Examination Report No.: 50-528/OL-92-01

Facility Licensee: Palo Verde Nuclear Generating Station, Units 1, 2, and 3 (PVNGS)

Facility Docket Nos.: 50-528, 50-529, and 50-530

Facility License Nos.: NPF-41, NPF-51, and NPF-74

Licensed operator initial examinations administered at PVNGS, near Wintersburg, Arizona:

Examiners:
T. Meadows, Chief Examiner
L. Briggs, Senior Examiner, RI
P. Isaksen, INEL
S. Johnson, INEL
J. Hanek, INEL

Approved by: PPNalt for
L. F. Miller, Jr., Chief
Reactor Safety Branch

7/22/92
Date Signed

Summary:

During the period of June 1-12, 1992, the NRC administered initial licensing examinations to fourteen Reactor Operator (RO) and three Senior Reactor Operator (SRO) candidates at the Palo Verde Nuclear Generating Station, Units 1, 2, and 3 (PVNGS).

One of the RO candidates failed the written portion of the examination, and was denied a license.

All of the other candidates were issued a letter informing them that they had passed the examination. Their licenses are being held in abeyance pending NRC receipt of written notification from Arizona Public Service Company (APS) that the emergency operating procedures (EOPs) upon which the candidates were trained and examined have been implemented. This notification will also affirm that the candidates are proficient in the use of these procedures.

Since the last initial examination in June of 1991, the examiners noted a substantial improvement in the quality and range of available simulator capabilities. The licensee's training and operating staffs provided good technical support in the validation of the examination test materials. They were particularly helpful during the validation reviews of the written examinations and simulator scenarios. The licensee's support in providing examination reference materials and other logistical assistance was excellent.

One concern was identified by the examiners during a simulated scenario event. The licensee's alarm response procedures did not provide adequate guidance for operators to mitigate the slow depressurization of the reactor coolant system (RCS) due to a stuck open pressurizer spray valve. Also lacking was guidance regarding the preferred tripping combination for the reactor coolant pumps (RCP's) in this event to reduce further depressurization. This issue was unresolved, and will be further addressed in NRC Inspection Report 50-528/92-22.

bcc w/enclosure 1 only:
L. Miller, RV
H. Wong, RV
M. Blume, RV
K. Perkins, RV
R. Zimmerman, RV
D. Coe, Senior Resident Inspector
B. Faulkenberry, RV
J. Martin, RV
B. Gallo, NRR/LOLB
K. Thompson, Project Manager, NRR/PDV
T. Quay, NRR/PDV
L. Briggs, RI
F. Jaggar, EG&G

CONCURRENCES:

[REQUEST COPY	] REQUEST COPY	] REQUEST COPY]
[YES / NO	] YES / NO	] YES / NO]
MEADOWS	HWONG _____	LMILLER _____ PDN 7/22/92
7/16/92 DBP 7/22/92	7/ /92	7/17/92

[COPY TO PDR YES/NO]
[COPY TO DCS YES/NO]

ENCLOSURE 1

EXAMINATION REPORT

50-528/OL-92-01

REPORT DETAILS

1. **Personnel Contacted**

NRC Personnel:

+*T. Meadows, Chief Examiner
L. Briggs, NRC Examiner
+J. Sloan, Resident Inspector
P. Isaksen, INEL Examiner
S. Johnson, INEL Examiner
J. Hanek, INEL Examiner

PVNGS Personnel:

+J. Levine, Vice President, Nuclear Production
*E. Firth, Manager, Nuclear Training
*T. Cannon, Supervisor, Simulator Support
*J. Dennis, Manager, Operation Standards
*L. Florence, Senior Advisor, Operation Standards
*J. Niedermeyer, Unit 1 Shift Supervisor
*G. Clyde, Nuclear Licensing
*P. Coffin, Compliance
*A. Peroutka, Supervisor, Licensed Operator Initial Training
*E. Shouse, Simulator Support, Senior Test Operator
*B. Picchiottino, Senior Instructor, Licensed Operator Initial Training
L. Clyde, Unit 3 Operations Manager
D. Bjorklund, Initial Licensing Instructor
R. Nunez, Manager, Licensed Operator Initial Training
D. Ensign, Senior Reactor Operator, Unit 2 Operations
M. Saba, Simulator Support

* Identifies personnel present at the June 12, 1991 exit meeting.
\+ Mr. Levine and Mr. Sloan were only present at the pre-exit briefing with the Chief Examiner on June 11, 1992

2. **Written Test**

The licensee reviewed the RO and SRO written examinations the in the regional office April 30 through May 1, 1992. The licensee review team consisted of a licensed SRO and a certified SRO instructor.

The examinations were administered on June 1, 1992 to fourteen RO candidates and three SRO upgrade candidates at the site. The examination's administrative arrangements were satisfactory.

At the conclusion of the examination, the Chief Examiner conducted a post examination review with the same licensee staff indicated above.There were no further comments or changes from the as given examination.

One of the RO candidates failed the written portion of the examination, and was denied a license.

All of the other candidates were issued a letter informing them that they had passed the examination. Their licenses are being held in abeyance pending written notification of the NRC that the emergency operating procedures (EOPs) the operators were trained and examined on have been implemented. This notification will also confirm that these candidates are proficient with these procedures before going on shift.

Although these candidates passed their examinations, some of them exhibited significant knowledge deficiencies.

The examiners conducted a post examination comparison of the SRO test results and found that thirty percent or more of the SRO candidates did not know: the normal power supply for charging pump CHE-PO1, the charging pump status upon a loss of offsite power following diesel generator (DG) breaker closure, the proper methods for containment ventilation, or major valve positioning upon a loss of control air.

The examiners conducted a post examination comparison of the RO test results and found that thirty percent or more of the RO candidates did not know: the actions that an operator must take before departing from technical specification requirements (10CFR 50.54x), the events that would cause a control element assembly (CEA) withdrawal prohibit (CWP) to occur, the normal power supply for charging pump CHE-PO1, the charging pump status upon a loss of offsite power following diesel generator (DG) breaker closure, the relationship between excore nuclear instrumentation and the core protection calculators (CPCs), the shut off pressure head of the low pressure safety injection system (LPSI), the control element drive mechanism control system (CEDMC) electrical upper/lower limits, or the affects of an auxiliary feed actuation signal (AFAS) on DG operations.

Some questions in refueling area were also missed by the RO candidates. However, these were of a lower importance rating for ROs.

The examiners noted that RO knowledge deficiencies in the area of refueling operations was also indicated in the walkthrough examinations.

3. Operating Test

During the period of June 2-12, 1992, the NRC administered initial licensing operating examinations to fourteen Reactor Operator (RO) and three Senior Reactor Operator (SRO) candidates on site. The examination consisted of both a multi-unit walkthrough test and evaluation in the licensee's simulation facility. The simulator's performance was good, with no major fidelity problems. Enclosure 3, "Simulator Facility Report," documents the simulator performance.

Since the last initial examination in June of 1991, the examiners noted an substantial improvement in the quality and range of available simulator capabilities. The licensee's training and operating staffs provided good technical support in the validation of the examination test materials. They were particularly helpful during the validation reviews of the written examinations and simulator scenarios. The licensee's support in providing examination reference materials and other logistical assistance onsite was excellent.

All of the candidates passed this portion of the examination. Since the licensee chose to implement the EOP version that the candidates were being trained on after the examinations, the NRC held their licenses in abeyance pending implementation of the later version of the EOPs. This was primarily due to the extensive revision of the EOPs.

4. Examiners' Concerns and Observations

In an enclosure to a letter received from Mr. J. Levine, Vice President Nuclear Production dated, June 19, 1992, the licensee responded to all of the issues discussed at the exit meeting. This letter, which is included as Attachment B, identifies the commitments the licensee made to the Chief Examiner at the exit meeting. The letter also outlines a proposed schedule for issue evaluation and proposed corrective action.

Procedure Deficiency Concern - Reactor Coolant System (RCS) Depressurization (Open):

During a simulator scenario event, involving a stuck open pressurizer spray valve, the examiners identified an apparent procedure deficiency. The alarm response procedure implemented by the licensee to address this event was 41AL-1RK4A, "PZR PRESS HI-LO." The adequacy of this procedure was the concern.

The alarm response procedure only directed operators to try

to close the spray valve with the control room handswitch, and to enter the emergency operating procedures (EOPs) on low RCS pressure. As a result, the NRC examiners observed three different groups of applicant reactor operators attempt to mitigate this event in the facility simulator, using different strategies.

a. One group isolated control air to the affected spray valve, which promptly closed the valve and stopped the RCS depressurization. Appropriately, this was accomplished before EOP entry was necessary.

b. A second group tripped all reactor coolant pumps (RCPs) to reduce spray flow. EOP entry was required on low RCS pressure.

c. A third group tripped two RCPs and left two running, in accordance with the EOPs. However, the EOPs did not specify which RCPs to trip for this specific event. This crew failed to consider the consequences of the further RCS depressurization that would result by leaving the RCP running in the loop with the stuck open spray valve.

All of the above strategies mitigated the event and the EOPs were correctly used. However, some challenged the plant safety systems and RCS subcooling margin more than others.

Therefore, the examiners determined that the alarm response procedures did not provide adequate guidance to mitigate a slow RCS depressurization event.

The licensee acknowledged this concern and revised the alarm response procedure. This new revision was given to the Chief Examiner at the pre-exit meeting on June 11, 1992. It contained additional guidance to isolate control air in an attempt to cause the spray valve to fail closed. It also provided timely RCP tripping strategy. Therefore, the Chief Examiner determined that the procedure was adequate.

This issue will be further addressed in NRC Inspection Report 50-528/92-22.

<u>Harsh Containment Detection Concern (Closed)</u>:

Another item of concern was that the licensee's intent to implement new EOPs in August 17, 1992 that require operators to use non-class temperature instrumentation for harsh containment determination. This instrumentation is necessary to determine the correct mitigating EOP strategy.

Adequate justification for using non-class instrumentation was not initially provided to the Chief Examiner at the exit meeting. Therefore, additional licensee evaluation was requested.

This information was received on July 10, 1992, and was evaluated by NRC staff. From this and additional telephone communication with the licensee, the Chief Examiner determined that temperature elements used at PVNGS to detect harsh containment conditions were probably adequate. The licensee provided documentation which indicated that they were environmentally qualified in February 1984, by Honeywell, Inc.

The qualification standards employed by Honeywell exceeded the temperature, pressure, and duration conditions necessary for the instruments to detect EOP harsh containment conditions. For example, the PVNGS EOPs identify containment conditions as harsh if air space temperature exceeds 150 degrees F. The instrument qualification certifies these instruments at greater than 400 degrees F for 30 days, in a 100 percent humidity high pressure atmosphere. Therefore, the Chief Examiner considered this issue closed.

Examiner Observations:

The examiners also observed that:

a. The candidates' communication abilities were acceptable during the simulator examinations. However, some of the RO candidates failed to ensure that their verbal communications were clearly received by other crew members. They did not aggressively solicit feedback from each other.

   The licensee stated that in the Fall of 1992 that they will engage a communications professional to enhance operator training in this area. See Attachment B for further details.

b. During a simulator scenario event involving a failure of control element assembly calculator (CEAC) number 1, the plant monitoring system (PMS) data point on panel B06 indicated that the trouble was with CEAC B. CEAC B correctly identifies that electrical division B supplies power to CEAC 1. However, this human factor deficiency caused some initial confusion to the examination candidate crew, and was noted by the examiners.

The licensee indicated that they would generate an engineering evaluation request (EER) on June 19, 1992 to correct the problem. Also, the licensee's incident investigation program would track corrective actions taken. Operating crews would then be appropriately trained.

c. During the walkthrough portion of the examination, the examiners found that procedure 4xAO-xZZ44, Appendix H, "Shutdown Outside the Control Room due to Fire or Smoke," appeared to be difficult for the operators to use. (The "x" in the procedure designator means that the procedure is generic to Units 1, 2, and 3.) The formula, or method, for determining adequate reactor coolant system (RCS) subcooling margin was not clearly specified. Also, guidance to verify that steam generator (SG) levels would support natural circulation RCS conditions was not provided.

The licensee stated that they had identified this deficiency, and that the procedure was currently being revised. Implementation of the new procedure was scheduled for October 1992, after appropriate operator training.

d. During an evaluated job performance measure (JPM) involving an inter-system leak cued from a radiation monitoring system alarm, the examiners noted that applicants were required to use three separate procedures before arriving at the appropriate mitigating actions. The three procedures were 74RM-9EF41 (RMS Alarm Response), 41AO-1ZZ14 (Excessive RCS Leakrate), and 41AO-1ZZ29 (RCP Motor Emergency).

The licensee stated that they would complete their evaluation of this JPM scenario by August 3, 1992. Appropriate corrective actions will then be taken to possibly streamline the procedure transition process.

e. Section 3.0 of alarm response procedure 4xAO-ZZ26, Irradiated Fuel Damage, instructed operators to attempt to reposition damaged fuel in its original "stable configuration" back inside the reactor core before taking actions to evacuate personnel. This placement action may not be appropriate, especially if the fuel is already out of its immediate core configuration area.

The licensee stated that they would evaluate changing this procedure by September 1, 1992.

f. The PVNGS units were not consistent in their methods for controlling the access to the cabinets containing safety significant operating procedures at the Remote Shutdown Panels. The examiners understood that some security was necessary to keep procedures from being pilfered. However, the security measures should not impede operator access under emergency conditions. Two of the three PVNGS units locked the cabinets: one unit used a lock with the key maintained by the shift supervisor (SS), the other unit used a combination lock with no clear means of determining the combination. The third unit had no lock on its cabinet.

The licensee implemented a night order for all three units requiring the cabinets to be secured with combination locks and posted with the appropriate SS telephone extension. The licensee said that more formal permanent controls would be implemented to ensure ready operator access after the completion of their evaluation in August 3, 1992.

5. <u>Exit Meeting</u>

An exit meeting was held by the NRC examiners with representatives of the licensee's staff on June 12, 1992 to discuss the NRC findings.

The licensee staff acknowledged the examination team's findings, and indicated that steps were being taken in these areas to correct the deficiencies. The licensee committed to respond to the NRC concerns and forwarded a written response (Attachment B) documenting their commitments. This document also outlined the licensee's schedule for resolving the issues.

ATTACHMENT A

POST WRITTEN EXAMINATION REVIEW AND COMMENT RESOLUTION

Arizona Public Service Company
PALO VERDE NUCLEAR GENERATING STATION
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

JAMES M. LEVINE
VICE PRESIDENT
NUCLEAR PRODUCTION

102-02173-JML/TRB/GEC
June 11, 1992

Mr. John B. Martin
Regional Administrator, Region V
U. S. Nuclear Regulatory Commission
1450 Maria Lane, Suite 210
Walnut Creek, CA 94596-5368

Dear Mr. Martin:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Operator License Examination
File: 92-005-419.06; 92-056-026

Arizona Public Service Company (APS) is providing the following information in accordance with the guidance of Examiner Standard ES-201, "PRE-EXAMINATION ACTIVITIES," (NUREG-1021, Revision 6, dated June 1, 1990):

1. APS has conducted a review of both the "Nuclear Regulatory Commission Operator Licensing Examination" and "Nuclear Regulatory Commission Senior Operator Licensing Examination," administered to PVNGS operator and senior operator candidates on June 1, 1992, and has no comments on the written examination.

2. The Pre-Examination and Post-Examination Security Agreements executed by those APS employees allowed access to the NRC prepared examination material prior to and/or during the examination will be provided to the Chief Examiner following completion of the examination in accordance with the guidance of ES-201.

Additionally, APS is evaluating the concerns identified by the NRC Chief Examiner during the course of the examination and expects to provide responses to these concerns by June 19, 1992.

Please contact Gary Clyde at (602) 393-5760 with any questions.

Sincerely,

James M. Levine

JML/TRB/GEC/gec

cc: W. F. Conway (9082)
D. F. Kirsch
G. W. Johnston
T. R. Meadows
R. M. Cross
D. H. Coe
A. C. Gehr
A. H. Gutterman

ATTACHMENT B

PVNGS RESPONSE TO CONCERNS & OBSERVATIONS IDENTIFIED DURING OPERATOR LICENSING EXAMINATION

RECEIVED
NRC
REGION V

'92 JUN 22 A9:59

Arizona Public Service Company
PALO VERDE NUCLEAR GENERATING STATION
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

JAMES M. LEVINE
VICE PRESIDENT
NUCLEAR PRODUCTION

Mr. John B. Martin
Regional Administrator, Region V
U. S. Nuclear Regulatory Commission
1450 Maria Lane, Suite 210
Walnut Creek, CA 94956-5368

102-02178-JML/TRB/PJC
June 19, 1992

Dear Mr. Martin:

Subject: **Palo Verde Nuclear Generating Station (PVNGS)**
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Response to Concerns and Observations
Identified During Operator License Examination
FILE: 92-070-026

Initial and upgrade examinations were conducted at PVNGS from June 1 through June 12, 1992. During the course of the examinations, Mr. Meadows identified three concerns and several observations to Arizona Public Service Company (APS) management. At the exit meeting, Mr. Meadows requested that APS respond to those concerns and observations. APS' response is provided in the enclosure.

Should you have any questions concerning this response, please contact me.

Sincerely,

James M. Levine

JML/TRB/PJC/mh

Enclosure

cc: W. F. Conway
L. F. Miller
D. H. Coe

ENCLOSURE

RESPONSE TO NRC CONCERNS AND OBSERVATIONS
From
PVNGS Initial License Examination-June 1992

NRC CONCERNS

1. NRC CONCERN

An evaluated simulator scenario involving a stuck open spray valve elicited a different response from each of three operating crews. PVNGS does not have an abnormal operating procedure for this event, and the alarm response procedure did not provide specific guidance for operator response and mitigating actions (e.g., isolation of instrument air to containment; which reactor coolant pumps to secure following a decision to trip the reactor).

APS RESPONSE

The alarm response procedure in question, 4xAL-xRK4A, was revised to include guidance for operator actions to mitigate depressurization of the reactor coolant system. Guidance is also provided for tripping the reactor and the reactor coolant pumps and entering the Emergency Operating Procedures (EOPs) if the transient continues. The guidance provided is consistent with that provided in the EOPs. The revised procedure became effective on June 12, 1992. Training on the revision has been incorporated into the requalification cycle which commences on July 6, 1992. The Chief Examiner was provided with a copy of the revised procedure.

2. NRC CONCERN

During an evaluated simulator scenario, non-class instruments for containment temperature and pressure were used to determine whether or not harsh environmental conditions existed in containment. This is considered an engineering issue, as well as, a procedural one.

APS RESPONSE

An evaluation of the use of non-class instruments to determine harsh environmental conditions in containment is being performed in accordance with the APS Incident Investigation Program. Completion of the evaluation is expected by September 15, 1992.

Revision 0.2 of the Emergency Operating Procedure Technical Guidelines, which is scheduled for implementation on August 17, 1992, will contain instructions to the operators to use class 1E containment pressure instruments in conjunction with the non-class containment temperature instruments to determine the existence of harsh conditions in containment. Training on the revised procedure will be incorporated into the upcoming requalification cycle and into initial training.

3. NRC CONCERN

Reactor Operator training for refueling activities needs to be strengthened.

APS RESPONSE

The PVNGS Initial Training Program will be revised to upgrade the training conducted on the refueling floor for reactor operators. The Training Department expects to implement the upgraded program for the next initial training course which is currently scheduled to begin in August 1992.

NRC OBSERVATIONS

1. NRC OBSERVATION

Although operating crew communications during evaluated simulator scenarios were passable, the performance of some of the initial license candidates indicated a need for additional communications training. In particular, the feedback necessary to provide closed loop communications was not consistently used.

APS RESPONSE

PVNGS has expanded efforts to enhance operator performance in the areas of communications and command and control, and improvement is being seen. The following initiatives have been undertaken to assure continued improvement:

- In the Fall of 1992 a communications professional will be engaged to provide additional training and to reinforce existing skills in communications and teamwork for both the PVNGS Training staff and the operating crews.

- Both the Training staff and Operations management will continue to emphasize the need for effective closed loop communications during initial and continuing training simulator scenarios.

- Both the Training staff and Operations management will continue to evaluate operators' communications and command and control skills during initial and continuing training evaluation scenarios.

- The Training staff and Operations management will continue to emphasize the need for and use of effective closed loop communications on shift.

2. NRC OBSERVATION

During an evaluated simulator scenario in which Control Element Assembly Calculator (CEAC) 1 failed, the Plant Monitoring System data point on the B06 panel display indicated trouble with CEAC B. This incident created some question among the operating crew as to which CEAC had actually failed and is considered to be a human factors issue.

APS RESPONSE

An Engineering Evaluation Request (EER) was generated to evaluate the CEAC identification problem. The EER was dispositioned on June 19, 1992, and work documents are being initiated to correct the problem. The corrective actions will be tracked in accordance with the Incident Investigation Program. When the change is implemented, the operating crews will be briefed.

3. NRC OBSERVATION

During an evaluated job performance measure (JPM), the Examiner found that Appendix H of 4xAO-xZZ44, Shutdown Outside the Control Room Due to Smoke or Fire, appeared to be difficult for the operators to use.

APS RESPONSE

The procedure is currently being revised. The revision resolves NRC comments regarding Appendix H. Approval is expected by July 30, 1992. Implementation of the revised procedure is scheduled for October 1992, following completion of training. The Chief Examiner was provided with a copy of the draft procedure.

4A. NRC OBSERVATION

During an evaluated JPM involving an inter-system leak cued from a radiation monitoring system alarm, the Examiner found that the operator was required to use three separate procedures before he arrived at the appropriate mitigating actions. The procedures involved were 74RM-9EF41 (RMS Alarm Response), 41AO-1ZZ14 (Excessive RCS Leakrate), and 41AO-1ZZ29 (Reactor Coolant Pump and Motor Emergency).

APS RESPONSE

APS will evaluate the progression of operator actions needed for this scenario to determine if the process can be streamlined. Completion of the evaluation is expected by August 3, 1992. Corrective actions, as appropriate, will be developed and implemented based upon the results of the evaluation.

4B. NRC OBSERVATION

Section 3.0 of the abnormal response procedure 4xAO-xZZ26, Irradiated Fuel Damage, instructs operators to attempt to position damaged fuel in a stable configuration before it instructs them to evacuate personnel from the affected area.

APS RESPONSE

APS will evaluate the procedure to ensure the placement of evacuation instructions within the procedure is clear and appropriate. Completion of the evaluation is expected by September 1, 1992. Corrective actions, as appropriate, will be developed and implemented prior to the next refueling outage.

5. NRC OBSERVATION

The units are not consistent in their methods for control of accessibility to the Remote Shutdown Panel procedures which are kept in cabinets in the Remote Shutdown Room.

APS RESPONSE

A night order has been implemented in all three units requiring the cabinets which contain the Remote Shutdown procedures to be secured with combination locks and to be posted with the Shift Supervisor's telephone extension.

APS is evaluating this accessibility issue to determine appropriate permanent controls that will ensure operators have ready access to the Remote Shutdown procedures. The evaluation is expected to be complete by August 3, 1992. Corrective actions, as appropriate, will be developed and implemented based upon the results of the evaluation.

ENCLOSURE 2

ANSWER SHEETS AND ANSWER KEYS (RO & SRO)

50-528/OL-92-01

ENCLOSURE 3

SIMULATION FACILITY REPORT

50-528/OL-92-01

Enclosure 3

SIMULATION FACILITY REPORT

Facility Licensee: Palo Verde Nuclear Generating Station, Units 1, 2 and 3

Facility Docket Nos: 50-528/529/530

Operating Tests Administered on: June 2-12, 1992

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

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

None

MASTER

RO

KEY

A N S W E R K E Y

MULTIPLE CHOICE

001	b
002	b
003	a
004	c
005	d
006	c
007	b
008	d
009	b
010	d
011	c
012	b
013	c
014	b
015	b
016	a
017	c
018	d.
019	c
020	d
021	b
022	c
023	d
024	d
025	a
026	d
027	c
028	a
029	c
030	d
031	a
032	c
033	a
034	c
035	c
036	b
037	b
038	c
039	b
040	d
041	c
042	d
043	c.
044	b
045	a

A N S W E R K E Y

046	b	069	b
047	c	070	d
048	b	071	a
049	d	072	a
050	a	073	d
051	d	074	b
052	b	075	c
053	d	076	b
054	c	077	a
055	c	078	c
056	b	079	d
057	b	080	b
058	c	081	d
059	b	082	c
060	c	083	b
061	b	084	d
062	d	085	c
063	c	086	c
064	a	087	c
065	a	088	b
066	d	089	d
067	d	090	c
068	b	091	c

A N S W E R K E Y

092 b

093 b

094 a

095 d

096 b

097 b

098 c

099 c

100 d

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

ANSWER: 001 (1.00)

b.

REFERENCE:

14AC-OFPO2, "Emergency Notification and Response", section 3.1.1 page 5.

[3.5/4.2]

194001K116 ..(KA's)

ANSWER: 002 (1.00)

b.

REFERENCE:

20AC-OSK04, "Protected Area Personnel Access Control" page 9.

[3.1/3.4]

194001K105 ..(KA's)

ANSWER: 003 (1.00)

a.

REFERENCE:

40AC-OZZ06, "Locked Valve and Breaker Control" page 6

[3.6/3.7]

194001K101 ..(KA's)

ANSWER: 004 (1.00)

c.

REFERENCE:

40AC-9OP02 "Conduct of Shift Operations" page 13.

[3.1/4.1]

194001A112 ..(KA's)

ANSWER: 005 (1.00)

d.

REFERENCE:

40AC-9OP02, "Conduct of Shift Operations" page 20, 10 CFR 50.54 (x)&(y).

[4.1/3.9].

194001A102 ..(KA's)

ANSWER: 006 (1.00)

c.

REFERENCE:

40AC-9OP15 Station Tagging and Clearance page 27.

[3.7/4.1]

194001K102 ..(KA's)

ANSWER: 007 (1.00)

b.

REFERENCE:

40AC-9OP15, Station Tagging and Clearance page 48.

[3.7/4.1]

194001K102 ..(KA's)

ANSWER: 008 (1.00)

d.

REFERENCE:

40 AC-90-P15, "Station Tagging and Clearance" page 51.

[3.7/4.1]

194001K102 ..(KA's)

ANSWER: 009 (1.00)

b.

REFERENCE:

EPIP-18 , "Emergency Exposure Guidelines" page 11. 10 CFR 20

[2.8/3.4]

194001K103 ..(KA's)

ANSWER: 010 (1.00)

d.

REFERENCE:

40DP-OOP02, "Relay Resetting" page 9.

[3.6/3.7]

194001K107 ..(KA's)

ANSWER: 011 (1.00)

c.

REFERENCE:

40DP-90P05, "Control Room Data Sheet Instructions" pages 5&6.

[3.4/3.4]

194001A106 ..(KA's)

ANSWER: 012 (1.00)

b.

REFERENCE:

75AC-9RP01, "Radiation Exposure and Access Control" pages 24&25.

[2.8/3.4]

194001K103 ..(KA's)

ANSWER: 013 (1.00)

c.

REFERENCE:

75AC-9RP01, "Radiation Exposure and Access Control" page 18

[2.8/3.4]

194001K103 ..(KA's)

ANSWER: 014 (1.00)

b.

REFERENCE:

41AO-1ZZ29 "Reactor Coolant Pump and Motor Emergency" page 7. NKL01 EO16.

[2.6/2.8]

003000K101 ..(KA's)

ANSWER: 015 (1.00)

b.

REFERENCE:

41AL-1RK4A, Window 4A09B, Label CWP page 124. NKLL21 page 47 EO12

[3.7/3/8]

001000K407 ..(KA's)

ANSWER: 016 (1.00)

a.

REFERENCE:

ARP 41AL-1RK4A page 122 window 4A08B. NKL21 EO26.

[4.1/3.9]

001000A402 ..(KA's)

ANSWER: 017 (1.00)

c.

REFERENCE:

41AO-1ZZ11 "Dropped or Slipped CEA" pages 4 and 6. CEA No. 5 T-MOD 191-SF-006 rod bottom light disabled. NKL21 EO13.

[3.8/3.8]

001000K402 ..(KA's)

ANSWER: 018 (1.00)

d.

REFERENCE:

41OP-1CH01, "CVCS Normal Operations" NKL-32 EO32

[3.3/3.5]

004000K202 ..(KA's)

ANSWER: 019 (1.00)

c.

REFERENCE:

41OP-1CH01 "CVCS Normal Operations" page 108. NKL09 EO34

[3.6/4.0]

004000K101 ..(KA's)

ANSWER: 020 (1.00)

d.

REFERENCE:

41AO-1ZZ37 "Loss of Letdown Flow" page 7. NKL09 page 42 EO06

[3.6/3.9]

004000K511 ..(KA's)

ANSWER: 021 (1.00)

b.

REFERENCE:

41EP-1RO06 "Loss of Offsite Power" page 5, PO actions. NKL-09 EO32 page 94.

[4.1/4.3]

004010A205 ..(KA's)

ANSWER: 022 (1.00)

c.

REFERENCE:

41OP-1SA01, "BOP ESFAS Modules Operation" page 27. NKL28 page 11 EO02

[4.1/4.2]

013000A302 ..(KA's)

ANSWER: 023 (1.00)

d.

REFERENCE:

41OP-1SA01, "BOP ESFAS Modules Operation" page 23. System Description ESFAS, page SA-20.

[3.8/3.8]

013000G007 ..(KA's)

ANSWER: 024 (1.00)

d.

REFERENCE:

SD ESFA, page SA-16. 40EP-9RO02 "Loss of Coolant Accident" page 8.

[4.5/4.7]

013000A403 ..(KA's)

ANSWER: 025 (1.00)

a.

REFERENCE:

41AO-1ZZ16, "Loss of Non-Class 120 VAC 1E Instrument Power" pages 26 and 38. NKL32 page 16 EO02

[3.3/3.7]

015000K201 ..(KA's)

ANSWER: 026 (1.00)

d.

REFERENCE:

NKL32 pages 38&39 EO07

[4.1/4.2]

015000K101 ..(KA's)

ANSWER: 027 (1.00)

c.

REFERENCE:

41OP-1SH01 "QSPDS User's Guide" page 6. NKL36 page 22 EO03.

[3.8/4.1]

017020A402 ..(KA's)

ANSWER: 028 (1.00)

a.

REFERENCE:

NKL36 page 9. Tech. Spec. Table 3.3.10

[2.8/3.4]

017000G005 ..(KA's)

ANSWER: 029 (1.00)

c.

REFERENCE:

41AO-1ZZ28, "Inadvertent SIAS and/or CIAS" page 20. NKL-29 page 22.

[4.1/4.3]

022000A301 ..(KA's)

ANSWER: 030 (1.00)

d.

REFERENCE:

41AO-1ZZ43 "Reactor Power Cutback" page 5. NKL23 E001

[3.1/3.2]

059000G001 ..(KA's)

ANSWER: 031 (1.00)

a.

REFERENCE:

Core Data Book Unit 1 Cycle 3 page 173

[3.0/3.3]

059000A207 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

NKL04 page 27. EO02 41OP-1FT01 and 02 "Feedwater Pump Turbine "A" and "B" pages 7 and 6b.

[3.1/3.4]

059000A205 ..(KA's)

ANSWER: 033 (1.00)

a.

REFERENCE:

41EP-1EO01, "Emergency Operations" Appendix O page 2.

[3.9/4.2]

061000A101 ..(KA's)

ANSWER: 034 (1.00)

c.

REFERENCE:

41 EP-1RO05 "Loss of all Feedwater" Appendix B page 18.

[3.6/3.9]

061000K501 ..(KA's)

ANSWER: 035 (1.00)

c.

REFERENCE:

41OP-1EC01, Essential Chilled Water Train "A" (EC) Appendix C page 5.

[4.1/4.2]

013000A302 ..(KA's)

ANSWER: 036 (1.00)

b.

REFERENCE:

41OP-1RD01 "Radwaste Drains" page 6.

[3.8/3.7]

068000A404 ..(KA's)

ANSWER: 037 (1.00)

b.

REFERENCE:

41EP-1E001, "Emergency Operations" Appendix B, page 2
NOTE 1 USE RCS Th WITH RCPs RUNNING THEREFORE 652-609 = 43 degrees F.

[3.9/4.1]

002000A104 ...(KA's)

ANSWER: 038 (1.00)

c.

REFERENCE:

41OP-1ZZ01, "Cold Shutdown to Hot Standby Mode 5 to Mode 3", Appendix A page 1.

[3.2/3.4]

002000K105 ..(KA's)

ANSWER: 039 (1.00)

b.

REFERENCE:

41OP-1RC01, "Reactor Coolant Pump Operation" page 7. NKL01 page 74 EO08

[4.0/4.2]

002000K511 ..(KA's)

ANSWER: 040 (1.00)

d.

REFERENCE:

41OP-1ZZ10 "Hot Standby to Cold Shutdown Mode 3 to Mode 5" page 30.

[3.0/3.5]

002000K612 ..(KA's)

ANSWER: 041 (1.00)

c.

REFERENCE:

40 EP-9RO01, "Reactor Trip" page 5 Facility Exam Bank 129 (Partial)

[4.1/4.3]

006050K402 ..(KA's)

ANSWER: 042 (1.00)

d.

REFERENCE:

41A)-1ZZ30, "Inadvertent CSAS" page 7

[4.5/4.3]

026000A401 ..(KA's)

ANSWER: 043 (1.00)

c.

REFERENCE:

40EP-9RO-01, Reactor Trip Appendix A page 5

[3.9/4.2]

006020K406 ..(KA's)

ANSWER: 044 (1.00)

b.

REFERENCE:

41EP-1E001, "Emergency Operations" Appendix C pages 2 and 3. NKL12 page 19.

[3.6/3.9]

006000K603 ..(KA's)

ANSWER: 045 (1.00)

a.

REFERENCE:

System Description Reactor Coolant System, page RC-10. NKL10 page 16

[3.2/3.6]

010000K603 ..(KA's)

ANSWER: 046 (1.00)

b.

REFERENCE:

41AL-RK4A, PZR PRESS HI-LO alarm, window 4A01B page 100.

[3.5/3.5]

010000G001 ..(KA's)

ANSWER: 047 (1.00)

c.

REFERENCE:

41AL-RK4A, PZR PRESS HI-LO alarm, window 4A01B page 100. NKL 10 page 43 E016

[4.0/4.1]

010000K302 ..(KA's)

ANSWER: 048 (1.00)

b.

REFERENCE:

System Description Reactor Coolant System page RC-4. NKL11 page 36 E011

[3.4/3.5]

011000K105 ..(KA's)

ANSWER: 049 (1.00)

d.

REFERENCE:

Tech Spec Bases page B2-5

[3.1/3.3]

012000K502 ..(KA's)

ANSWER: 050 (1.00)

a.

REFERENCE:

NKL39 page 60 E003 NO CONTROLLED REFERENCE AVAILABLE

[3.1/3.3]

012000K404 ..(KA's)

ANSWER: 051 (1.00)

d.

REFERENCE:

41OP-1SF01, "Control Element Drive Mechanism Control System Operation"
NKL21 page 52 E013

[3.2/3.6]

014000A102 ..(KA's)

ANSWER: 052 (1.00)

b.

REFERENCE:

NKL page 9 E001

[2.8/2.9]

016000K403 ..(KA's)

ANSWER: 053 (1.00)

d.

REFERENCE:

System Description Fuel Pool Cooling and Cleanup System page pc-6

[2,9/3.2]

033000K401 ..(KA's)

ANSWER: 054 (1.00)

c.

REFERENCE:

41OP-1DG01 "Emergency Diesel Generator A" page 5 NKL 52 page 27 E005

[3.5/4.0]

064000K411 ..(KA's)

ANSWER: 055 (1.00)

c.

REFERENCE:

41AL-1DGB1 page 11, Window DGB 07A "OVERSPEED ENGINE"

[3.9/4.2]

064000K402 ..(KA's)

ANSWER: 056 (1.00)

b.

REFERENCE:

41OP-1PB01, "4.16KV Class IE Power (PB)" appendix C page 31.

[3.5/3.6]

062000A305 ..(KA's)

ANSWER: 057 (1.00)

b.

REFERENCE:

41-OP1SI01, "Shutdown Cooling Initiation" page 8.

[2.7/2.9]

005000A201 ..(KA's)

ANSWER: 058 (1.00)

c.

REFERENCE:

41-OP1SI01, "Shutdown Cooling Initiation" page 13

[3.5/3.9]

005000K411 ..(KA's)

ANSWER: 059 (1.00)

b.

REFERENCE:

41OP-1ZZ01 "Cold Shutdown to Hot Standby Mode 5 to Mode 3" Appendix A page 1.

[2.7/2.9]

007000A102 ..(KA's)

ANSWER: 060 (1.00)

c.

REFERENCE:

41OP-1NC01 "Nuclear Cooling Water (NC)" page 6

[3.2/3.0]

008000A301 ..(KA's)

ANSWER: 061 (1.00)

b.

REFERENCE:

78OP-9FX03 "Spent Fuel Handling Machine" page 21, Appendix A page 1 of 2.

[2.7/2.9]

034000G010 ..(KA's)

ANSWER: 062 (1.00)

d.

REFERENCE:

41OP-1SF05 "Operation of the Steam Bypass Control System" page 12

[2.9/3.1]

041020A406 ..(KA's)

ANSWER: 063 (1.00)

c.

REFERENCE:

41A0-1ZZ02 "Load Rejection" Appendix L page 1

[2.7/2.9]

041020K603 ..(KA's)

ANSWER: 064 (1.00)

a.

REFERENCE:

41AO-1ZZ03 "Loss of Turbine Cooling Water" page 5.

[2.6/2.8]

045000G010 ..(KA's)

ANSWER: 065 (1.00)

a.

REFERENCE:

AOP 41AO-1ZZ35, "Continuous CEA Withdrawal" page 4. NKL21 EO26.

[3.9/4.0]

000001G010 ..(KA's)

ANSWER: 066 (1.00)

d.

REFERENCE:

41AO-1ZZ29, "Reactor Coolant Pump And Motor Emergency" page 16.
NKL01 EO 15.

[4.0/4.2]

000015A122 ..(KA's)

ANSWER: 067 (1.00)

d.

REFERENCE:

41AO-1ZZ29 "Reactor Coolant Pump and Motor Emergency" page 16.
NKL01 EO 14.

[3.4/3.4]

000015G010 ..(KA's)

ANSWER: 068 (1.00)

b.

REFERENCE:

NLK09 Chemical and Volume Control System, pages 190 and 191. EO 78

[4.1/4.4]

000024K301 ..(KA's)

ANSWER: 069 (1.00)

b.

REFERENCE:

41AO-1ZZ01 "Emergency Boration" page 9 and appendix B. NLK09 EO 63d.

[3.4/4.2]

000024A204 ..(KA's)

ANSWER: 070 (1.00)

d.

REFERENCE:

41AO-1ZZ11, " Dropped or Slipped CEA" page 6. NKL21 pages 98 and 120 .
EO 19

[3.7/3.9]

000003A201 ..(KA's)

ANSWER: 071 (1.00)

a.

REFERENCE:

NKL21 page 35 EO 004.

[3.6/3.4]

000005A101 ..(KA's)

ANSWER: 072 (1.00)

a.

REFERENCE:

41EP-1E001, "Emergency Operations" page 10 of 23.

[3.7/3.7]

000011A213 ..(KA's)

ANSWER: 073 (1.00)

d.

REFERENCE:

41A001ZZ04, "Loss OF Plant Cooling Water" page 13. NKL31 E004.

[3.6/3.6]

000026A103 ..(KA's)

ANSWER: 074 (1.00)

b.

REFERENCE:

41EP-1E001, Emergency Operations page 4.

[4.1/4.2]

000029G012 ..(KA's)

ANSWER: 075 (1.00)

c.

REFERENCE:

41EP-1E001, Emergency Operations page 4.

[4.5/4.5]

000029G010 ..(KA's)

ANSWER: 076 (1.00)

b.

REFERENCE:

41EP-1E001, Emergency Operations page 13.

[4.0/4.6]

000074A206 ..(KA's)

ANSWER: 077 (1.00)

a.

REFERENCE:

40EP-9RO04, "Excess Steam Demand" page 20, 40DP-9AP06, "Emergency Operations" page 34.

[4.5/4.7]

000040K304 ..(KA's)

ANSWER: 078 (1.00)

c.

REFERENCE:

Tech Spec Bases 3.4.1.1

[4.1/4.4]

000040K105 ..(KA's)

ANSWER: 079 (1.00)

d.

REFERENCE:

41AO-1ZZ07, "Loss Of Condenser Vacuum" page 5. NKL07 E001-5

[2.8/3.1]

000051K301 ..(KA's)

ANSWER: 080 (1.00)

b.

REFERENCE:

41EP-1RO07 Blackout page 7.

[4.3/4.6]

000055K302 ..(KA's)

ANSWER: 081 (1.00)

d.

REFERENCE:

41EP-1RO07, "Blackout" page 8.

[3.6/3.7]

000055G007 ..(KA's)

ANSWER: 082 (1.00)

c.

REFERENCE:

40DP-9AP13, "Blackout Technical Guideline" page 19. NKL E020

[3.8/4.1]

000055G006 ..(KA's)

ANSWER: 083 (1.00)

b.

REFERENCE:

41AO-1ZZ27, "Shutdown Outside Control Room" page 18

[3.9/4.0]

000068K201 ..(KA's)

ANSWER: 084 (1.00)

d.

REFERENCE:

41EP-1ZZ01, "Emergency Operations" page 5 and 40DP-9AP08. "Loss of Coolant Guidelines" page 23. 41EP-1E001, "Emergency Operations" Appendix D, RCP NPSH Curves

[4.1/4.2]

000011K314 ..(KA's)

ANSWER: 085 (1.00)

c.

REFERENCE:

41EP-1RO07, "Station Blackout" page 7.

[3.9/4.0]

000055G012 ..(KA's)

ANSWER: 086 (1.00)

c.

REFERENCE:

40DP-9APO-6, "Emergency Operation Guideline" page 29.

[4.1/4.3]

000007G011 ..(KA's)

ANSWER: 087 (1.00)

c.

REFERENCE:

41EP-1E001, "Emergency Operations" Appendix B. 409R001, "Reactor Trip" page 4. NKS32 E006.

[3.6/3.6]

000007G007 ..(KA's)

ANSWER: 088 (1.00)

b.

REFERENCE:

NKS32-01 page 12 E001

[3.4/3.7]

000008A212 ..(KA's)

ANSWER: 089 (1.00)

d.

REFERENCE:

40DP-9AP08, "Loss of Coolant Technical Guideline" page 31. NKS32 EO 20

[3.7/3.9]

000009G007 ..(KA's)

ANSWER: 090 (1.00)

c.

REFERENCE:

40EP-9R002, "Loss of Coolant Accident" page 12 step 3.17.

[3.5/3.8]

000022K302 000009A107 ..(KA's)

ANSWER: 091 (1.00)

c.

REFERENCE:

41AO-1ZZ22, "Loss of Shutdown Cooling" page 10.

[3.4/3.7]

000025A207 ..(KA's)

ANSWER: 092 (1.00)

b.

REFERENCE:

System Description Volume 1 RC page RC-20

[2.6/2.8]

000027K203 ..(KA's)

ANSWER: 093 (1.00)

b.

REFERENCE:

40EP-9AP09, Steam Generator Tube Rupture Technical Guidelines pages 72&74.

[3.7/4.0]

000037K305 ..(KA's)

ANSWER: 094 (1.00)

a.

REFERENCE:

40DP-9AP09 "Steam Generator Tube Rupture Technical Guideline" page 63.
NKS32 E005

[4.2/4.5]

000038K306 ..(KA's)

ANSWER: 095 (1.00)

d.

REFERENCE:

41EP-1RO05, "Loss of All Feedwater" page 4.

[3.2/3.2]

000054G010 ..(KA's)

ANSWER: 096 (1.00)

b.

REFERENCE:

74RM-9EF41, "Radiation Monitoring System Alarm Response" page 37.
System Description, Gaseous Radwaste System page 4.

[3.7/4.2]

000060A205 ..(KA's)

ANSWER: 097 (1.00)

b.

REFERENCE:

41AO-1ZZ17, "Loss of 125 VDC Class 1E Electrical Power" page 5.

[4.0/4.2]

000058K302 ..(KA's)

ANSWER: 098 (1.00)

c.

REFERENCE:

41AO-1ZZ53, "Loss of Refueling Pool and/or Spent Fuel Pool Level" page 14 step 2.8.1.

[3.1/3.7]

000036A104 ..(KA's)

ANSWER: 099 (1.00)

c.

REFERENCE:

41AO-1ZZ06, "Loss of Instrument Air" page 10.

[2.8/3.2]

000065A207 ..(KA's)

ANSWER: 100 (1.00)

d.

REFERENCE:

41AO-1ZZ06, "Loss of Instrument Air" page 12

[2.9/3.3]

000065A208 ..(KA's)

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

RO Exam PWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
001	1.00	9000001
002	1.00	9000003
003	1.00	9000004
004	1.00	9000006
005	1.00	9000007
006	1.00	9000009
007	1.00	9000010
008	1.00	9000011
009	1.00	9000013
010	1.00	9000014
011	1.00	9000015
012	1.00	9000017
013	1.00	9000018
014	1.00	9000020
015	1.00	9000021
016	1.00	9000022
017	1.00	9000023
018	1.00	9000024
019	1.00	9000025
020	1.00	9000026
021	1.00	9000027
022	1.00	9000028
023	1.00	9000029
024	1.00	9000030
025	1.00	9000031
026	1.00	9000032
027	1.00	9000033
028	1.00	9000034
029	1.00	9000035
030	1.00	9000036
031	1.00	9000037
032	1.00	9000038
033	1.00	9000040
034	1.00	9000041
035	1.00	9000043
036	1.00	9000045
037	1.00	9000046
038	1.00	9000047
039	1.00	9000048
040	1.00	9000049
041	1.00	9000050
042	1.00	9000051
043	1.00	9000052
044	1.00	9000053
045	1.00	9000054
046	1.00	9000055
047	1.00	9000056
048	1.00	9000057
049	1.00	9000058

RO Exam PWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
050	1.00	9000059
051	1.00	9000060
052	1.00	9000061
053	1.00	9000063
054	1.00	9000064
055	1.00	9000065
056	1.00	9000068
057	1.00	9000069
058	1.00	9000070
059	1.00	9000071
060	1.00	9000072
061	1.00	9000073
062	1.00	9000074
063	1.00	9000075
064	1.00	9000076
065	1.00	9000077
066	1.00	9000078
067	1.00	9000079
068	1.00	9000080
069	1.00	9000081
070	1.00	9000082
071	1.00	9000083
072	1.00	9000084
073	1.00	9000085
074	1.00	9000087
075	1.00	9000088
076	1.00	9000089
077	1.00	9000092
078	1.00	9000093
079	1.00	9000094
080	1.00	9000095
081	1.00	9000096
082	1.00	9000097
083	1.00	9000099
084	1.00	9000100
085	1.00	9000101
086	1.00	9000102
087	1.00	9000104
088	1.00	9000105
089	1.00	9000106
090	1.00	9000107
091	1.00	9000108
092	1.00	9000110
093	1.00	9000113
094	1.00	9000115
095	1.00	9000117
096	1.00	9000118
097	1.00	9000119
098	1.00	9000121

RO Exam PWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
099	1.00	9000123
100	1.00	9000124

	100.00	

	100.00	

RO Exam PWR Reactor

Organized by KA Group

PLANT WIDE GENERICS

QUESTION	VALUE	KA
005	1.00	194001A102
011	1.00	194001A106
004	1.00	194001A112
003	1.00	194001K101
008	1.00	194001K102
007	1.00	194001K102
006	1.00	194001K102
012	1.00	194001K103
013	1.00	194001K103
009	1.00	194001K103
002	1.00	194001K105
010	1.00	194001K107
001	1.00	194001K116

PWG Total	13.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
016	1.00	001000A402
017	1.00	001000K402
015	1.00	001000K407
014	1.00	003000K101
019	1.00	004000K101
018	1.00	004000K202
020	1.00	004000K511
021	1.00	004010A205
022	1.00	013000A302
035	1.00	013000A302
024	1.00	013000A403
023	1.00	013000G007
026	1.00	015000K101
025	1.00	015000K201
028	1.00	017000G005
027	1.00	017020A402
029	1.00	022000A301
032	1.00	059000A205
031	1.00	059000A207
030	1.00	059000G001
033	1.00	061000A101
034	1.00	061000K501
036	1.00	068000A404

R O Exam PWR Reactor

Organized by KA Group

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
PS-I Total	23.00	

Group II

QUESTION	VALUE	KA
037	1.00	002000A104
038	1.00	002000K105
039	1.00	002000K511
040	1.00	002000K612
044	1.00	006000K603
043	1.00	006020K406
041	1.00	006050K402
046	1.00	010000G001
047	1.00	010000K302
045	1.00	010000K603
048	1.00	011000K105
050	1.00	012000K404
049	1.00	012000K502
051	1.00	014000A102
052	1.00	016000K403
042	1.00	026000A401
053	1.00	033000K401
056	1.00	062000A305
055	1.00	064000K402
054	1.00	064000K411

PS-II Total	20.00	

Group III

QUESTION	VALUE	KA
057	1.00	005000A201
058	1.00	005000K411
059	1.00	007000A102
060	1.00	008000A301
061	1.00	034000G010
062	1.00	041020A406
063	1.00	041020K603
064	1.00	045000G010

PS-III Total	8.00	

RO Exam PWR Reactor

Organized by KA Group

PLANT SYSTEMS

QUESTION	VALUE	KA
PS Total	51.00	

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
071	1.00	000005A101
066	1.00	000015A122
067	1.00	000015G010
069	1.00	000024A204
068	1.00	000024K301
073	1.00	000026A103
092	1.00	000027K203
078	1.00	000040K105
077	1.00	000040K304
079	1.00	000051K301
082	1.00	000055G006
081	1.00	000055G007
085	1.00	000055G012
080	1.00	000055K302
083	1.00	000068K201
076	1.00	000074A206

EPE-I Total	16.00	

Group II

QUESTION	VALUE	KA
065	1.00	000001G010
070	1.00	000003A201
087	1.00	000007G007
086	1.00	000007G011
088	1.00	000008A212
089	1.00	000009G007
072	1.00	000011A213
084	1.00	000011K314
090	1.00	000022K302
091	1.00	000025A207
075	1.00	000029G010
074	1.00	000029G012
093	1.00	000037K305
094	1.00	000038K306
095	1.00	000054G010

R O Exam PWR Reactor

Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group II

QUESTION	VALUE	KA
097	1.00	000058K302
096	1.00	000060A205
EPE-II Total	17.00	

Group III

QUESTION	VALUE	KA
098	1.00	000036A104
099	1.00	000065A207
100	1.00	000065A208
EPE-III Total	3.00	
EPE Total	36.00	
Test Total	100.00	

MASTER

Nuclear Regulatory Commission
Operator Licensing
Examination

This document is removed from
Official Use Only category on
date of examination.

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
REACTOR OPERATOR LICENSE
REGION 5

CANDIDATE'S NAME: _____

FACILITY: Palo Verde 1, 2, & 3

REACTOR TYPE: PWR-CE80

DATE ADMINISTERED: 92/06/01

INSTRUCTIONS TO CANDIDATE:

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

TEST VALUE	CANDIDATE'S SCORE	%	
100.00	_____	___ %	TOTALS
	FINAL GRADE		

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

_____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

3. Restroom trips are to be limited and only one 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.

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

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

6. Mark your answers on the answer sheet provided. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

7. 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.

8. 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.

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

10. Show all calculations, methods, or assumptions used to obtain an answer to any short answer questions.

11. Partial credit may be given except on multiple choice questions. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

12. 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.

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

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

15. Ensure all information you wish to have evaluated as part of your answer is on your answer sheet. Scrap paper will be disposed of immediately following the examination.

16. To pass the examination, you must achieve a grade of 80% or greater.

17. There is a time limit of four (4) hours for completion of the examination.

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

QUESTION: 001 (1.00)

A person discovering a fire in any power block shall immediately:

a. fight the fire until help arrives.

b. notify Security at extension 4444.

c. notify the Control Room at extension X206.

d. notify the site Fire Department at extension 1612.

QUESTION: 002 (1.00)

Individuals losing their ACAD/ID badge inside the protected area shall:

a. immediately go to the Security Headquarters Building.

b. immediately notify their supervisor and the Security Shift Supervisor.

c. return to the area where they last used their ACAD/ID badge and commence a search.

d. report it when they exit the protected area at the end of the work day.

QUESTION: 003 (1.00)

An operator removing the sealing device and changing the position of a valve from its locked position shall:

a. return the removed seal to the S.S. for disposal.

b. dispose of the removed seal.

c. return the removed seal to the S.S. for reuse.

d. remove the seal, in such a way that it can be reused to lock the valve in the position it was changed to.

QUESTION: 004 (1.00)

A Reactor Operator in training may manipulate controls at the remote shutdown panel which do NOT affect reactivity or power level:

a. when ordered by the Reactor Operator in the control room during power operation.

b. when performing a surveillance test procedure while in communication with the Reactor Operator.

c. only under the direct supervision of a Licensed Operator.

d. when directed by the Shift Technical Advisor during Emergency Operating Procedure (EOP) usage.

QUESTION: 005 (1.00)

During operation an emergency situation arises for which no procedure is available; and action which departs from Technical Specifications is required immediately to protect the health and safety of the public.

Which ONE of the following describes the course of action the Reactor Operator is authorized to take?

a. Immediately takes whatever action is required without further direction.

b. Notifies the Assistant Shift Supervisor of his intent and performs the required action without further direction.

c. Obtains approval from the Shift Supervisor and only the Shift Supervisor prior to taking any action.

d. Obtains approval from the Shift Supervisor or Assistant Shift Supervisor prior to taking any action.

QUESTION: 006 (1.00)

Which ONE of the following states the requirement for hanging a Red Danger Tag on a piece of equipment which already has a Yellow Caution Tag hanging on it?

a. The yellow caution tag must be cleared and removed before the red danger tag is installed.

b. The red danger tag is hung, the yellow caution tag is replaced with a new one having the same tag number as the red danger tag.

c. The red danger tag is hung with the yellow caution tag, provided there are no tagging conflicts.

d. The yellow caution tag must be temporarily lifted and replaced after the red danger tag is removed.

QUESTION: 007 (1.00)

A clearance is being prepared for a Unit 1 lighting circuit. The controlled copy of the drawing indicates a Field Change Request (FCR) is N/A.

This indicates to the clearance preparer that:

a. the FCR is not applicable and does not require review.

b. it is not applicable to the "13 series prints" but may have been implemented in one or two of the units.

c. the FCR has been changed to a Drawing Change Notice (DCN) and does not require review.

d. the FCR has been completed on all three units and is no longer applicable.

QUESTION: 008 (1.00)

An air operated valve which fails OPEN:

a. cannot be used as an isolation boundary under any circumstances.

b. can be used as an isolation boundary if the air supply is red tagged open.

c. can be used as an isolation boundary if a backup air supply is connected to the valve operator to keep it closed.

d. can be used as an isolation boundary only if it is manually ("jacked") closed and the jacking device becomes the tagging control point.

QUESTION: 009 (1.00)

During emergency conditions the whole body exposure limit for "corrective or protective actions" is:

a. 5 rem.

b. 25 rem.

c. 75 rem

d. 100 rem.

QUESTION: 010 (1.00)

Which ONE of the following states the requirement for resetting a 786 Lockout Relay on a 4.16KV Class 1E bus from the control room?

a. The operator may attempt one reset without obtaining any additional authorization.

b. The cause of the relay trip must be verified locally to NOT have been the result of a phase-to-phase fault prior to attempting reset, with its associated Diesel Generator out of commission.

c. The Diesel Generator must be verified to NOT be supplying the bus prior to attempting reset.

d. Reset may be attempted only if the Diesel Generator is supplying the bus.

QUESTION: 011 (1.00)

Which ONE of the following Control Room Data Sheet entries shall NOT be skipped and shall be taken within one hour of the time specified?

a. Main Generator gross megawatts

b. RCP seal pressures

c. RCP shaft vibration data

d. Safety Injection Tanks pressure and level

QUESTION: 012 (1.00)

The minimum level of management which can authorize an Auxiliary Operator to exceed the PVNGS quarterly radiation exposure administrative limit of 1.0 Rem is:

a. Shift Supervisor

b. Radiation Protection Manager

c. Respective Unit Plant Manager

d. ALARA Committee Chairman

QUESTION: 013 (1.00)

A High Radiation Area is an area where a major portion of the whole body could receive in 1 hour a dose in excess of:

a. 1 mRem but less than 10 mRem.

b. 10 mRem but less than 100 mRem.

c. 100 mRem but less than 1000 mRem.

d. 1000 mRem.

QUESTION: 014 (1.00)

During operation at 100% power Reactor Coolant Pump (RCP) 1A upper thrust bearing is approaching the high temperature alarm setpoint (230 degrees F.). The thrust bearing oil level indicated on the PMS is low (<71%).

Based on the available indications the operator should:

a. trip the reactor and then trip RCP.

b. turn and hold the oil lift pump handswitch to the on position.

c. immediately trip the RCP.

d. commence a rapid power reduction.

QUESTION: 015 (1.00)

Which ONE of the following parameters exceeding the setpoint on all channels will DIRECTLY generate a Control Element Assembly (CEA) Withdrawal Prohibit (CWP)?

a. Tavg/Tref Deviation

b. Local Power Density

c. CEA Position Deviation

d. CEA Group Sequence

QUESTION: 016 (1.00)

Given the following plant conditions:

- reactor power 65%
- Tavg 586 degrees F
- Tref 579 degrees F.
- CEAs in Manual Sequential for ASI control
- "T AVG - T REF HI-LO" annunciator alarm lit on panel B04

Based on the available indications the operator should:

a. increase boron concentration.

b. decrease turbine load.

c. withdraw CEAs.

d. commence a boron dilution.

QUESTION: 017 (1.00)

Unit 1 was operating at 100% power when the following indications and alarms are observed in the control room:

- reactor power decreasing
- "T AVG - T REF HI-LO" annunciator alarm lit on panel B04
- CEA number 5 lower electrical limit (LEL) light illuminated on panel B04
- CWP alarm on annunciator 4A9B
- pressurizer level and pressure decreasing

Which ONE of the following describes the cause of the above indications?

a. Continuous CEA insertion is occurring

b. A CEA has slipped and stopped at an unknown position

c. A CEA has dropped all the way into the core

d. A Group of CEAs has dropped all the way into the core

QUESTION: 018 (1.00)

Which ONE of the following is the "Normal" power supply to Charging Pump #3, CHE-P01?

a. PGA-L31

b. PGB-L32

c. PGA-L33

d. PGB-L36

QUESTION: 019 (1.00)

The Charging Pump Mode Selector switch is in position CH-2, -3, -1 with pressurizer level on program.

Which ONE of the following describes the response if Charging Pump 2 trips and pressurizer level decreases to 23% (100") below program?

a. Charging pumps 3 and 1 will start on level deviation.

b. Charging pump 1 will start due to an interlock signal from Charging pump 2, Charging pump 3 will start on level deviation.

c. Charging pump 1 will start on level deviation signal.

d. Charging pump 3 will start on level deviation signal.

QUESTION: 020 (1.00)

Thirty minutes have elapsed since letdown isolated. AOP 41AO-1ZZ37, "Loss of Letdown Flow" cautions the operator to open the letdown control valve bypass, CHN-HV-526 for at least four minutes before restoring letdown flow.

The reason for this caution is to minimize:

a. thermal shock to the VCT

b. flashing in the letdown heat exchanger

c. thermal shock to the non-regenerative heat exchanger

d. thermal shock to the charging nozzle

QUESTION: 021 (1.00)

Following a Loss of Offsite Power (LOP) with no Safety Injection Actuation Signal (SIAS) all running Charging Pumps are "load shed".

After the diesel generator output breaker closes, all of the previously running charging pumps:

a. will restart immediately.

b. must be manually restarted.

c. will restart following a 40 second time delay .

d. will automatically restart as required by the pressurizer level control system.

QUESTION: 022 (1.00)

A Control Room Essential Filtration Actuation Signal (CREFAS) is received via a "cross trip" from Fuel Building Essential Ventilation Actuation Signal (FBEVAS) initiation.

Which ONE of the following BOP ESFAS Module indications on the CREFAS Module tell the operator that the actuation was from a "cross trip"?

a. "MAN" and "ACTUATE indications light

b. "TEST" "TRIP" AND "ACTUATE" indications light

c. Only the "ACTUATE" indication lights

d. Control room annunciator alarm with no module indications

QUESTION: 023 (1.00)

The BOP ESFAS, CRVIAS trip module, Train "A" is placed in "BYPASS".

Which ONE of the following describes the response if CRVIAS trip module, Train "B" is then also placed in "BYPASS"?

a. Both CRVIAS trip modules Train "A" and Train "B" will be bypassed

b. Only CRVIAS trip module Train "A" will be bypassed

c. Only CRVIAS trip module Train "B" will be bypassed

d. No CRVIAS trip modules will be bypassed

QUESTION: 024 (1.00)

Which ONE of the following is an automatic action associated with a Recirculation Actuation Signal (RAS) signal?

a. High Pressure Safety Injection pumps trip

b. Refueling Water Tank outlet valves close

c. Charging pumps trip

d. ECCS suction from Containment Sump valves open.

QUESTION: 025 (1.00)

Which ONE of the following describes the power supply to the Excore Nuclear Instrumentation Startup Channels?

a. NNN-D11 and D12

b. NNN-D15 and D16

c. PNA-D25 and D26

d. PNA-D27 and D28

QUESTION: 026 (1.00)

Which ONE of the following describes how the Excore Nuclear Instruments provide "raw neutron power" data to the Core Protection Calculators?

a. Through the linear power circuit and its summing amplifier.

b. From the control channel detectors COLSS ASI calculation amplifier output.

c. From the individual log safety channel detector power output amplifier signals.

d. Through the individual subchannel linear amplifiers which receive a signal from each safety channel detector.

QUESTION: 027 (1.00)

Which ONE of the following levels indicated by the Qualified Safety Parameter Display System (QSPDS) is invalid with reactor coolant pump(s) operating during accident conditions?

a. Steam Generator

b. Pressurizer

c. Plenum reactor vessel

d. Upper head reactor vessel

QUESTION: 028 (1.00)

Which ONE of the following inputs to the Qualified Safety Parameter Display System (QSPDS) is required as part of the Technical Specification Accident Monitoring Instrumentation?

a. Core exit thermocouples

b. Reactor power

c. ESFAS actuation status

d. Containment Hydrogen concentration

QUESTION: 029 (1.00)

During operation at 100% power, surveillance testing results in an inadvertent Safety Injection Actuation Signal (SIAS).

Which ONE of the following describes the operation of the Containment Normal Air Cooling Units, HCA-A01A-D?

a. Trip on a SIAS and cannot be restarted until the SIAS is reset

b. Remain running on a SIAS

c. Trip on a SIAS but can be overridden and restarted without resetting the SIAS

d. Trip on a SIAS and automatically restart when the SIAS is reset

QUESTION: 030 (1.00)

Unit 1 is operating at 100% power with the Reactor Power Cutback System (RPCS) "AUTO ACTUATE OUT OF SERVICE" pushbutton lit, when Reactor Feedwater Pump Turbine "B" trips.

Which ONE of the following describes the required operator action?

a. Push the "Loss of Feed Pump" pushbutton on RPCS

b. Trip the reactor and verify turbine trips

c. Push the "Drop Subgroups" pushbutton on RPCS and manually lower turbine power to 65% with the load limiter potentiometer

d. Simultaneously push the "Loss of Feed pump" and "Drop Subgroups" pushbuttons on the RPCS

QUESTION: 031 (1.00)

Unit 1 is operating at 72% power with the Reactor Power Cutback System in service when Reactor Feed Water Pump Turbine "B" trips.

Which ONE of the following describes the CEA Subgroups which will drop into the Core?

a. None

b. Subgroup 4

c. Subgroup 4 and 5

d. Subgroup 4, 5, and 22

QUESTION: 032 (1.00)

While operating at 100% power Feed Water Pump suction pressure drops to less than 255 psig.

Which ONE of the following describes the automatic response of the feedwater system?

a. Both pumps trip immediately

b. Both pumps trip after 10 second time delay

c. One pump trips after 10 second time delay, one pump trips after 15 second time delay

d. Both pumps trip after 15 second time delay

QUESTION: 033 (1.00)

After an automatic Auxiliary Feedwater Actuation Signal (AFAS) the following plant conditions exist:

- #1 steam generator (SG) level 36% WR and increasing
- #2 steam generator level 41% WR and steady
- auxiliary feedwater (AFW) regulating valves are overridden and throttled to maintain level
- #1 SG auxiliary feedwater isolation valve is NOT overridden and is open
  #2 SG auxiliary feedwater isolation valve is NOT overridden and is closed

The AFAS signal may:

a. not be reset because the SG levels have not been above 40% WR level.

b. be reset since the AFW regulating valves are in override.

c. not be reset since the AFW isolation valves are not overridden.

d. be reset because the SG levels are being maintained within the required band.

QUESTION: 034 (1.00)

Following an automatic initiation of Auxiliary Feedwater both Steam Generators (SG) levels have increased to 4% WR and the Secondary Operator has overridden and throttled the AFW valves.

Which ONE of the following describes the minimum required feedflow the Secondary Operator must establish to at least one SG?

a. Enough flow to maintain SG level

b. Enough flow to ensure RCS Tc is stable

c. At least 250 gpm (0.14 mlbm/hr)

d. Greater than 500 gpm (0.28 mlbm/hr) but less than 900 gpm (0.5 mlbm/hr)

QUESTION: 035 (1.00)

Following an automatic initiation on a Safety Injection Actuation Signal the "A" Essential Chilled Water Compressor trips on low chilled water temperature.

The control room operator should:

a. trip the chill water pump.

b. secure the "A" train ECCS equipment if the "B" train equipment is available.

c. allow the compressor to automatically restart when the chilled water trip clears.

d. direct the auxiliary operator to provide alternate ECCS room cooling by closing doors and realigning ventilation.

QUESTION: 036 (1.00)

Which ONE of the following describes the response of the Containment Radwaste Sump System to an automatic Engineered Safeguards Features Actuation Signal (ESFAS)?

a. Inboard and outboard containment isolation valves, RD-UV-23 and RD-UV-24 close on a CIAS and cannot be overridden open

b. Inboard and outboard containment isolation valves, RD-UV-23 and RD-UV-24 close on a CIAS and can be overridden open

c. Inboard containment isolation valve, RD-UV-23 closes on a CIAS, outboard valve RD-UV-24 closes on a SIAS, each valve can be overridden open

d. Inboard and outboard containment isolation valves, RD-UV-23 and RD-UV-24 close on a MSIS and cannot be overridden open

QUESTION: 037 (1.00)

Given the following data:

- The plant is at full power	100%
- T-hot	609 degrees F
- T-cold	554 degrees F
- Highest CET	613 degrees F
- Pressurizer temperature	652 degrees F
- Pressurizer pressure	2250 psia

Determine the most accurate value for bulk RCS reactor subcooling.

a. 39.0 degrees F

b. 43.0 degrees F

c. 70.5 degrees F

d. 98.0 degrees F

QUESTION: 038 (1.00)

During a Reactor Coolant System (RCS) heatup a pressurizer steam bubble is being formed. The nitrogen (N2) overpressure in the pressurizer is being reduced by venting the pressurizer to the Reactor Drain Tank (RDT).

Which ONE of the following describes the expected response of the RDT to the final phase of venting (most N2 is purged)?

a. Level will increase with no change in pressure

b. Level will not change but pressure will increase

c. Level and pressure will increase

d. Level will decrease and pressure will remain the same

QUESTION: 039 (1.00)

When starting the first reactor coolant pump with pressurizer level less than 33%, Steam Generator secondary side temperature may not exceed any RCS cold leg temperature by 20 degrees F.

Which ONE of the following is the basis for that precaution?

a. Prevents excessive heat transfer from the RCS to the S/G

b. Minimize potential for lifting Low Temperature Over Pressure (LTOP) relief valves

c. Minimize thermal shock to the S/G tubes which could result in possible S/G Tube Rupture

d. Prevents thermal shock to the RCP which could result in possible RCP Seal Failure

QUESTION: 040 (1.00)

During Cooldown, Technical Specifications require the LTOP's to be in service when the lowest indicated RCS Tc decreases to less than:

a. 227 degrees F.

b. 224 degrees F

c. 220 degrees F

d. 214 degrees F

QUESTION: 041 (1.00)

Following a Safety Injection Actuation Signal (SIAS) when the High Pressure Safety Injection (HPSI) throttle criteria are met, the HPSI injection valves are throttled one injection valve per train at a time.

Which ONE of the following is the bases for only throttling one HPSI injection valve per train at a time?

a. Maintain the minimum flow requirements of the HPSI delivery curve

b. Ensure acceptable RCS subcooling is met

c. Minimize potential valve damage

d. Minimize pressurizer level perturbations

QUESTION: 042 (1.00)

Following an inadvertent initiation of Containment Spray (CS) the operator stops the CS pumps.

Which ONE of the following describes how the CS pumps would be restarted if a Safety Injection Actuation Signal (SIAS) was received?

a. Resetting the CSAS on Panel B05

b. They will automatically restart

c. Place the control switch in the start position to override the SIAS signal then release the switch. Place the handswitch to the stop position to clear the "anti-pump" condition.

d. Place the control switch in the start position to override the SIAS signal then release the switch. Place the handswitch to the start position again.

QUESTION: 043 (1.00)

Following a Reactor Trip with a SIAS, the Safety Injection termination criteria are met.

Which ONE of the following actions is required to stop the HPSI pumps?

a. Reset the SIAS actuation, then place the handswitch in stop

b. Hold the handswitch to stop for greater than 5 seconds.

c. Take the handswitch to start then to stop

d. Hold the handswitch in stop while resetting the SIAS actuation

QUESTION: 044 (1.00)

Following a Loss Of Coolant Accident (LOCA), RCS pressure is at 250 psia and dropping.

Which ONE of the following statements describes the "NORMAL" status of Emergency Core Cooling INJECTION flow?

a. Constant HPSI flow exists with no LPSI flow.

b. Increasing HPSI flow exists with no LPSI flow.

c. Constant HPSI and LPSI flow exists.

d. Increasing HPSI and LPSI flow exists.

QUESTION: 045 (1.00)

With the Pressurizer Spray Valves fully closed, there is still approximately 6-10 gpm continuous flow through the spray bypass valves.

Which ONE of the following is the PRIMARY purpose of this flow?

a. Minimize the thermal shock to the spray nozzle.

b. Ensure pressurizer heaters remain energized.

c. Maintain spray valve inlet and outlet piping isothermal.

d. Maintain Pressurizer and RCS loop boron concentrations equalized.

QUESTION: 046 (1.00)

During operation at 100% power the following indications are noted:

- PZR PRESS HI-LO alarm, window 4A01B lit
- actual pressurizer pressure is 2265 psia and increasing
- all pressurizer heaters indicate on

Which ONE of the following is the first priority operator action required?

a. Initiate auxiliary spray

b. Switch pressurizer pressure control to the unaffected channel with handswitch RCN-HS-100

c. Manually initiate normal pressurizer spray flow

d. Deenergize all pressurizer heaters

QUESTION: 047 (1.00)

During operation at 100% power the Pressurizer pressure transmitter for the controlling channel fails high.

Which ONE of the following is the first AUTOMATIC plant response which occurs to mitigate the transient?

a. High pressure reactor trip

b. Spray valves close

c. Low Pressure reactor trip and SIAS

d. Backup heaters energize

QUESTION: 048 (1.00)

During operation at 100% power the "normally running" Charging Pump stops and letdown flow increases to maximum.

Which ONE of the following instrument failures is the cause of these conditions? (assume selected channel is failed)

a. Tc input to reactor regulating system fails high

b. Tc input to reactor regulating system fails low

c. Pressurizer level transmitter output fails low

d. Pressurizer level controller loses power

QUESTION: 049 (1.00)

The Local Power Density (LPD) - High trip is calculated in the Reactor Protection System.

Which ONE of the following information inputs is NOT used "directly" in the LPD calculation?

a. Axial power distribution

b. CEA positions

c. Radial peaking factors

d. RCS pressure

QUESTION: 050 (1.00)

Which ONE of the following pressures is used as the input to the Supplementary Protection System (SPS)?

a. Pressurizer

b. Containment

c. RCS Hot Leg

d. Steam Generator

QUESTION: 051 (1.00)

Which ONE of the following describes the operation of the CEDMCS Electrical Limit?

a. The Lower Electrical limit stops individual CEA insertion at 3/4 inches and lights the "white" control panel light.

b. The Lower Electrical limit stops individual CEA insertion at 5.25 inches and lights the "green" control panel light.

c. The Upper Electrical limit stops individual CEA withdrawal at 145.5 inches and lights the "yellow" control panel light.

d. The Upper Electrical limit stops individual CEA withdrawal at 152 5/8 inches and lights the "red" control panel light.

QUESTION: 052 (1.00)

Which ONE of the following Main Feed Pump parameters is used for initiation of a Reactor Power Cutback on a Loss of Main Feed Pump event?

a. Pump discharge pressure

b. Turbine control oil pressure

c. Pump discharge valve position

d. Turbine steam stop valve position

QUESTION: 053 (1.00)

Which ONE of the following describes the design feature(s) that prevents a pipe break outside of the Spent Fuel Pool (SFP) from draining the pool

a. All SFP Cooling system supply and return piping is provided with siphon breaker holes.

b. The SFP primary makeup valve automatically opens on low level and can makeup inventory at a rate equal to the largest calculated pipe break.

c. The SFP Cooling system valves, piping and pumps are located above the normal operating level.

d. Piping either penetrates the SFP wall above the water level or has siphon breaker holes for pipes which run below the water level.

QUESTION: 054 (1.00)

Diesel Generator "A" is operating in parallel with offsite power in the "test" mode when a Auxiliary Feedwater Actuation Signal (AFAS) is received?

Which ONE of the following describes the automatic response of the Diesel Generator (DG)?

a. DG continues running in "test" mode output breaker trips open

b. DG continues running in "test" mode in parallel with offsite power

c. Output breaker trips open, DG transfers to "emergency" mode and output breaker remains open

d. Output breaker trips open, DG transfers to "emergency" mode and output breaker closes

QUESTION: 055 (1.00)

Which ONE of the following Diesel Generator (DG) Trips will stop the DG in both the "emergency" and "test" modes?

a. Crankcase pressure high

b. Low turbocharger lube oil pressure

c. Engine overspeed

d. Excessive engine vibration

QUESTION: 056 (1.00)

The Green indicating light intensity on a 4.16KV Class 1E breaker cabinet indicates brighter than the light on an adjacent breaker.

This indicates the breaker:

a. is open with the closing springs not charged.

b. has opened because of a 786 protective relay actuation.

c. has a blown fuse in the control power trip coil circuit.

d. has a blown fuse in the control power closing circuit.

QUESTION: 057 (1.00)

While operating in Shutdown Cooling the Operator is cautioned not to operate the Low Pressure Safety Injection (LPSI) pumps in the "rumble range".

This is to prevent damage caused by:

a. exceeding the maximum design flow rate.

b. excessive pump vibration.

c. operation below the minimum required flow rate.

d. cavitation caused by high suction temperature.

QUESTION: 058 (1.00)

Following Boron equalization, Shutdown Cooling Train "A" is aligned to a standby condition. SIA-UV-669, LPSI Pump Mini-flow Recirculation valve is closed.

Which ONE of the following describes how a LPSI pump recirculation flow path is provided?

a. The shutdown cooling train flow control valve is opened.

b. The shutdown cooling train temperature control and warmup valves are opened.

c. The shutdown cooling heat exchanger bypass and warmup valves are open.

d. The shutdown cooling train flow control and temperature control valves are open.

QUESTION: 059 (1.00)

When venting the Pressurizer to the Reactor Drain Tank (RDT) to establish a steam bubble, the RDT pressure is maintained less than 100 psia by venting:

a. directly to the containment atmosphere.

b. via a hose to the refueling purge system.

c. via a hose to the power access purge system .

d. directly to the containment hydrogen control system.

QUESTION: 060 (1.00)

Which ONE of the following indications or alarms would indicate that the Standby Nuclear Cooling (NC) Water pump has automatically started?

a. NC header pressure indicates 90 psig

b. High system pressure annunciator alarm in the control room

c. Amber light indicated for the standby pump

d. NC pump motor current indicates less than 135 AMPS

QUESTION: 061 (1.00)

If the Spent Fuel Handling Machine hoist is NOT at the "Up" Limit and a fuel bundle is on the hoist, Bridge and Trolley movement is:

a. unrestricted.

b. restricted to slow speed.

c. restricted to fast speed.

d. stopped.

QUESTION: 062 (1.00)

Which ONE of the following describes the Unit 3 Steam Bypass Control System (SBCS) valve alignment when operating at 90% power?

a. All SBCS valves in "Auto with an automatic permissive"

b. All SBCS valves in "Auto with a manual permissive"

c. 7 SBCS valves in "Auto with an automatic permissive" and 1 in "Manual with a manual permissive"

d. 7 SBCS valves in "Auto with an automatic permissive" and 1 in Automatic with Mode Select Switch OFF

QUESTION: 063 (1.00)

When operating the Atmospheric Dump Valves, the minimum required controller demand signal to initiate valve movement is:

a. 10%.

b. 20%.

c. 30%.

d. 50%.

QUESTION: 064 (1.00)

Following a loss of Turbine Cooling Water (TCW) the Main Turbine is stopped as rapidly as possible by opening the condenser vacuum breakers.

The PRIMARY reason for rapidly stopping the turbine is to:

a. prevent bearing babbitt damage.

b. remove the largest heat load from the TC system.

c. prevent turbine rotor bow.

d. place it on the turning gear as rapidly as possible.

QUESTION: 065 (1.00)

Verification checks provide positive indication of a Continuous CEA withdrawal event occurring.

Which ONE of the following indications require the operator to manually trip the reactor after selecting SB (Standby) on the CEDM MODE SELECT switch?

a. CEA motion in the outward direction

b. One CEA has a inward subgroup deviation of greater than 9.9 inches

c. A twelve fingered CEA is misaligned greater than 6.6 inches

d. A CWP has occurred due to the CEA deviation

QUESTION: 066 (1.00)

Reactor Coolant Pump (RCP) 1A is operating with the following indications?

- suction pressure 2250 psig.
- controlled bleedoff flow 10 gpm
- controlled bleedoff temperature 185 degrees F.

Which ONE of the following indicates the status of the RCP seals?

a. No seals have failed

b. Only NO. 1 seal has failed

c. Only NO. 2 Seal has failed

d. NO. 1 and 2 seals have failed

QUESTION: 067 (1.00)

During operation at 100% power RCP 2A experiences a simultaneous loss of Nuclear Cooling Water (NCW) and Seal Injection.

Which ONE of the following describes the required operator action (in the proper order) if seal injection and NCW cannot be restored?

a. Trip the reactor and stop the RCP after 10 minutes

b. Stop the RCP, trip the reactor and close the bleed-off valve after 1 minute

c. Stop the RCP, trip the reactor within 10 minutes

d. Trip the reactor, stop the RCP within 1 minute and close the bleed-off valve.

QUESTION: 068 (1.00)

Which ONE of the following conditions require Immediate Boration?

a. One hour after a reactor power cutback and CEAs are lower than the transient insertion limit

b. One CEA is NOT fully inserted following a reactor trip

c. Two Charging Pumps are declared inoperable in Mode 6

d. Shutdown margin is 2% in mode 3

QUESTION: 069 (1.00)

Which ONE of the following Boration flowpaths is utilized if the Refueling Water Tank (RWT) level is 65%?

a. Gravity feed from the RWT to the charging pumps via CH-HV-536, RWT to Charging Pumps valve.

b. Gravity feed from the RWT to the Safety Injection System (SIS), to the charging pumps via CH-HV-327, SIS to Charging Pumps valve.

c. Gravity feed from the Spent Fuel Pool to the SIS, to the charging pumps via CH-HV-327, SIS to Charging Pumps valve.

d. Gravity feed from the RWT through CH-V164, Boric Acid Filter Bypass valve, to the charging pumps via CH-UV-514, Boric Acid Makeup to Charging Pumps valve.

QUESTION: 070 (1.00)

While operating at 100% power a CEA slips from 148.5 inches to 137.75 inches.

The pulse counter for that CEA would now indicate:

a. 0.0 inches

b. 137.75 inches

c. 140.75 inches

d. 148.5 inches

QUESTION: 071 (1.00)

A mechanically bound CEA which can be withdrawn, CANNOT be driven into the core because:

a. CEA insertion is normally accomplished by gravity only.

b. the force exerted by the lower gripper is less than the force exerted by the upper gripper.

c. the pulldown coils are a weaker magnet than the lifting coils.

d. the load transfer coil cannot transfer the load of the CEA to engage the lower gripper.

QUESTION: 072 (1.00)

Which ONE of the following post trip indications would determine that a Loss of Coolant Accident was occurring instead of an Main Steam Line Break inside containment?

a. RCS subcooling decreasing

b. RCS pressure decreasing

c. Containment pressure increasing

d. Pressurizer level decreasing

QUESTION: 073 (1.00)

Following a loss of Plant Cooling Water supply to the Nuclear Cooling Water (NCW) system, the Essential Cooling Water (ECW) System is used to supply critical NCW loads.

Which ONE of the following loads CANNOT be supplied by this lineup?

a. Essential Chiller

b. Spent Fuel Pool Heat Exchanger

c. Normal Chiller

d. Letdown Heat Exchanger

QUESTION: 074 (1.00)

The Safety Function Status Check with the highest priority is:

a. Containment Atmosphere Control

b. Reactivity Control

c. RCS Inventory and Pressure Control

d. Maintenance of Vital Auxiliaries

QUESTION: 075 (1.00)

Following receipt of an automatic Reactor Trip signal from low Steam Generator Level, all CEAs failed to insert.

The FIRST operator action taken to establish Reactivity Control is:

a. insert CEAs manually from the CEDMCS operators module.

b. borate the RCS at greater than 40 gpm.

c. manually trip the reactor.

d. de-energize L03 and L10, CEDM-MG power supply load centers.

QUESTION: 076 (1.00)

Under accident conditions with the Auxiliary Spray valves open for pressure control, the presence of a steam void in the Reactor Vessel Upper Head would be indicated by a:

a. decrease in pressurizer pressure.

b. increase in pressurizer level.

c. decrease in pressurizer pressure with a small decrease in pressurizer level.

d. increase in pressurizer pressure with a small increase in pressurizer level.

QUESTION: 077 (1.00)

For a steam line rupture inside containment, 40EP-9RO04, "Excess Steam Demand" requires stabilizing RCS Tc using the Atmospheric Dump valve on the intact Steam Generator after the uncontrolled cooldown has been terminated.

Which ONE of the following is the reason for stabilizing RCS Tc?

a. Prevent an uncontrolled pressurizer level increase, resulting in solid plant operations and possible lifting of a pressurizer safety valve

b. Promote development of natural circulation flow if the RCPs are stopped

c. Ensure minimum RCS subcooling margin is maintained

d. Prevent RCS cooldown and heatup rate from exceeding the limits in Technical Specifications

QUESTION: 078 (1.00)

Which ONE of the following is the most restrictive condition established for Shutdown Margin with the reactor operating at 100% power?

a. Single CEA ejection

b. Steam Line Break inside Containment at BOL

c. Steam Line Break inside Containment at EOL

d. Excessive Steam Generator Feed Flow

QUESTION: 079 (1.00)

The Unit is rapidly reducing Reactor power and Main Turbine load in response to a loss of Condenser vacuum. Reactor Power cannot be reduced fast enough to equalize with Turbine load and maintain TAVE=TREF +/- 2 degrees F.

Which ONE of the following manual operator actions is required?

a. Initiate a Reactor Power Cutback.

b. Trip the Reactor and Main Turbine.

c. Open the Turbine Bypass Valves to the Condenser.

d. Open the Turbine Bypass Valves to the Atmosphere.

QUESTION: 080 (1.00)

During a "Station Blackout" the RCS should be cooled ONLY enough to maintain subcooling.

The basis for that criteria is to prevent:

a. exceeding the steam driven Auxiliary Feed Pump capacity.

b. emptying the pressurizer, causing a loss of subcooling.

c. losing all auxiliary feed water capability as steam pressure is lowered.

d. a reactor restart accident during cooldown.

QUESTION: 081 (1.00)

The following conditions exist following a station blackout:

- Diesel Generator (DG) "A" tagged out for maintenance
- Diesel Generator "B" has started but the output breaker has not automatically closed

Why is it necessary to close the DG "B" output breaker within 15 minutes or secure the DG?

a. Oil accumulation in the cylinders will cause damage when running unloaded.

b. Provide power to a fuel oil transfer pump to refill the day tank.

c. Conserve the batteries to allow energizing alternate sources of power.

d. Provide power for the essential spray pond system which supplies DG cooling.

QUESTION: 082 (1.00)

Which ONE of the following must be used during a station blackout to determine CEA position?

a. Rod Bottom Lights

b. Lower Electrical Limit Lights

c. CPC/CEAC remote operator display

d. CEAC CRT indication

QUESTION: 083 (1.00)

Which ONE of the following evolutions CANNOT be performed from the Remote Shutdown Panel?

a. Open Auxiliary Spray Valves

b. Shutdown Main Feedwater Pumps

c. Initiate an MSIS

d. Open Atmospheric Steam Dump Valves

QUESTION: 084 (1.00)

Following a Reactor Trip from 100% power, Pressurizer pressure is 1300 psia, RCS Tc is 550 degrees F.

Which ONE of the following is the required operator action?

(41EP-1E001, "Emergency Operations" Appendix D, RCP NPSH Curve is attached for your use)

a. Trip one RCP.

b. Trip two RCPs.

c. Trip three RCPs.

d. Trip ALL RCPs.

QUESTION: 085 (1.00)

During a Station Blackout RCS subcooling is maintained using the:

a. Steam Bypass Control Valves.

b. Main Steam Safety Valves.

c. Atmospheric Dump Valves.

d. Pressurizer Safety Valves.

QUESTION: 086 (1.00)

Following a "relatively uncomplicated reactor trip" from 100% power, what is the minimum expected value indicated pressurizer level should attain?

a. 3%

b. 7%

c. 10%

d. 15%

QUESTION: 087 (1.00)

Given the following information after a Reactor Trip and using the appropriate "Normal and Harsh Containment RCS P/T Limits" curve (attached):

- RCS Th 525 degrees F
- Pressurizer Pressure 1200 psia
- Pressurizer Level 31%
- Reactor Vessel Upper Head Level 16%
- Steam Generator #1 WR level 4% with the ADV open and auxiliary feedwater flow established
- Containment Temperature 148 Degrees F.

Determine which parameter will NOT allow HPSI throttling.

a. Subcooled Margin

b. Reactor Vessel Upper Head Level

c. Pressurizer Level

d. Steam Generator Level

QUESTION: 088 (1.00)

Which ONE of the following parameters would indicate that an RCS leak was through a stuck open Pressurizer Safety Valve instead of from the Cold Leg?

a. Containment temperature increasing

b. Pressurizer level increasing

c. Containment sump level increasing

d. Pressurizer pressure decreasing

QUESTION: 089 (1.00)

Following a LOCA the Primary Operator should not anticipate and prematurely initiate a Recirculation Actuation Signal (RAS) because:

a. the LPSI pumps are required to provide sufficient decay heat removal until the time calculated for the Refueling Water tank (RWT) level to decrease to the RAS setpoint.

b. the charging pumps must be placed in the "pull to lock" position after the RAS but they are still required for RCS inventory control until RWT level decreases to the RAS setpoint.

c. the entire contents of the RWT to the RAS setpoint are required to be injected to ensure adequate shutdown margin is obtained.

d. the entire contents of the RWT to the RAS setpoint are required to prevent air binding pumps and consequently the loss of both heat removal loops.

QUESTION: 090 (1.00)

If the charging pumps are still required for auxiliary spray or RCP seal injection following a RAS signal, the suction is transferred to the:

a. Reactor water makeup tank.

b. Boric acid makeup tank.

c. Spent Fuel Pool.

d. Volume control tank.

QUESTION: 091 (1.00)

Which ONE of the following parameters would be the most likely cause of an "air bound" LPSI pump, while running for Shutdown Cooling in mid-loop operations?

a. Throttled flow.

b. Rising suction temperature.

c. Lowering suction level in the hot leg.

d. Lowering suction pressure in the hot leg.

QUESTION: 092 (1.00)

The unit is operating at 100% power when a failure of the controlling Pressurizer pressure instrument caused Pressurizer Pressure to decrease to 2200 psia. The Pressurizer Pressure Controller RCN-PIC-100 was taken to MANUAL.

Which ONE of the following is required to increase pressure back to the normal range?

a. Increase the controller output.

b. Decrease the controller output.

c. Raise the pressure setpoint adjustment.

d. Lower the pressure setpoint adjustment.

QUESTION: 093 (1.00)

During a Steam Generator Tube Leak the operators are cautioned to maintain Reactor Coolant System (RCS) Pressure approximately equal (+/- 50 psi) to Steam Generator (SG) pressure during the cooldown and depressurization.

The reason for this is to:

a. reduce the feed water inventory required for the affected SG.

b. reduce the leak rate from the RCS to the SG.

c. prevent the possibility of Boron dilution and SG chemicals entering the RCS.

d. prevent the loss of RCS subcooling.

QUESTION: 094 (1.00)

During a Steam Generator (SG) Tube Rupture event, it is important to maintain the level in the isolated SG 72-80% WR during cooldown, in order to:

a. minimize the radioactive release in the event of a stuck open SG atmospheric dump valve.

b. reduce flow impingement on undamaged tubes surrounding the rupture.

c. increase the static head in order to reduce break flow.

d. enhance natural circulation flow during the cooldown.

QUESTION: 095 (1.00)

Given the following plant conditions following an automatic reactor trip on low Steam Generator level:

- RCS pressure 1850 psia
- RCS Th 567 degrees F.
- the CRS has entered 41EP-1R005, "Loss of All Feedwater"

Which ONE of the following is the correct Reactor Coolant Pump (RCP) alignment for these conditions?

a. 1A 2A

b. 1B 2B

c. All RCP's operating

d. All RCP's stopped

QUESTION: 096 (1.00)

A High Alarm Trip (RED) is received on "WASTE GAS DECAY TANK DISCHARGE MONITOR," RU-12.

Which ONE of the following is a "FIRST PRIORITY ACTION"?

a. Verify automatic isolation of the Radwaste Building Exhaust System.

b. Verify Waste Discharge valves GRN-UV-34A and 34B have automatically closed.

c. Dispatch an operator to manually close Waste Discharge valves GRN-UV-34A and 34B.

d. Verify Waste Gas Header Flow Control Valve, GRN-FV-33 automatically closes.

QUESTION: 097 (1.00)

During operation at 100% power a loss of 125 VDC Class 1E Bus PKB-M42 occurs.

The operator is required to manually Trip the Reactor because:

a. the associated reactor trip breakers have lost control power and will not trip.

b. the main steam isolation valves fail closed.

c. IA-UV-2, Instrument Air Containment Isolation valve fails closed.

d. the RCP's will also be required to be tripped.

QUESTION: 098 (1.00)

During Refueling a total failure of the Steam Generator Nozzle Dam Seals occurs which results in a loss of Refueling Pool level with an irradiated fuel assembly above the Reactor Core.

Which ONE of the following describes the preferred location the fuel assembly should be placed in?

a. Upender and then lowered to the horizontal position

b. Any available location in the core

c. Its original location in the core

d. Deep area of the pool just above the floor

QUESTION: 099 (1.00)

A gradual loss of Instrument Air (IA) pressure is occurring and system pressure has decreased to 80 psig.

Which ONE of the following describes all of the automatic functions which should have occurred?

a. Standby IA compressor starts

b. IA nitrogen backup valve opens

c. Standby IA compressor starts and IA nitrogen backup valve opens

d. Standby IA compressor starts, IA nitrogen backup valve opens and atmospheric dump valves fail closed.

QUESTION: 100 (1.00)

Which ONE of the following air operated Feed Water valves does NOT fail "AS-IS" following a loss of Instrument Air?

a. Economizer Control

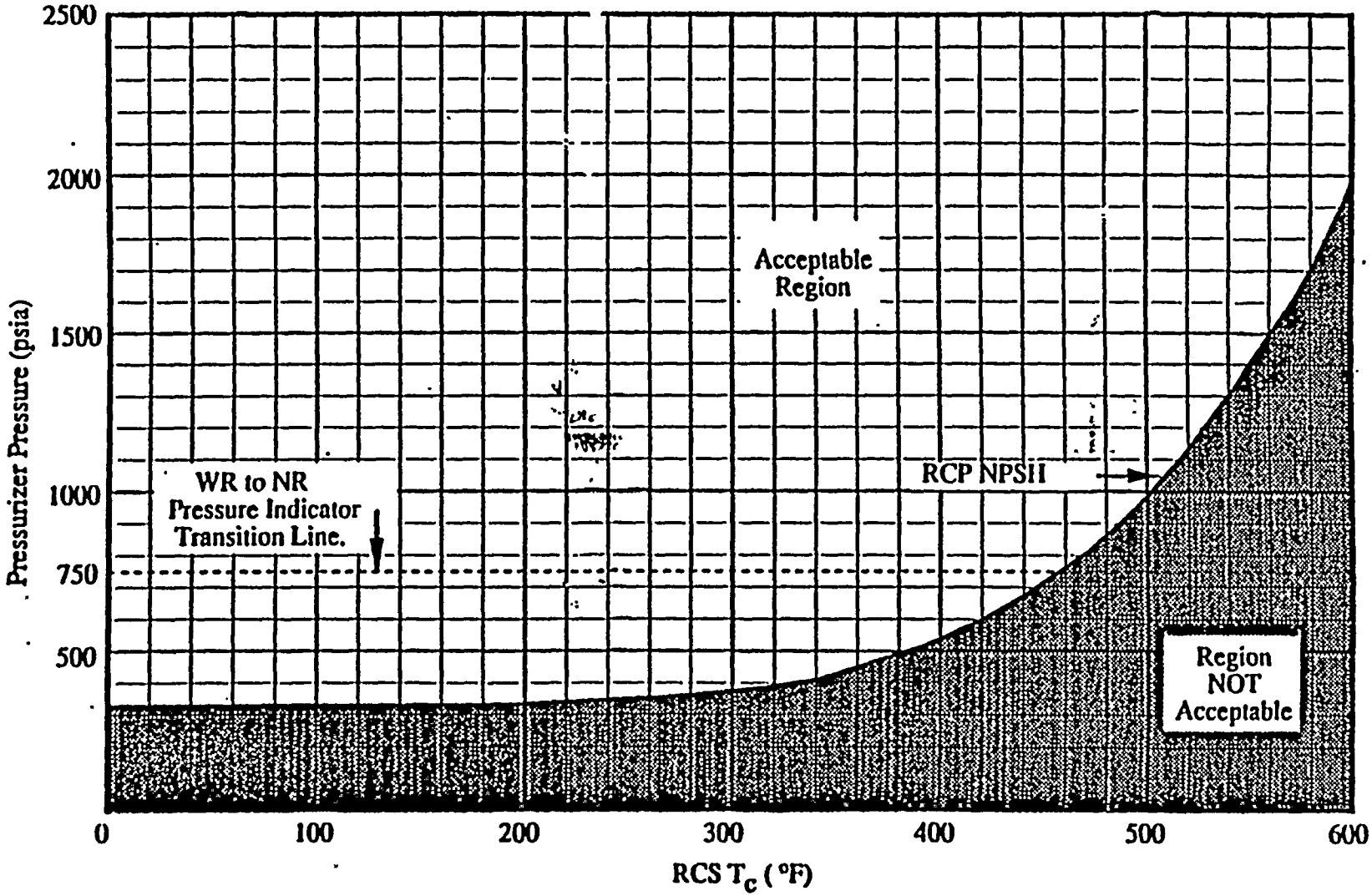
b. Economizer Isolation

c. Downcomer Control

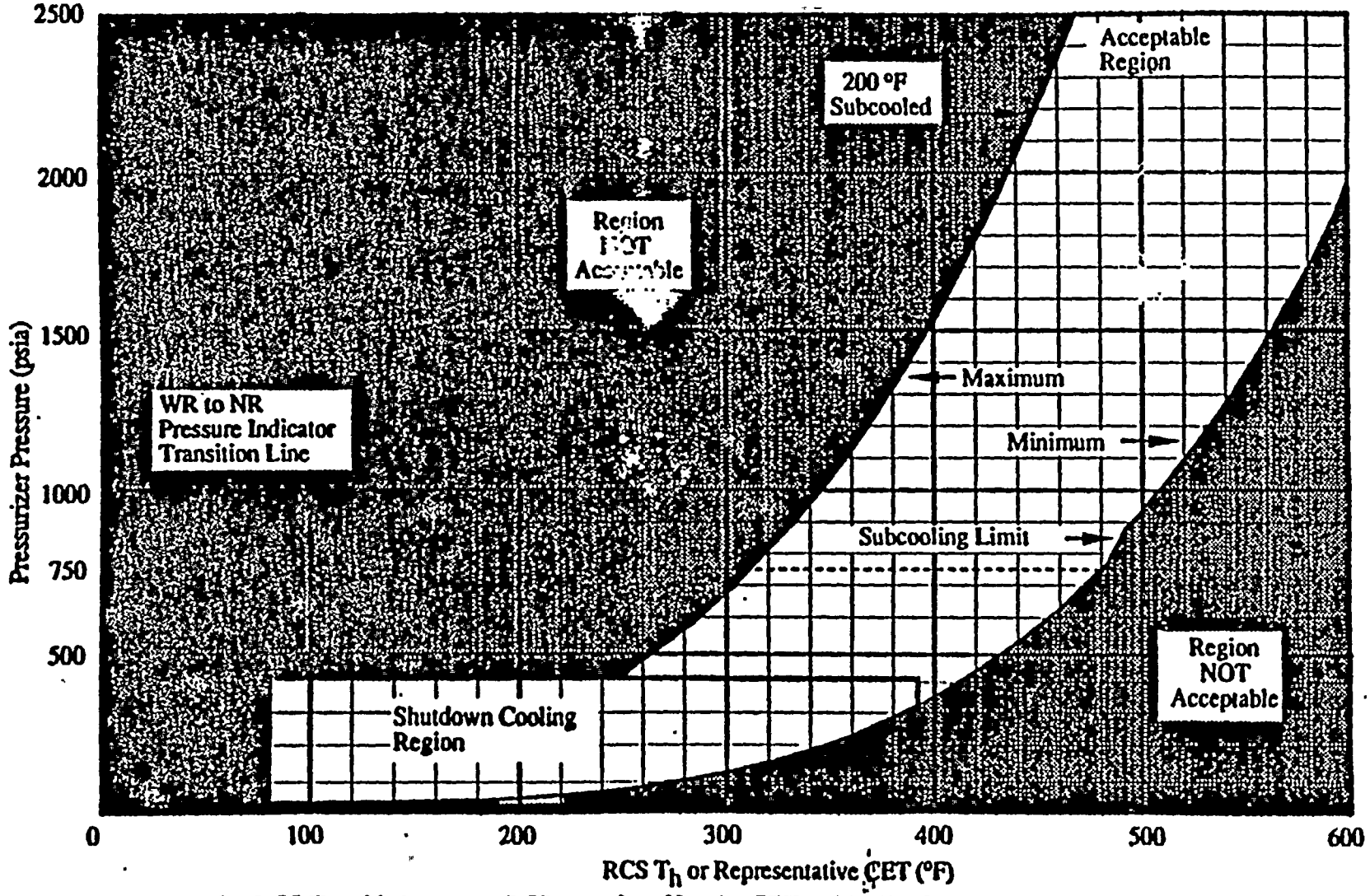
d. Downcomer Isolation

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

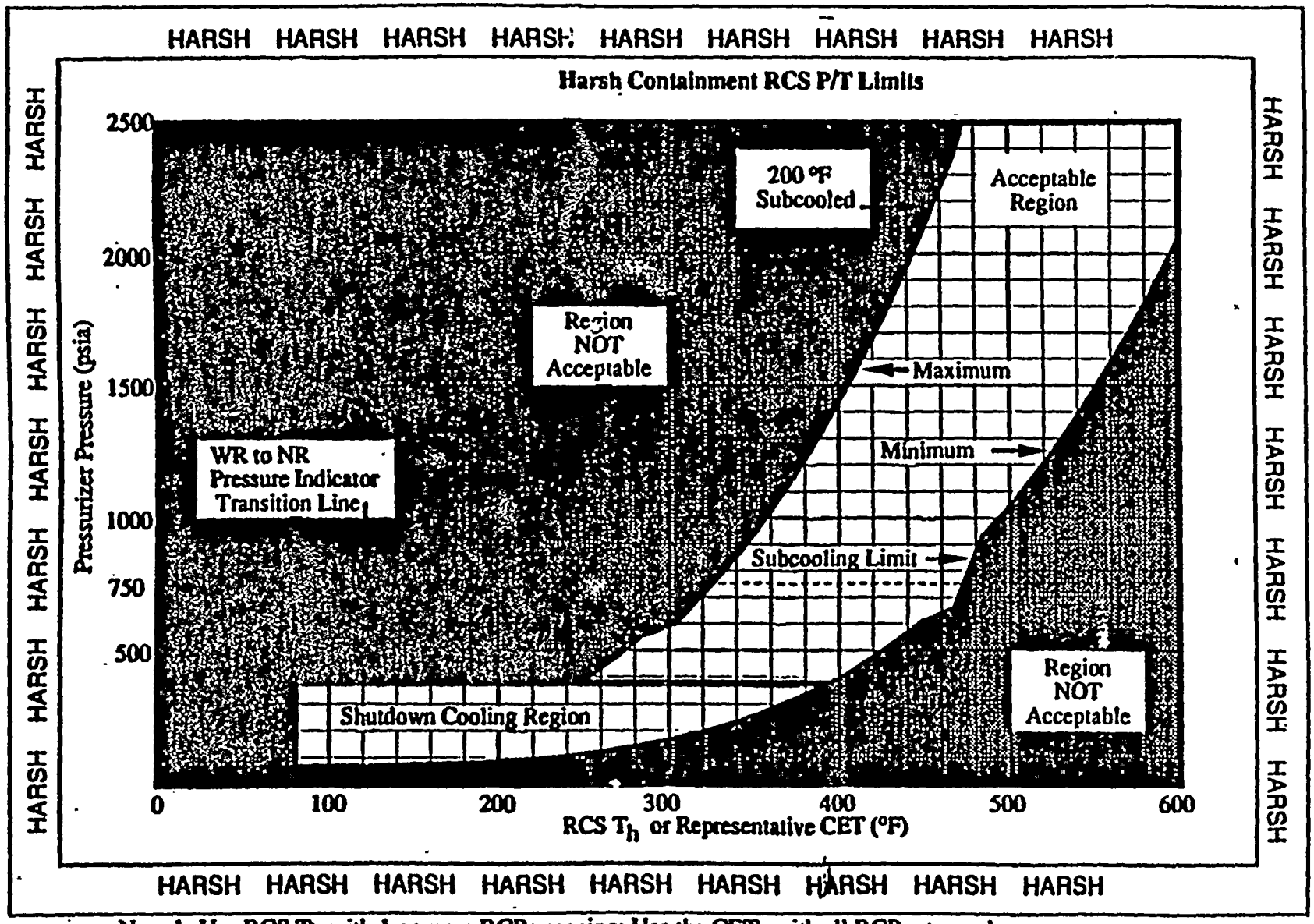
RCP NPSH Curve



NORMAL CONTAINMENT RCS P/T LIMITS



Note 1- Use RCS T_h with 1 or more RCPs running; Use the CETs with all RCPs stopped.
Note 2- Use this curve with containment temperature below 150°F.



Note 1- Use RCS T_h with 1 or more RCPs running; Use the CETs with all RCPs stopped.
Note 2- Use this curve with containment temperature above 150°F.

MASTER

SRO

KEY

A N S W E R K E Y

MULTIPLE CHOICE

001	b	023	a
002	d	024	d
003	b	025	c
004	d	026	a
005	c	027	c
006	d	028	a
007	d	029	c
008	c	030	b
009	b	031	a
010	d	032	c
011	a	033	b
012	b	034	b
013	d	035	b
014	c	036	b
015	a	037	c
016	b	038	b
017	b	039	d
018	c	040	b
019	d	041	a
020	b	042	b
021	d	043	c
022	d	044	d
		045	a

A N S W E R K E Y

046	b	069	c
047	c	070	b
048	d	071	a
049	c	072	d
050	c	073	a
051	c	074	c
052	a	075	d
053	b	076	b
054	b	077	d
055	b	078	c
056	d	079	a
057	c	080	b
058	a	081	d
059	d	082	b
060	d	083	c
061	b	084	b
062	b	085	d
063	d	086	c
064	a	087	c
065	a	088	c
066	d	089	a
067	c	090	d
068	b	091	b

A N S W E R K E Y

D92 c

D93 a

D94 c

D95 b

D96 d

D97 c

D98 d

D99 d

100 d

[illegible] 8/30/92 [illegible]

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

ANSWER: 001 (1.00)

b.

REFERENCE:

14AC-OFP02, "Emergency Notification and Response", section 3.1.1 page 5.

[3.5/4.2]

194001K116 ..(KA's)

ANSWER: 002 (1.00)

d.

REFERENCE:

40AC-90P02 "Conduct of Operations", section 2.5 page 9.

[3.5/4.2]

194001K116 ..(KA's)

ANSWER: 003 (1.00)

b.

REFERENCE:

20AC-OSK04, "Protected Area Personnel Access Control" page 9.

[3.1/3.4]

194001K105 ..(KA's)

ANSWER: 004 (1.00)

d.

REFERENCE:

40AC-9OP02 "Conduct of Shift Operations" page 6

[2.5/3.4]

194001A103 ..(KA's)

ANSWER: 005 (1.00)

c.

REFERENCE:

40AC-9OP02 "Conduct of Shift Operations" page 13.

[3.1/4.1]

194001A112 ..(KA's)

ANSWER: 006 (1.00)

d.

REFERENCE:

40AC-9OP02, "Conduct of Shift Operations" page 20, 10 CFR 50.54 (x)&(y).

[4.1/3.9].

194001A102 ..(KA's)

ANSWER: 007 (1.00)

d.

REFERENCE:

40AC-9OP02, Conduct of Shift Operations, page 37.

[3.1/3.4]

194001K105 ..(KA's)

ANSWER: 008 (1.00)

c.

REFERENCE:

40AC-90P15 Station Tagging and Clearance page 27.

[3.7/4.1]

194001K102 ..(KA's)

ANSWER: 009 (1.00)

b.

REFERENCE:

40AC-90P15, Station Tagging and Clearance page 48.

[3.7/4.1]

194001K102 ..(KA's)

ANSWER: 010 (1.00)

d.

REFERENCE:

4D AC-90-P15, "Station Tagging and Clearance" page 51.

[3.7/4.1]

194001K102 ..(KA's)

ANSWER: 011 (1.00)

a.

REFERENCE:

40AC-90P16, "Shift Turnover" page 14.

[2.8/4.1]

194001A111 ..(KA's)

ANSWER: 012 (1.00)

b.

REFERENCE:

EPIP-18 , "Emergency Exposure Guidelines" page 11. 10 CFR 20

[2.8/3.4]

194001K103 ..(KA's)

ANSWER: 013 (1.00)

d.

REFERENCE:

40DP-OOP02, "Relay Resetting" page 9.

[3.6/3.7]

194001K107 ..(KA's)

ANSWER: 014 (1.00)

c.

REFERENCE:

40DP-90P05, "Control Room Data Sheet Instructions" pages 5&6.

[3.4/3.4]

194001A106 ..(KA's)

ANSWER: 015 (1.00)

a.

REFERENCE:

74AC-9CY04, "Systems Chemistry Specifications" page 6.

[2.5/2.9]

194001A114 ..(KA's)

ANSWER: 016 (1.00)

b.

REFERENCE:

75AC-9RP01, "Radiation Exposure and Access Control" pages 24&25.

[2.8/3.4]

194001K103 ..(KA's)

ANSWER: 017 (1.00)

b.

REFERENCE:

75AC-9RP01, "Radiation Exposure and Access Control" page 14.

[3.3/3.5]

194001K104 ..(KA's)

ANSWER: 018 (1.00)

c.

REFERENCE:

41AO-1ZZ11 "Dropped or Slipped CEA" pages 4 and 6. CEA No. 5 T-MOD 191-SF-006 rod bottom light disabled. NKL21 E013.

[3.8/3.8]

001000K402 ..(KA's)

ANSWER: 019 (1.00)

d.

REFERENCE:

41OP-1CH01, "CVCS Normal Operations" NKL-32 E032

[3.3/3.5]

004000K202 ..(KA's)

ANSWER: 020 (1.00)

b.

REFERENCE:

41EP-1RO06 "Loss of Offsite Power" page 5, PO actions. NKL-09 E032 page 94.

[4.1/4.3]

004010A205 ..(KA's)

ANSWER: 021 (1.00)

d.

REFERENCE:

41OP-1SA01, "BOP ESFAS Modules Operation" page 23. System Description ESFAS, page SA-20.

[3.8/3.8]

013000G007 ..(KA's)

ANSWER: 022 (1.00)

d.

REFERENCE:

SD ESFA, page SA-16. 40EP-9R002 "Loss of Coolant Accident" page 8.

[4.5/4.7]

013000A403 ..(KA's)

ANSWER: 023 (1.00)

a.

REFERENCE:

41AO-1ZZ16, "Loss of Non-Class 120 VAC 1E Instrument Power" pages 26 and 38. NKL32 page 16 E002

[3.3/3.7]

015000K201 ..(KA's)

ANSWER: 024 (1.00)

d.

REFERENCE:

NKL32 pages 38&39 E007

[4.1/4.2]

015000K101 ..(KA's)

ANSWER: 025 (1.00)

c.

REFERENCE:

41OP-1SH01 "QSPDS User's Guide" page 6. NKL36 page 22 EO03.

[3.8/4.1]

017020A402 ..(KA's)

ANSWER: 026 (1.00)

a.

REFERENCE:

NKL36 page 9. Tech. Spec. Table 3.3.10

[2.8/3.4]

017000G005 ..(KA's)

ANSWER: 027 (1.00)

c.

REFERENCE:

41AO-1ZZ28, "Inadvertent SIAS and/or CIAS" page 20. NKL-29 page 22.

[4.1/4.3]

022000A301 ..(KA's)

ANSWER: 028 (1.00)

a.

REFERENCE:

Core Data Book Unit 1 Cycle 3 page 173

[3.0/3.3]

059000A207 ..(KA's)

ANSWER: 029 (1.00)

c.

REFERENCE:

NKL04 page 27. E002 41OP-1FT01 and 02 "Feedwater Pump Turbine "A" and "B" pages 7 and 6b.

[3.1/3.4]

059000A205 ..(KA's)

ANSWER: 030 (1.00)

b.

REFERENCE:

Technical Specification Bases 3.7.1.3, page B 3/4 7-2 NKL08 page 15 E001

[2.7/3.8]

061000G006 ..(KA's)

ANSWER: 031 (1.00)

a.

REFERENCE:

41EP-1E001, "Emergency Operations" Appendix O page 2.

[3.9/4.2]

061000A101 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

41 EP-1R005 "Loss of all Feedwater" Appendix B page 18.

[3.6/3.9]

061000K501 ..(KA's)

ANSWER: 033 (1.00)

b.

REFERENCE:

Technical Specification 3.4.5.1

[2.9/3.5]

072000G011 ..(KA's)

ANSWER: 034 (1.00)

b.

REFERENCE:

41AO-1ZZ42 "Loss of Heating Ventilation and Air Conditioning (HVAC)" page 11.a.

[3.2/3.4]

022000G010 ..(KA's)

ANSWER: 035 (1.00)

b.

REFERENCE:

41OP-1RD01 "Radwaste Drains" page 6.

[3.8/3.7]

D68000A404 ..(KA's)

ANSWER: 036 (1.00)

b.

REFERENCE:

41EP-1E001, "Emergency Operations" Appendix B, page 2
NOTE 1 USE RCS Th WITH RCPs RUNNING THEREFORE 652-609 = 43 degrees F.

[3.9/4.1]

D02000A104 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

41OP-1ZZ01, "Cold Shutdown to Hot Standby Mode 5 to Mode 3", Appendix A page 1.

[3.2/3.4]

D02000K105 ..(KA's)

ANSWER: 038 (1.00)

b.

REFERENCE:

41OP-1RC01, "Reactor Coolant Pump Operation" page 7. NKL01 page 74 E008

[4.0/4.2]

D02000K511 ..(KA's)

ANSWER: 039 (1.00)

d.

REFERENCE:

41OP-1ZZ10 "Hot Standby to Cold Shutdown Mode 3 to Mode 5" page 30.

[3.0/3.5]

D02000K612 ..(KA's)

ANSWER: 040 (1.00)

b.

REFERENCE:

41EP-1E001, "Emergency Operations" Appendix C pages 2 and 3. NKL12 page 19.

[3.6/3.9]

D06000K603 ..(KA's)

ANSWER: 041 (1.00)

a.

REFERENCE:

System Description Reactor Coolant System, page RC-10. NKL10 page 16

[3.2/3.6]

D10000K603 ..(KA's)

ANSWER: 042 (1.00)

b.

REFERENCE:

41AL-RK4A, PZR PRESS HI-LO alarm, window 4A01B page 100.

[3.5/3.5]

010000G001 ..(KA's)

ANSWER: 043 (1.00)

c.

REFERENCE:

41AL-RK4A, PZR PRESS HI-LO alarm, window 4A01B page 100. NKL 10 page 43 E016

[4.0/4.1]

010000K302 ..(KA's)

ANSWER: 044 (1.00)

d.

REFERENCE:

Tech Spec Bases page B2-5

[3.1/3.3]

012000K502 ..(KA's)

ANSWER: 045 (1.00)

a.

REFERENCE:

NKL39 page 60 E003 NO CONTROLLED REFERENCE AVAILABLE

[3.1/3.3]

012000K404 ..(KA's)

ANSWER: 046 (1.00)

b.

REFERENCE:

NKL page 9 E001

[2.8/2.9]

D16000K403 ..(KA's)

ANSWER: 047 (1.00)

c.

REFERENCE:

Tech. Spec. 4.5.2.d. 3NKL12 page 35 E003

[3.1/3.6]

D26000K402 ..(KA's)

ANSWER: 048 (1.00)

d.

REFERENCE:

System Description Fuel Pool Cooling and Cleanup System page pc-6

[2,9/3.2]

D33000K401 ..(KA's)

ANSWER: 049 (1.00)

c.

REFERENCE:

410P-1DG01 "Emergency Diesel Generator A" page 5 NKL 52 page 27 E005

[3.5/4.0]

064000K411 ..(KA's)

ANSWER: 050 (1.00)

c.

REFERENCE:

41AL-1DGB1 page 11, Window DGB 07A "OVERSPEED ENGINE"

[3.9/4.2]

064000K402 ..(KA's)

ANSWER: 051 (1.00)

c.

REFERENCE:

EPIP-02 Emergency Classification page 9

[3.3/3.6]

086000G015 ..(KA's)

ANSWER: 052 (1.00)

a.

REFERENCE:

Technical Specification Bases 3/4.4.5.2

[2.7/3.7]

035000G006 ..(KA's)

ANSWER: 053 (1.00)

b.

REFERENCE:

41OP-1PB01, "4.16KV Class IE Power (PB)" appendix C page 31.

[3.5/3.6]

062000A305 ..(KA's)

ANSWER: 054 (1.00)

b.

REFERENCE:

41-OP1SI01, "Shutdown Cooling Initiation" page 8.

[2.7/2.9]

005000A201 ..(KA's)

ANSWER: 055 (1.00)

b.

REFERENCE:

41OP-1ZZ01 "Cold Shutdown to Hot Standby Mode 5 to Mode 3" Appendix A page 1.

[2.7/2.9]

007000A102 ..(KA's)

ANSWER: 056 (1.00)

d.

REFERENCE:

41OP-1SF05 "Operation of the Steam Bypass Control System" page 12

[2.9/3.1]

041020A406 ..(KA's)

ANSWER: 057 (1.00)

c.

REFERENCE:

41AO-1ZZ02 "Load Rejection" Appendix L page 1

[2.7/2.9]

041020K603 ..(KA's)

ANSWER: 058 (1.00)

a.

REFERENCE:

AOP 41AO-1ZZ35, "Continuous CEA Withdrawal" page 4. NKL21 EO26.

[3.9/4.0]

000001G010 ..(KA's)

ANSWER: 059 (1.00)

d.

REFERENCE:

41AO-1ZZ29, "Reactor Coolant Pump And Motor Emergency" page 16.
NKL01 EO 15.

[4.0/4.2]

000015A122 ..(KA's)

ANSWER: 060 (1.00)

d.

REFERENCE:

41AO-1ZZ29 "Reactor Coolant Pump and Motor Emergency" page 16.
NKL01 EO 14.

[3.4/3.4]

D00015G010 ..(KA's)

ANSWER: 061 (1.00)

b.

REFERENCE:

NLK09 Chemical and Volume Control System, pages 190 and 191. EO 78

[4.1/4.4]

D00024K301 ..(KA's)

ANSWER: 062 (1.00)

b.

REFERENCE:

41AO-1ZZ01 "Emergency Boration" page 9 and appendix B. NLK09 EO 63d.

[3.4/4.2]

D00024A204 ..(KA's)

ANSWER: 063 (1.00)

d.

REFERENCE:

41AO-1ZZ11, " Dropped or Slipped CEA" page 6. NKL21 pages 98 and 120 . EO 19

[3.7/3.9]

000003A201 ..(KA's)

ANSWER: 064 (1.00)

a.

REFERENCE:

NKL21 page 35 EO 004.

[3.6/3.4]

000005A101 ..(KA's)

ANSWER: 065 (1.00)

a.

REFERENCE:

41EP-1E001, "Emergency Operations" page 10 of 23.

[3.7/3.7]

000011A213 ..(KA's)

ANSWER: 066 (1.00)

d.

REFERENCE:

41A001ZZ04, "Loss OF Plant Cooling Water" page 13. NKL31 E004.

[3.6/3.6]

000026A103 ..(KA's)

ANSWER: 067 (1.00)

c.

REFERENCE:

NKL31, Essential Cooling Water LP, E005.

[3.2/3.5]

000026K301 ..(KA's)

ANSWER: 068 (1.00)

b.

REFERENCE:

41EP-1E001, Emergency Operations page 4.

[4.1/4.2]

000029G012 ..(KA's)

ANSWER: 069 (1.00)

c.

REFERENCE:

41EP-1E001, Emergency Operations page 4.

[4.5/4.5]

000029G010 ..(KA's)

ANSWER: 070 (1.00)

b.

REFERENCE:

41EP-1E001, Emergency Operations page 13.

[4.0/4.6]

000074A206 ..(KA's)

ANSWER: 071 (1.00)

a.

REFERENCE:

41EP-1E001, "Emergency Operations" page 16.

[3.7/4.2]

000074A204 ..(KA's)

ANSWER: 072 (1.00)

d.

REFERENCE:

Tech Spec Bases 3/4.6.2.2. NKL E003.a.

[2.5/3.8]

000069G004 ..(KA's)

ANSWER: 073 (1.00)

a.

REFERENCE:

40EP-9R004, "Excess Steam Demand" page 20, 40DP-9AP06, "Emergency Operations" page 34.

[4.5/4.7]

000040K304 ..(KA's)

ANSWER: 074 (1.00)

c.

REFERENCE:

Tech Spec Bases 3.4.1.1

[4.1/4.4]

D00040K105 ..(KA's)

ANSWER: 075 (1.00)

d.

REFERENCE:

41AO-1ZZ07, "Loss Of Condenser Vacuum" page 5. NKL07 E001-5

[2.8/3.1]

D00051K301 ..(KA's)

ANSWER: 076 (1.00)

b.

REFERENCE:

41EP-1RO07 Blackout page 7.

[4.3/4.6]

D00055K302 ..(KA's)

ANSWER: 077 (1.00)

d.

REFERENCE:

41EP-1RO07, "Blackout" page 8.

[3.6/3.7]

000055G007 ..(KA's)

ANSWER: 078 (1.00)

c.

REFERENCE:

40DP-9AP13, "Blackout Technical Guideline" page 19. NKL E020

[3.8/4.1]

000055G006 ..(KA's)

ANSWER: 079 (1.00)

a.

REFERENCE:

41AO-1ZZ44, "Shutdown Outside the Control Room Due To Fire And/Or Smoke" page 6.

[4.2/4.5]

000068K318 ..(KA's)

ANSWER: 080 (1.00)

b.

REFERENCE:

41AO-1ZZ27, "Shutdown Outside Control Room" page 18

[3.9/4.0]

000068K201 ..(KA's)

ANSWER: 081 (1.00)

d.

REFERENCE:

41EP-1ZZ01, "Emergency Operations" page 5 and 40DP-9AP08. "Loss of Coolant Guidelines" page 23. 41EP-1E001, "Emergency Operations" Appendix D, RCP NPSH Curves

[4.1/4.2]

000011K314 ..(KA's)

ANSWER: 082 (1.00)

b.

REFERENCE:

40DP-9APO-6, "Emergency Operation Guideline" page 41.

[4.0/4.6]

000007K301 ..(KA's)

ANSWER: 083 (1.00)

c.

REFERENCE:

41EP-1E001, "Emergency Operations" Appendix B. 409R001, "Reactor Trip" page 4. NKS32 E006.

[3.6/3.6]

000007G007 ..(KA's)

ANSWER: 084 (1.00)

b.

REFERENCE:

NKS32-01 page 12 E001

[3.4/3.7]

000008A212 ..(KA's)

ANSWER: 085 (1.00)

d.

REFERENCE:

40DP-9AP08, "Loss of Coolant Technical Guideline" page 31. NKS32 EO 20

[3.7/3.9]

000009G007 ..(KA's)

ANSWER: 086 (1.00)

c.

REFERENCE:

40EP-9RO02, "Loss of Coolant Accident" page 12 step 3.17.

[3.5/3.8]

000022K302 000009A107 ..(KA's)

ANSWER: 087 (1.00)

c.

REFERENCE:

41AO-1ZZ22, "Loss of Shutdown Cooling" page 10.

[3.4/3.7]

000025A207 ..(KA's)

ANSWER: 088 (1.00)

c.

REFERENCE:

Core Data Book, Unit 1 cycle 3 page 105.
MODE 4 = 210 degrees F. 210 - 124 = 86 degrees F. delta T.
6.5 degrees F. per hour heatup rate 86/6.5 = 13.23 minutes.

[3.4/3.6]

000025G007 ..(KA's)

ANSWER: 089 (1.00)

a.

REFERENCE:

41AO-1ZZ24, "NI System Malfunction" page 5 NKL32 E013

[2.8/3.3]

000032G008 ..(KA's)

ANSWER: 090 (1.00)

d.

REFERENCE:

41AO-1ZZ24 "NI System Malfunction" page 7 Tech. Spec Table 3.3.1 action 2. NKL32 E013

[2.7/3.3]

000033G003 ..(KA's)

ANSWER: 091 (1.00)

b.

REFERENCE:

40EP-9AP09, Steam Generator Tube Rupture Technical Guidelines pages 72&74.

[3.7/4.0]

000037K305 ..(KA's)

ANSWER: 092 (1.00)

c.

REFERENCE:

40EP-9RO03, "Steam Generator Tube Rupture" page 23. NKS32 EO10

[4.4/4.6]

000038A203 ..(KA's)

ANSWER: 093 (1.00)

a.

REFERENCE:

40DP-9AP09 "Steam Generator Tube Rupture Technical Guideline" page 63. NKS32 EO05

[4.2/4.5]

000038K306 ..(KA's)

ANSWER: 094 (1.00)

c.

REFERENCE:

41AO-1ZZ06, "Loss of Instrument Air" page 14.

[3.7/3.9]

000065K308 ..(KA's)

ANSWER: 095 (1.00)

b.

REFERENCE:

41AO-1ZZ17, "Loss of 125 VDC Class 1E Electrical Power" page 5.

[4.0/4.2]

D00058K302 ..(KA's)

ANSWER: 096 (1.00)

d.

REFERENCE:

NKL11 Pressurizer level control system, page 35 E011.

[2.8/2.9]

D00028A105 ..(KA's)

ANSWER: 097 (1.00)

c.

REFERENCE:

41AO-1ZZ53, "Loss of Refueling Pool and/or Spent Fuel Pool Level" page 14 step 2.8.1.

[3.1/3.7]

D00036A104 ..(KA's)

ANSWER: 098 (1.00)

d.

REFERENCE:

40DP-9AP07, Reactor Trip Technical Guideline page 53. 40EP-9R001, "Reactor Trip" page 19. 40DP-9AP06, Emergency Operations Technical Guidelines page 242 item 6, contingency actions. NKS32 Reactor Trip, page 24 E010.

[3.3/3.4]

000056A251 ..(KA's)

ANSWER: 099 (1.00)

d.

REFERENCE:

41AO-1ZZ06, "Loss of Instrument Air" page 12

[2.9/3.3]

000065A208 ..(KA's)

********** END OF EXAMINATION **********

[SRO]
[KEY]

QUESTION: 100 (1.00)

Following a LOCA, which of the following conditions would ENSURE that the Containment Atmospheric Control Safety Function Criteria is met?

a. One spray header is functioning and delivering 3200 gpm.

b. Containment pressure is 9.5 psig.

c. One spray header is functioning and delivering 3200 gpm, and CIAS has actuated.

d. Containment pressure increased and then stabilized at 8.0 psig, and CIAS has actuated.

*ANSWER
d.
*REFERENCE
41EP-1RD02, "Loss of Coolant Accident" Appendix C, Shift Supervisor Overview page 11.

[3.7/3.9]

MASTER

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U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
SENIOR OPERATOR LICENSE
REGION 5

CANDIDATE'S NAME: _____

FACILITY: Palo Verde 1, 2, & 3

REACTOR TYPE: PWR-CE80

DATE ADMINISTERED: 92/06/01

INSTRUCTIONS TO CANDIDATE:

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

TEST VALUE	CANDIDATE'S SCORE	%	
100.00 ~~99.00~~ 5/30/92	_____	___ %	TOTALS
	FINAL GRADE		

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

_____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

3. Restroom trips are to be limited and only one 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.

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

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

6. Mark your answers on the answer sheet provided. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

7. 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.

8. 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.

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

10. Show all calculations, methods, or assumptions used to obtain an answer to any short answer questions.

11. Partial credit may be given except on multiple choice questions. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

12. 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.

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

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

15. Ensure all information you wish to have evaluated as part of your answer is on your answer sheet. Scrap paper will be disposed of immediately following the examination.

16. To pass the examination, you must achieve a grade of 80% or greater.

17. There is a time limit of four (4) hours for completion of the examination.

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

QUESTION: 001 (1.00)

A person discovering a fire in any power block shall immediately:

a. fight the fire until help arrives.

b. notify Security at extension 4444.

c. notify the Control Room at extension X206.

d. notify the site Fire Department at extension 1612.

QUESTION: 002 (1.00)

Which ONE of the following individuals may be assigned as the Fire Team Advisor by the Shift Supervisor?

a. Qualified Radiation Protection Technician

b. Nuclear Operations Technician

c. Shift Technical Advisor (STA)

d. Reactor Operator

QUESTION: 003 (1.00)

Individuals losing their ACAD/ID badge inside the protected area shall:

a. immediately go to the Security Headquarters Building.

b. immediately notify their supervisor and the Security Shift Supervisor.

c. return to the area where they last used their ACAD/ID badge and commence a search.

d. report it when they exit the protected area at the end of the work day.

QUESTION: 004 (1.00)

Which ONE of the following is an EXCLUSIVE responsibility of the Unit 1 Shift Supervisor?

a. Control all station loading as directed by the Generation Control Center (GCC).

b. All tagging operations in the Water Reclamation Facility (WRF).

c. Periodically monitor shift turnover at the WRF.

d. Station security on backshifts, weekends and holidays.

QUESTION: 005 (1.00)

A Reactor Operator in training may manipulate controls at the remote shutdown panel which do NOT affect reactivity or power level:

a. when ordered by the Reactor Operator in the control room during power operation.

b. when performing a surveillance test procedure while in communication with the Reactor Operator.

c. only under the direct supervision of a Licensed Operator.

d. when directed by the Shift Technical Advisor during Emergency Operating Procedure (EOP) usage.

QUESTION: 006 (1.00)

During operation an emergency situation arises for which no procedure is available; and action which departs from Technical Specifications is required immediately to protect the health and safety of the public.

Which ONE of the following describes the course of action the Reactor Operator is authorized to take?

a. Immediately takes whatever action is required without further direction.

b. Notifies the Assistant Shift Supervisor of his intent and performs the required action without further direction.

c. Obtains approval from the Shift Supervisor and only the Shift Supervisor prior to taking any action.

d. Obtains approval from the Shift Supervisor or Assistant Shift Supervisor prior to taking any action.

QUESTION: 007 (1.00)

The Missile Shield Doors on the 140 ft elevation between the Control and Corridor Buildings shall both be fully shut:

a. when the main turbine is reset with a severe thunder storm watch in effect.

b. only when steam is present downstream of the MSIVs with a tornado alert in effect.

c. when the main generator is connected to the grid with a severe thunder storm watch in effect.

d. anytime a tornado alert is in effect.

QUESTION: 008 (1.00)

Which ONE of the following states the requirement for hanging a Red Danger Tag on a piece of equipment which already has a Yellow Caution Tag hanging on it?

a. The yellow caution tag must be cleared and removed before the red danger tag is installed.

b. The red danger tag is hung, the yellow caution tag is replaced with a new one having the same tag number as the red danger tag.

c. The red danger tag is hung with the yellow caution tag, provided there are no tagging conflicts.

d. The yellow caution tag must be temporarily lifted and replaced after the red danger tag is removed.

QUESTION: 009 (1.00)

A clearance is being prepared for a Unit 1 lighting circuit. The controlled copy of the drawing indicates a Field Change Request (FCR) is N/A.

This indicates to the clearance preparer that:

a. the FCR is not applicable and does not require review.

b. it is not applicable to the "13 series prints" but may have been implemented in one or two of the units.

c. the FCR has been changed to a Drawing Change Notice (DCN) and does not require review.

d. the FCR has been completed on all three units and is no longer applicable.

QUESTION: 010 (1.00)

An air operated valve which fails OPEN:

a. cannot be used as an isolation boundary under any circumstances.

b. can be used as an isolation boundary if the air supply is red tagged open.

c. can be used as an isolation boundary if a backup air supply is connected to the valve operator to keep it closed.

d. can be used as an isolation boundary only if it is manually ("jacked") closed and the jacking device becomes the tagging control point.

QUESTION: 011 (1.00)

Which ONE of the following Shift Supervisor turnover items is normally completed after turnover is complete?

a. Control Room Log reviews

b. Vital Key inventory

c. Review SESS panels status

d. Unit Log review

QUESTION: 012 (1.00)

During emergency conditions the whole body exposure limit for "corrective or protective actions" is:

a. 5 rem.

b. 25 rem.

c. 75 rem

d. 100 rem.

QUESTION: 013 (1.00)

Which ONE of the following states the requirement for resetting a 786 Lockout Relay on a 4.16KV Class 1E bus from the control room?

a. The operator may attempt one reset without obtaining any additional authorization.

b. The cause of the relay trip must be verified locally to NOT have been the result of a phase-to-phase fault prior to attempting reset, with its associated Diesel Generator out of commission.

c. The Diesel Generator must be verified to NOT be supplying the bus prior to attempting reset.

d. Reset may be attempted only if the Diesel Generator is supplying the bus.

QUESTION: 014 (1.00)

Which ONE of the following Control Room Data Sheet entries shall NOT be skipped and shall be taken within one hour of the time specified?

a. Main Generator gross megawatts

b. RCP seal pressures

c. RCP shaft vibration data

d. Safety Injection Tanks pressure and level

QUESTION: 015 (1.00)

Which ONE of the following RCS parameters Technical Specification limits is ONLY applicable when Tavg is greater than 250 degrees F?

a. Dissolved Oxygen

b. Chloride

c. Fluoride

d. Specific Activity

QUESTION: 016 (1.00)

The minimum level of management which can authorize an Auxiliary Operator to exceed the PVNGS quarterly radiation exposure administrative limit of 1.0 Rem is:

a. Shift Supervisor

b. Radiation Protection Manager

c. Respective Unit Plant Manager

d. ALARA Committee Chairman

QUESTION: 017 (1.00)

If the Unit Operations Shift Supervisor determines that plant conditions have changed such that they are affecting the working conditions established for a Radiation Exposure Permit (REP), he may:

a. revise the REP.

b. request Radiation Protection (RP) to revise the REP.

c. terminate the REP and reissue a new REP.

d. suspend the REP and utilize direct RP coverage.

QUESTION: 018 (1.00)

Unit 1 was operating at 100% power when the following indications and alarms are observed in the control room:

- reactor power decreasing
- "T AVG - T REF HI-LO" annunciator alarm lit on panel B04
- CEA number 5 lower electrical limit (LEL) light illuminated on panel B04
- CWP alarm on annunciator 4A9B
- pressurizer level and pressure decreasing

Which ONE of the following describes the cause of the above indications?

a. Continuous CEA insertion is occurring

b. A CEA has slipped and stopped at an unknown position

c. A CEA has dropped all the way into the core

d. A Group of CEAs has dropped all the way into the core

QUESTION: 019 (1.00)

Which ONE of the following is the "Normal" power supply to Charging Pump #3, CHE-P01?

a. PGA-L31

b. PGB-L32

c. PGA-L33

d. PGB-L36

QUESTION: 020 (1.00)

Following a Loss of Offsite Power (LOP) with no Safety Injection Actuation Signal (SIAS) all running Charging Pumps are "load shed".

After the diesel generator output breaker closes, all of the previously running charging pumps:

a. will restart immediately.

b. must be manually restarted.

c. will restart following a 40 second time delay .

d. will automatically restart as required by the pressurizer level control system.

QUESTION: 021 (1.00)

The BOP ESFAS, CRVIAS trip module, Train "A" is placed in "BYPASS".

Which ONE of the following describes the response if CRVIAS trip module, Train "B" is then also placed in "BYPASS"?

a. Both CRVIAS trip modules Train "A" and Train "B" will be bypassed

b. Only CRVIAS trip module Train "A" will be bypassed

c. Only CRVIAS trip module Train "B" will be bypassed

d. No CRVIAS trip modules will be bypassed

QUESTION: 022 (1.00)

Which ONE of the following is an automatic action associated with a Recirculation Actuation Signal (RAS) signal?

a. High Pressure Safety Injection pumps trip

b. Refueling Water Tank outlet valves close

c. Charging pumps trip

d. ECCS suction from Containment Sump valves open.

QUESTION: 023 (1.00)

Which ONE of the following describes the power supply to the Excore Nuclear Instrumentation Startup Channels?

a. NNN-D11 and D12

b. NNN-D15 and D16

c. PNA-D25 and D26

d. PNA-D27 and D28

QUESTION: 024 (1.00)

Which ONE of the following describes how the Excore Nuclear Instruments provide "raw neutron power" data to the Core Protection Calculators?

a. Through the linear power circuit and its summing amplifier.

b. From the control channel detectors COLSS ASI calculation amplifier output.

c. From the individual log safety channel detector power output amplifier signals.

d. Through the individual subchannel linear amplifiers which receive a signal from each safety channel detector.

QUESTION: 025 (1.00)

Which ONE of the following levels indicated by the Qualified Safety Parameter Display System (QSPDS) is invalid with reactor coolant pump(s) operating during accident conditions?

a. Steam Generator

b. Pressurizer

c. Plenum reactor vessel

d. Upper head reactor vessel

QUESTION: 026 (1.00)

Which ONE of the following inputs to the Qualified Safety Parameter Display System (QSPDS) is required as part of the Technical Specification Accident Monitoring Instrumentation?

a. Core exit thermocouples

b. Reactor power

c. ESFAS actuation status

d. Containment Hydrogen concentration

QUESTION: 027 (1.00)

During operation at 100% power, surveillance testing results in an inadvertent Safety Injection Actuation Signal (SIAS).

Which ONE of the following describes the operation of the Containment Normal Air Cooling Units, HCA-A01A-D?

a. Trip on a SIAS and cannot be restarted until the SIAS is reset

b. Remain running on a SIAS

c. Trip on a SIAS but can be overridden and restarted without resetting the SIAS

d. Trip on a SIAS and automatically restart when the SIAS is reset

QUESTION: 028 (1.00)

Unit 1 is operating at 72% power with the Reactor Power Cutback System in service when Reactor Feed Water Pump Turbine "B" trips.

Which ONE of the following describes the CEA Subgroups which will drop into the Core?

a. None

b. Subgroup 4

c. Subgroup 4 and 5

d. Subgroup 4, 5, and 22

QUESTION: 029 (1.00)

While operating at 100% power Feed Water Pump suction pressure drops to less than 255 psig.

Which ONE of the following describes the automatic response of the feedwater system?

a. Both pumps trip immediately

b. Both pumps trip after 10 second time delay

c. One pump trips after 10 second time delay, one pump trips after 15 second time delay

d. Both pumps trip after 15 second time delay

QUESTION: 030 (1.00)

Technical Specification Bases requires sufficient water volume available in the Unit 1 Condensate Storage Tank (CST) to maintain the Unit 1 RCS at HOT STANDBY conditions for:

a. 8 hours followed by an orderly cooldown to 325 degrees F.

b. 8 hours followed by an orderly cooldown to 350 degrees F.

c. 4 hours followed by an orderly cooldown to 325 degrees F.

d. 4 hours followed by an orderly cooldown to 350 degrees F.

QUESTION: 031 (1.00)

After an automatic Auxiliary Feedwater Actuation Signal (AFAS) the following plant conditions exist:

- #1 steam generator (SG) level 36% WR and increasing
- #2 steam generator level 41% WR and steady
- auxiliary feedwater (AFW) regulating valves are overridden and throttled to maintain level
- #1 SG auxiliary feedwater isolation valve is NOT overridden and is open
  #2 SG auxiliary feedwater isolation valve is NOT overridden and is closed

The AFAS signal may:

a. not be reset because the SG levels have not been above 40% WR level.

b. be reset since the AFW regulating valves are in override.

c. not be reset since the AFW isolation valves are not overridden.

d. be reset because the SG levels are being maintained within the required band.

QUESTION: 032 (1.00)

Following an automatic initiation of Auxiliary Feedwater both Steam Generators (SG) levels have increased to 4% WR and the Secondary Operator has overridden and throttled the AFW valves.

Which ONE of the following describes the minimum required feedflow the Secondary Operator must establish to at least one SG?

a. Enough flow to maintain SG level

b. Enough flow to ensure RCS Tc is stable

c. At least 250 gpm (0.14 mlbm/hr)

d. Greater than 500 gpm (0.28 mlbm/hr) but less than 900 gpm (0.5 mlbm/hr)

QUESTION: 033 (1.00)

In addition to the Containment Sump level and flow monitoring system, which of the following is required for the Reactor Coolant leakage detection system to be operable?

a. Containment Power Access Purge Exhaust RU-37 and RU-38

b. Containment Building Atmosphere Particulate monitor RU-1

c. Containment High Radiation Monitor RU-148 or RU-149

d. Primary Coolant Activity Monitor RU-150 or RU-151

QUESTION: 034 (1.00)

During operation at 100% power, Control Element Drive Mechanism (CEDM) cooling is lost and cannot be restored.

Which ONE of the following is required?

a. Open the reactor trip breakers and achieve an RCS temperature of less than 500 degrees F. within 4 hours.

b. Have the CEA's deenergized within 40 minutes.

c. Reduce reactor power as rapidly as possible using only boration, then open the reactor trip breakers.

d. Immediately open the reactor trip breakers

QUESTION: 035 (1.00)

Which ONE of the following describes the response of the Containment Radwaste Sump System to an automatic Engineered Safeguards Features Actuation Signal (ESFAS)?

a. Inboard and outboard containment isolation valves, RD-UV-23 and RD-UV-24 close on a CIAS and cannot be overridden open

b. Inboard and outboard containment isolation valves, RD-UV-23 and RD-UV-24 close on a CIAS and can be overridden open

c. Inboard containment isolation valve, RD-UV-23 closes on a CIAS, outboard valve RD-UV-24 closes on a SIAS, each valve can be overridden open

d. Inboard and outboard containment isolation valves, RD-UV-23 and RD-UV-24 close on a MSIS and cannot be overridden open

QUESTION: 036 (1.00)

Given the following data:

- The plant is at full power	100%
- T-hot	609 degrees F
- T-cold	554 degrees F
- Highest CET	613 degrees F
- Pressurizer temperature	652 degrees F
- Pressurizer pressure	2250 psia

Determine the most accurate value for bulk RCS reactor subcooling.

a. 39.0 degrees F

b. 43.0 degrees F

c. 70.5 degrees F

d. 98.0 degrees F

QUESTION: 037 (1.00)

During a Reactor Coolant System (RCS) heatup a pressurizer steam bubble is being formed. The nitrogen (N2) overpressure in the pressurizer is being reduced by venting the pressurizer to the Reactor Drain Tank (RDT).

Which ONE of the following describes the expected response of the RDT to the final phase of venting (most N2 is purged)?

a. Level will increase with no change in pressure

b. Level will not change but pressure will increase

c. Level and pressure will increase

d. Level will decrease and pressure will remain the same

QUESTION: 038 (1.00)

When starting the first reactor coolant pump with pressurizer level less than 33%, Steam Generator secondary side temperature may not exceed any RCS cold leg temperature by 20 degrees F.

Which ONE of the following is the basis for that precaution?

a. Prevents excessive heat transfer from the RCS to the S/G

b. Minimize potential for lifting Low Temperature Over Pressure (LTOP) relief valves

c. Minimize thermal shock to the S/G tubes which could result in possible S/G Tube Rupture

d. Prevents thermal shock to the RCP which could result in possible RCP Seal Failure

QUESTION: 039 (1.00)

During Cooldown, Technical Specifications require the LTOP's to be in service when the lowest indicated RCS Tc decreases to less than:

a. 227 degrees F.

b. 224 degrees F

c. 220 degrees F

d. 214 degrees F

QUESTION: 040 (1.00)

Following a Loss Of Coolant Accident (LOCA), RCS pressure is at 250 psia and dropping.

Which ONE of the following statements describes the "NORMAL" status of Emergency Core Cooling INJECTION flow?

a. Constant HPSI flow exists with no LPSI flow.

b. Increasing HPSI flow exists with no LPSI flow.

c. Constant HPSI and LPSI flow exists.

d. Increasing HPSI and LPSI flow exists.

QUESTION: 041 (1.00)

With the Pressurizer Spray Valves fully closed, there is still approximately 6-10 gpm continuous flow through the spray bypass valves.

Which ONE of the following is the PRIMARY purpose of this flow?

a. Minimize the thermal shock to the spray nozzle.

b. Ensure pressurizer heaters remain energized.

c. Maintain spray valve inlet and outlet piping isothermal.

d. Maintain Pressurizer and RCS loop boron concentrations equalized.

QUESTION: 042 (1.00)

During operation at 100% power the following indications are noted:

- PZR PRESS HI-LO alarm, window 4A01B lit
- actual pressurizer pressure is 2265 psia and increasing
- all pressurizer heaters indicate on

Which ONE of the following is the first priority operator action required?

a. Initiate auxiliary spray

b. Switch pressurizer pressure control to the unaffected channel with handswitch RCN-HS-100

c. Manually initiate normal pressurizer spray flow

d. Deenergize all pressurizer heaters

QUESTION: 043 (1.00)

During operation at 100% power the Pressurizer pressure transmitter for the controlling channel fails high.

Which ONE of the following is the first AUTOMATIC plant response which occurs to mitigate the transient?

a. High pressure reactor trip

b. Spray valves close

c. Low Pressure reactor trip and SIAS

d. Backup heaters energize

QUESTION: 044 (1.00)

The Local Power Density (LPD) - High trip is calculated in the Reactor Protection System.

Which ONE of the following information inputs is NOT used "directly" in the LPD calculation?

a. Axial power distribution

b. CEA positions

c. Radial peaking factors

d. RCS pressure

QUESTION: 045 (1.00)

Which ONE of the following pressures is used as the input to the Supplementary Protection System (SPS)?

a. Pressurizer

b. Containment

c. RCS Hot Leg

d. Steam Generator

QUESTION: 046 (1.00)

Which ONE of the following Main Feed Pump parameters is used for initiation of a Reactor Power Cutback on a Loss of Main Feed Pump event?

a. Pump discharge pressure

b. Turbine control oil pressure

c. Pump discharge valve position

d. Turbine steam stop valve position

QUESTION: 047 (1.00)

Which ONE of the following describes the purpose of the Trisodium Phosphate (TSP) located in containment storage baskets?

a. To reduce the final pH of the water following a LOCA.

b. To scavenge Iodine following a LOCA.

c. To minimize the corrosion cracking of certain metal components.

d. To minimize boron precipitation following a LOCA.

QUESTION: 048 (1.00)

Which ONE of the following describes the design feature(s) that prevents a pipe break outside of the Spent Fuel Pool (SFP) from draining the pool

a. All SFP Cooling system supply and return piping is provided with siphon breaker holes.

b. The SFP primary makeup valve automatically opens on low level and can makeup inventory at a rate equal to the largest calculated pipe break.

c. The SFP Cooling system valves, piping and pumps are located above the normal operating level.

d. Piping either penetrates the SFP wall above the water level or has siphon breaker holes for pipes which run below the water level.

QUESTION: 049 (1.00)

Diesel Generator "A" is operating in parallel with offsite power in the "test" mode when a Auxiliary Feedwater Actuation Signal (AFAS) is received?

Which ONE of the following describes the automatic response of the Diesel Generator (DG)?

a. DG continues running in "test" mode output breaker trips open

b. DG continues running in "test" mode in parallel with offsite power

c. Output breaker trips open, DG transfers to "emergency" mode and output breaker remains open

d. Output breaker trips open, DG transfers to "emergency" mode and output breaker closes

QUESTION: 050 (1.00)

Which ONE of the following Diesel Generator (DG) Trips will stop the DG in both the "emergency" and "test" modes?

a. Crankcase pressure high

b. Low turbocharger lube oil pressure

c. Engine overspeed

d. Excessive engine vibration

QUESTION: 051 (1.00)

Which ONE of the following fire events would require an Emergency Plan Classification in accordance with Tab 4, EPIP-03, "Fire and/or Security Compromise"?

a. Bomb threat received by telephone in the control room

b. Fire in a materials staging area which is extinguished in less than 10 minutes by the site fire team

c. Control room evacuated due to smoke and control established at the remote shutdown panel

d. Site fire team less than the minimum required composition for greater than 2 hours

QUESTION: 052 (1.00)

Which ONE of the following is the Bases of the Technical Specification limit for leakage through any one Steam Generator (SG) of 720 gallons per day?

a. Ensures that SG tube integrity is maintained in the event of a LOCA

b. Ensures that the radiation release from the condenser off gas remains below the alarm level

c. This is the minimum leakage detectable with the normal sample requirements

d. At leakage rates greater than this the RCS boron will adversely affect SG chemistry parameters

QUESTION: 053 (1.00)

The Green indicating light intensity on a 4.16KV Class 1E breaker cabinet indicates brighter than the light on an adjacent breaker.

This indicates the breaker:

a. is open with the closing springs not charged.

b. has opened because of a 786 protective relay actuation.

c. has a blown fuse in the control power trip coil circuit.

d. has a blown fuse in the control power closing circuit.

QUESTION: 054 (1.00)

While operating in Shutdown Cooling the Operator is cautioned not to operate the Low Pressure Safety Injection (LPSI) pumps in the "rumble range".

This is to prevent damage caused by:

a. exceeding the maximum design flow rate.

b. excessive pump vibration.

c. operation below the minimum required flow rate.

d. cavitation caused by high suction temperature.

QUESTION: 055 (1.00)

When venting the Pressurizer to the Reactor Drain Tank (RDT) to establish a steam bubble, the RDT pressure is maintained less than 100 psia by venting:

a. directly to the containment atmosphere.

b. via a hose to the refueling purge system.

c. via a hose to the power access purge system .

d. directly to the containment hydrogen control system.

QUESTION: 056 (1.00)

Which ONE of the following describes the Unit 3 Steam Bypass Control System (SBCS) valve alignment when operating at 90% power?

a. All SBCS valves in "Auto with an automatic permissive"

b. All SBCS valves in "Auto with a manual permissive"

c. 7 SBCS valves in "Auto with an automatic permissive" and 1 in "Manual with a manual permissive"

d. 7 SBCS valves in "Auto with an automatic permissive" and 1 in Automatic with Mode Select Switch OFF

QUESTION: 057 (1.00)

When operating the Atmospheric Dump Valves, the minimum required controller demand signal to initiate valve movement is:

a. 10%.

b. 20%.

c. 30%.

d. 50%.

QUESTION: 058 (1.00)

Verification checks provide positive indication of a Continuous CEA withdrawal event occurring.

Which ONE of the following indications require the operator to manually trip the reactor after selecting SB (Standby) on the CEDM MODE SELECT switch?

a. CEA motion in the outward direction

b. One CEA has a inward subgroup deviation of greater than 9.9 inches

c. A twelve fingered CEA is misaligned greater than 6.6 inches

d. A CWP has occurred due to the CEA deviation

QUESTION: 059 (1.00)

Reactor Coolant Pump (RCP) 1A is operating with the following indications?

- suction pressure 2250 psig.
- controlled bleedoff flow 10 gpm
- controlled bleedoff temperature 185 degrees F.

Which ONE of the following indicates the status of the RCP seals?

a. No seals have failed

b. Only NO. 1 seal has failed

c. Only NO. 2 Seal has failed

d. NO. 1 and 2 seals have failed

QUESTION: 060 (1.00)

During operation at 100% power RCP 2A experiences a simultaneous loss of Nuclear Cooling Water (NCW) and Seal Injection.

Which ONE of the following describes the required operator action (in the proper order) if seal injection and NCW cannot be restored?

a. Trip the reactor and stop the RCP after 10 minutes

b. Stop the RCP, trip the reactor and close the bleed-off valve after 1 minute

c. Stop the RCP, trip the reactor within 10 minutes

d. Trip the reactor, stop the RCP within 1 minute and close the bleed-off valve.

QUESTION: 061 (1.00)

Which ONE of the following conditions require Immediate Boration?

a. One hour after a reactor power cutback and CEAs are lower than the transient insertion limit

b. One CEA is NOT fully inserted following a reactor trip

c. Two Charging Pumps are declared inoperable in Mode 6

d. Shutdown margin is 2% in mode 3

QUESTION: 062 (1.00)

Which ONE of the following Boration flowpaths is utilized if the Refueling Water Tank (RWT) level is 65%?

a. Gravity feed from the RWT to the charging pumps via CH-HV-536, RWT to Charging Pumps valve.

b. Gravity feed from the RWT to the Safety Injection System (SIS), to the charging pumps via CH-HV-327, SIS to Charging Pumps valve.

c. Gravity feed from the Spent Fuel Pool to the SIS, to the charging pumps via CH-HV-327, SIS to Charging Pumps valve.

d. Gravity feed from the RWT through CH-V164, Boric Acid Filter Bypass valve, to the charging pumps via CH-UV-514, Boric Acid Makeup to Charging Pumps valve.

QUESTION: 063 (1.00)

While operating at 100% power a CEA slips from 148.5 inches to 137.75 inches.

The pulse counter for that CEA would now indicate:

a. 0.0 inches

b. 137.75 inches

c. 140.75 inches

d. 148.5 inches

QUESTION: 064 (1.00)

A mechanically bound CEA which can be withdrawn, CANNOT be driven into the core because:

a. CEA insertion is normally accomplished by gravity only.

b. the force exerted by the lower gripper is less than the force exerted by the upper gripper.

c. the pulldown coils are a weaker magnet than the lifting coils.

d. the load transfer coil cannot transfer the load of the CEA to engage the lower gripper.

QUESTION: 065 (1.00)

Which ONE of the following post trip indications would determine that a Loss of Coolant Accident was occurring instead of an Main Steam Line Break inside containment?

a. RCS subcooling decreasing

b. RCS pressure decreasing

c. Containment pressure increasing

d. Pressurizer level decreasing

QUESTION: 066 (1.00)

Following a loss of Plant Cooling Water supply to the Nuclear Cooling Water (NCW) system, the Essential Cooling Water (ECW) System is used to supply critical NCW loads.

Which ONE of the following loads CANNOT be supplied by this lineup?

a. Essential Chiller

b. Spent Fuel Pool Heat Exchanger

c. Normal Chiller

d. Letdown Heat Exchanger

QUESTION: 067 (1.00)

The ECW to NCW cross-tie valves EWA-UV-145 and EWA-UV-65 will automatically close upon receipt of a:

a. AFAS signal.

b. ECW radiation monitor alarm.

c. SIAS signal.

d. ECW surge tank "B" low level alarm.

QUESTION: 068 (1.00)

The Safety Function Status Check with the highest priority is:

a. Containment Atmosphere Control

b. Reactivity Control

c. RCS Inventory and Pressure Control

d. Maintenance of Vital Auxiliaries

QUESTION: 069 (1.00)

Following receipt of an automatic Reactor Trip signal from low Steam Generator Level, all CEAs failed to insert.

The FIRST operator action taken to establish reactivity Control is:

a. insert CEAs manually from the CEDMCS operators module.

b. borate the RCS at greater than 40 gpm.

c. manually trip the reactor.

d. de-energize L03 and L10, CEDM-MG power supply load centers.

QUESTION: 070 (1.00)

Under accident conditions with the Auxiliary Spray valves open for pressure control, the presence of a steam void in the Reactor Vessel Upper Head would be indicated by a:

a. decrease in pressurizer pressure.

b. increase in pressurizer level.

c. decrease in pressurizer pressure with a small decrease in pressurizer level.

d. increase in pressurizer pressure with a small increase in pressurizer level.

QUESTION: 071 (1.00)

During accident conditions Subcooled Natural Circulation flow has stopped and Steam Generator (SG) tube voiding is suspected.

The preferred method of removing voiding in the (SG) tubes is to:

a. cool the SG by increased steaming.

b. feed and drain the SG.

c. increase reactor pressure to 2300 psig.

d. open the reactor vessel head vents to the reactor drain tank.

QUESTION: 072 (1.00)

The Containment Spray Chemical Addition Tank has been inadvertently drained while in Mode 1.

The consequences of this action would be increased:

a. corrosion of containment materials during a LOCA.

b. containment pressure during a LOCA.

c. formation of hydrogen during a LOCA.

d. containment iodine levels during a LOCA.

QUESTION: 073 (1.00)

For a steam line rupture inside containment, 40EP-9R004, "Excess Steam Demand" requires stabilizing RCS Tc using the Atmospheric Dump valve on the intact Steam Generator after the uncontrolled cooldown has been terminated.

Which ONE of the following is the reason for stabilizing RCS Tc?

a. Prevent an uncontrolled pressurizer level increase, resulting in solid plant operations and possible lifting of a pressurizer safety valve

b. Promote development of natural circulation flow if the RCPs are stopped

c. Ensure minimum RCS subcooling margin is maintained

d. Prevent RCS cooldown and heatup rate from exceeding the limits in Technical Specifications

QUESTION: 074 (1.00)

Which ONE of the following is the most restrictive condition established for Shutdown Margin with the reactor operating at 100% power?

a. Single CEA ejection

b. Steam Line Break inside Containment at BOL

c. Steam Line Break inside Containment at EOL

d. Excessive Steam Generator Feed Flow

QUESTION: 075 (1.00)

The Unit is rapidly reducing Reactor power and Main Turbine load in response to a loss of Condenser vacuum. Reactor Power cannot be reduced fast enough to equalize with Turbine load and maintain TAVE=TREF +/- 2 degrees F.

Which ONE of the following manual operator actions is required?

a. Initiate a Reactor Power Cutback.

b. Trip the Reactor and Main Turbine.

c. Open the Turbine Bypass Valves to the Condenser.

d. Open the Turbine Bypass Valves to the Atmosphere.

QUESTION: 076 (1.00)

During a "Station Blackout" the RCS should be cooled ONLY enough to maintain subcooling.

The basis for that criteria is to prevent:

a. exceeding the steam driven Auxiliary Feed Pump capacity.

b. emptying the pressurizer, causing a loss of subcooling.

c. losing all auxiliary feed water capability as steam pressure is lowered.

d. a reactor restart accident during cooldown.

QUESTION: 077 (1.00)

The following conditions exist following a station blackout:

- Diesel Generator (DG) "A" tagged out for maintenance
- Diesel Generator "B" has started but the output breaker has not automatically closed

Why is it necessary to close the DG "B" output breaker within 15 minutes or secure the DG?

a. Oil accumulation in the cylinders will cause damage when running unloaded.

b. Provide power to a fuel oil transfer pump to refill the day tank.

c. Conserve the batteries to allow energizing alternate sources of power.

d. Provide power for the essential spray pond system which supplies DG cooling.

QUESTION: 078 (1.00)

Which ONE of the following must be used during a station blackout to determine CEA position?

a. Rod Bottom Lights

b. Lower Electrical Limit Lights

c. CPC/CEAC remote operator display

d. CEAC CRT indication

QUESTION: 079 (1.00)

After a fire and establishing control at the Remote Shutdown Panel (RSP) it is necessary to close the control room isolation dampers, HJB-M01, M02, M03, and M55.

Why is it necessary to shut these dampers immediately after establishing control at the RSP?

a. Ensure that the smoke does not make the RSP uninhabitable.

b. Ensure they are shut before the fire makes them inoperable.

c. Closing the isolation dampers from the RSP will also trip the control room essential air handling units.

d. Isolates the vital cable spreading room to maximize the effectiveness of the fire suppression system.

QUESTION: 080 (1.00)

Which ONE of the following evolutions CANNOT be performed from the Remote Shutdown Panel?

a. Open Auxiliary Spray Valves

b. Shutdown Main Feedwater Pumps

c. Initiate an MSIS

d. Open Atmospheric Steam Dump Valves

QUESTION: 081 (1.00)

Following a Reactor Trip from 100% power, Pressurizer pressure is 1300 psia, RCS Tc is 550 degrees F.

Which ONE of the following is the required operator action?

(41EP-1E001, "Emergency Operations" Appendix D, RCP NPSH Curve is attached for your use)

a. Trip one RCP.

b. Trip two RCPs.

c. Trip three RCPs.

d. Trip ALL RCPs.

QUESTION: 082 (1.00)

During implementation of the CRS Safety Function Flowchart; the Secondary Operator is directed to stabilize RCS Tc below 570 degrees F.

This action ensures that:

a. RCS heat removal is through the steam generators.

b. the main steam safety valves will not lift.

c. core heat production is reduced to decay heat only.

d. adequate Shutdown Margin is maintained.

QUESTION: 083 (1.00)

Given the following information after a Reactor Trip and using the appropriate "Normal and Harsh Containment RCS P/T Limits" curve (attached):

- RCS Th 525 degrees F
- Pressurizer Pressure 1200 psia
- Pressurizer Level 31%
- Reactor Vessel Upper Head Level 16%
- Steam Generator #1 WR level 4% with the ADV open and auxiliary feedwater flow established
- Containment Temperature 148 Degrees F.

Determine which parameter will NOT allow HPSI throttling.

a. Subcooled Margin

b. Reactor Vessel Upper Head Level

c. Pressurizer Level

d. Steam Generator Level

QUESTION: 084 (1.00)

Which ONE of the following parameters would indicate that an RCS leak was through a stuck open Pressurizer Safety Valve instead of from the Cold Leg?

a. Containment temperature increasing

b. Pressurizer level increasing

c. Containment sump level increasing

d. Pressurizer pressure decreasing

QUESTION: 085 (1.00)

Following a LOCA the Primary Operator should not anticipate and prematurely initiate a Recirculation Actuation Signal (RAS) because:

a. the LPSI pumps are required to provide sufficient decay heat removal until the time calculated for the Refueling Water tank (RWT) level to decrease to the RAS setpoint.

b. the charging pumps must be placed in the "pull to lock" position after the RAS but they are still required for RCS inventory control until RWT level decreases to the RAS setpoint.

c. the entire contents of the RWT to the RAS setpoint are required to be injected to ensure adequate shutdown margin is obtained.

d. the entire contents of the RWT to the RAS setpoint are required to prevent air binding pumps and consequently the loss of both heat removal loops.

QUESTION: 086 (1.00)

If the charging pumps are still required for auxiliary spray or RCP seal injection following a RAS signal, the suction is transferred to the:

a. Reactor water makeup tank.

b. Boric acid makeup tank.

c. Spent Fuel Pool.

d. Volume control tank.

QUESTION: 087 (1.00)

Which ONE of the following parameters would be the most likely cause of an "air bound" LPSI pump, while running for Shutdown Cooling in mid-loop operations?

a. Throttled flow.

b. Rising suction temperature.

c. Lowering suction level in the hot leg.

d. Lowering suction pressure in the hot leg.

QUESTION: 088 (1.00)

Given the following conditions and the attached Core Data Book Curves:

- Unit 1 is in Mode 5 with the RCS drained to the centerline of the hot leg (21 inches).
- One LPSI pump is OOS and the operating LPSI pump has tripped.
- The core exit temperature was 124 degrees F at the time of the LPSI pump trip.
- At the time of the loss of LPSI pump the reactor had been shutdown 9 days from 100% full power EOC (End Of Cycle).

Determine the amount of time it would take to make an uncontrolled entry into mode 4 (EPIP Alert Level) from the time the LPSI pump was lost.

a. 8 - 10 minutes.

b. 10 - 12 minutes.

c. 12 - 14 minutes

d. 14 - 16 minutes

QUESTION: 089 (1.00)

The Reactor is in MODE 6 with Core alterations in progress when one Startup Channel Neutron Flux Monitor is declared inoperable.

Which ONE of the following immediate actions is required?

a. Suspend all operations involving core alterations or positive reactivity changes.

b. Emergency borate RCS until a boron concentration of 2150 ppm is established.

c. Immediately evacuate the refueling area until the startup channel neutron flux monitor is restored to operable.

d. Immediately contact Reactor Engineering to perform an overall core reactivity balance.

QUESTION: 090 (1.00)

Unit 3 is in Mode 2 when the middle detector of an Excore Safety Channel fails.

Which ONE of the following describes all of the RPS functional units which must be bypassed in accordance with Technical Specifications?

a. Variable Overpower - Local Power Density High

b. DNBR Low - Logarithmic Power Level High

c. Logarithmic Power Level High - DNBR Low - Local Power Density High

d. Logarithmic Power Level High - Variable Overpower - DNBR Low - Local Power Density High

QUESTION: 091 (1.00)

During a Steam Generator Tube Leak the operators are cautioned to maintain Reactor Coolant System (RCS) Pressure approximately equal (+/- 50 psi) to Steam Generator (SG) pressure during the cooldown and depressurization.

The reason for this is to:

a. reduce the feed water inventory required for the affected SG.

b. reduce the leak rate from the RCS to the SG.

c. prevent the possibility of Boron dilution and SG chemicals entering the RCS.

d. prevent the loss of RCS subcooling.

QUESTION: 092 (1.00)

If a tube rupture is indicated in BOTH Steam Generators (SG) which ONE of the following is used to identify the MOST affected SG.

a. Main steam line radiation monitor

b. SG blowdown radiation monitor

c. SG sample activity

d. Condenser off gas radiation monitor

QUESTION: 093 (1.00)

During a Steam Generator (SG) Tube Rupture event, it is important to maintain the level in the isolated SG 72-80% WR during cooldown, in order to:

a. minimize the radioactive release in the event of a stuck open SG atmospheric dump valve.

b. reduce flow impingement on undamaged tubes surrounding the rupture.

c. increase the static head in order to reduce break flow.

d. enhance natural circulation flow during the cooldown.

QUESTION: 094 (1.00)

When instrument air pressure is lost, RCS Tave and Secondary Pressure is to be controlled using:

a. manual control of turbine bypass valves SGN-PV-1001 through 1006.

b. manual control of turbine bypass valves SGN-PV-1007 and 1008.

c. the atmospheric dump valves SGA-HV-178, 179, 184, and 185.

d. the main steam safety valves.

QUESTION: 095 (1.00)

During operation at 100% power a loss of 125 VDC Class 1E Bus PKB-M42 occurs.

The operator is required to manually Trip the Reactor because:

a. the associated reactor trip breakers have lost control power and will not trip.

b. the main steam isolation valves fail closed.

c. IA-UV-2, Instrument Air Containment Isolation valve fails closed.

d. the RCP's will also be required to be tripped.

QUESTION: 096 (1.00)

During operation at 100% power and Tave selected to "average" in the Reactor Regulating System (RRS), a Tc RTD fails low. RCS pressure was 2250 psia at the time of the failure.

Which one of the following is an expected automatic response of the pressurizer level control system?

a. Standby charging pump starts

b. All pressurizer heaters deenergize

c. Always running charging pump stops

d. Letdown flow increases to maximum

QUESTION: 097 (1.00)

During Refueling a total failure of the Steam Generator Nozzle Dam Seals occurs which results in a loss of Refueling Pool level with an irradiated fuel assembly above the Reactor Core.

Which ONE of the following describes the preferred location the fuel assembly should be placed in?

a. Upender and then lowered to the horizontal position

b. Any available location in the core

c. Its original location in the core

d. Deep area of the pool just above the floor

QUESTION: 098 (1.00)

Following a Reactor Trip and Loss of Forced Circulation, natural circulation (NC) flow cannot be accurately verified for a short interval.

Which ONE of the following parameters is indication that the Heat Removal Safety Function is being accomplished during that interval?

a. T-hot and Tc are tracking together and rising.

b. RCS subcooling margin is 18 degrees F.

c. Delta-T between the core exit thermocouple (CET) temperature and T-hot is 15 degrees F.

d. RCS loop Delta T is 50 degrees F.

QUESTION: 099 (1.00)

Which ONE of the following air operated Feed Water valves does NOT fail "AS-IS" following a loss of Instrument Air?

a. Economizer Control

b. Economizer Isolation

c. Downcomer Control

d. Downcomer Isolation

QUESTION: 100 (1.00)

Following a LOCA, which of the following conditions would ENSURE that the Containment Atmospheric Control Safety Function Criteria is met?

a. One spray header is functioning and delivering 3200 gpm.

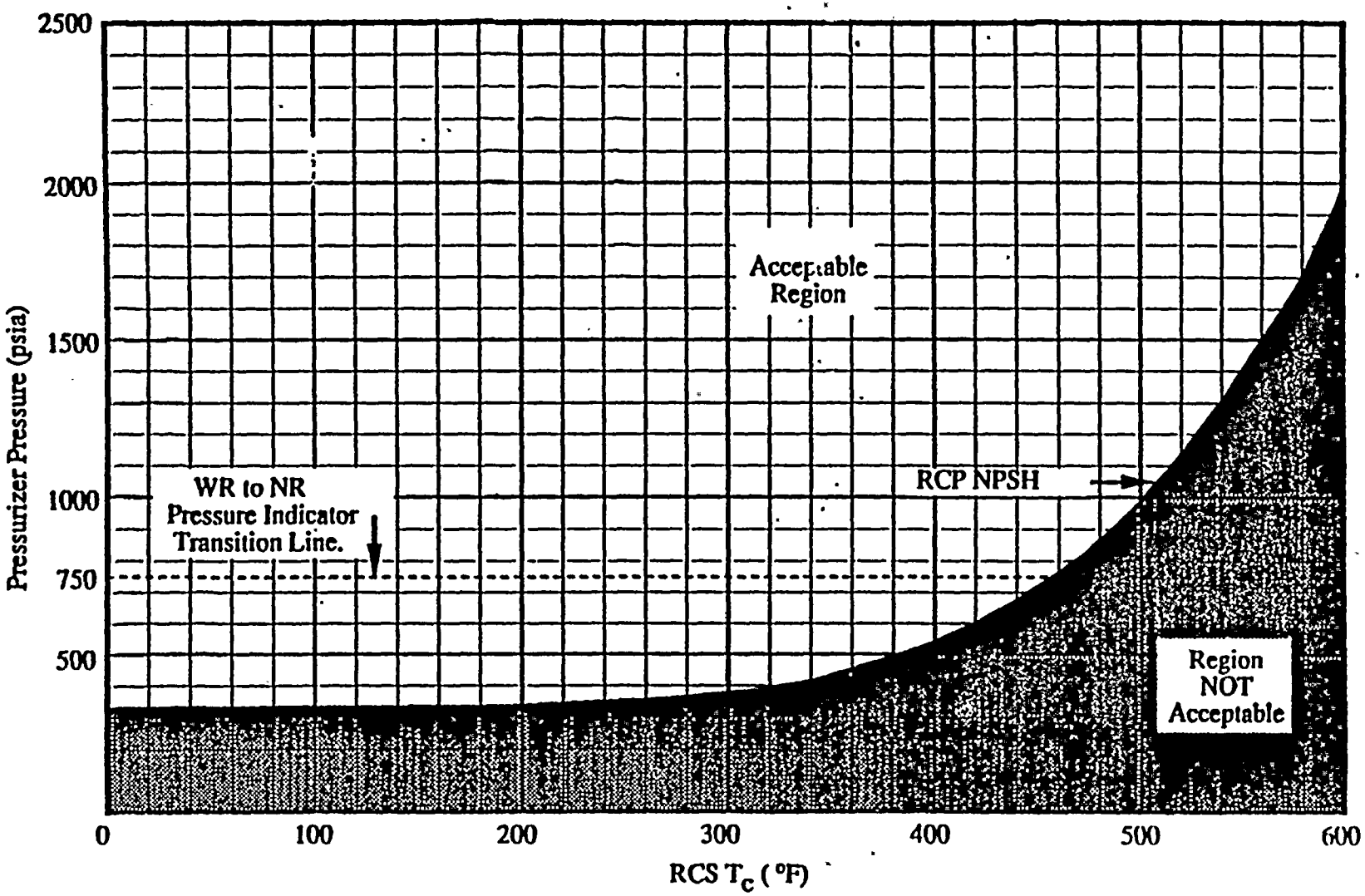
b. Containment pressure is 9.5 psig.

c. One spray header is functioning and delivering 3200 gpm, and CIAS has actuated.

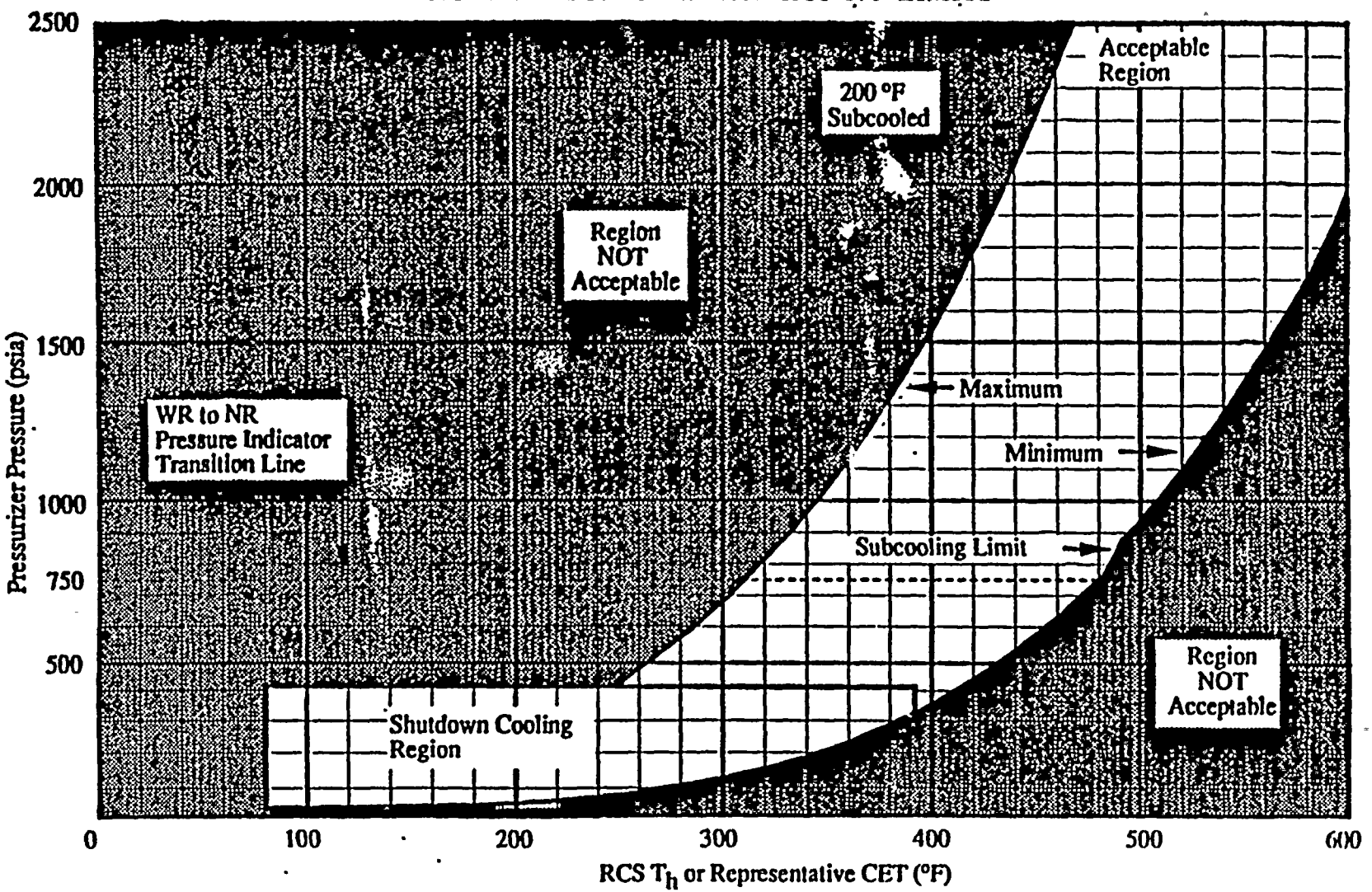
d. Containment pressure increased and then stabilized at 8.0 psig, and CIAS has actuated.

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

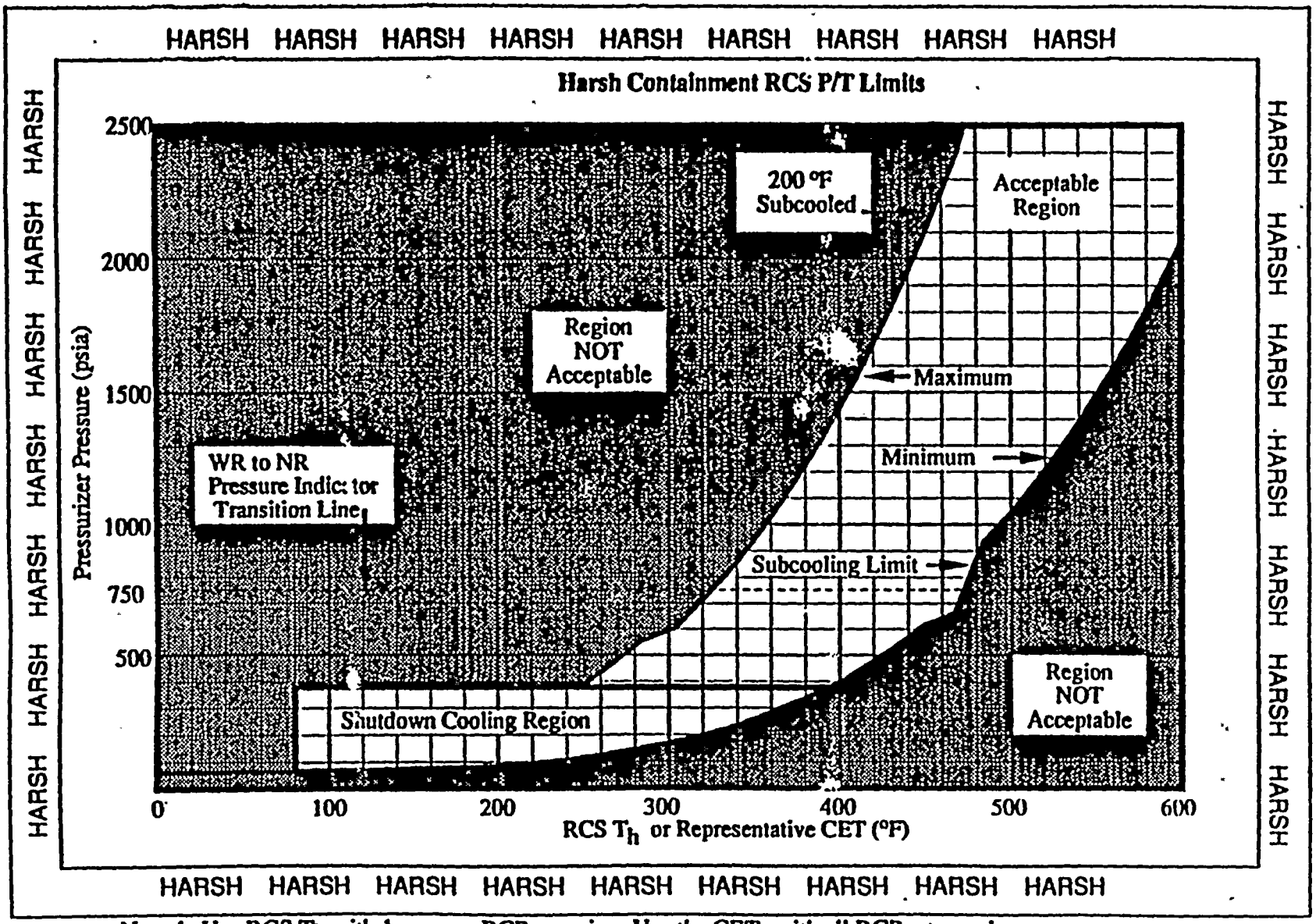
RCP NPSH Curve



NORMAL CONTAINMENT RCS P/T LIMITS



Note 1- Use RCS T_h with 1 or more RCPs running; Use the CETs with all RCPs stopped.
Note 2- Use this curve with containment temperature below 150°F.



Note 1- Use RCS T_h with 1 or more RCPs running; Use the CETs with all RCPs stopped.
Note 2- Use this curve with containment temperature above 150°F.

TABLE 2.18.1

KEY REACTOR CORE PARAMETERS FOLLOWING A LOSS OF SHUTDOWN COOLING DURING MIDLOOP OPERATION

TIME AFTER REACTOR SHUTDOWN (days)	DECAY HEAT LOAD (MWth)	HEATUP RATE (degF/min)	TIME TO BOIL (min)	MAKEUP WATER FLOWRATE (gpm)	TIME TO CORE UNCOVERY (min)
5	12.7	7.5	12.0	92.0	89.0
15	8.6	5.4	17.9	65.0	137.0
30	6.4	4.3	23.6	46.0	167.0
50	4.7	3.2	32.7	33.0	234.0
80	3.5	2.4	45.1	25.0	312.0

Plant Conditions

500 EFPD at 100% RTP

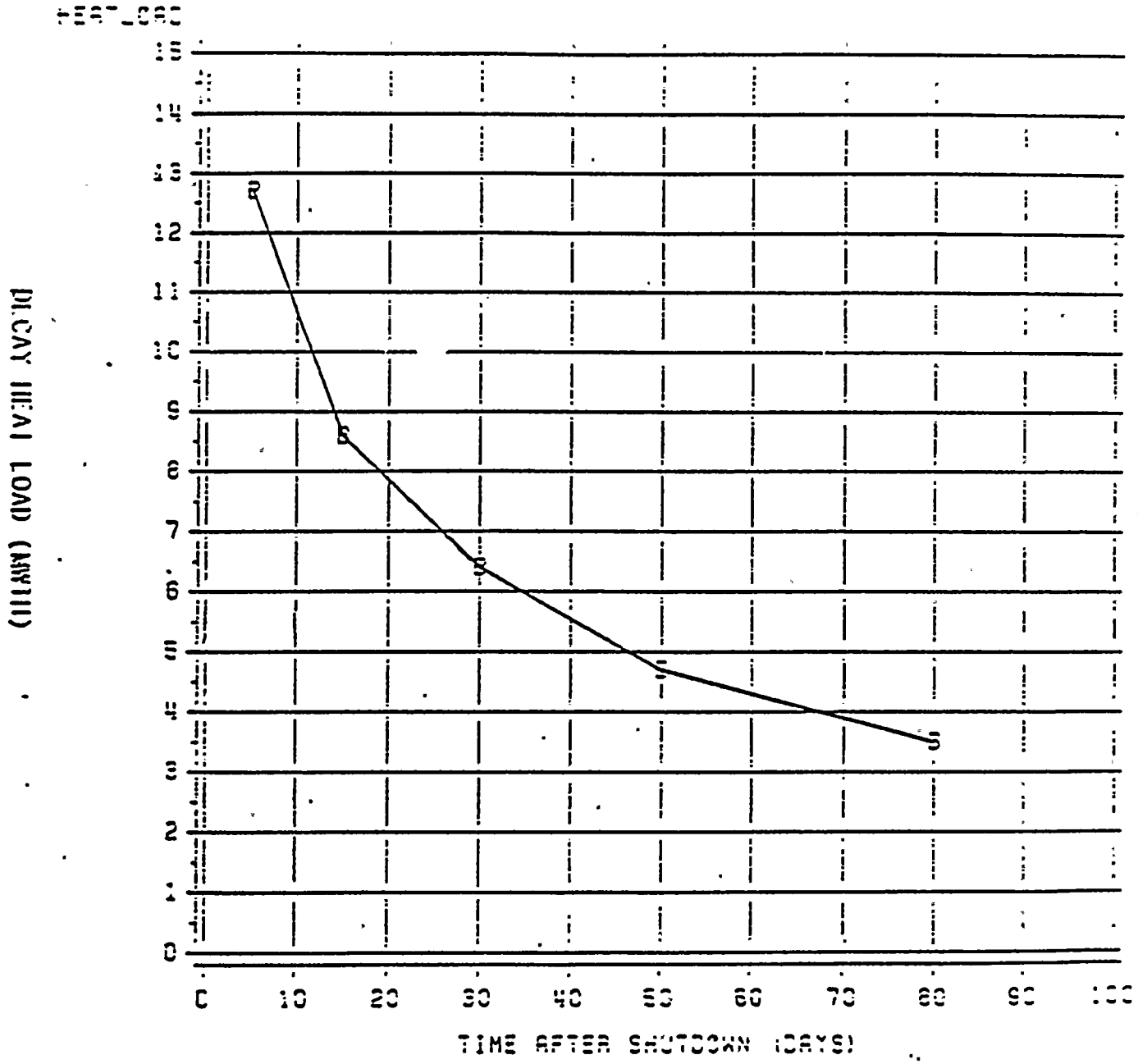
Comments: T2Z18Z1

Reference Source Of Data

162-03189-PFC/HAT

FOR INFORMATION ONLY

CURVE 2.18.1
DECAY HEAT LOAD BASED ON 500 EFPD AT 100 PERCENT POWER

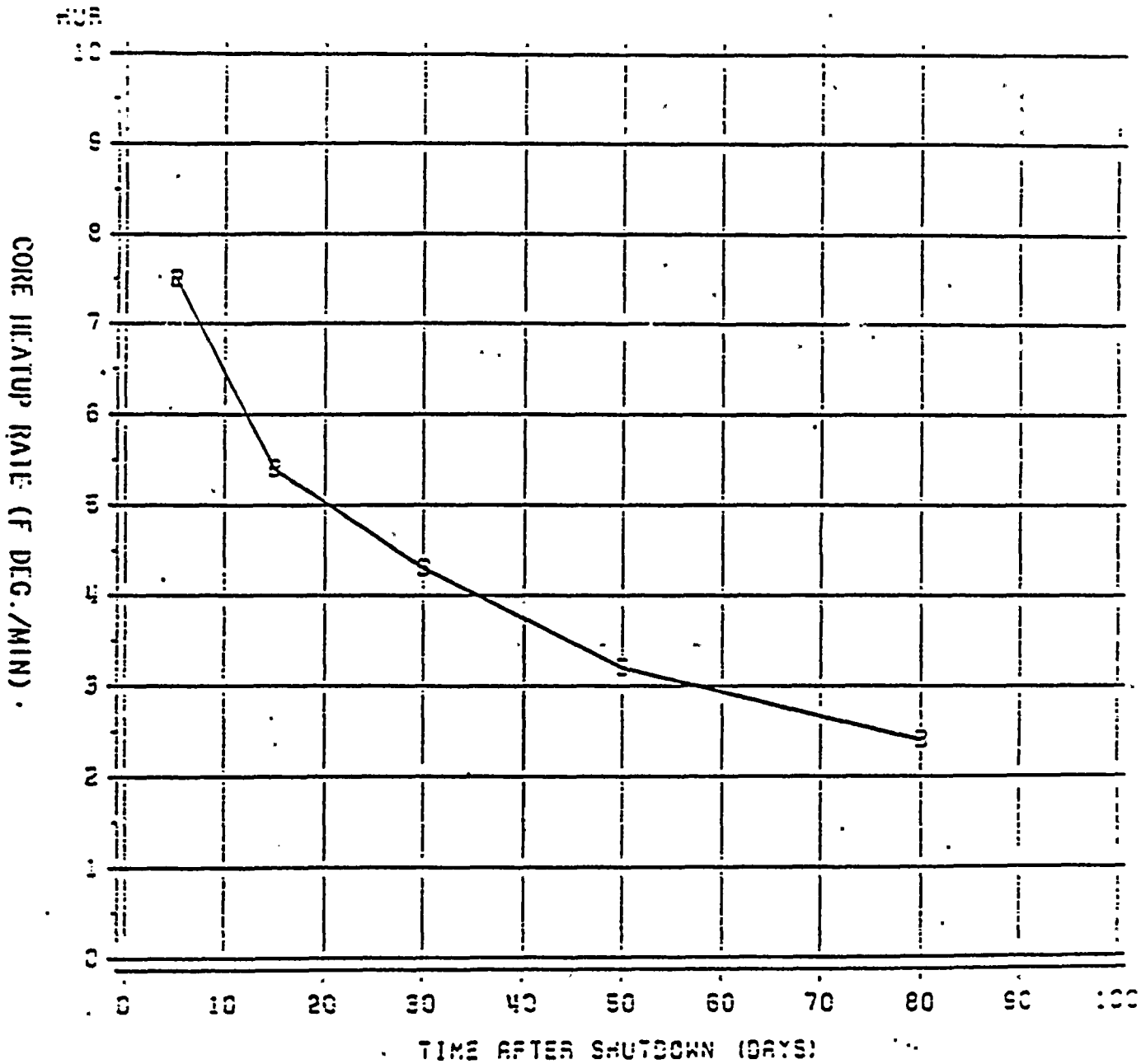


PLANT CONDITIONS
AFTER 500 EFPD @ 100% RTP
COMMENTS: 0221821

REFERENCE SOURCE OF DATA
182-03189-PFC/HAT

FOR INFORMATION ONLY

CURVE 2.18.2
CORE HEATUP RATE FOLLOWING A LOSS OF SDC DURING MIDLOOP OPERATIONS

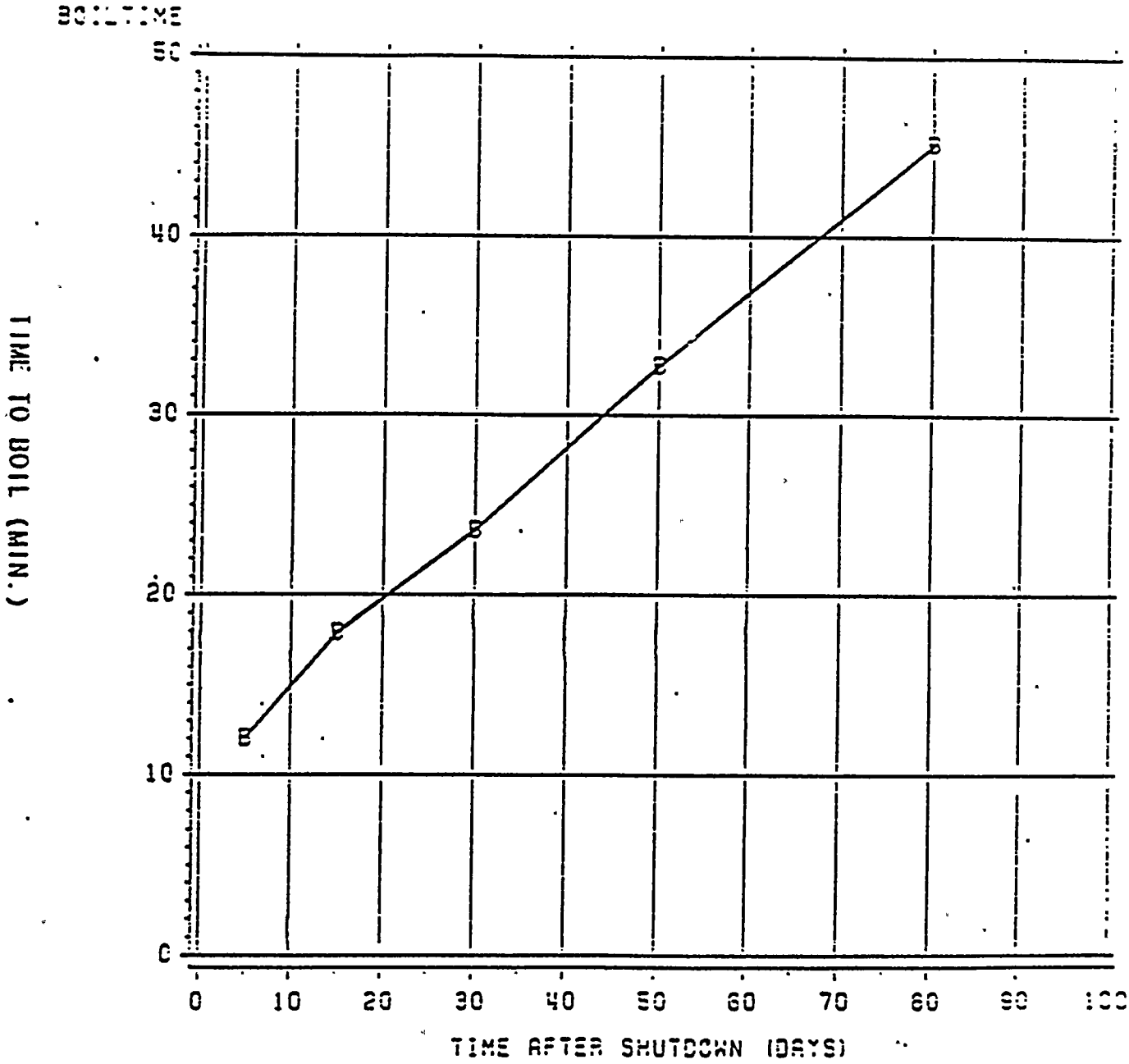


PLANT CONDITIONS
AFTER 500 EFPD @ 100% RTP
COMMENTS: C2218Z2

REFERENCE SOURCE OF DATA
162-03189-PFC/HAC

FOR INFORMATION ONLY

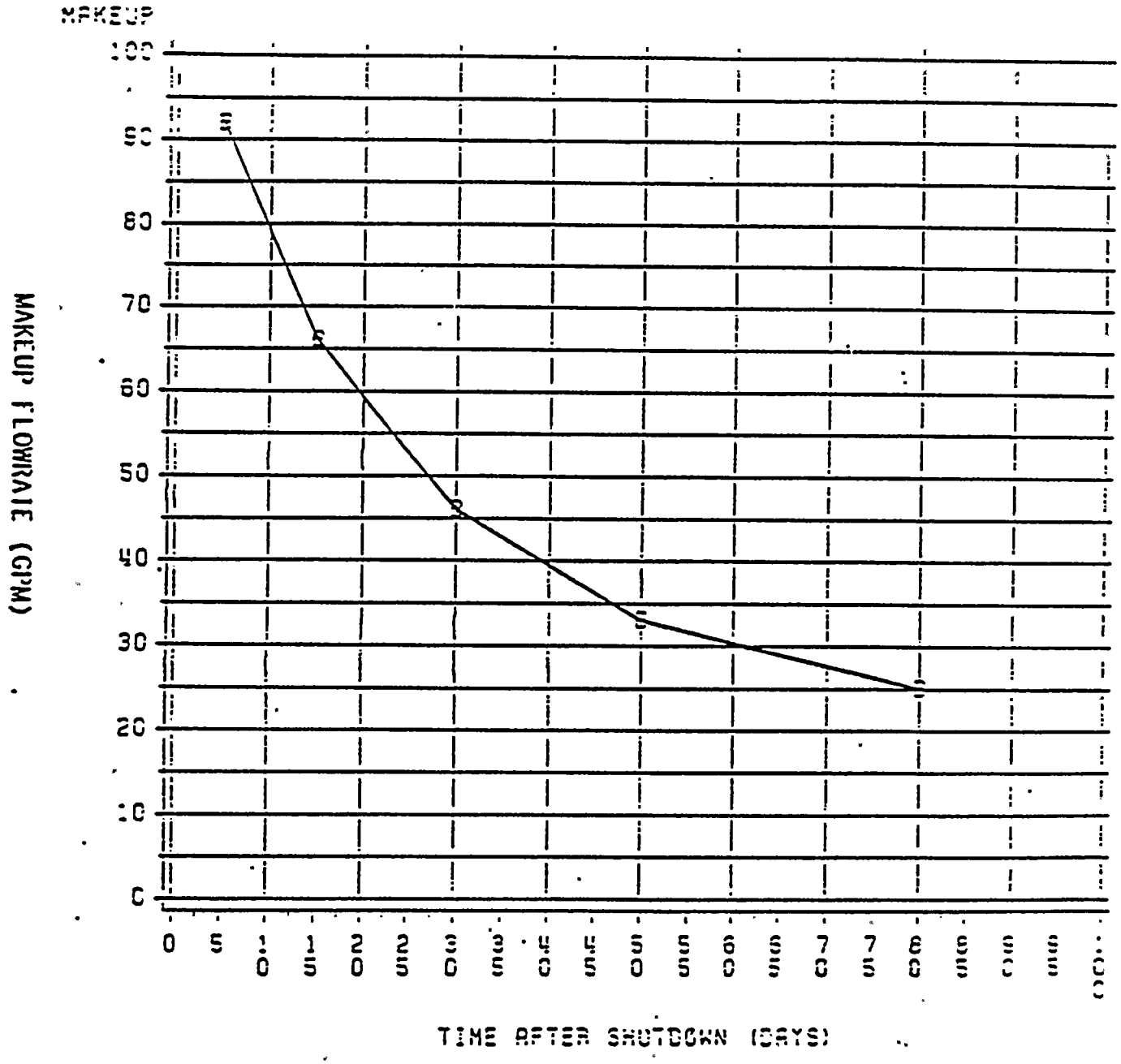
CURVE 2.18.3
TIME TO BOIL COOLANT FOLLOWING A LOSS OF SDC DURING MIDLOOP OPERATIONS



PLANT CONDITIONS
AFTER 500 EFPD @ 100% RTP
COMMENTS: C221823

REFERENCE SOURCE OF DATA
162-03189-PFC/HAT

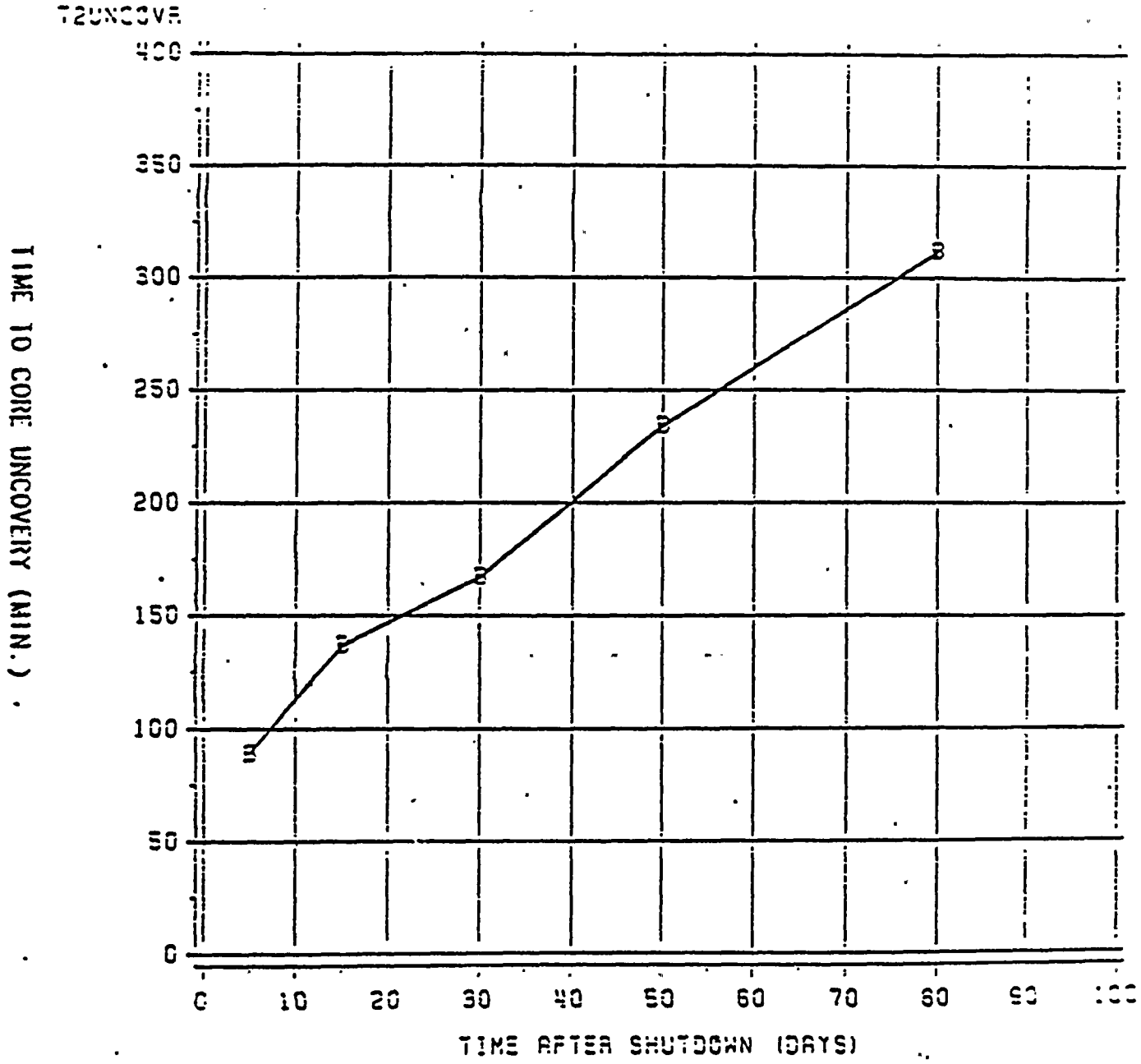
CURVE 2.18.4
MAKEUP FLOWRATE REQUIRED FOLLOWING A LOSS OF SDC DURING MIDLOOP OPERATIONS



PLANT CONDITIONS
AFTER 500 EFPD @ 100% RTP
COMMENTS: C2218Z4

REFERENCE SOURCE OF DATA
162-03189-PFC/HAT

CURVE 2.18.5
TIME TO CORE UNCOVERY FOLLOWING A LOSS OF SDC DURING MIDLOOP OPERATIONS



PLANT CONDITIONS
AFTER 500 EFPD @ 100% RTP
COMMENTS: C2Z1825

REFERENCE SOURCE OF DATA
162-03189-PFC/HAT