U.S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT

EXAMINATION REPORT NO.	50-336/88-17(OL)
FACILITY DOCKET NO.	50-336
FACILITY LICENSE NO.	DPR-65
LICENSEE:	Northeast Nuclear Energy Company P.O.Box 270 Hartford, Connecticut 06141
FACILITY:	Millstone Unit 2
EXAMINATION DATES:	July 20-21, 1988

CHIEF EXAMINER: BSN — 7 Sept 88
Barry S. Norris, Senior Operations Engineer — Date

APPROVED BY: P W Eselgroth — 9-12-88
Peter W. Eselgroth, Chief, PWR Section, Operations Branch, DRS — Date

SUMMARY: Written and operating examinations were administered to three Senior Reactor Operator (SRO) upgrade candidates. All three SROs passed the examinations and were issued their licenses.

DETAILS

TYPE OF EXAMINATIONS: Replacement

EXAMINATION RESULTS:

	SRO Pass/Fail
Written	3 / 0
Operating	3 / 0
Overall	3 / 0

CHIEF EXAMINER AT SITE: B. S. Norris, USNRC, RI

˜HER EXAMINERS: R. B. Eaton, USNRC, NRR
J. A. Prell, USNRC, RI
D. M. Silk, USNRC, RI

1. GENERIC DEFICIENCIES:

The following is a summary of generic deficiencies noted on the examinations. This information is being provided to aid the licensee in upgrading license and requalification training programs. No licensee response is required.

a. Operating Examination

(1) During the simulator examination, the candidates dismissed two instances of spurious alarms with "Guess it's a simulator problem." Although their reaction was in fact correct, this type of action on the part of the operators would tend to distract from the training atmosphere in the simulator.

(2) During transients, the board operators tend to focus on the digital readouts on the nuclear instrumentation cabinets for extended periods of time (30-40 seconds) while ignoring the other indications on the panels.

b. Written Examination

There were no generic weaknesses identified during the grading of the written examinations.

2. EXAMINATION REVIEW:

a. Written Examination

The written examination was reviewed by the training staff in two parts. Prior to the administration of the examination, it was reviewed for technical accuracy, applicability to the Millstone Unit 2 facility, and assurance that the answer key was consistent with the question. Twenty of the fifty questions were rewritten as a result of this review. As can be seen in Attachment 1 (SRO Written Examination with Answer Key), most of the changes were minor in nature: terminology, or addition/deletion of a word or phrase for clarification; one question (7.02) was rewritten to reflect the actual responsibilities of the SRO at Millstone Unit 2.

The new questions were retyped and pages replaced in the examinations before they were given to the candidates. The examination, as attached, was modified to show the original question and the question as rewritten.

Additionally, the answer key was modified to reflect any technical changes identified during the facility review of the examination.

There was disagreement between the NRC examiners and the facility reviewers as to whether or not two questions (8.03 and 8.11) on Technical Specification basis were required knowledge (see Attachment 2). NUREG-1021, Operator Licensing Examiner Standards, ES-402, "Scope of Written Examinations Administered to Senior Reactor Operators - Power Reactors," paragraph A.4 states in part "... Questions concerning the Technical Specifications will require a thorough knowledge of ... the basis for the requirements ..."

The facility reviewers for the written examination were R. Cimmino, J. Parillo, and M. Wilson.

b. Operating Examination

The scenarios for the simulator portion of the operating examination were also reviewed by the training staff prior to being administered to the candidates. One event had to be changed to due a modeling problem with one of the malfunctions.

The facility reviewers for the simulator scenarios were R. Burnside, J. Parillo, and D. Wright.

3. PERSONNEL PRESENT AT EXIT MEETING:

NRC Personnel

B. S. Norris - Chief Examiner
P. J. Habinghorst - Resident Inspector

Facility Personnel
R. Burns, Jr. - Simulator Instructor
C. Clememt - Unit 3 Superintendent
D. Pantalone - Program Coordinator
J. Parillo - Assistant Supervisor, Operator Training
J. Smith - Unit 2 Operations Supervisor
M. Wilson - Supervisor, Operator Training

4. SUMMARY OF COMMENTS MADE AT EXIT MEETING:

a. The NRC and the facility agreed that the pre-administration review of the examinations was an improvement over the previous method of review in that the candidates would be examined on their knowledge and not on their ability to interpret questions that were not clear and specific to Millstone Unit 2.

b. The NRC expressed concern about three items within the plant:

(1) During transients the RO's tend to focus their attention on the digital readouts on the NI cabinets rather than the entire control boards. Although this was noticed only in the simulator, training frequently is a reflection of actual practice in the plant. (see paragraph 1.a(2) above)

The facility noted the comment and stated they would reinforce to the operators that the digital readouts were not to be used to the exclusion of the remainder of the control boards.

(2) Technical Specification 3.1.2.2.1 states "Two flow paths from the boric acid storage tanks via either a boric acid pump or gravity feed connection and a charging pump to the Reactor Coolant System and one associated heat tracing circuit shall be OPERABLE. (Modes 1, 2, and 3)" Although there is a management interpretation of what determines two separate paths if one boric acid pump is out of service, the interpretation is not in writing. This could lead to confusion and even contradiction between shifts.

The facility stated that will review remedies implemented by other facilities.

(3) In the case of an evacuation of the control room, it is not clear within the procedures which of the shutdown panels the operators should go to. One procedure directs the operators to the Hot Shutdown panel, and another directs them to the Fire Shutdown panel.

The facility response was that it is the responsibility of the Shift Supervisor to determine which of the panels would provide better control of the plant. This determination would be based partially on whether or not it was an Appendix R fire.

Attachments:
1. SRO Written Examination and Answer Key
2. Facility Comments on Written Examinations after Facility Review

ATTACHMENT 1

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

FACILITY: MILLSTONE 2

REACTOR TYPE: PWR-CE

DATE ADMINISTERED: 88/07/20

EXAMINER: PRELL, J.

CANDIDATE: Master Answer Key

INSTRUCTIONS TO CANDIDATE:

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

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	25.00	______	______	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00	______	______	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00	______	______	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	______	______	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00		______ Final Grade	______ %	Totals

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

3. Use black ink or dark pencil <u>only</u> to facilitate legible reproductions.

4. Print your name in the blank provided on the cover sheet of the examination.

5. Write your answers on the examination paper. If more paper is required, use the paper provided. For each extra piece of paper used, print your name in the upper right hand corner and identify the answer, for example, 1.4, 6.3.

6. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.

7. Use abbreviations only if they are commonly used in facility <u>literature</u>.

8. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

9. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.

10. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

11. If parts of the examination are not clear as to intent, ask questions of the <u>examiner</u> only.

12. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

13. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

(2) Exam aids - figures, tables, etc.

b. Turn in your copy of the examination and all pages used to answer the examination questions.

c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 5.01 (1.00)

Following a reactor trip on low SG level, EXPLAIN why automatic initiation of the AFW system is delayed 3 minutes and 25 seconds.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.02 (2.00)

DEFINE and PROVIDE the basis for the Power Dependent Insertion Limit (PDIL)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.03 (1.00)

A reactor startup is in progress. Shutdown Groups A and B are fully withdrawn. Keff = 0.95 and the countrate is 50 cps and stable. The control room operator begins regulating rod withdrawal until he has added 3 percent delta k/k reactivity to the core. As a result of his actions, has the shutdown margin INCREASED, DECREASED, or REMAINED THE SAME? Explain your answer. (1.0)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.04 (2.40)

QUESTION AS ORIGINALLY WRITTEN

Identify FOUR changes in core reactivity which take place immediately following a reactor trip and during the nest 24 hours. STATE the reason for each change. Assume the reactor was at 100 percent power, equilibrium conditions, prior to the trip and that the plant will remain in hot standby following the trip.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

IDENTIFY FOUR changes in core reactivity which take place during a reactor trip and the next 24 hours. STATE the reason for each change. Assume the reactor was at 100 percent power, equilibrium conditions, prior to the trip and that the plant will remain in hot standby following the trip.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.05 (2.00)

INDICATE whether the MTC will become LESS NEGATIVE, MORE NEGATIVE, or REMAIN CONSTANT under the following conditions.

a. BOC, with a reduction in moderator temperature

b. BOC, HFP with a boron addition

c. core ages between BOC and EOC

d. fission product poison buildup

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.06 (2.00)

QUESTION AS ORIGINALLY WRITTEN

Assume the reactor is in a steady state condition with no Xenon Oscillations for the core conditions identified below. MATCH each core condition to one of the four basic axial power distribution shapes expected for that condition.

a. BOC, hot full power	1. top peaked
b. EOC, hot full power	2. bottom peaked
c. BOC, hot zero power	3. saddle or hourglass shape
d. EOC, stuck rod partially inserted	4. cosine shape

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Assume the reactor is in a steady state condition with no Xenon Oscillations for the core conditions identified below. MATCH each core condition to one of the four basic axial power distribution shapes expected for that condition.

a. BOC, hot full power	1. top peaked
b. EOC, hot full power	2. bottom peaked
c. BOC, hot zero power	3. saddle or hourglass shape
d. EOC, 100% power, group 7 at 145 steps	4. cosine shape

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.07 (1.50)

QUESTION AS ORIGINALLY WRITTEN

Assume there is an electrical load increase from 70 percent to 90 percent. Indicate how the following parameters would change (INCREASE, DECREASE, or REMAIN THE SAME). Assume steady state to steady state. EXPLAIN your answer.

a. Tave.

b. rho

c. Pressurizer pressure

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Assume there is an electrical load increase from 70 percent to 90 percent. Indicate how the following parameters would change (INCREASE, DECREASE, or REMAIN THE SAME). Assume steady state to steady state. EXPLAIN your answer.

a. Tave.

b. Reactivity

c. Pressurizer pressure

QUESTION 5.08 (3.00)

QUESTION AS ORIGINALLY WRITTEN

Millstone Unit 2 is in Mode 3, BOC, boron concentration is 900 ppm, all shutdown groups are withdrawn, and the actual reactivity present in the core is minus 4 percent delta k/k. A dilution of the boron concentration increases source range counts from 100 cps to 196 cps. Coincident with the dilution, Xenon reactivity changes have added plus 1000 pcm to the core. CALCULATE the final boron concentration. Assume a constant differential boron worth of 100 pcm/ppm. State all assumptions and show all work.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Millstone Unit 2 is in Mode 3, BOC, boron concentration is 900 ppm, all shutdown groups are withdrawn, and the actual reactivity present in the core is minus 4 percent delta k/k. A dilution of the boron concentration increases source range counts from 100 cps to 196 cps. Coincident with the dilution, Xenon reactivity changes have added plus 1 percent delta k/k to the core. CALCULATE the final boron concentration. Assume a constant inverse boron worth of 100 ppm/percent delta k/k. State all assumptions and show all work.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.09 (.60)

Answer TRUE or FALSE

As the core ages, differential boron worth decreases.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.10 (2.50)

Assume the plant is stable on natural circulation with the following conditions:

Thot - Tcold = 30 degrees F

Thot subcooled margin = 35 degrees F subcooled

Steam Generator pressure = 650psia

CALCULATE the RCS pressure and LIST any assumptions made.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.11 (2.00)

QUESTION AS ORIGINALLY WRITTEN

Answer the following statements as TRUE or FALSE

a. Single phase natural circulation (NC) can take up to 15 minutes to initially establish.

b. For single phase NC, the flow rate should be greater than 5 percent.

c. A sudden decrease in RCS flowrate may indicate a transit from single phase NC to two phase NC.

d. During single phase NC, cool water circulates from the steam generator, through the cold leg, up through the core, into the head area of the reactor vessel, into the hot leg and back to the steam generator.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Answer the following statements as TRUE or FALSE

a. Single phase natural circulation (NC) can take up to 15 minutes to initially establish.

b. For single phase NC, the flow rate should be greater than 5 percent.

c. A sudden decrease in RCS flowrate may indicate a transit from single phase NC to two phase NC.

d. During single phase NC, the majority of the cool water circulates from the steam generator, through the cold leg, up through the core, into the head area of the reactor vessel, into the hot leg and back to the steam generator.

QUESTION 5.12 (2.00)

QUESTION AS ORIGINALLY WRITTEN

For each of the pump characteristics identified below, IDENTIFY whether it applies to a positive displacement pump (PDP), a centrifugal charging pump (CCP), BOTH or NEITHER.

a. flow varies with the pump discharge head

b. requires a discharge flowpath when starting in order to prevent excess pressure buildup

c. discharge head is directly proportional to the pump speed squared

d. high inlet fluid velocity creates a low pressure which is used to draw the inlet fluid into the mixing chamber

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

For each of the pump characteristics identified below, IDENTIFY whether it applies to a positive displacement pump (PDP), a centrifugal pump (CP), BOTH or NEITHER.

a. flow varies with the pump discharge head

b. requires a discharge flowpath when starting in order to prevent excess pressure buildup

c. discharge head is directly proportional to the pump speed squared

d. high inlet fluid velocity creates a low pressure which is used to draw the inlet fluid into the mixing chamber

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.13 (3.00)

INDICATE for each of the following parameter changes whether it will cause an INCREASE, DECREASE, or NO CHANGE in the DNBR. Assume all other parameters remain constant.

a. Decrease in the core inlet temperature

b. Decrease in RCS pressure

c. ASI changes from 0 to minus 0.3

d. Increase in reactor coolant flow

e. Reactor power increase

f. Loading high enriched fuel near the core center

(***** END OF CATEGORY 05 *****)

QUESTION 6.01 (2.00)

QUESTION AS ORIGINALLY WRITTEN

What is the purpose of each of the following Control Element Drive System (CEDS) interlocks?

a. Metroscope Inhibit from Shutdown (MISH)

b. Metroscope Inhibit from Regulating Group (MIRG)

c. CEA Motion Inhibit (CMI)

d. CEA Withdrawal Prohibit (CWP)

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

What rod motion is prevented by each of the following Control Element Drive System (CEDS) interlocks? Include setpoints for each interlock.

a. Metroscope Inhibit from Shutdown (MISH)

b. Metroscope Inhibit from Regulating Group (MIRG)

c. CEA Motion Inhibit (CMI)

d. CEA Withdrawal Prohibit (CWP)

QUESTION 6.02 (2.40)

QUESTION AS ORIGINALLY WRITTEN

MATCH each of the below RCP seal pressure conditions to the appropriate seal status.

	RCS	MIDDLE	UPPER	VAPOR	SEAL STATUS
a.	2250	1125	1125	50	1. NORMAL
b.	2250	1500	750	50	2. FIRST SEAL FAILS
c.	2250	1125	50	50	3. SECOND SEAL FAILS
d.	2250	2250	1125	50	4. THIRD SEAL FAILS

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

MATCH each of the below RCP seal pressure conditions to the appropriate seal status.

	RCS	MIDDLE	UPPER	VAPOR	SEAL STATUS
a.	2250	1125	1125	50	1. NORMAL
b.	2250	1500	750	50	2. LOWER SEAL FAILS
c.	2250	1125	50	50	3. MIDDLE SEAL FAILS
d.	2250	2250	1125	50	4. UPPER SEAL FAILS

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.03 (1.00)

QUESTION AS ORIGINALLY WRITTEN

Indicate if the following statements are TRUE or FALSE:

a. During plant heatup, the excess letdown flow path can be used to draw a bubble in the pressurizer. This lineup is preferable to normal letdown due to the increased driving head.

b. While operating in the shutdown cooling mode, the back pressure control valves (2CH-201P and Q) are used for controlling pressure.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Indicate if the following statements are TRUE or FALSE:

a. During plant heatup, the excess letdown flow path can be used to draw a bubble in the pressurizer. This lineup is preferable to normal letdown due to the increased flowrate.

b. While operating in the shutdown cooling mode, the back pressure control valves (2CH-201P and Q) are used for controlling PRESSURE.

QUESTION 6.04 (3.00)

QUESTION AS ORIGINALLY WRITTEN

Indicate if the following statements are TRUE or FALSE and JUSTIFY your answer.

a. During a plant cooldown, with RCS pressure at 1650 psia and the SIAS blocked per procedure, if a large LOCA were to occur, the SIAS would be prevented.

b. A CSAS is initiated whenever containment pressure exceeds 27 psig.

c. The ESAS sensor cabinet monitors the following parameters: RWST level, pressurizer pressure, pressurizer level, containment pressure, emergency bus voltage, steam generator pressure, fuel handling area radiation monitors, containment radiation monitors and steam generator level.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Indicate if the following statements are TRUE or FALSE and JUSTIFY your answer.

a. During a plant cooldown, with RCS pressure at 1650 psia and the SIAS blocked per procedure, if a large LOCA were to occur, the SIAS would be prevented.

b. A CSAS will be initiated if thw ONLY input signal to the CSAS is containment pressure of 27 psig.

c. The ESAS sensor cabinet monitors the following parameters: RWST level, pressurizer pressure, pressurizer level, containment pressure, emergency bus voltage, steam generator pressure, fuel handling area radiation monitors, containment radiation monitors and steam generator level.

QUESTION 6.05 (1.20)

One function of the incore analysis program (INCA) is to calculate six (6) different incore power values used for estimating incore power distribution. IDENTIFY any four (4) of these values calculated by INCA.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.06 (2.00)

QUESTION AS ORIGINALLY WRITTEN

Assume the plant is at 100 % power, steady state conditions, when an instrument failure causes the main Feedwater Regulating Valve to No. 1 steam generator (SG) to close. Before the operators can recover, the plant trips on low steam generator level. All other systems are functioning normal.

a. Assuming no operator action, what is providing RCS heat removal the first two minutes after the reactor trip ? (0.50)

b. Again assuming no further operator action, what automatic actions occur ensuring RCS heat removal via the SGs five minutes after the trip ? (1.0)

c. Ten minutes after the reactor trip, water from the Condensate Storage Tank is lost. What is the secondary source of water available to the operators to help ensure continued RCS heat removal ? (0.5)

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Assume the plant is at 100 % power, steady state conditions, when an instrument failure causes the main Feedwater Regulating Valve to No. 1 steam generator (SG) to close. Before the operators can recover, the plant trips on low steam generator level. All other systems are functioning normal.

a. Assuming no operator action, what system/component is providing RCS heat removal the first two minutes after the reactor trip ? (0.50)

b. Again assuming no further operator action, what automatic actions occur ensuring RCS heat removal via the SGs five minutes after the trip ? (1.0)

c. Ten minutes after the reactor trip, water from the Condensate Storage Tank is lost. What is the secondary source of water available to the operators to help ensure continued RCS heat removal ? (0.5)

QUESTION 6.07 (2.40)

a. DEFINE the purpose/function of the Safety Injection Tanks (SITs)? (1.0)

b. DESCRIBE the operational concern associated with operating the SITs with a pressure greater than 250 psig? (1.4)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.08 (1.00)

Prior to starting the fourth RCP during a plant heatup, the RCS temperature should be (LESS THAN, GREATER THAN, or EQUAL TO) 500 degrees F. EXPLAIN your answer.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.09 (2.00)

LIST any five of the seven conditions which cause either the EDG to trip or the EDG output breaker to open after an emergency start following a loss of normal power.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.10 (3.00)

Give the set point and the Technical Specification (TS) Bases for each of the following RPS trips.

a. Reactor Coolant flow
b. low SG level
c. high PZR pressure
d. high containment pressure

QUESTION 6.11 (2.50)

With the plant at 100 percent power, a small leak occurs from the RCS to the RBCCW System.

a. What TWO indications does the operator have available to him to identify that this leak exists?

b. List the three (3) possible sources of this leakage.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.12 (2.50)

DRAW the power supplies to the facility 1 vital ac panel, 2-VIAC-1. Include both the normal and alternate power supplies.

(***** END OF CATEGORY 06 *****)

QUESTION 7.01 (3.00)

QUESTION AS ORIGINALLY WRITTEN

A 24 year old female worker, expecting to get pregnant, is scheduled to work in a Radiation Area where she will be exposed to a field of 50 mrem/hour. Her lifetime exposure history per NRC Form 4 indicates she has received a lifetime wholebody exposure of 9,600 mrem. During this present quarter, the woman has received 200 mrems of wholebody radiation. CALCULATE how long the woman can work in the Radiation Area before exceeding her quarterly limit. LIST all considerations used in arriving at your answer.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Answer the following questions concerning radiological exposure control practice at Milstone Station:

a. As a radiation worker, your quarterly whole body exposure limit has been increased to the maximum allowed by Millstone Administrative procedures. During the current quarter, you have received a whole body exposure of 1.5 Rem. How long could you stay in an area with a dose rate of 200 mrem/hr without exceeding your administrative quarterly exposure limit? Show all work. State all assumptions. (1.00)

b. List FOUR of the areas where a Radiation Work Permit is required for entry. (2.00)

QUESTION 7.02 (2.90)

QUESTION AS ORIGINALLY WRITTEN

ANSWER the following questions concerning a loss of Instrument Air (AOP-2563):

a. Explain why PZR level is controlled by securing the operating charging pump? (0.5)

b. With the operating charging pump secured, HOW is water made up to the RCS to make up for RCP bleedoff? (0.5)

c. IDENTIFY four design considerations that make a total loss of instrument air extremely unlikely? (1.2)

d. A manual reactor trip must be initiated if instrument air falls below what value? (0.7)

QUESTION AS REWRITTEN DURING THE PRE-EXAM REVIEW

ANSWER the following questions concerning a loss of Instrument Air (AOP-2563):

a. Explain why PZR level is controlled by securing the operating charging pump? (0.5)

b. With the operating charging pump secured, HOW is water AUTOMATICALLY made up to the RCS to make up for RCP bleedoff? (0.5)

c. IDENTIFY four design considerations that make a total loss of instrument air extremely unlikely? (1.2)

d. A manual reactor trip must be initiated if instrument air falls below what value? (0.7)

QUESTION 7.03 (1.00)

Answer the following in accordance with AOP2564, "Loss of RBCCW." Assume the plant is at 100 percent power, steady state conditions. At 14:20 the following alarms are annunciated:

containment air recirc. coolers (HI TEMP)
spent fuel pool heat exchanger (HI TEMP)
letdown heat exchanger (HI TEMP)
Rx vessel concrete support cooling coils (LO FLOW)
RBCCW-A pump (TRIP)

Also RCP-A bearing oil temperature is at 185 degrees F and slowly trending higher. The operator immediately begins actions to restore RBCCW flow. At 14:24 the RCP-A bearing oil temperature is at 190 degrees F, its seal temperature is at 225 degrees F and its bleedoff temperature is 200 degrees F. WHAT actions, if any, should the operator take? EXPLAIN your answer.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.04 (2.50)

Assume that SG No.1 Blowdown Rad Monitor has given a high alarm.

After 20 minutes the operator trips the reactor because the CVCS is not able to maintain PZR level. He completes the standard post trip actions of EOP2525 and transitions to EOP2534, "SG Tube Rupture."

a. WHY is the operator cautioned in EOP2534 not to use the steam driven AFW pump? (1.0)

b. EOP2534 requires the operator to reduce Th to less than 520 degrees Fahrenheit before manually isolating SG No.1. WHAT is the reason for this? (1.0)

c. Why does EOP-2534 caution the operator that if ruptures are detected in both Steam Generators to only isolate one of the SGs? (0.5)

QUESTION 7.05   (1.00)

In accordance with AOP-2571, "Inadvertent ECCS Actuation," once an ECCS actuation has been determined to have been inadvertent, why is the operator cautioned on using the equipment handswitch to override an ESAS signal?

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.06 (3.00)

A reactor trip occurs 100 percent power. EOP2525, "Standard Post Trip Actions," is entered.

a. LIST three actions/expected responses which are used to ensure and verify that a turbine trip has occurred? (1.5)

b. Four CEAs remain stuck out. How long must the operator borate? State any assumptions. (1.5)

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.07 (3.00)

QUESTION AS ORIGINALLY WRITTEN

Answer the following questions in accordance with Technical Specifications for refueling operations:

a. WHICH nuclear instrumentation must be operable? Include WHERE and WHAT type of indication each must provide? (1.0)

b. WHAT TWO reactivity conditions must be met for the RCS and the refueling canal? (1.0)

c. GIVE the TWO bases why at least one shutdown cooling loop must be operable. (1.0)

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Answer the following questions in accordance with Technical Specifications for refueling operations:

a. WHICH nuclear instrumentation must be operable? Include WHERE and WHAT type of indication each must provide? (1.0)

b. WHAT reactivity condition must be met for the RCS and the refueling canal? (1.0)

c. GIVE the TWO bases why at least one shutdown cooling loop must be operable. (1.0)

(3.00)

h AOP-2558, "Emergency Boration" :

conditions which may require emergency boration? (1.2)

teps required to initiate emergency boration? Include
tions. (1.8)

pleted. IDENTIFY
of EOP-2532,

is required to
ing? (2.0)

QUESTION 7.10 (1.60)

Assume there has been a reactor trip and the post trip actions of EOP-2525 have been completed. Plant conditions indicate that an excess steam event has occurred causing the RCS temperature to go below 500 degrees F. The operators have entered EOP-2536, "Excess Steam Demand", and isolated the faulted SG. A CAUTION statement alerts the operators to the potential for Pressurized Thermal Shock (PTS) of the reactor vessel. Per EOP-2536, LIST the actions the operators should take in order to meet the Pressure/Thermal limits of TS and IDENTIFY the systems used.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.11 (1.00)

In accordance with AOP-2552, "Cooldown from Outside the Control Room", IDENTIFY the reason for the CAUTION that the Shutdown heat exchanger flow control valve be cracked open VERY SLOWLY.

(***** END OF CATEGORY 07 *****)

QUESTION 8.01 (1.00)

The following question is applicable to ACP-QA-2.06A,"Station Tagging". MATCH the following tag definitions with the appropriate colored tag.

a. Placed on equipment to prevent reenergizing if it trips.	1. Red tag
b. Placed on mechanical or electrical equipment which, if operated, would endanger personnel and/or equipment.	2. Blue tag
c. Placed on a device or piece of equipment as a caution if other tags do not apply.	3. Green stripped hold tag
d. Placed on equipment to indicate that the equipment is to be energized only by order of the individual to whom the tag is issued.	4. Yellow tag

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.02 (1.00)

In accordance with ACP-OA-2.06B, "Jumper, Lifted Lead and Bypass Control", LIST under what condition the Shift Supervisor can grant an exception to the requirement for performing an independent verification of a installed/removed jumper device.

QUESTION 8.03 (1.00)

QUESTION AS ORIGINALLY WRITTEN

TS Section 3.9.6 places a maximum loading limit on containment building cranes used in the "fuel only" region. Give the TS Basis for this requirement.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

TS Section 3.9.6 places a maximum loading limit on containment refueling machine cranes used in the "fuel only" region. Give the TS Basis for this requirement.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.04 (3.00)

QUESTION AS ORIGINALLY WRITTEN

In accordance with ACP-QA-3.02, "Station Procedures and Forms", would you, as a licensed on-shift SRO, approve the following proposed changes? Justify your answer.

a. A proposed change to an I&C surveillance would substitute a particular model of digital volt meter with a newer model of digital volt meter.

b. A proposed change to an Operations Dept. valve lineup procedure would add steps to the procedure that would provide useful information to the operator for locating and identifying certain valves

c. A proposed change to an Operations Dept. Abnormal Operating Procedure would reduce the conditions under which the procedure would be applicable.

d. A proposed change to a Maintenance procedure would add a new QC hold point.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

In accordance with ACP-QA-3.02, "Station Procedures and Forms", would you, as a licensed on-shift SRO, approve the following proposed changes? Justify your answer.

a. A proposed change to an I&C surveillance would substitute a particular model of digital volt meter with a functionally equivalent newer model of digital volt meter.

b. A proposed change to an Operations Dept. valve lineup procedure would add steps to the procedure that would provide useful information to the operator for locating and identifying certain valves

c. A proposed change to an Operations Dept. Abnormal Operating Procedure would reduce the conditions under which the procedure would be applicable.

d. A proposed change to a Maintenance procedure would add a new QC hold point.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.05 (3.00)

In accordance with TS, WHAT is the relationship between Limiting Conditions for Operations (LCO), Limiting Safety System Setting (LSSS) and Safety Limits in terms of preventing releases of radioactivity to the environment.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.06 (2.00)

QUESTION AS ORIGINALLY WRITTEN

Listed below are the dates two monthly surveillances were performed during 1987. Also listed, when applicable, are the dates when the operational status of the trains changed. IDENTIFY whether or not the surveillance requirements were met. EXPLAIN your answer. Note: a calendar is attached for your use.

a.	1/2/87 surv.	2/2/87 surv.	3/9/87 surv.	4/9/87 surv.	5/15/87 surv.	
b.	1/7/87 surv.	2/7/87 surv.	3/10/87 surv.	4/9/87 declared inoperable due to maintenance	4/28/87 surv.	4/29/87 declared operable

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

Listed below are the dates two monthly surveillances were performed during 1987. Also listed, when applicable, are the dates when the operational status of the trains changed. IDENTIFY whether or not the surveillance requirements were met. EXPLAIN your answer. Note: a calendar is attached for your use. CONSIDER EACH CASE SEPARATELY.

a.	1/2/87 surv.	2/2/87 surv.	3/9/87 surv.	4/9/87 surv.	5/16/87 surv.	
b.	1/7/87 surv.	2/7/87 surv.	3/10/87 surv.	4/9/87 declared inoperable due to maintenance	4/28/87 surv.	4/29/87 declared operable

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.07 (2.00)

QUESTION AS ORIGINALLY WRITTEN

The Shift Supervisor has approved a Unit 2 Discharge permit for liquid radioactive waste per ACP6.03, "Radioactive Liquid Waste Discharge Policy", and the required dilution flow has been established. Suddenly effluent monitors detect, at the site boundary, levels corresponding to 2 R/hour whole body exposure. The concentration is determined to exceed the limits specified in 10CFR20.403. Attached is a copy of Millstone Unit 2 Emergency Action Levels and EPIP Form 4701-4, "Reportability".

a. IDENTIFY what immediate actions, including reporting requirements, the Shift Supervisor should take.

b. WHAT level should this event be classified? JUSTIFY your answer by listing the section of the Emergency Action Level Table used to classify this event.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

The Shift Supervisor has approved a Unit 2 Discharge permit for liquid radioactive waste per ACP6.03, "Radioactive Liquid Waste Discharge Policy", and the required dilution flow has been established. Suddenly effluent monitors detect, at the site boundary, levels corresponding to 2 R/hour whole body exposure. The concentration is determined to exceed the limits specified in 10CFR20.403. Attached is a copy of Millstone Unit 2 Emergency Action Levels and EPIP Form 4701-4, "Reportability".

a. IDENTIFY what immediate actions, including reporting requirements, the Shift Supervisor should take.

b. WHAT NRC level should this event be classified? JUSTIFY your answer by listing the section of the Emergency Action Level Table used to classify this event.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.08 (1.50)

QUESTION AS ORIGINALLY WRITTEN

In accordance with ACP-QA-10.05, "Log Book Requirements, (Control Room)", INDICATE whether the following statements are TRUE or FALSE with respect to Control Room Logs.

a. The on-duty Supervising Control Operator (SCO) may make entries into the SS Log.

b. The SS log can have a blank line between entries.

c. In the event the computer is unavailable, the SCO may make entries into the BOP log.

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

In accordance with ACP-QA-10.05, "Log Book Requirements, (Control Room)", INDICATE whether the following statements are TRUE or FALSE with respect to Control Room Logs.

a. The on-duty Supervising Control Operator (SCO) may make entries into the SS Log.

b. The SS log can have a blank line between entries.

c. In the event the computer is unavailable, the FPO must take manual logs.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.09 (1.50)

In accordance with SHP-4912, "Radiation Work Permit Completion and Flow Control", LIST the THREE SS/SCO responsibilities associated with their signing Radiation Work Permits.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.10 (2.50)

In accordance with Technical Specifications for fire protection and the AOP-2579 series "Contingency Fire Procedures for ...Appendix "R" Fire...":

a. WHAT is the minimum number of Fire Brigade members required to be maintained onsite at all times? (0.5)

b. WHAT is the maximum number of members of the MINIMUM OPERATING SHIFT CREW, as required by TS, which can be assigned to the Fire Brigade? (0.5)

c. WHAT is the maximum amount of time that the number of members of the Fire Brigade can be less than the minimum required by TS? (0.5)

d. IDENTIFY the two locations where the two way radios, used for backup communications, are stored? (1.0)

QUESTION 8.11 (2.00)

TS Section 3.4.4.6 states that " the pressurizer shall be "OPERABLE" with ...at least two groups of pressurizer heaters having at least 130 KW." WHAT is the TS Basis for this requirement?

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.12 (1.50)

QUESTION AS ORIGINALLY WRITTEN

In accordance with ACP-QA-9.02, "Station Surveillance Program", assume that the Maintenance Dept. comes to the Shift Supervisor (SS) with a Work Order to perform a surveillance on HPSI pump "A" that will require that it be taken out of service.

a. LIST what action(s) the SS must perform prior to signing the Surveillance Data Sheet authorizing the work. (0.5)

b. Upon completion of the surveillance by Maintenance, IDENTIFY the two actions the SS must perform prior to checking the correct "Acceptance Criteria Met" block and signing the Surveillance Data Sheet accepting HPSI pump "A". (1.0)

QUESTION AS REWRITTEN DURING PRE-EXAM REVIEW

In accordance with ACP-QA-9.02, "Station Surveillance Program", assume that the Maintenance Dept. comes to the Shift Supervisor (SS) with a Work Order to perform work on HPSI pump "A" that will require that it be taken out of service.

a. LIST what action(s) the SS must perform prior to authorizing the work. (0.5)

b. Upon completion of the work by Maintenance, IDENTIFY the two actions the SS must perform on the Surveillance Data Sheet prior to checking the correct "Acceptance Criteria Met" block and signing the Surveillance Data Sheet accepting HPSI pump "A". (1.0)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.[illegible] (1.00)

An I&C [illegible]n had been trouble shooting a problem with the rod control system [illegible] control room operator requests permission from the SCO to allow t[illegible]ician to step the rods in one step. The plant is at 90 percent [illegible] SHOULD the SCO grant permission to allow the technician to operate t[illegible] controls? JUSTIFY WHY or WHY NOT permission should be granted.

QUESTION 8.14 (2.00)

According to Technical Specifications, Identify the FOUR criteria that determine when containment integrity exists?

(***** END OF CATEGORY 08 *****)
(************* END OF EXAMINATION ***************)

ANSWER 5.01 (1.00)

Allows the delayed neutron population to decrease sufficiently so as to avoid an unacceptable return to power situation. [0.5] ~~[1.0]~~ and design considerations could be exceeded for containment temp. or pres. for a steam line break inside containment. [0.5]

REFERENCE

1. LP-2322, EO - 4
2. SD-2322, p - 10
3. PDCR 2-182-79

K/A 035000K109 3.8/4.5
035000K301 4.4/4.6
035000SG04 3.4/3.4

035000K109 035000K301 035000SG04 ...(KA'S)

ANSWER 5.02 (2.00)

Height above which rods must be maintained [0.5] in order to:
1. provide adequate shutdown margin [0.5]
2. limit ejected rod worth [0.5]
3. limit peaking factors [0.5]

REFERENCE

1. LP-2117, p-~~272~~.4,52 EO - ~~18~~ 1h
2. Exam bank 2117, q - 353

K/A 192005K115 3.4/3.9
192005K115 ...(KA'S)

ANSWER 5.03 (1.00)

REMAINED THE SAME [0.4] The SDM is not dependent on rod height. [0.6]

REFERENCE

1. LP-2116G, p-35 EO-1b
2. Exam bank 2116B, q - 111

K/A 192002 K1.14 3.8/3.9
192003 K1.07 3.0/3.0

ANSWERS -- MILLSTONE 2 -88/07/20-PRELL, J.

192002K114 192003K107 ...(KA'S)

ANSWER 5.04 (2.40)

[any four @ 0.6 each]

1. power drops promptly [0.3] due to negative rho inserted by the CEAs [0.3]
2. positive rho is added [0.3] as moderator temperature decreases [0.3]
3. negative rho is added during the next 10 hours [0.3] due to the Xenon buildup [0.3]
4. positive rho is added between 8-10 hours to 24 hours [0.3] due to Xenon decay [0.3]
5. positive rho [0.3] fuel temp decrease [0.3]
6. negative rho [0.3] samarium build up [0.3]

REFERENCE

1. LP-2116E, p-10,11,12 EO-2
2. LP-2116F, p-17,18 EO-4,5

K/A 192008K124 3.5/3.6
192006K107 3.4/3.4
192008K123 2.9/3.1
192008K124 ...(KA'S)

ANSWER 5.05 (2.00)

a. LESS NEGATIVE (more positive)
b. ~~MORE~~ LESS NEGATIVE
c. MORE NEGATIVE
d. MORE NEGATIVE [0.5 each]

REFERENCE

LP-2116E, p-16,19,21 EO-03

K/A 192004K106 3.1/3.1
192004K106 ...(KA'S)

ANSWER 5.06 (2.00)

a. 4
b. 3
c. 1
d. 2 [0.50 each]

REFERENCE

1. LP-2117, p-23,24,25 EO-1h,16
2. LP-2116D, TP-3

K/A 192005K114 3.2/3.5
192005K110 3.0/3.3
192005K115 192005K1.1 ...(KA'S)

ANSWER 5.07 (1.50)

a. Increases [0.2] - due to program Tave. [0.3]
b. ~~Decreases [0.2] - due to MTC [0.3]~~
c. No change [0.2] - due to heaters cycling on and off [0.3]
No change [0.2] - due to steady state to steady state $\rho = 0$ [0.3]

REFERENCE

1. LP-2304A, p-3,4 EO-5

K/A 010000A105 2.8/2.9
010000A106 3.1/3.2
010000A107 3.7/3.7
011000A101 3.5/3.6
011000A104 3.1/3.3
010000A105 010000A106 010000A107 011000A101 011000A104 ...(KA'S)

ANSWER 5.08 (3.00)

rho 1 = (k1 - 1) / k1 = -0.04 [0.25]
k1 = 1 / (1 - (-0.04)) = 0.9615 [0.25]
100 (1 - 0.9615) = 196 (1 - k2) [0.25]
k2 = 0.9804 [0.25]
rho 2 = (0.9804 - 1) / 0.9804 = -0.02 [0.25]
rho 2 - rho 1 = -0.02 - (-0.04) = 0.02 = ~~2000pcm~~ 2% ΔK/K [0.5]
1% ΔK/K ~~1000pcm~~ is due to Xenon, therefore remaining 1% ΔK/K ~~1000pcm~~ is due to the boron. [0.5]
~~1000pcm = 1000pcm / (10pcm / ppm) = 100ppm~~ [0.5]
1% ΔK/K = (1% ΔK/K)/(100 ppm / % ΔK/K) = 100 ppm
900ppm - 100ppm = 800ppm the new boron concentration [0.25]

REFERENCE
1. LP-2116B EO-4

K/A 192008K104 3.8/3.8
192007K105 3.0/3.2
192008K104 ...(KA'S)

ANSWER 5.09 (.60)

False [0.60]

REFERENCE
1. LP-2116D, p-15 EO-7

K/A 192007K104 3.1/3.4
192007K104 ...(KA'S)

ANSWER 5.10 (2.50)

SGpress.= 650psia = 495 degrees F sat. [0.5]
Tc will follow SG temp., thus Tc = 495 degrees F [0.5]
Th = 30 + Tc = 525 degrees F [0.5]
PZRtemp. = Th + 35 = 525 + 35 = 560 degrees F [0.5]
PZRpress. = sat. pressure for 560 degrees F = 1133.4psia [0.5]

REFERENCE
1. LP-2121C, p - 7,8,9,10,11 EO-5
2. Exam bank 2121C, q - 1172

K/A 193003K125 3.3/3.4
193003K117 3.0/3.2
193003K117 193003K125 ...(KA'S)

ANSWER 5.11 (2.00)

a. True
b. False
c. True
d. False [0.5 each]

REFERENCE
1. LP-2121J, p-14,18,19,22 EO-4,5,6,8b

K/A 193008K122 4.2/4.2
193008K122 ...(KA'S)

ANSWER 5.12 (2.00)

a. ~~C~~CP
b. PDP
c. BOTH
d. NEITHER [0.5 each]

REFERENCE
1. LP-2121E, p-17,19,21,32 EO-2a,2b,2c,4

K/A 193006K115 3.1/3.3
193006K115 ...(KA'S)

ANSWER 5.13 (3.00)

a. Increase
b. Decrease
c. Decrease
d. Increase
e. Decrease
f. Decrease [0.50 each]

*Reference

1. LP-2121I, p - 20 EO - 4
2. LP-2117, p - 29

K/A 193008K105 3.4/3.6

REFERENCE
193008K105 ...(KA'S)

ANSWER 6.01 (2.00)

(± 10 steps)

a. prevents the withdrawal or insertion of regulating CEA [0.25] if any shutdown CEA is lower than 163 steps [0.25]
b. prevents withdrawal or insertion of any shutdown CEA [0.25] if any regulating CEA is above 13 steps [0.25] (±5 steps)
c. inhibits both insertion and withdrawal motion for any of the 61 CEAs [0.25] with the occurrence of a PDIL, out of sequence overlap, group deviation, MIRG or MISH [0.25]
d. prevents all CEA group withdrawal [0.25] when 2 out of 4 RPS channels of high power or thermal margin low pressure exceed their pre-trip setpoints [0.25]

REFERENCE

1. SD-2302A, p-19,30,32 EO-6

K/A 001000K402 3.8/3.8
001000K504 ...(KA'S)

ANSWER 6.02 (2.40)

a. 3
b. 1
c. 4
d. 2 [0.6 each]

REFERENCE

1.LP-2301, P-35 EO-06

K/A 003000K103 3.3/3.6
003000K103 3.5/3.9
003000A201 003000K103 ...(KA'S)

ANSWER 6.03 (1.00)

a. True
b. False [0.5 each]

REFERENCE
1. SD-2304, p-37 EO-3,10
2. OP2304F, p-5

K/A 004010K505 3.8/4.2
004010K505 ...(KA'S)

ANSWER 6.04 (3.00)

a. False [0.5]-SIAS would still occur on high containment pressure [0.5]
b. False [0.5]-also need a SIAS present [0.5]
c. False [0.5]- does NOT monitor PZR level [0.5]

REFERENCE

1. LP-2384, p-13,16 EO-6,11

K/A 013000K101 4.2/4.4
013000K412 3.7/3.9
013000K101 013000K412 ...(KA'S)

ANSWER 6.05 (1.20)

1. Tq-Azimuthal tilt
2. ASI-Axial shape index
3. Fr-Integrated radial peaking factor
4. FrT-Integrated radial peaking factor total
5. Fxy-Planar radial peaking factor
6. FxyT- Planar radial peaking factor total
7. LHR- Linear Heat Rate [any 4 at 0.3 each]

REFERENCE

LP2117, p-43,44,F30 EO-9

K/A 193009K106 2.8/3.7
193000K106 ...(KA'S)

ANSWER 6.06 (2.00)

a. Water steaming off via the Steam Dump system [0.5]

b. Both motor driven AFW pumps start [0.5] and both AFW regulating valves open. [0.5] (equivalent answer is "AFW initiates")

c. Plant fire water [0.5]

REFERENCE
1. LP-2316A, p - 17 EO-14
2. SD-2322, p - 7,8,9,10 EO-9,10
3. OP-2322, p - 9

K/A 061000K406 4.0/4.2
061000K501 3.6/3.9
035010K101 4.2/4.5
035010K109 3.8/4.5
035010K301 4.4/4.6
035010K402 3.2/3.5

ANSWER 6.07 (2.40)

a. Provide a reliable means of rapidily reflooding the core following a large-break LOCA. [1.0]

b. Could introduce Nitrogen into the RCS [0.7] which could result in gas binding of the heat removal flow path. [0.7]

REFERENCE

1. SD-2306, p - 1,4
2. LP-2306-9 EO - 1,17

K/A 006000K602 3.4/3.9
006020A107 3.5/3.7
006000SG04 3.5/3.8
006000SG07 3.8/3.9
006000K602 006000SG04 006000SG07 006020A107 ...(KA'S)

ANSWER 6.08 (1.00)

Greater than [0.5] Prevents uplifting of the core internals due to the higher density of the RCS below 500 degrees F. [0.5]

REFERENCE

1. LP-2301, p-19 EO-12

K/A 002000A105 3.4/3.7
002000K501 3.1/3.4
002000A105 002000K501 ...(KA'S)

ANSWER 6.09 (2.00)

1. Lube oil pressure low
2. Engine overspeed
3. Voltage restraint overcurrent
4. Generator differential overcurrent
5. Start failure
6. Generator under frequency
7. DG trip/shutdown [any 5 of 7, 0.4 pts each]

REFERENCE

1. LP-2346, p-78,79 EO-2

K/A 064000K402 3.9/4.2
064000K402 ...(KA'S)

ANSWER 6.10 (3.00)

a. 1.greater than or equal to 91.7% [0.25]
   2.protects core from a DNBR 1.3 or less if a RCP is lost (or reduction in RCS flow) [0.50]

b. 1.greater than or equal to 36 percen. [0.25]
   2.assures that a minimum inventory of secondary cooling exists after a trip OR protects RCS from inadequate cooling and overpressurization [0.50]

c. 1.less than or equal to 2400 psia [0.25]
   2.to protect RCS from overpressurization due to loss of load that does not cause a Rx trip [0.50]

d. 1.less than or equal to 4.75 psig [0.25]
   2.to assure a Rx trip occurs concurrently with a SIAS [0.50]

REFERENCE

LP-2380-1, p-4,5 EO-9
TS 2.2.1, p - 2-3, 2-4, 2-5, B2-4, B2-5, B2-6

K/A 012000K402 3.9/4.3
012000K402 ...(KA'S)

ANSWER 6.11 (2.50)

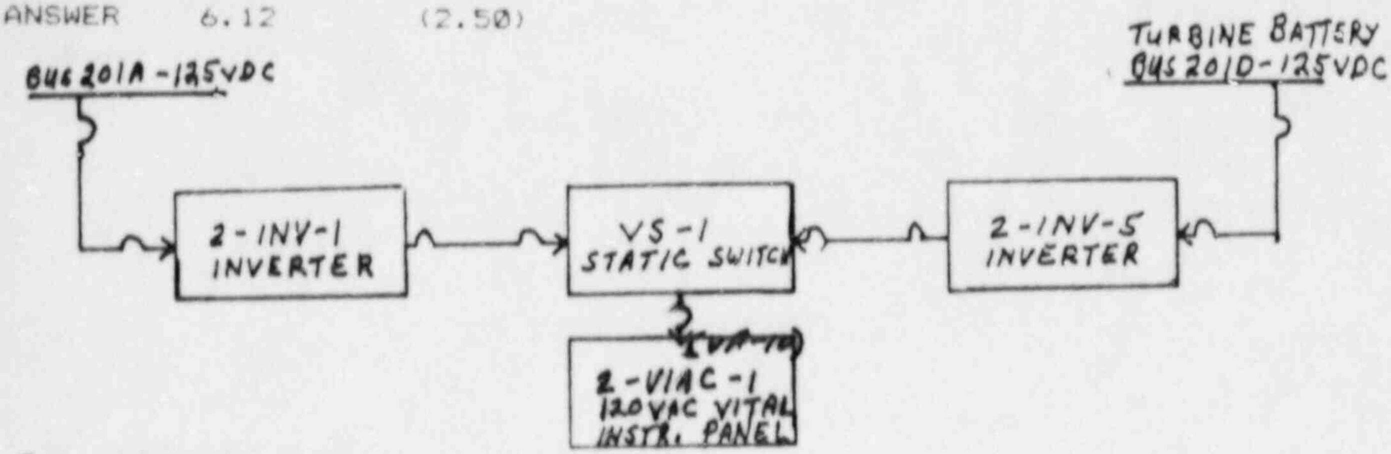
a. Radiation monitor increase or alarm, surge tank level increase, Sample results on RBCCW, or Temp. ↑ on components ~~[0.5 each]~~ [any 2 @ 0.5 each]

b. Letdown heat exchanger, RCPs, primary sample coolers [0.5 each]

REFERENCE

1. LP-2230A, p-7 & T-A EO-11
2. AOP 2568, p-3
K/A 008000K102 3.3/3.4
008000K104 3.3/3.3

ANSWER 6.12 (2.50)



See attached sketch [2.5]

REFERENCE

1. SD-2345, p - 3, f - 1,2
2. LP-2345, EO-2

K/A 062000K104 3.7/4.2
062000K104 ...(KA'S)

ANSWERS -- MILLSTONE 2 -88/07/20-PRELL, J.

ANSWER 7.01 (3.00)

a. Admin Limit = 2500 mrem/qtr [0.5]
2500 - 1500 = 1000 mrem [0.25]
1000 ÷ 200 = 5 hours [0.25]

b. High Radiation Area, Neutron Radiation Area, Contaminated Area, Containment, High Airborne Concentration, Noble Gas Exposure Area [any 4 @ 0.5 each]

~~5(N-18) = 30 rem (total lifetime limit allowed) [0.75]~~

~~30 - 9.6 = 20.4 = 20400 mrem left before she exceeds lifetime limit [0.75]~~

~~500 mrem/qtr, allowed female, expected pregnancy, with documented current quarterly exposure [0.5]~~

~~500 mrems - 200 mrems = 300 mrems left this quarter [0.5]~~

~~300 mrem is more limiting therefore 300 / 50mrems/hr = 6 hours [0.5]~~

REFERENCE

1. SHP-4902, P-7,8,9
2. SHP-4906, p-6,7

EO - M2-OP-LOIOJT-ADMIN-SHP-II.B
M2-OP-LOIOJT-ADMIN-SHP-II.A.1
II.C.1

K/A 194001K104 3.3/3.5
194001K104 ...(KA'S)

ANSWER 7.02 (2.90)

a. letdown flow control valves fail closed on loss of instrument air [0.5]

b. the backup charging pump cycles on and off [0.5]

c. 1. redundancy of instrument air compressors
2. station to instrument air cross-tie
3. Unit 1 to Unit 2 cross-tie
4. the use of flow limiting orifices throughout the air system
5. Excess flow check valves [any 4 @ 0.3 each]

d. 80psig [0.7]

REFERENCE

1. AOP2563, p-3,6 EO-1
2. LP 2332A/B, p-26

K/A 000065K303 2.9/3.4
000065K304 3.0/3.2
000065A103 2.9/3.1
000065A206 3.6/4.2
000065A103 000065A206 000065K303 000065K304 ...(KA'S)

ANSWER 7.03 (1.00)

Trip the reactor [0.5] RCP bleedoff temperature greater than 195 degrees F. [0.5]

REFERENCE

1. AOP2564, p-2,3 EO-1b,c,d;7

K/A 008000K102 3.3/3.4
008000K301 3.4/3.5
008000K102 008000K301 ...(KA'S)

ANSWER 7.04 (2.50)

a. it would result in an unmonitored radioactive release [1.0]

b. this minimizes the potential for the subsequent lifting of the SG safeties [1.0]

c. SGs are vital for RCS heat removal [0.5]

REFERENCE

1. AOP2569, p-2 EO-2
2. EOP2534, p-6,9 EO-3,5

K/A 000037K305 3.7/4.0
000037K307 4.2/4.4
000037K305 000037K307 ...(KA'S)

ANSWERS -- MILLSTONE 2 -88/07/20-PRELL, J.

ANSWER 7.05 (1.00)

Future ESAS signals will not be processed for that equipment until the ESAS modules have been reset. [1.0]

REFERENCE

1. AOP 2571, p-3,4 EO-1c,2

K/A 006000K413 4.1/4.3
006000K405 4.3/4.4
006000K418 3.3/3.8
006000K405 006000K413 006000K418 ...(KA'S)

ANSWER 7.06 (3.00)

a. 1. Depress the turbine trip button
2. Verify steam admission valves are closed
3. Verify the generator megawatts equal zero [0.5 each]

b. two charging pumps running [0.5]
18 minutes / stuck rod [0.5]
(# rods stuck - 1) X 18 = 54 minutes [0.5]

Note: if assume 1CP, 36 min/rod
or if assume 3 CP, 12 min/rod

REFERENCE

1. EOP2525, p-3,8 EO-4

K/A 000007K301 4.0/4.6
000024K301 4.1/4.4
000024K302 4.2/4.4
000007K301 000024K301 000024K302 ...(KA'S)

ANSWER 7.Ø7 (3.ØØ)

a. Two source range neutron flux monitors (wide range log NI) [Ø.5]
Each must provide a continuous visual indication in the control room [Ø.25] and one an audible indication inside containment. [Ø.25]

b. *The more limiting of* *[0.2]*
Keff </= Ø.95 [Ø.~~8~~4Ø]
Cb >/= 172Ø ppm [Ø.~~8~~4Ø]

c. 1. Ensures sufficient cooling capacity to remove decay heat [Ø.25] and maintain less than 14Ø degrees F. [Ø.25]
2. Ensures sufficient coolant circulation to minimize affects of a boron dilution incident [Ø.25] and prevent boron stratification. [Ø.25]

REFERENCE

1. TS 3.9.2 and 3.9.1, B3/4.9.8 EO-*Tech Specs Generic TPG 606*

K/A ØØØØ36SGØ3 2.9/3.8
ØØØØ36SGØ4 2.6/3.8
Ø33ØØØK3Ø2 Ø34ØØØA4Ø1 Ø34ØØØK6Ø2 ...(KA'S)

ANSWER 7.08 (3.00)

a. 1. Two or more CEAs have not inserted following a trip
2. An unanticipated reactor cooldown has been initiated
3. PDIL alarm is annunciated
4. While shutdown, an unexplained increase in reactivity is noted
[any 3, 0.4 each]

b. 1. Open BA pump discharge to charging pump suction valve(2CH-514) [0.3]
2. Start both BA pumps [0.3]
3. Close BA pump Recirc. valves (2-CH-510 and 2-CH-511) [0.3]
4. Start all available charging pumps [0.3]
OR
IF BA pumps fail to start
2a. then open BA gravity feed valves to charging pump suction (2-CH-508 and 2-CH-509) [0.3]
3a. then close VCT outlet valve (2-CH-501) [0.3]

REFERENCE

1. AOP2558, p-2,3 EO-1a, OJT

K/A 000024EK301 4.1/4.4
000024EK302 4.2/4.4
000024EK30 ...(KA'S)

ANSWERS -- MILLSTONE 2 -88/07/20-PRELL, J.

ANSWER 7.09 (3.00)

a. 1. Abnormal change in PZR level
   2. Decreasing PZR pressure
   3. Increasing containment pressure
   4. High containment radiation
   5. Unbalanced charging/letdown flows [any 4, 0.25 each]

b. 1. PZR level increases greater than expected using Aux. spray
   2. PZR level increases slower than expected for existing HPSI and charging flow
   3. Unheated thermocouples in the upperhead indicate saturated conditions
   4. RX vessel level is less than 100 percent [0.5 each]

REFERENCE

1. EOP2532, p-2,10,11 EO-1b,4c,6b

K/A 000011EK312 4.4/4.6
000011EA114 3.9/4.1
000011EA201 4.2/4.7
000011A114 000011A201 000011K312 ...(KA'S)

ANSWER 7.10 (1.60)

1. By the SD+TBS or atmosphere dump valves on the operating SG [0.25]

2. Control heat removal [0.3] by careful addition of feedwater to operating SG [0.25] ~~and by the steam dump and bypass or atmospheric dump valves on the operating SG. [0.25]~~

3. Control of RCS repressurization [0.3] via HPSI [0.25] and CVCS [0.25].

REFERENCE

1. EOP-2536, p - 13
2. LP-2536, EO - 10

K/A 000040K106 3.7/3.8
000040A106 4.0/4.1
000040SG03 3.1/3.9
000040A106 000040K106 000040SG03 ...(KA'S)

ANSWER 7.11 (1.00)

Prevents excessive thermal shock to the shutdown heat exchanger [1.0]

REFERENCE

1. AOP-2552, p - 18 EO-None Exists

K/A 004000K509 3.7/4.2
004000SG10 3.1/3.4
004010A401 3.6/3.1
004010K622 2.3/2.6
004010K403 3.1/3.6
004000K509 004000SG10 004010A401 004010K403 004010K622
...(KA'S)

ANSWER 8.01 (1.00)

a. 3
b. 1
c. 4
d. 2 [0.25 each]

REFERENCE

1. ACP-QA-2.06A, p-3,4 EO-5

K/A 194001K102 3.7/4.1
194001K102 ...(KA'S)

ANSWER 8.02 (1.00)

If verification would result in significant radiation exposure [1.0]

REFERENCE

1. ACP-QA-2.06B, p-10 EO-5e,6a

K/A 194001K102 3.7/4.1
194001K102 ...(KA'S)

ANSWER 8.03 (1.00)

The core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. [1.0]

REFERENCE

1. TS 3.9.6, p - 3/4 9-6, B3/4 9-2
2. SD-2303A, p - 12 EO-6

K/A 034000SG06 2.5/3.6
034000SG06 ...(KA'S)

ANSWER 8.04 (3.00)

a. Yes [0.25], Non-intent [0.5]
b. Yes [0.25], Non-intent [0.5]
c. No [0.25], Intent [0.5]
d. No [0.25], Intent [0.5]

REFERENCE

1. ACP-QA-3.02, p-5,6,27,30 EO-3,7,10

K/A 194001A102 4.1/3.9
194001A102 ...(KA'S)

ANSWER 8.05 (3.00)

LCOs indicate the lowest performance level of equipment required for safe operation of the plant [1.0]. If automatic actuation of protective equipment related to those variables having significant safety functions occurs prior to the LSSS, then the Safety Limits will not be exceeded [1.0]. If Safety Limits are not exceeded then fuel and RCS integrity will be maintained [1.0].

REFERENCE

1. 10CFR50.36(c)
2. TS EO-5

K/A 002020SG05 3.6/4.1
002020SG06 2.6/3.8
002020SG05 002020SG06 ...(KA'S)

ANSWER 8.06 (2.00)

a. Reqs. were NOT met [0.3] - The combined time interval for the last three consecutive surveillance intervals exceeded 3.25 times the specified surveillance. [0.7]

b. Reqs. were met [0.3] - Surveillance requirements do not have to be performed on inoperable equipment. [0.7]

REFERENCE

1. TS-4.02;4.03; 4.3.3.3
2. TPG 2400,2600/Generic "Surveillance Procedures"; EO - 1

K/A 016000SG05 2.9/3.5
016000SG05 ...(KA'S)

ANSWER 8.07 (2.00)

a. 1. terminate the discharge [0.5]
   2. classify the event [0.5]
   3. notify the NRC (within one hour) [0.5]

b. General emergency [0.5]

REFERENCE

1. ACP-6.03, p - 5
2. EPIP 4701, p - 3, F - 4701-2,4701-4 EO-1
3. EPIP 4112, p - 2

K/A 000068SG01 3.3/4.1
000068SG02 2.9/4.1
000068SG01 000068SG02 ...(KA'S)

ANSWER 8.08 (1.50)

a. True
b. False
c. ~~False~~ True [0.5 each]

REFERENCE

1. ACP-OA-10.05, p-4,7 EO-M2-OP-LOIOJT-ADMIN-ACP
II K.1.c
III A.6

K/A 194001A106 3.4/3.4
194001A106 ...(KA'S)

ANSWER 8.09 (1.50)

1. That no plant evolutions are planned which could change the radiological conditions in the area listed on the RWP. [0.5]

2. That HP will be notified whenever any plant evolution takes place that changes the radiological conditions in the area listed on the RWP. [0.5]

3. That the plant is not jeopardized by the work indicated on the RWP. [0.5]

REFERENCE

1. SHP-4912, p-4 EO-14a

K/A 194001K104 3.3/3.5
194001K104 ...(KA'S)

ANSWER 8.10 (2.50)

a. 5 [0.5]

b. 1 [0.5]

c. 2 hours [0.5]

d. 1. SS Staff Assistants Office ~~[0.5]~~ [any 2 @ 0.5 each]
2. Hot Shutdown Panel Storage Box ~~[0.5]~~
3. Turnout lockers outside CR
4. Fire lockers in Aux Bldy (@ 14.6 level near H.P.)

REFERENCE

1. AOP2579A,B,C,D,...; p - 3
2. LP2579A,B,C,D,...; p - 7 EO-2
3. TS 6.2.2, p - 6-1, 6-4
4. AOP 2559 p-3
K/A 194001K116 3.5/4.2
194001K116 ...(KA'S)

ANSWER 8.11 (2.00)

Provides assurance that these heaters can be energized during a loss of offsite power condition [1.0] in order to establish and maintain natural circulation ~~at HOT STANDBY.~~ ~~[1.0]~~ [0.5] and control RCS pressure [0.5]

REFERENCE

1. TS3.4.4.6, p - 3/4 4-4, B3/4 4-2 EO-18 (TS)
2. LP-2304A, p - 49 EO-14

K/A 010000SG05 3.2/3.8
010000K201 3.0/3.4
010000K301 3.8/3.9
010000K603 3.2/3.6
010000K201 010000K301 010000K603 010000SG05 ...(KA'S)

ANSWER 8.12 (1.50)

a. Ensure the operability of HPSI pump "B" [0.5]

b. Ensure all systems/components are returned to an operable condition [0.5] and verify that the data on the Surveillance Data Sheet is in compliance with the acceptance criteria. [0.5]

REFERENCE

1. ACP-9.02, p - 10,12 EO-M2-OP-LOIOIT-ADMIN-ACP II. 4
2. TS3.5.2a

K/A 006000SG01 4.1/4.3
006000SG05 3.5/4.2
006000SG01 006000SG05 ...(KA'S)

ANSWER 8.13 (1.00)

The SCO should NOT grant permission [0.50] because only licensed operators [0.25] or individuals in training under the direct supervision of a licensed operator can operate the controls [0.25].

REFERENCE

1. ACP-QA-6.01, p-9,10,11 EO-M2-OP-SRO-ADMIN-LICRESP-EO-6
2. 10CFR55.11; 10CFR50.54(i); 10CFR55.3
3. IN88-20

K/A 194001A103 2.5/3.4
194001A103 ...(KA'S)

ANSWER 8.14 (2.00)

1. All penetrations required to be closed during accident conditions are either:

   a. capable of being closed by an operable containment isolation valve system [0.5]

   b. closed by manual valves, blind flanges or deactivated automatic valves secured in their closed positions. [0.5]

2. the equipment hatch is closed and sealed [0.5]

4. the airlock is operable (pursuant to TS 3.6.1.3) [0.5]

REFERENCE

1. TS, p1-2
2. LP2313, p-11,12 EO-13a

K/A 000069A201 3.7/4.3
000069A201 ...(KA'S)

EQUATION SHEET

$f = ma$

$v = s/t$

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

$w = mg$

$s = v_o t + 1/2at^2$

$E = mC^2$

$a = (v_f - v_o)/t$

$KE = 1/2mv^2$

$v_f = v_o + at$

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

$PE = mgh$

$\omega = \theta/t$

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

$W = v\Delta P$

$\Delta E = 931\Delta m$

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

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

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

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

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

$\dot{Q} = UA(T_{ave} - T_{stm})$

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

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

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

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

$TVL = 1.3/\mu$

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

$HVL = 0.693/\mu$

$SUR = 26.06/T$

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

$T = 1.44\ DT$

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

$SUR = 26 \dfrac{\lambda_{eff}\,\rho}{\bar{\beta} - \rho}$

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

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

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

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

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

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

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

$\Sigma = N\sigma$

$I_1 d^2{}_1 = I_2 d^2{}_2$

$LMTD = \dfrac{\Delta T_2 - \Delta T_1}{\ln \Delta T_2/\Delta T_1}$

WATER PARAMETERS

1 gal. = 8.345 lbm

1 gal. = 3.78 liters

1 ft^3 = 7.48 gal.

Density = 62.4 lbm/ft^3

Density = 1 gm/cm^3

Heat of vaporization = 970 Btu/lbm

Heat of fusion = 144 Btu/lbm

1 Atm = 14.7 psi = 29.9 in. Hg

1 ft. H_2O = 0.4335 lbf/in^2

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

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

MISCELLANEOUS CONVERSIONS

1 Curie = 3.7×10^{10} dps

1 kg = 2.21 lbm

1 hp = 2.54×10^3 BTU/hr

1 Mw = 3.41×10^6 Btu/hr

1 Btu = 778 ft-lbf

1 inch = 2.54 cm

°F = (9/5°C) + 32

°C = 5/9 (°F - 32

1987

JANUARY

S	M	T	W	T	F	S
				1	2	3
4	5	6	7	8	9	10
11	12	13	14	15	16	17
18	19	20	21	22	23	24
25	26	27	28	29	30	31

FEBRUARY

1	2	3	4	5	6	7
8	9	10	11	12	13	14
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MARCH

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APRIL

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MAY

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JUNE

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JULY

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AUGUST

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SEPTEMBER

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OCTOBER

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NOVEMBER

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DECEMBER

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1988

JANUARY

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FEBRUARY

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MARCH

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APRIL

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MAY

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JUNE

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JULY

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AUGUST

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SEPTEMBER

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OCTOBER

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NOVEMBER

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DECEMBER

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1989

JANUARY

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FEBRUARY

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MARCH

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APRIL

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MAY

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JUNE

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JULY

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AUGUST

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SEPTEMBER

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OCTOBER

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NOVEMBER

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DECEMBER

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24	25	26	27	28	29	30
31						

CALENDAR YEARS

Stephen E. Scace
FORM APPROVED BY STATION SUPERINTENDENT

3/7/88
EFFECTIVE DATE

88-09
SORC MTG. NO.

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
		NOTE 1: This form does not duplicate the EAL tables (4701-1, 4701-2, 4701-3). This form provides information for incident classification concerning NRC and State of Connecticut reporting criteria.		
		NOTE 2: During all Immediate Notifications to the NRC, USE ONLY 10 CFR 50.72 Reporting Criteria. DO NOT USE State Posture Codes. (See also EPIP 4112-3)		
A.1 Immediate	50.72(a)(1)(i)	I. The declaration of any of the Emergency Classes specified in the approved Emergency Plan.		
		a. General Emergency (with major breach of containment).	ALPHA	
		b. General Emergency (without major breach of containment).	BRAVO	
		c. Site Area Emergency	CHARLIE-TWO	
		d. Alert	CHARLIE-ONE	
		e. Unusual Event (with an unplanned radioactive release).	DELTA-TWO	

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
A.1 Immediate	50.72(a)(1)(i) (continued)	f. Unusual Event (with no radioactive release).	DELTA-ONE	
		g. Radioactive material unaccounted for and which is in excess of 10CFR30.71, Schedule B as applicable in 10CFR20.402 and 10CFR20.10.	FOX	
		h. Radioactive material transportation accident. 1. In Connecticut outside a fixed facility.	GOLF	
		II. Non-Emergency Events - One Hour Reports		
	50.72(b)(1)(i)	• (A) The initiation of any nuclear plant shutdown required by the plant's Technical Specifications. (B) Any deviation from the plant's Technical Specifications authorized pursuant to 10CFR50.54(x).	ECHO*	See 10CFR50.73(a)(2)(i)
	50.72(b)(1)(ii)	• Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or results in the nuclear power plant being: (A) In an unanalyzed condition that significantly compromises plant safety; (B) In a condition that is outside the design basis of the plant; or (C) In a condition not covered by the plant's operating and emergency procedures.	ECHO*	See 10CFR50.73(a)(2)(ii)

*Unless reported as higher classification (i.e. ALPHA, BRAVO, CHARLIE-ONE, CHARLIE-TWO DELTA-ONE, DELTA-TWO)

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA		EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
A.1 Immediate	50.72(b)(1)(iii)	•	Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.	ECHO*	See 10CFR50.73(a)(2)(iii)
	50.72(b)(1)(iv)	•	Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the Reactor Coolant System as a result of a valid signal.	DELTA-ONE*	See 10CFR50.73(a)(2)(iv)
	50.72(b)(1)(v)	•	Any event that results in a major loss of emergency assessment capability, offsite response capability, or communications capability (e.g., significant portion of Control Room indication, Emergency Notification System, or Offsite Notification system).	DELTA-ONE*	No Comparable 10CFR50.73 Requirement. NOTE: The loss of a system required to support the EOF under accident conditions is not reportable (Ref. NP-S-4832)
	50.72(b)(1)(vi)	•	Any event that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.	ECHO*	See 10CFR50.73(a)(2)(x)

*Unless reported as higher classification (i.e. ALPHA, BRAVO, CHARLIE-ONE, CHARLIE-TWO DELTA-ONE, DELTA-TWO)

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
A.1 Immediate		III. Non Emergency Events - Four Hour Reports NOTE: ALTHOUGH THESE ARE CALLED FOUR HOUR REPORTS PER 10 CFR 50.72 THEY SHALL BE REPORTED TO THE NRC WITHIN ONE HOUR AND IDENTIFIED AS FOUR HOUR REPORTS TO THE NRC.		
	50.72(b)(2)(i)	• Any event found while the reactor is shutdown, that had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety.	ECHO*	See 10CFR50.73(a)(2)(ii)
	50.72(b)(2)(ii)	• Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that results from and is part of the preplanned sequence during testing or reactor operation need not be reported.	ECHO*	See 10CFR50.73(a)(2)(iv)

*Unless reported as higher classification
(i.e. ALPHA, BRAVO, CHARLIE-ONE, CHARLIE-TWO
DELTA-ONE, DELTA-TWO)

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
A.1 Immediate	50.72(b)(2)(iii)	• Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition, (B) Remove residual heat, (C) Control the release of radioactive material, or (D) Mitigate the consequences of an accident.	ECHO*	See 10CFR50.73(a)(2)(v) and 10CFR50.73(a)(2)(vi) Examples of (B): 1) Unplanned loss of all Fuel Pool Cooling that can not be restored within one hour. 2) Unplanned loss of Shutdown Cooling (SDC) can not be restored within one hour, when SDC is required to be in service

*Unless reported as higher classification (i.e. ALPHA, BRAVO, CHARLIE-ONE, CHARLIE-TWO DELTA-ONE, DELTA-TWO)

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
A.1 Immediate	50.72(b)(2)(iv) (continued)	• (A) Any airborne radioactive release that exceeds 2 times the applicable concentrations of the limits specified in Appendix B, Table II of 10CFR Part 20 in unrestricted areas, when averaged over a time period of one hour. (B) Any liquid effluent release that exceeds 2 times the limiting combined Maximum Permissible Concentration (MPC) (See Note 1 of Appendix B to 10CFR Part 20) at the point entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour. (Immediate notifications made under this paragraph also satisfy the requirements of paragraphs (a)(2) and (b)(2) of §20.403 of 10CFR, Part 20).	DELTA-TWO*	10CFR20.403(a)(2). The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in App. B, Table II of 10CFR, Part 20. See 10CFR50.73(a)(2)(viii) and 10CFR50.73(a)(2)(ix). 10CFR20.403(b)(2). The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in App. B, Table II of 10CFR Part 20.

*Unless reported as higher classification (i.e. ALPHA, BRAVO, CHARLIE-ONE, CHARLIE-TWO DELTA-ONE, DELTA-TWO)

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
A.1 Immediate	50.72(b)(2)(v) (continued)	• Any event requiring the transport of a radioactivity contaminated person to an offsite facility for treatment.	DELTA-ONE*	No Comparable 10CFR50.73 Requirement.
	50.72(b)(2)(vi)	• Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactive contaminated materials.	ECHO*	No Comparable 10CFR50.73 Requirement. For Environmental Event, see EPIP 4503.
Public Tailorment		IV. Reports Required by State and/or Local Governments ~~NOTE: AUTHORITIES NOTIFIED TO THE NRC WITHIN ONE HOUR~~, IDENTIFY AS FOUR HOUR REPORTS TO THE NRC.		
	50.72(b)(2)(vi)	• Minor occurrences of general interest and no radioactive release. a. Incidents that require outside assistance; such as: 1. Fire (NOTE: If fire lasts longer than 10 minutes within a Unit. Refer to EAL's.) 2. Civil disturbances. 3. Bomb threats. 4. Oil Spills that contaminate coastal water	ECHO*	No Comparable 10CFR50.73 Requirement. For Oil Spills, see EPIP 4112-2.

*Unless reported as higher classification (i.e. ALPHA, BRAVO, CHARLIE-ONE, CHARLIE-TWO DELTA-ONE, DELTA-TWO)

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
C Public Interest	50.72(b)(2)(vi) (continued)	5. Railroad Car Derailment/Chlorine.		See applicable EPIP Procedures for additional actions.
		b. Derating caused by regulatory action.		
		c. Any unscheduled shutdown estimated to last more than 48 hours.		
		d. Derating (greater than 50 percent) caused by equipment malfunction lasting more than 72 hours.		
		e. Scheduled shutdown for testing, maintenance, or refueling expected to last more than 72 hours and confirmed by NEPEX/CONVEX.	ECHO*	No Comparable 10CFR50.73 Requirement.
		f. An environmental condition which may be of general interest (fish kill or entrapments above normal environmental sample analysis but not exceeding Tech. Specs., etc.)		
	50.72(b)(2)(vi)	• Any incident involving licensed material (by product, source, or special nuclear material) which may have caused or threatens to cause:	ECHO*	10CFR20.403(a)(1), (3), and (4) Requirements.
		a. Exposure of the whole body of any individual of 25 Rems or more of radiation; exposure of the skin of the whole body of any individual of 150 Rems or more of radiation; or exposure of feet, ankles, hands, or forearms of any individual of 375 Rems or more of radiation.		

*Unless reported as higher classification
(i.e. ALPHA, BRAVO, CHARLIE-ONE, CHARLIE-TWO
DELTA-ONE, DELTA-TWO)

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
C Public Interest	50.72(b)(2)(vi) (continued)	b. A loss of one working week or more of the operation of any facilities affected. c. Damage to any property in excess of $200,000.		
	50.72(b)(2)(vi)	• Any incident involving licensed material (by product, source, or special nuclear material) which may have caused or threatens to cause: a. Exposure of the whole body of any individual of 5 Rems or more of radiation; exposure of the skin of the whole body of any individual of 30 Rems or more of radiation; or exposure of feet, ankles, hands, or forearms of any individual of 75 Rems or more of radiation. b. A loss of one day or more of the operation of any facilities affected. c. Damage to any property in excess of $2,000.	ECHO*	10CFR20.403(a)(1), (3), and (4) Requirements.

*Unless reported as higher classification (i.e. ALPHA, BRAVO, CHARLIE-ONE, CHARLIE-TWO DELTA-ONE, DELTA-TWO)

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
B 30 Day LER		V. Events Requiring 30 Day Report to NRC NOTE: THE REQUIRED 30 DAY REPORT MAY ALSO REQUIRE AN IMMEDIATE REPORT UNDER 10CFR50.72 (SEE A.1 ABOVE).		
	50.73(a)(2)(i)	• (A) The completion of any nuclear plant shutdown required by the plant's Technical Specifications; or (B) Any operation or condition prohibited by the plant's Technical Specifications; or (C) Any deviation from the plant's Technical Specifications authorized pursuant to 10CFR50.54(X).	N/A N/A	See 10CFR50.72(b)(1)(i) for Immediate Reporting Requirements Example of (B): Missed Surveillances
	50.73(a)(2)(ii)	• Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or that resulted in the nuclear power plant being: (A) In an unanalyzed condition that significantly compromises plant safety; (B) In a condition that was outside the design basis of the plant; or (C) In a condition not covered by the plant's operating and emergency procedures.	N/A	See 10CFR50.72(b)(1)(ii) or 10CFR50.72(b)(2)(i) for Immediate Reporting Requirements

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA		EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
B 30 Day LER	50.73(a)(2)(iii) (continued)	•	Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.	N/A	See 10CFR50.72(b)(1)(iii) for Immediate Reporting Requirements
	50.73(a)(2)(iv)	•	Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported.	N/A	See 10CFR50.72(b)(1)(iv) or 10CFR50.72(b)(2)(ii) for Immediate Reporting Requirements
	50.73(a)(2)(v)	•	Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition, (B) Remove residual heat, (C) Control the release of radioactive material, or (D) Mitigate the consequences of an accident.	N/A	See 10CFR50.72(b)(2)(iii) for Immediate Reporting Requirements

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REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
B 30 Day LER	50.73(a)(2)(vi) (continued)	• Events covered in paragraph 50.73(a)(2)(v) above may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.	N/A	See 10CFR50.72(b)(2)(iii) for Immediate Reporting Requirements
	50.73(a)(2)(vii)	• Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat; (C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident.	N/A	No Comparable 10CFR50.72 Reporting Requirements.

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
B 30 Day LER	50.73(a)(2)(viii) (continued)	• (A) Any airborne radioactivity release that exceeds 2 times the applicable concentrations of the limits specified in Appendix B, Table II of 10CFR Part 20 in unrestricted areas, when averaged over a time period of one hour. (B) Any liquid effluent release that exceed 2 times the limiting combined Maximum Permissible Concentration (MPC) (See Note 1 of Appendix B to 10CFR Part 20) at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour.	N/A	See 10CFR50.72(b)(2)(iv) for Immediate Reporting Requirements
	50.73(a)(2)(ix)	• Reports submitted to the Commission in accordance with Paragraph (A)(2)(viii) above. Also meet the effluent release reporting requirements of Paragraph 20.405(A)(5) of 10CFR Part 20.	N/A	See 10CFR50.72(b)(2)(iv) for Immediate Reporting Requirements
	50.73(a)(2)(x)	• Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.	N/A	See 10CFR50.72(b)(1)(vi) for Immediate Reporting Requirements

REPORTABILITY

PIR CATEGORY	NRC (10 CFR) REPORTING CRITERIA	EVENT DESCRIPTION	STATE POSTURE CODE	COMMENTS, NOTES, EXAMPLES, AND ADDITIONAL GUIDANCE
D Not Reportable to NRC	N/A	VI. Non-Reportable Events Requiring Further Plant Review/Investigation and Documentation • Certain plant incidents or occurrences that are not reportable to the NRC, but require action and review by the plant staff to correct underlying causes and prevent recurrences. a. Any deviations from Section 6 (Administrative Controls) of Technical Specifications, unless identified in Sections I, II, III, IV, or V above. b. Incidents associated with systems and equipment not addressed by Technical Specifications.	N/A	

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