Which of the following statements is the other basis for this pressure limit?

a. It prevents an uncontrolled radioactive release path if primary to secondary leakage exists.

b. It maintains Tave within the assumed limits of the FSAR to ensure accurate Nuclear Instrument indication.

c. It assures that the Turbine Driven AFW pump can supply adequate flow to restore SG level and remove decay heat.

d. It reduces the amount of cyclic stress on the Steam Generators caused by excessive SRV actuation.

*ANSWER
A

*REFERENCE
Blackout Simulator Lesson Plan, EO-4 (vol 12, tab 22, page 5).
KA 000055K302 4.6

*QUESTION
5.04 (3.0)

MATCHING

PVNGS Unit 1 has been operating at 100% for several weeks. A BLACKOUT HAS JUST OCCURRED. 41RO-1ZZ09, Blackout, is being performed.

Write the letter of the correct TIME PERIOD next to the CONDITION listed below. Only one (1) TIME PERIOD is correct for each CONDITION, but the same TIME PERIOD may be used once, more than once, or not at all.
(0.5 points each)

TIME PERIODS:

a. 15 minutes OR LESS.
b. 20 minutes.
c. 30 minutes.
d. 60 minutes.
e. 90 minutes.
f. 120 minutes OR MORE.

CONDITIONS:

1. ____ The length of time an EDG can run unloaded, without cooling water during Blackout conditions.

2. ____ The length of time Class 1E instrumentation can be powered by the associated batteries during Blackout conditions (according to 41RO-1ZZ09).

3. ____ The MAXIMUM length of time after loss of forced RCS flow for indications of natural circulation to be apparent.

4. ____ The MAXIMUM amount of time to have the Technical Support Center activated after declaring this emergency event.

5. ____ The length of time a secured RCP in hot standby can last, without cooling water nor seal injection, before seal damage occurs (control bleedoff valve is shut).

6. ____ The approximate length of time it will take to uncover the reactor core if NO source of Steam Generator feed can be initiated.

*ANSWER (0.5 each)
1. D 2. E 3. A 4. D 5. B 6. D
(#6: actual value in 41AO-1ZZ44 Appendix L is 59 minutes)

*REFERENCE
Blackout Simulator Lesson Plan, EO-9, EO-10, (#1,2: vol 12, tab 22); Emergency Operations, Appendix E (#3: vol 12, tab 1); 41AO-1ZZ29, RCP and Motor Emergency, 9.1 (#5: vol 10, tab 20); 41AO-1ZZ44 Shutdown from Outside the Control Room due to Fire and/or Smoke, Appendix L (#6: vol 11, tab15).
KA 000055K302 4.6

*QUESTION
5.05 (2.0)

Standard Appendix E to 41EP-1ZZ01, Emergency Operations, provides seven (7) plant conditions that are used to verify natural circulation cooling in the Reactor Coolant System (RCS). Two conditions are Th and Tc - STABLE OR DECREASING.

What are four (4) of the remaining five (5) plant conditions (include parameter and magnitudes)?

*ANSWER (0.5 each for any four; 2.0 maximum. +/- 10% OK for setpoints)

RCS subcooling > 28 degrees F.

Reactor Vessel outlet plenum full.

Core exit thermocouples & Th trending consistently.

No (or not suspected) RCS voiding.

RCS delta T < 57 degrees F (also accept full power delta T) and stable.

REFERENCE
PVNGS 41EP-1ZZ01, Emergency Operations, Appendix E (vol 12, tab 1).
KA 000055K102 4.4

*QUESTION
5.06 (2.5)

Emergency Response Organization and Staffing, 16AC-OEP01, identifies six (6) responsibilities for the Emergency Coordinator that CANNOT be delegated. One of theses responsibilities is to notify emergency response/support personnel, NRC, and State/County agencies.

What are the five (5) remaining responsibilities that CANNOT be delegated? (0.5 each)

*ANSWER (0.5 each)

Activate emergency response organizations (for Alert or higher emergency classifications).

Declare changes in the emergency classification level.

Provide protective action recommendations (to offsite emergency management agencies).

Authorize emergency exposures (in excess of 10CFR20 limits, but not greater than emergency exposure limits).

Determine the necessity for evacuating (various) site personnel.

*REFERENCE
Emergency Response Organization and Staffing, 16AC-OEP01, 2.1.1 (vol 16, tab
18)
KA 000011G002 4.3

*QUESTION
5.07 (1.0)

[The following plant data is identical for Questions 5.07 through 5.10.]

Reactor power	10 E-4 % decreasing
PZR pressure	1800 psia
PZR level	0%
RVLMS level	3 HJTC sensors uncovered
T hot	580 F
T cold	530 F
Highest CET	590 F (all are increasing slowly)
Steam Generator level	35% wide range
Steam Generator pressure	700 psig
RCPs running	None
SI flow (total)	60 gpm/leg
Containment pressure	10 psig
Containment temperature	215 F

MULTIPLE CHOICE (Select the correct answer)

[Use the Steam Tables attached to this exam.]

A Loss of Coolant Accident has occurred and current plant status is shown by the plant indications above. Verification of Adequate Core Cooling is in progress per 41RO-1ZZ07, Loss of Coolant Accident.

Which of the following choices include the current RCS subcooling margin?

a. Greater than 38 F subcooled.

b. Between 28 and 38 F subcooled.

c. Between 18 and 28 F subcooled.

d. Less than 18 F subcooled.

0915 General Comment to candidates: Questions 5.07 - 5.10 refer to the current time "right now" based on the information provided.

*ANSWER
B   T sat (1800 psia) = 621.0

T sat (1800 psig) - CET =

621 - 590 = 31.0

*REFERENCE
PVNGS 41RO-1ZZ07, Loss of Coolant Accident (vol 12, tab 17).
KA 000074K101 4.7

*QUESTION
5.08 (1.0)

[The following plant data is identical for Questions 5.07 through 5.10.]

Reactor power	10 E-4 % decreasing
PZR pressure	1800 psia
PZR level	0%
RVLMS level	3 HJTC sensors uncovered
T hot	580 F
T cold	530 F
Highest CET	590 F (all are increasing slowly)
Steam Generator level	35% wide range
Steam Generator pressure	700 psig
RCPs running	None
SI flow (total)	60 gpm/leg
Containment pressure	10 psig
Containment temperature	215 F

MULTIPLE CHOICE (Select the correct answer)

A Loss of Coolant Accident has occurred and current plant status is shown by the plant indications above. Verification of Adequate Core Cooling is in progress per 41RO-1ZZ07, Loss of Coolant Accident. Natural circulation has NOT been verified. Four plant parameters are used to verify adequate core cooling; one of them is Subcooling Margin.

Which of the following Subcooling Margins is required to verify adequate core cooling?

a. Subcooling Margin must be greater than 60 F.

b. Subcooling Margin must be greater than 33 F.

c. Subcooling Margin must be greater than 28 F.

d. Subcooling Margin must be Greater than 0 F.

*ANSWER
D

*REFERENCE
PVNGS 41RO-1ZZ07, Loss of Coolant Accident, 7.1 (vol 12, tab 17, page 16).
KA 000074A201 4.9

*QUESTION
5.09 (1.0)

[The following plant data is identical for Questions 5.07 through 5.10.]

Reactor power	10 E-4 % decreasing
PZR pressure	1800 psia
PZR level	0%
RVLMS level	3 HJTC sensors uncovered
T hot	580 F
T cold	530 F
Highest CET	590 F (all are increasing slowly)
Steam Generator level	35% wide range
Steam Generator pressure	700 psig
RCPs running	None
SI flow (total)	60 gpm/leg
Containment pressure	10 psig
Containment temperature	215 F

MULTIPLE CHOICE (Select the correct answer)

A Loss of Coolant Accident has occurred and current plant status is shown by the plant indications above. Verification of Adequate Core Cooling is in progress per 41RO-1ZZ07, Loss of Coolant Accident. Natural circulation has NOT been verified and Reflux Boiling is the only method of decay heat removal.

Which of the following actions is recommended by 41RO-1ZZ07 to enhance reflux boiling natural circulation?

a. Initiate Containment Spray.

b. Energize the Pressurizer heaters.

c. Raise Steam Generator level.

d. Open a Pressurizer Main Spray Valve.

*ANSWER
C (Note that the required SCM is 60 F due to high Cntmt temperature.)

*REFERENCE
41RO-1ZZ07, Loss of Coolant Accident, 18.5 (vol 12, tab 17, page 24).
KA 000074K212 2.4

*QUESTION
5.10 (1.0)

[The following plant data is identical for Questions 5.07 through 5.10.]

Reactor power	10 E-4 % decreasing
PZR pressure	1800 psia
PZR level	0%
RVLMS level	3 HJTC sensors uncovered
T hot	580 F
T cold	530 F
Highest CET	590 F (all are increasing slowly)
Steam Generator level	35% wide range
Steam Generator pressure	700 psig
RCPs running	None
SI flow (total)	60 gpm/leg
Containment pressure	10 psig
Containment temperature	215 F

MULTIPLE CHOICE (Select the correct answer)

A Loss of Coolant Accident has occurred and current plant status is shown by the plant indications above. 41RO-1ZZ07, Loss of Coolant Accident, states the criteria to determine if adequate core cooling has been established. Reference Figure 5.10 attached.

ASSUME current RCS subcooling is ADEQUATE.

Which of the following statements correctly states the status of core cooling?

a. Adequate core cooling exists.

b. Adequate core cooling does NOT exist because Safety Injection flow is too low.

c. Adequate core cooling does NOT exist because CET temperatures must be stable or decreasing.

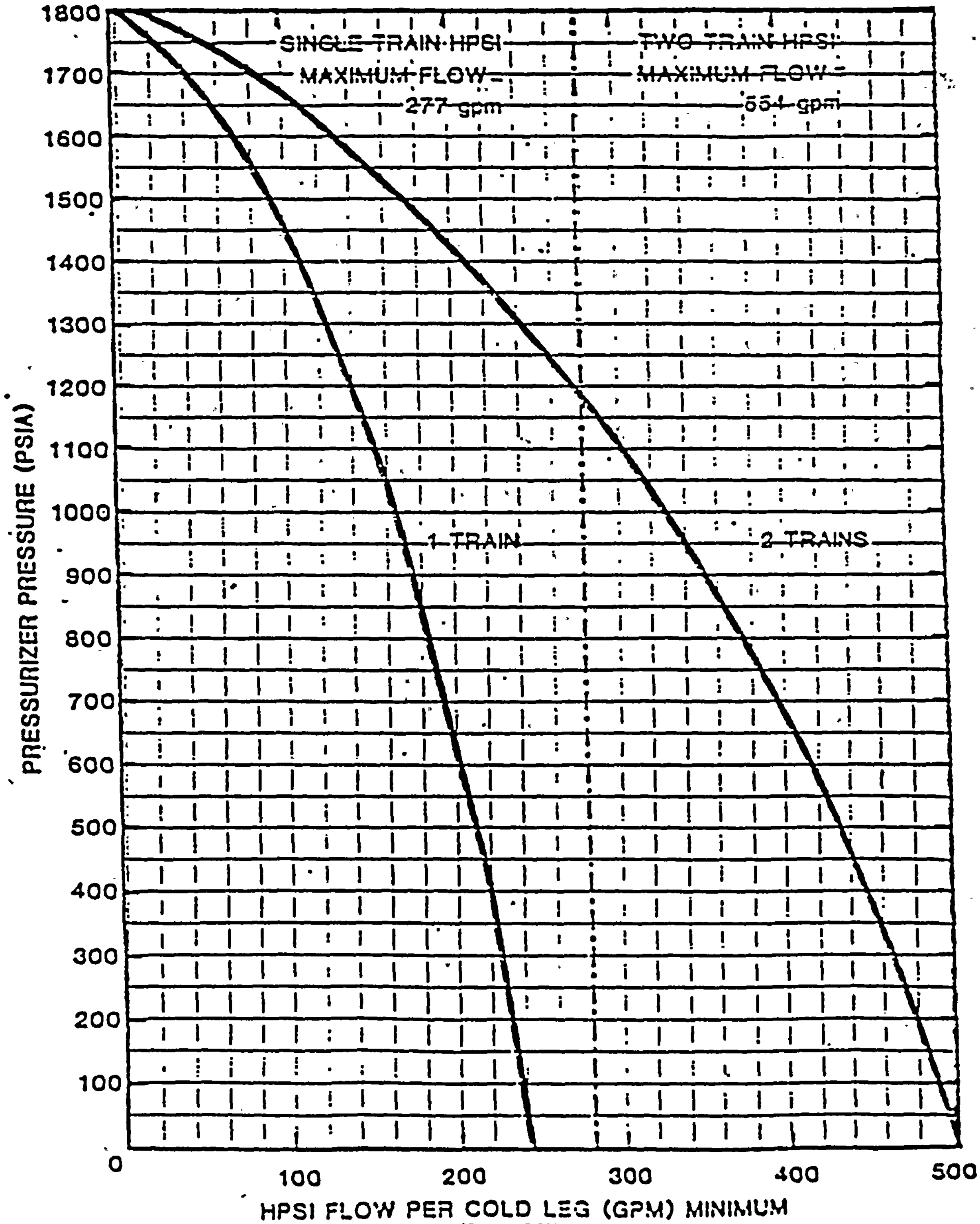
d. Adequate core cooling does NOT exist because Reactor vessel level is too low.

*ANSWER
A

*REFERENCE
PVNGS 41RO-1ZZ07, Loss of Coolant Accident, 7.1 (vol 12, tab 17, page 16).
KA 000009A239 4.7

WORKING COPY

PALO VERDE NUCLEAR GENERATING STATION MANUAL	PROCEDURE NO. 41EP-1ZZ01	APPENDIX B Page 2 of 5
EMERGENCY OPERATIONS	REVISION 3	Page 19 of 179



INFORMATION AID. - CONSULT ODG-52

*QUESTION
5.11 (1.0)

MULTIPLE CHOICE (Select the correct answer)

PVNGS Unit 1 is operating at full power when a toxic gas (carbon dioxide) forces Control Room evacuation per 41AO-1ZZ27, Shutdown Outside Control Room.

Which of the following actions is required prior to leaving the Control Room?

a. Initiate a Main Steam Isolation Signal.

b. Trip the turbine driven AFW pump.

c. Trip both Main Feed Pump Turbines.

d. Trip two Reactor Coolant pumps.

*ANSWER
D

*REFERENCE
PVNGS 41AO-1ZZ27 and 1ZZ44, Shutdown Outside the Control Room, Initial Operator Actions (vol 10 tab 16, and vol 11 tab 15)
KA 000068K318 4.5

*QUESTION
5.12 (1.0)

MULTIPLE CHOICE (Select the correct answer)

PVNGS Unit 1 is operating at rated thermal power. RCP 2A Upper Thrust Bearing Temperature is at its alarm setpoint, reading 230 F and is increasing at 2 F per minute. PVNGS 41AO-1ZZ29, Reactor Coolant Pump and Motor Emergency, is being performed.

Which of the following Verification actions are required by 41AO-1ZZ29?

a. Stabilize the plant, monitor the other RCPs for high temperature indications, and comply with Technical Specification 3.4.1.1.

b. Notify the GCC, reduce reactor power, and if temperature exceeds the trip value manually trip the reactor and RCP 2A.

c. Commence reactor shutdown, be in Hot Standby within 1 hour, and verify that Thermal Margin requirements are met.

d. Immediately initiate a manual trip of the reactor, and proceed to 41EP-1ZZ01, Emergency Operations.

*ANSWER
B

*REFERENCE
PVNGS 41AO-1ZZ29, Reactor Coolant Pump and Motor Emergency, 1.0 (vol 10, tab 20).
KA 000015K303 4.0

*QUESTION
5.13 (1.0)

MULTIPLE CHOICE (Select the correct answer)

PVNGS Unit 2 was in Mode 1 when Steam Generator #1 suffered a Main Steam Line (MSL) rupture. SIAS has actuated, RCPs 1A and 2A have been tripped, and the reactor has been tripped. S/G #1 has been isolated, but the RCS cooldown rate is still excessive. 42RO-2ZZ03, Loss of Secondary Coolant is being performed.

Which of the following actions is directed by 42RO-2ZZ03 to help regain control of RCS cooldown rate?

a. Trip RCP 1B.

b. Isolate Steam Generator #2.

c. Restore blowdown to Steam Generator #1.

d. Reduce Pressurizer pressure to prevent Pressurized Thermal Shock.

*ANSWER
C

*REFERENCE
PVNGS 41RO-1ZZ03, Loss of Secondary Coolant, 4.2 (vol 12, tab 9, p 8).
KA 000040K304 4.7

*QUESTION
5.14 (1.5)

41RO-1ZZ03, Loss of Secondary Coolant, identifies four general criteria that must be met prior to starting a RCP when no other RCP is running. One of these conditions is that the RCS is greater than 28 F subcooled.

What are the remaining three (3) general criteria that must be met (Do NOT list RCP starting interlocks)? (0.5 each)

*ANSWER (0.5 each)

Adequate RCS pressure (under control and capable of supporting RCP operation per Standard Appendix F).

Adequate RCS inventory (greater than 33% w/o voiding OR as allowed by Standard Appendix CC w/ voiding OR SI flow established).

At least one Steam Generator available (for RCS heat removal).

*REFERENCE
PVNGS 41RO-1ZZ03, Loss of Secondary Coolant, 10.0 (vol 12, tab 9, p 19).
000040K304 4.7

*QUESTION
5.15 (1.0)

MULTIPLE CHOICE (Select the correct answer)

PVNGS is recovering from a Main Steam Line rupture on Steam Generator #1 and has been performing a natural circulation cooldown. All conditions to restart a RCP have been met and RCP 2A has just been started. Immediately after pump start Pressurizer pressure and level rapidly decreased causing a loss of all RCS subcooling.

Which of the following actions should be taken to recover from this situation?

a. Immediately trip the RCP.

b. Energize the Pressurizer heaters.

c. Increase S/G steam flow.

d. Monitor RCP parameters and restore RCS pressure.

*ANSWER
A

*REFERENCE
PVNGS 41RO-1ZZ03, Loss of Secondary Coolant, 10.2.2 (vol 12, tab 9, p 21).
000040K304 4.7

*QUESTION
5.16 (3.0)

PVNGS 41RO-1ZZ01, Reactor Trip, identifies the four (4) criteria that must be met prior to throttling Safety Injection flow and the bases for these criteria. Two (2) of these criteria are that the RCS be subcooled greater than 28 F and that Pressurizer level be greater than 33% and controllable.

a. What are the other two (2) criteria to throttle SI flow? (1.0)

b. What are the two (2) bases for the Pressurizer level criteria? (1.0)

c. What are the two (2) bases for the RCS Subcooling criteria? (1.0)

*ANSWER

a. (0.75 each)

1) Reactor vessel level indicates void restricted to upper head.

2) One S/G is capable of (maintaining) heat removal. (Also accept one S/G is capable of steaming and being fed.)

b. Assures adequate RCS inventory (0.5) and operation of the Pressurizer heaters (0.5).

c. To maintain adequate core cooling (0.5) and a coolable core geometry (0.5).

*REFERENCE
PVNGS 41RO-1ZZ01, Reactor Trip, Appendix C (vol 12, tab 5, p 29).
000009K324 4.6

*QUESTION
5.17 (1.0)

MULTIPLE CHOICE (Select the correct answer)

Reactor Trip, 41RO-1ZZ01, directs the operator to trip two RCPs if SIAS actuation occurs.

Which of the following statements is the basis for this action?

a. This minimizes RCS coolant loss if a small break LOCA has occurred, and also maintains forced circulation cooling.

b. This reduces the reactivity addition rate during the Design Basis Accident MSL rupture.

c. This reduces the severity of most analyzed transients, including Blackout, Total Loss of RCS Flow, and ATWS.

d. This places the plant in the optimal condition to recover from an inadvertent SIAS initiation.

*ANSWER
A

*REFERENCE
Not provided.
KA 000007K301 4.6

*QUESTION
5.18 (1.0)

MULTIPLE CHOICE (Select the correct answer)

PVNGS Unit #2 is operating at full power with all systems operable and in automatic. An RCS leakage calculation using 42AO-2ZZ08, Steam Generator Tube Leak, has just been completed. They determined that S/G #1 has a 20 gpm tube leak.

Which of the following symptoms are identified in 42AO-2ZZ08 as an indication of a S/G tube leak?

a. Higher S/G level and feed flow-steam flow mismatch in the affected S/G.

b. Decreasing Pressurizer level and pressure.

c. Decreasing VCT level and charging-letdown flow mismatch.

d. High Containment radiation levels.

*ANSWER
C

*REFERENCE
PVNGS 41AO-1ZZ08, Steam Generator Tube Leak, 1.1 (vol 9, tab 15, p 3)
KA 000037A204 3.7

*QUESTION
5.19 (2.0)

41AO-1ZZ08, Steam Generator Tube Leak, requires the reactor to be shutdown if either one of two limits is exceeded. One (1) of these conditions is S/G activity greater than 0.1 uCi/cc Dose Equivalent Iodine.

a. What is the other condition (including limit) that would require plant shutdown according to 41AO-1ZZ08? (1.0)

b. What two (2) concurrent accidents make up the design basis accident of concern identified in the Technical Specification basis for S/G Activity? (1.0)

*ANSWER

a. S/G tube leakage (also accept primary to secondary leakage; 0.5) greater than 1 gpm (0.5). (also 720 gpd)

b. A Main Steam Line Rupture (concurrent with; 0.5) a loss of Off-site Electrical Power (0.5).

*REFERENCE
41AO-1ZZ08, Steam Generator Tube Leak (vol 9, tab 15); PVNGS T.S. Bases, 3/4.7.1.4 ACTIVITY.
KA 000037K305 4.0

*QUESTION
5.20 (1.00)

PVNGS procedure 41RO-1ZZ08, Small Loss of Coolant Accident (LOCA), provides specific operator actions for a small LOCA. One of the objectives of procedure 41RO-1ZZ08 is to shutdown the plant for repairs.

What are two (2) out of the three remaining objectives of procedure 41RO-1ZZ08? (0.50 each)

*ANSWER

Any two (2) of the following: (.50 each - 1.00 total)

1. Provide adequate core cooling.

2. Restore and maintain RCS inventory control.

3. Prevent inadvertent activity release.

*REFERENCE
Procedure 41RO-1ZZ08, Objectives, Page 3
Lesson Number NLC32-03-RC-008, E.O. 2, Page 6
KA 000037K305 3.7/4.0

*QUESTION
5.21 (1.00)

MULTIPLE CHOICE (Select the correct answer)

PVNGS Unit #1 is operating at full power when a loss of Instrument Air occurs. Instrument Air pressure has just decreased below 80 psig.

Which one of the following valves will have failed CLOSED due to low Instrument Air pressure?

a. S/G Economizer Feedwater Control Valves.

b. Main Steam Isolation Valves.

c. Steam Bypass Control Valves.

d. S/G Downcomer Feedwater Control Valves.

*ANSWER

c. (1.00)

*REFERENCE
41AO-1ZZ06, 2.6.2.1
Lesson Number NLC32-03-RC-024-00,E05
KA 000065A208 2.9/3.3

*QUESTION
5.22 (1.00)

MULTIPLE CHOICE (Select the correct answer)

During a Unit #1 Reactor Startup (Mode 2) one Linear Power Channel becomes inoperable and is placed in BYPASS. Later, a second Linear Power channel fails high. Reactor power was at 1% when the second failure occurred.

Which one of the following statements is correct?

a. The reactor will trip.

b. One of the two inoperable channels must be returned to OPERABLE status prior to entering Mode 1.

c. The startup may continue if the second failed channel is placed in TRIP within one (1) hour.

d. The reactor must be shutdown because Technical Specification 3.0.3 has been violated.

*ANSWER

c. (1.0)

*REFERENCE
41AO-1ZZ24 step 4.1 and T/S Table 3.3-1, Action 3
KA 000033G008 3.9

*QUESTION
5.23 (1.00)

MULTIPLE CHOICE (Select the correct answer)

PVNGS Unit #1 is recovering from a small break LOCA which occurred during power operations. 41RO-1ZZ08, Small Loss of Coolant Accident, is being performed.

Which one of the following conditions would require the operators to change procedures to 41RO-1ZZ07, Loss of Coolant Accident?

a. HPSI becomes inoperable and RCS pressure decreases below 1837 psia.

b. Containment Integrity is lost.

c. Pressurizer level indicates off scale low.

d. SI actuates and RCS subcooling is less than 28 degrees F.

*ANSWER

d.

*REFERENCE
41RO-1ZZ08, CAUTION after step 2.1, page 6
Lesson Number NLC32-03-RC-008-00, E04
KA 000009S011 4.3/4.3

*QUESTION
5.24 (1.00)

MULTIPLE CHOICE (Select the correct answer)

41AO-1ZZ22, Loss of Shutdown Cooling (SDC), provides several symptoms to determine what type of event caused the loss of cooling.

Which one of the following symptoms indicates that a loss of RCS Coolant inventory caused a loss of SDC?

a. Automatic isolation of the SDC System.

b. Minimum SDC flow is 6000 GPM.

c. SDC System Delta-T across the inservice heat exchanger is a maximum.

d. Fluctuating current and/or flow on SDC pumps.

*ANSWER

d. (1.00)

*REFERENCE
41AO-1ZZ22 step 2.1.2, page 6
Lesson Number NLC31-03-RC-032-02, Objective E03
KA 000025A102 3.8/3.9

*QUESTION
5.25 (1.00)

6

PVNGS Unit #1 is in Mode 5 performing core alterations and the Refueling Purge System is in service. A fuel element has just been dropped onto the reactor core and appears to be badly damaged; a Refueling Floor radiation alarm is alarming. 41AO-1ZZ26, Irradiated Fuel Damage, is being performed. The Shift Supervisor has initiated EPIP-02.

What are two (2) of the remaining three (3) actions required by 41AO-1ZZ26 for Isolation and Evaluation of the Area?

*ANSWER

(any 2, 0.5 each)

1. Verify or manually initiate CPIAS.

2. Notify Radiation Protection (and have them conduct surveys as necessary).

3. Evacuate personnel from the area.

*REFERENCE
41AO-1ZZ26 Step 2.3, Page 6 of 8.
NLC31-03-RC-041-00, Objective E02
KA 000036A101 3.3/3.8

CATEGORY 6

PLANT SYSTEMS (30%) AND PLANT-WIDE
GENERIC RESPONSIBILITIES (13%)

*QUESTION
6.01 (2.00)

Procedure 40AC-9OP15, Station Tagging and Clearance, provides guidance in the use of danger tags and methods for the electrical and mechanical isolation of equipment.

A. What two conditions require that a Red danger Tag be issued? (.25 each -0.50 total)

B. What two conditions require that a Blue tag be issued? (.25 each - 0.50 total)

C. Where are red tags that are associated with a breaker hung when the circuit breaker is removed from its cubicle? (1.0)

*ANSWER

A. RED- if equipment operation would endanger personnel (0.25) or damage equipment (0.25).

B. BLUE- to control system operation for troubleshooting (0.25) or testing (0.25).

C. On the cubicle door. (1.0)

*REFERENCE
Station tagging and clearance Procedure 40AC-9OP15, A-Page 77, B-Page 78
C-Page 57. Lesson Number NLC31-03-RC-062-00, A&B-Lesson Objective E03
C-Lesson Objective E09
KA 194001K102 4.1/4.1

*QUESTION
6.02 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The station tagging and clearance procedure provides guidance for temporary lifting of danger tags and methods for the electrical and mechanical isolation of equipment.

Whose concurrence is required for temporary lifts?

a. Shift Supervisor

b. Shift Supervisor and all acceptors

c. Responsible supervisor and all acceptors

d. Plant Manager and Shift Supervisor

*ANSWER

c.

*REFERENCE
Station tagging and clearance procedure 40AC-9OP15, Page 28
Lesson Number NLC31-03-RC-062-00, E08
KA 194001K102 4.1/4.1

*QUESTION
6.03 (1.0)

MULTIPLE CHOICE (Select the correct answer)

The station tagging and clearance procedure provides guidance for temporarily lifting danger tags.

Who maintains custody of the lifted danger tags after they have been removed from plant equipment?

a. Responsible Supervisor

b. Shift Supervisor

c. Shift supervisor and all acceptors

d. Shift Technical Advisor and reactor operator

*ANSWER

a.

*REFERENCE
Station tagging and clearance procedure 40AC-9OP15, Page 28
Lesson Number NLC31-03-RC-062-00, Lesson Enabling Objective E08
KA 194001K102 4.1/4.1

*QUESTION
6.04 (1.00)

Deleted

Procedure 75AC-9RP01, Radiation Exposure and Access Control, lists four (4) criteria for continuous coverage by Radiation Protection personnel when potentially extreme radiological conditions exist in a work area. One condition is in an area with high radiation levels.

What are two out of the three other radiological conditions where continuous coverage by RP personnel are required?
(Two at 0.50 each - 1.00 total)

*ANSWER
(Any two at .50 each -1.00 total)

A. 1. High potential for uptake of radioactivity (such as high loose surface contamination and high particulate/iodine airborne radioactivity).
2. Hot particle control areas.
3. Radiological conditions which are unknown or may change significantly or rapidly.

*REFERENCE
Radiation Exposure and Access Control Procedure 75AC-9RP01
A-Page 16
KA 194001K104 3.5/3.5

Question Deleted

*QUESTION
6.05 (2.00)

In order to maintain personnel radiation exposures within the limits established by 10 CFR 20, administrative restrictions are applied to the rate of dose accumulation.

a. What is Palo Verde's maximum whole body QUARTERLY Administrative Exposure Limits? w/o Special management approval. (give to candidates) (0.50)

b. What is the maximum lifetime dose in Rems for a ~~person~~ 34 years of age with NRC form 4 on file? (1.00)
IAW 10 CFR 20 Limits)

c. What is the highest level of approval required for a person to exceed 2500 mrem / quarter or 4000 mrem / year ? (0.50)

*ANSWER

A. Whole Body: 1.0 rem/quarter (0.50)

B. Lifetime dose= (5X(N-18))= 5X(34-18)= 5X16=80 Rems (1.00)

C. The ALARA Committee (chairman). (0.5).
(0.50 total)

*REFERENCE
Radiation Exposure and Access Control, Procedure 75AC-9RP01
A-Page 21; B-Pages 24 and 25
A-Radiological Work Practices Handbook, Rev.1, 1986, Page B09
KA 194001K103 3.4/3.4

*QUESTION
6.06 (1.50)

Conduct of Shift Operations, 40AC-90P02, establishes requirements for the duties and activities performed by "on-shift" operating personnel.

A. What is the minimum licensed operator crew composition for Modes 1 thru 4? (.75)

B. What modes require that an STA be on shift duty? (0.25)

c. What is the minimum required licensed positions that must be filled when in modes 5 and 6? (0.50)

~~0.50 General Comment to ~~[illegible]~~ candidates~~
~~A) Question is referenced to 'w/o special management approval'~~

*ANSWER

A. Position	#of individuals to fill position
SS	1 (.25)
SRO	1 (.25)
RO	2 (.25)

B. Modes 1-4 (0.25)

C. At least one RO (SRO) shall be in the controls area(when fuel is in the vessel). (.50 total)
PER Technical Specification, when in MODE 5 or 6 one SS (with an SRO License) and one RO (with an RO License) are required to be part of the minimum shift complement.

*REFERENCE
Conduct of Shift Operations, 40AC-90P02; A-Page 11; B-Page 13; C-Page 13
Lesson Number NLC31-03-RC-060-00; A-Enabling Objective E01;
B-Enabling Objective E01.1;C-Enabling Objective E01.1
KA 194001A103 3.4/3.4

*QUESTION
6.7 (1.0)

According to 01AC-OIS01, Confined Space Entry, one of the three major hazards associated with Confined Spaces is oxygen deficiency.

What are the two (2) other major hazards for Confined Space entry? (0.50 each, 1.0 total)

*ANSWER

1. Air Contamination (0.50) (accept Toxic or poisonous)

2. Flammable or Explosive Atmospheres (0.50)

*REFERENCE
Confined Space Entry, Procedure 01AC-OIS01, page 5.
KA 194001K114 3.6/3.6

*QUESTION
6.08 (1.00)

MULTIPLE CHOICE (Select the correct answer)

According to 01AC-OIS01, Confined Space Entry, an oxygen deficient atmosphere must be ventilated prior to entry.

What is the minimum oxygen percentage that the confined space must have to allow entry? (1.00)

A. 18.5%

B. 19.0%

C. 19.5%

D. 20.0%

*ANSWER

C. 19.5% oxygen (1.00)

*REFERENCE
Confined Space Entry, Procedure 01AC-OIS01, Page 6
KA 194001K113 3.6/3.6

*QUESTION
6.09 (1.50)

According to the Radiological Work Practices Handbook, Radiation Exposure Permits (REP) are required for any job or task within the RCA, and any job involving the use of radioactive materials. A Tour REP is one of three types of REPs used at PVNGS.

A. What kind of jobs is a Tour REP issued for? (.25)

B. What is the maximum time period a Tour REP can be issued for? (.25)

C. What are the other two REPs? (0.50 each -1.00 total)

*ANSWER

Entries routine,

A. Jobs of a repetitive or continuous nature. (0.25)

B. one year. (0.25)

C. 1. Job REP (.50)

2. Emergency REP (.50)

*REFERENCE
Radiological Work Practices Handbook; A, Page F03;B,Page F03;
C, Page F04
KA 194001K105 3.4/3.4

*QUESTION
6.10 (1.00)

MULTIPLE CHOICE (Select the correct answer)

The Control Element Drive Mechanism Control System (CEDMCS) provides control signals and motive power to operate the Control Element Drive Mechanisms.

Which of the following pre-trips/conditions will cause a CEA Withdrawal Prohibit (CWP)? (1.00)

A. Either of TC's that input to RRS > 575 degrees F.

B. Tavg-Tref Hi within +6 degrees F from RRS.

C. High Pressurizer pressure pre-trip.

D. Reactor Power less than 15%.

*ANSWER

C. (1.00)

*REFERENCE
Training Article SF-CEDMCS Control Element Drive Mechanism Control System, Rev. 2, Page 46. Lesson Number NLC56-01-XC-11-01, Student Objective E06, Page 27.
KA 001000K407 3.7/3.8

*QUESTION
6.11 (2.00)

An Automatic Motion Inhibit (AMI) prevents both withdrawal and insertion of a regulating group in the Automatic Sequential(AS) mode. The AMI can originate in the Steam Bypass Control System. One of the five setpoints which cause an AMI is Reactor Power less than 15%.

What are the other four (4) conditions and their setpoints that can cause an AMI? (.25 condition & .25 setpoint-2.00 total)

*ANSWER

(For each item: 0.25 for setpoint, 0.25 for condition, 0.5 total)

1. Reactor power less than the AMI threshold with Turbine power less than 15%.

2. The Tave channels in RRS deviate by 5 degrees F (when selected to average).

3. The control channels deviate by 5% (when selected to average).

4. The TLI signals deviate by 5% (when selected to average).

*REFERENCE
Training Article SF-CEDMCS, Page 46.
Lesson Number NLC56-01-XC-11-01, Student Objective E06, Page 27.
KA 001000A404 3.6

*QUESTION
6.12 (1.00)

During normal Reactor Coolant Pump (RCP) operation, seal injection flow is about 6.6 GPM per pump and controlled bleedoff flow is about 4.0 GPM per pump.

a. What system is the 2.6 gpm difference in flow routed to? (0.50)

b. Where is any leakage past the reactor coolant pump vapor seal routed to? (0.5)

*ANSWER

1. Flow into the RCS (via pump casing or journal bearing). (0.50)

2. Flow (via 3rd stage leakage) to Reactor Drain Tank. (0.50)

*REFERENCE
Training Article RC-RCS Reactor Coolant System, Rev.3, Page RC-RCS-19
Lesson Number NLC21-00-RC-02, Student Objective C.7
KA 003000K404 2.8/3.1

*QUESTION
6.13 (1.00)

The plant is operating at full power and all systems are operable. The Reactor Coolant Pumps(RCP's) are operating normally.

What are the TWO (2) busses that supply power to the reactor coolant pumps? (1.0)

*ANSWER

(NAN-)S01 and (NAN)-S02 (0.5 points each max 1.0 points)

*REFERENCE
Training Article RC-RCS Reactor Coolant System, Rev.3, Page RC-RCS 20
Lesson Number NLC21-00-RC-02, Student Objective C.11, Page 16
KA 003000K201 3.1/3.1

*QUESTION
6.14 (1.00)

Given an inadvertent SIAS while operating at 100% power, steady state, in a normal lineup.

A. What changes will automatically occur in the CVCS due to the SIAS (do NOT list changes caused by the CIAS)? (0.5)

B. What CVCS component is automatically prevented from being operated immediately following this SIAS? (0.5)

*ANSWER

A. Letdown isolates. (0.5) (V515 & V516 CLOSE.

B. The non-running CCP. (0.5)

*REFERENCE
Training Article Chemical and Volume Control System,
Rev 02, Page CH-64, Para. 4.4.4
Lesson Number NLC21-00-XC-004, Student Objective 6.C.1, Page 79
KA 004010A205 4.1/4.3

*QUESTION
6.15 (2.00)

The plant is operating at full power and all systems are operable. The Engineered Safety Features Actuation System (ESFAS) is in its normal lineup.

A. What are the two (2) signals and setpoints that generate a Safety Injection Actuation Signal? (0.50 each -1.00 total)

B. What signal and setpoint generates the Recirculation Actuation Signal? (.50)

C. What signal and setpoint provide the Auxiliary Feedwater Actuation Signal? (.50)

*ANSWER

A. 1. Low pressurizer pressure(0.25)-setpoint 1837 (+/- 10) psia (0.25)

2. High containment pressure(0.25)-setpoint 3.0 (+/-0.3) psia (0.25)

B. Low Refueling Water Storage Tank (0.25)-setpoint 7.4% (+/- 1%) (0.25)

C. Steam generator level (0.25)-setpoint level < 25.8% (+/- 1%) (0.25)

*REFERENCE
Training Article SA-ESF-Engineered Safety Features Actuation System, Rev 1, Pages SA-ESF-7,8,9. Lesson Number NLC56-01-005-02, Student Objective E05, Pages 13, 16, 17
KA 013000K101 4.2/4.4

*QUESTION
6.16 (1.00)

The Reed Switch Position Transmitters provide axial position indication of all full length and part length Control Element Assemblies.

What two (2) actions does the Upper Electrical Limit switch perform? (1.00)

*ANSWER

1. Prevents (further individual or subgroup) outward motion. (0.50)
2. Turns on RED indicating light (on ROM). (0.50)

[Also accept stops individual rod motion / stops subgroup rod motion as separate answers for full credit.]

*REFERENCE
Training Article SR-CEDMCS, Rev2, Page 37
Lesson Number NLC56-01-XC-11-01, Page 20
KA 014000K401 2.5/2.7 KA 014000K402 2.5/2.7

*QUESTION
6.17 (1.0)

The Log 1 and Log 2 bistables are associated with the Log Safety channels.

A. What is the purpose of the Log 1 bistable during power ascension? (0.50)

B. What are the TWO (2) trips that the Log 2 bistable bypass during power reduction? (0.25 each-0.50 total)

*ANSWER

A. Allows bypassing the Hi Log trip. (0.50)

B. Allows bypassing the Low DNBR and High LPD trips. (0.25 each 0.5 pts total)

*REFERENCE
Training Article SE, Excore Nuclear Instrument System, Rev. 1, Page SE-29
Lesson Number NLC56-01-XC-012-02, Page 32
KA 015000K401 3.1/3.3

*QUESTION
6.18 (1.00)

The Non-Essential AFW pump is operating.

What are the FOUR (4) conditions or signals that will automatically trip the running Non-essential AFW pump, other than the manual trip? (0.25 ea. 1.0 pts)

*ANSWER (.25 each -1.00 total)

1. Pump suction pressure low.

2. Load shed. (ambiguity)

3. Electrical protection (overcurrent).

4. Safety injection actuation signal (channel A).

*REFERENCE
Lesson Number NLC22-00-XC-009-03, Student Objective E09, Page 36
KA 061000602 2.6/2.7

*QUESTION
6.19 (1.00)

The Condensate Storage Tank has a capacity of 550,000 gallons of which 330,000 gallons is dedicated for the AFW.

A. What is the bases for the Technical Specification minimum of 300,000 gallons of the 330,000 in the condensate storage tank? (0.50)

B. What is the back-up source of water for essential AFW pumps? (0.50)

*ANSWER

A. Maintain 8 hrs in hot standby (Mode 3) followed by (orderly) plant cooldown (to where the shutdown cooling takes over). (0.5)

B. Reactor Makeup Water Tank (0.50)

*REFERENCE
Lesson Number NLC22-00-XC-009-03, Student Objective E02,A-Page 12
B-Page 13.
KA 061000K401 3.9/4.2

*QUESTION
6.20 (1.00)

MULTIPLE CHOICE (Select the correct answer)

Which of the following basés is the primary purpose of the CEA shutdown groups?

a. Provide startup count rate when withdrawn.

b. Ensure radial flux profile is satisfactory to maintain thermal limits.

c. Ensure proper shutdown margin for reactor trip.

d. Provide axial flux shaping at full power.

*ANSWER

c. (1.00)

*REFERENCE
Lesson Number NLC56-01-XC-11-01, Objective E02
KA 001000K508 3.9/4.4

*QUESTION
6.21 (1.0)

MULTIPLE CHOICE (Select the correct answer)

During Reactor Coolant Pump startup, the fourth RCP shall not be started until RCS temperature is equal to or greater than 500 degrees F.

Which of the following statements gives the correct reason?

a. The RCS heatup rate may be exceeded.

b. This prevents excessive starting currents.

c. This limits reactor vessel stresses.

d. Core uplift may occur.

*ANSWER

d. (1.00)

*REFERENCE
41OP-1RC01, Limitations and Precautions 3.13, Page 8
Lesson Number NLC21-00-RC-02, Objective E13, Page 17
KA 003000S13 3.6/3.7

*QUESTION
6.22 (1.00)

MULTIPLE CHOICE (Select the correct answer)

Which one of the following Control Room indications denotes a loss of control power to a 4.16 KV Class 1E breaker?

a. Green lamp-normal brilliance at the breaker switch.

b. Green lamp-intense brilliance at the breaker switch.

c. Red lamp at the breaker switch.

d. Both green and red lamps extinguished at the breaker switch.

*ANSWER

d. (1.00)

*REFERENCE
T/A 4.16KVAC Class 1E Power System, Page PB-5
Lesson Number NLC22-00-RC-017-0A, Student Objective 05, Page 11
KA 062020K510 3.5/3.9

*QUESTION
6.23 (1.5)

Palo Verde's Technical Specifications require that RCS average temperature be maintained above the Minimum Temperature for Criticality (MTC) in Modes 1 and 2.

What are the three (3) bases for this limit? (0.5 each-1.50 total)

*ANSWER (0.5 each- 1.50 total)

The minimum temperature for criticality is based upon:

1. That the MTC is within it's analyzed temperature range.

2. The protective instrumentation is within it's normal operating range.

3. Consistency with the FSAR safety analysis.

*REFERENCE
Palo Verde Technical Specification Bases 3/4.1.1.4, Page B 3/4 1-1a
Lesson Number NLC 21-00-RC-02, Student Objective C, Page 29.
KA 002000G006 2.6/3.8

*QUESTION
6.24 (1.50)

What are the Technical Specification leakage limitations for the following types of Reactor Coolant System leakage? (0.50 each)

A. Identified.

B. Unidentified.

C. Primary to secondary leakage one steam generator maximum/day.

*ANSWER

A. 10 gpm. (0.50)

B. 1 gpm. (0.50)

C. 720 gallons per day through any one S.G. (0.50)

*REFERENCE
Palo Verde Technical Specification 3.4.5.2, Page 3/4 4-19
KA 002020K401 3.6/3.8

*QUESTION
6.25 (1.00)

The Normal Pressure setpoint override mode of operation for the backup heaters is initiated by taking the heater bank control switch (HS-100-4 thru 9) to "OFF". Assume you are in Unit 1.

A. What pressure must the pressurizer pressure be above to allow the operation of the override mode switch? (0.50)

B. What is the maximum pressure above which the override mode switch will not operate? (0.50)

*ANSWER

A. Pressure must be above 2200 psia. (+/-25 psia) (0.50)

B. Pressure must be below 2285 psia. (+/-25 psia) (0.50)

*REFERENCE
Training Article RC-PPCS Pressurizer Pressure Control System, Page RC-PPCS-11. Lesson Number NLC56-01-XC-010-2, Student Objective E05
KA 010000K603 3.2/3.6

*QUESTION
6.26 (3.00)

The Reactor Protection System (RPS) provides initiating actuation to protect the plant in the event of an anticipated operational occurrence and other postulated events.

What are the trip setpoints and the type of reactor or plant protection that is provided by each one of the ~~following~~ following RPS trips? (0.25 each setpoint) (0.5 each protection concept)

1. Variable Overpower trip.
2. High Log Power trip.
3. High steam generator level trip.
4. High Containment pressure trip.

*ANSWER

1. Variable Overpower trip-110%-(0.25)- protects the core against rapid positive reactivity addition excursions.(0.5)
2. High Log Power trip- ~~1.0%~~ ≤.01% -(0.25) protects integrity of fuel cladding (and RCS pressure boundary in event of unplanned criticality from a shutdown condition). (0.5)
3. High S.G. level trip-91%-(0.25)-protects turbine against excessive moisture carryover. (0.5)
4. High Containment pressure trip-3 psig-(0.25)-provides assurance reactor is tripped in event of containment pressurization due to a pipe break inside the containment. (0.5).

*REFERENCE
Lesson Number NLC56-01-XC-005-02, Student Objective E02, Pages 7,10,11
KA 012000A101 3.4

*QUESTION
6.27 (1.00)

MULTIPLE CHOICE (Select the correct answer)

The Supplementary Protection System(SPS) provides a diverse trip logic for the reactor.

What event is the SPS designed to protect/mitigate against?

A. Protects the integrity of fuel cladding in event of unplanned criticality.

B. Mitigate the effects of an Anticipated Transient Without Scram(ATWS).

C. Protects in the event of increase in secondary heat removal and subsequent RCS cooldown.

D. Prevents exceeding DNBR limits in the event of anticipated operational occurrence.

*ANSWER

B. (1.00)

*REFERENCE
Lesson Number NLC56-01-XC-005-02, Student Objective E06, Page 21
KA 012000A206 4.4/4.7

*QUESTION
6.28 (1.00)

MULTIPLE CHOICE (Select the correct answer)

The Supplementary Protection System (SPS) provides a diverse trip logic for the reactor.

Which of the following is the SPS designed to protect?

a. Fuel cladding protection.

b. Overpower protection.

c. Fuel Centerline peak temperature protection.

d. Reactor Coolant System Pressure boundary protection.

*ANSWER

d. (1.00)

*REFERENCE
T/A Reactor Protection Sytstem, Page SB-4
Lesson Number NLC56-01-XC-005-02, Student Objective E06, Page 21
KA 012000A206 4.4/4.7

*QUESTION 6.29 (1.00)

MULTIPLE CHOICE (Select the correct answer)

The Supplementary Protection System (SPS) provides a diverse trip logic for the reactor.

Which plant parameter is used to actuate SPS?

a. Reactor Coolant System Th temp HI.

b. Nuclear Power High Log trip.

c. Core Exit Thermocouple Hi.

d. Pressurizer Pressure.

*ANSWER

d. (1.00)

*REFERENCE
T/A Reactor Protection System, Page SB-4
Lesson Number NLC56-01-XC-005-02, Student Objective E07, Page 22.
KA 012000A206 4.4/4.7

*QUESTION 6.30   (1.00)

The Plant Protection System (PPS) causes the reactor trip switchgear (RTSG) breakers to automatically open, tripping the reactor, when certain key plant parameters are reached.

Why are both General Electric and Westinghouse Reactor Trip breakers used in the reactor trip switchgear?   (1.00)

*ANSWER

It helps prevent common mode failures.   (1.00)

*REFERENCE
T/A Reactor Protection System, Page SB-4
Lesson Number NLC56-01-XC-005-02, Student Objective E07, Page 23
KA 012000A206   4.4/4.7

*QUESTION
6.31 (1.00)

What is the design function of the charcoal in the charcoal filter in the exhaust air filtering unit of the Power Access Purge System?

*ANSWER

To remove fission product gases (will accept iodine also) (1.00)

*REFERENCE
Training Article Containment Purge System, Page CP-6
No student objective
KA 027000K501 3.1/3.4

*QUESTION
6.32 (3.0)

During emergency or abnormal conditions, operation of the Nuclear Cooling Water System (NCWS) is not required. However certain priority NCWS heat loads are provided with a backup source of cooling water by the Essential Cooling Water System. Two of the loads are the normal chillers, and the Reactor Coolant Pumps.

A. What are the three (3) other priority NCWS heat loads that are provided a backup source of cooling water? (0.5 each, total 1.5)

B. What are three (3) of the five (5) specific Reactor Coolant Pump heat loads supplied by the NCWS? (0.5 each, total 1.5)

*ANSWER

A. (Any three at .50 each -total 1.50)
   a. CEDM normal ACUs.
   b. Fuel Pool heat exchangers.
   c. Primary sample coolers.

B. (Any three at 0.5 each-total 1.5)
   a. Reactor Coolant Pump (RCP) seal coolers.
   b. RCP HP coolers.
   c. RCP thrust bearing oil coolers.
   d. RCP motor air .
   e. RCP motor oil coolers

*REFERENCE
Training Article NCWS, Rev. 3, A- Page NC-10, Para.4.3
B- Page NC-11, Para. 4.3
Lesson Number NLC22-00-XC-012-0A, A-Student Objective 12;
B-S.O.15.
KA 008000K301 3.4/3.5

*QUESTION 6.33 (1.00)

MULTIPLE CHOICE - (Select the correct answer)

Which one of the following defines a High Radiation Area?

a. >10 Rem to <100 Rem in one hour.

b. >10 mRem to <100 mRem in one hour.

c. >100 Rem to <1000 Rem in one hour.

d. >100 mRem to <1000 mRem in one hour.

*ANSWER

d. (1.00)

*REFERENCE
Radiation Exposure and Access Control Procedure 75AC-9RP01
Page 17.
KA 194001K104 3.5/3.5

END OF CATEGORY SIX
END OF WRITTEN EXAM