OPERATOR LICENSING EXAMINATION REPORT

Facility Licensee: Louisiana Power and Light Co.
142 Delaronde Street
New Orleans, Louisiana 70174

Facility Docket No.: 50-382

Operator licensing examinations at Waterford 3 Steam Electric Station near Taft, Louisiana

Chief Examiner: John L Pellet 7-16-84
J. L. Pellet, Examiner Date

Approved by: R. A. Cooley 7-16-84
R. A. Cooley, Section Chief Date

Summary

Operator Licensing Examinations conducted June 26 - 28, 1984

Operator licensing examinations were conducted at Waterford 3 during the week of June 24, 1984 for five senior reactor operator license candidates. All five candidates passed the examinations.

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WATERFORD 3 EXAMINATION REPORT

Details

1. <u>Persons Examined</u>

   SRO (instant) Candidates:
   Five SRO license candidates were examined on written and oral exams.

2. <u>Examiners:</u>

   J. L. Pellet, Chief Examiner, NRC
   R. Smith, NRC
   M. Robichaux, Proctor, NRC

3. <u>Examination Report</u>

   This examination report is composed of the sections listed below.

   A. Examination Review Meeting Comment Resolution

   B. Exit Meeting Summary

   C. Generic Comments

   D. Examination Master Copy (SRO Questions and Answers)

   Performance results for individual candidates are not included in this report because, as noted in the transmittal letter attached, examination reports are placed in NRC Public Document Room as a matter of course.

A. <u>Examination Review Meeting Comment Resolution</u>

In general, editorial comments or changes made during the exam, the exam review, or subsequent grading reviews are not addressed by this resolution section. This section reflects resolution of substantive comments made during the exam review. The modifications discussed below are included in the master exam key which is provided elsewhere in this report as are all other changes mentioned above but not discussed herein. The following personnel were present for the exam review:

WATERFORD 3 EXAMINATION REPORT

NRC	UTILITY
J. Pellet	D. Clark
R. Smith	M. James
M. Robichaux	J. Rein
	S. Whitley

COMMENTS

(1) 5.1 Answer value should be 0.25 for name & 0.75 for expl.
Resp.: ACCEPT.

(2) 5.4 Accept curve in place of text if correct and complete.
Resp.: ACCEPT.

(3) 5.9.b Correct answer is "slightly supercritical."
Resp.: ACCEPT.

(4) 5.11. For parts a & b delete the phrase "boiling in the core." For part d evaluate assumptions before grading answer.
Resp.: ACCEPT.

(5) 5.12 For f and g swap fuel centerline directions (e.g., incr. for decr. & vice versa)
Resp.: ACCEPT.

(6) 6.1 Answer c is two answers and for answer j "automatic" should not be required.
Resp.: ACCEPT.

(7) 6.2 Modify key to show: safety log channel only uses center detector, linear range is 1-200 or 0-200 per p.2 of ref., and control channel range is 1-125 or 0-125 per fig. 3 of ref.
Resp.: ACCEPT.

(8) 6.3 Per fig. 2 of the ref. add "CEDM cooling, RAB HVAC, Contain. fan coolers, Boron Mgmt. Sys., and LWMS" as correct answers.
Resp.: ACCEPT.

(9) 6.4 Delete "reducing shine."
Resp.: ACCEPT.

(10) 6.5 For ans. 1 change "ARSS" to "ARRS" and note typo "redio... ." For ans. 4 temperatures should not be required.
Resp.: ACCEPT.

(11) 6.7 For ans. 1 delete "opens emergency air intake" and add "initiates on SIAS." For ans. 2 change "remains" to "interlocked." Both per CWD 1153-S.
Resp.: ACCEPT.

WATERFORD 3 EXAMINATION REPORT

(12) 6.11 Add CSS and containment cooling sys. as correct ans.
Resp.: ACCEPT.

(13) 7.1 Accept "mode 6" for "head off/not tensioned," "mode 5" for "LT 200^{0}," and "modes 1-4" for "GT 200^{0}." Also ans. in part 6 shoulb be SDM or B_{conc} not with B_{conc}.
Resp.: ACCEPT.

(14) 7.1 Add ans. per OP-901-004, "Control Room Evacuation,": requires emergency boration if Rx tripped from outside CR or if GT 1 CEA stuck out (Also, per OP-902-00).
Resp.: ACCEPT.

(15) 7.2 Accept either 17.4 or 17.1 cont. press.
Resp.: ACCEPT.

(16) 7.3 Accept alarms for partial credit in place of indications.
Resp.: ACCEPT.

(17) 7.8 Ans. 5 should include 1050 # on SDBCS and ans. 6 should insert "(70% WR)" in front of "EFW."
Resp.: ACCEPT.

(18) 7.9 Accept ans. from rev. 1 of procedure also.
Resp.: ACCEPT. Key modified to show both answer sets.

(19) 7.10.b Replace "NOS" with "CRS & SS"
Resp.: ACCEPT.

(20) 7.11 Correct ans. is "Keep pump motor within nameplate data," per CE letter C-CE-7050 dated March 19, 1981.
Resp.: ACCEPT. Full credit for either answer.

(21) 8.2 Accept additional answers outside scope with no penalty.
Resp.: ACCEPT.

(22) 8.5 Note 720 gpd = 0.5 gpm.
Resp.: ACCEPT.

(23) 8.6 Question intent is not clear - a variety of answers may be received.
Resp.: ACCEPT. During exam announcement was made to answer as "When and how per T. S."

(24) 8.8 Question is poor because it requires rote memorization of specific numbers in T. S. without testing knowledge or understanding needed by an operator. If question is retained it should be graded with a wide tolerance band.

Resp.: NO. Question is appropriate and will be graded +/- 5%. Question review during grading indicates that no pass/fail decision resulted from this question.

(25) 8.9 Accept "peak LHR/LPD" for "peak LHGR."
Resp.: ACCEPT.

(26) 8.12 Times should not be required and "secondary calorimetric" should be added.
Resp.: ACCEPT.

B. EXIT MEETING SUMMARY

At the conclusion of the site visit examiners met with representatives of the plant staff to discuss the results of the examinations. The following personnel were present for the exit interview:

NRC	UTILITY	
J. Pellet	J. Somsel	L. Storz
R. Smith	M. Bourgeois	D. Packer
T. Flippo	D. Clark	J. Woods
M. Robichaux	C. Toth	R. Gimelli
	R. Barkhurst	

Mr. Smith starrted the discussion by stating that preliminary results indicated no candidates were not clear passes (all were clear passes). A general discussion then took place on the areas described below.

1. The results above are based on a preliminary review of oral results only. They represent the best judgement available at the time of the exit meeting.

2. In general, NRC attempts to return final exam results to the facility and individuals concerned within 30 days. Unofficial results should be available before that time.

3. The EOP's were much improved since the previous visit.

4. The plant looks much cleaner than during the last visit.

5. Procedures in general assume corrective actions are successful and do not guide the operator to alternate success paths. For example, the steam tube rupture procedure requires the operator to depressurize the RCS with auxiliary pressurizer sprays but does not give any alternate path if difficulty is encountered.

6. The loss of off-site power procedure assumes a reactor trip has occurred. However, if the reactor power cutback and steam dump systems perform as designed it would appear that a complete loss of off-site power could occur which would leave the plant on station (i.e., hotel or house) loads only. If the plant is designed to withstand a transient without tripping the reactor or turbine then procedures and operator training should not assume a trip occurs.

7. Drawings in the control room did not match procedures. Specifically, operator actions following a Recirculation Actuation Signal (RAS) did not agree with P&ID indication of automatic valve realignments.

C. GENERIC COMMENTS

During grading of the written examinations, the areas of generic weakness described below were identified based on the responses of the candidates as a whole. These areas should be regarded as potential areas of concern for the training program, but note that they were generated by an informal review of the exams, and, as such, may not reflect an accurate picture of the training program as a whole.

(1) Basic Theory
None identified

(2) Systems Design
- (a) HVAC control system design.
- (b) Steam generator blowdown flow rates.
- (c) Physical factors affecting hydrogen combustion inside containment.

(3) Procedures
- (a) Controls required by 10CFR20 for a high radiation area.

(4) Administrative Procedures
- (a) Power limits for inoperable secondary safety valves.

D. EXAMINATION MASTER COPY (SRO QUESTIONS AND ANSWERS)

The master examination, consisting of SRO questions, answers, and references is included in this report.

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

Facility: Waterford 3

Reactor Type: CE-PWR

Date Administered: 6-26-84

Examiner: R. SMITH

Candidate: ____________

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheets on top of 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	25			5. Theory of Nuclear Power Plant Operations, Fluids, and Thermodynamics
25	25			6. Plant Systems Design, Control and Instrumentation
25	25			7. Procedures - Normal, Abnormal, Emergency, and Radiological Control
25	25			8. Administrative Procedures Conditions, and Limitations
100				TOTALS

Final Grade ________ %

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

____
Candidate's Signature

5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS (25 pts)

5.1 Give the method and explain how a gamma photon interacts with matter to transfer energy. (3 methods) (3.00)

5.2 Describe the basic mechanism of natural circulation. (1.50)

5.3 The relative worth of a CEA is dependent on the neutron flux reaching it. List and <u>explain</u> four (4) factors or conditions which affect the flux reaching a particular CEA. (3.00)

5.4 What happens to the moderator temperature coefficient as boron concentration is increased? Explain why. (2.50)

5.5 Explain the importance of delayed neutrons in light water reactor technology. (1.00)

5.6 For a constant power level, why does fuel rod centerline temperature decrease over core life? (1.00)

5.7 Describe how and why centrifugal pump discharge flow (gpm) changes when: <u>(2.00)</u>

A. Throttling the suction valve (0.50)

B. Decreasing the fluid temperature (0.50)

C. Throttling the discharge valve, and (0.50)

D. Increasing pump speed. (0.50)

5.8 The value of the core delayed neutron fraction changes between the beginning of core life and the end of core life.

a. Does this value increase or decrease? (0.50)

b. Briefly explain why the two values are different. (1.50)

5.9 During a routine startup the reactor is subcritical with a stable count rate of 200 CPS on all source range instruments and the shutdown groups are fully withdrawn with a Keff of 0.95. The operator withdraws control banks until the count rate is 400 CPS then stops rod motion.

a. What will be the new Keff? Show all work. (1.00)

b. What will happen if the same amount of +reactivity is added again? Show all work. (1.00)

5.10 The core is operating in the nucleate boiling region. Reactor coolant pressure is increased. What effect does this have on heat transfer at the clad/coolant interface? Explain. (1.00)

5.11 After operating in natural circulation for 2 hours, a complete loss of natural circulation flow occurs. How will the following parameters change (increase, decrease, or remain the same)? Briefly explain your answer. (assume no further operator action) (3.00)

a. Core delta T

b. Core thermocouple temperature

c. Steam generator pressure

d. Reactor coolant system pressure

5.12 Provide FIVE physical factors, that change over core life, which affect the heat transfer capability of the fuel and the full power center line temperature. Indicate whether each factor will tend to INCREASE or DECREASE center line temperature over the life of the core. (3.00)

6. Plant Systems Design

6.1 List nine (9) of the twelve (12) functions that are provided by the Chemical and Volume Control System. (3.00)

6.2 Describe the eight (8) channels of excore nuclear instrumentation. Include number of detectors, type of detectors, and the range of each detector. Also explain the principal of power level detection for each detector. (12 answers) (4.00)

6.3 List six (6) systems which are physically connected to the Component Cooling Water System and cooled by it. (2.00)

6.4 Explain how the Containment Spray System can limit offsite dose or radiation post-accident. (2.00)

6.5 Give three (3) of the four (4) subsystems of the Containment Cooling and Ventilation System and explain the principal function of each. (3.00)

6.6 Give two (2) methods of removing or limiting the post-accident hydrogen concentration inside containment. (1.50)

6.7 Name the two (2) of the three (3) off-normal operating modes of the Control Room HVAC system, explain their major differences from normal, and state one signal which would initiate each. (3.00)

6.8 Give the maximum <u>and</u> normal blowdown rates in percentage of the maximum steaming rate for the Steam Generator Blowdown System. (1.00)

6.9 Place the list below in the proper sequence along a feedwater line starting at the main condenser and continuing downstream to a steam generator. (2.00)

a. Main Feedwater Flow Control Valve
b. High Pressure Feedwater Heaters
c. Main Feedwater Isolation Valve
d. Low Pressure Feedwater Heaters
e. Emergency Feedwater Tee
f. Condensate Pump
g. Steam Generator Feedwater Pump
h. Gland Steam Condenser

6.10 Explain how the presence of steam in a post-accident containment environment can affect air-hydrogen combustion. (2.00)

6.11 List three (3) Engineered Safety Features Sytems which provide direct support to the containment. (1.50)

7. PROCEDURES - NORMAL - ABNORMAL EMERGENCY AND RADIOLOGICAL

7.1 List conditions or events that require emergency boration. (4 answers required for full credit) (2.00)

7.2 List the immediate actions that must be taken for a reactor trip. (16 actions total-12 required for full credit) (4.00)

7.3 List the symptoms of a loss of coolant accident. (10 answers - 8 for full credit) (4.00)

7.4 List the precautions and controls that must be provided per 10 CFR 20 at each entrance or access point to a HIGH RADIATION AREA. (3 answers required) (1.50)

7.5 Provide immediate action that must be taken if two or more seals fail on a reactor coolant pump. (2 actions) (1.00)

7.6 List the actions that must be taken immediately to initiate emergency boration. (7 answers, 5 required for full credit) (2.50)

7.7 Provide the automatic actions that occur when there is a loss of charging flow. (4 answers) (2.00)

7.8. List the immediate actions that must be taken following a turbine trip. (8 steps, 6 required) (3.00)

7.9 List the immediate actions that must be taken on an inadvertant safety injection actuation. (2 steps required) (1.00)

7.10 (a) What is the purpose of a danger tag? (1.00)

(b) Who may authorize the installation and removal of danger tags? (1.00)

(c) What is the purpose of a caution tag? (1.00)

7.11 What is the reason reactor coolant pump 2A must not be operated, with one of more pumps at temperatures below 340°F. (1.00)

8. ADMINISTRATIVE PROCEDURES, CONTROLS, AND LIMITATIONS (25 pts)

8.1 List the conditions that must be met to ensure containment integrity per the technical specifications. (5 answers required for full credit) (3.0)

8.2 List the position, number of individuals required for minimum shift crew composition while in modes 1 (one) through 4 (four). (1 point for positions, 1 point for numbers). (2.00)

8.3 Who may be designated to assume the control room function in the absence of the shift supervisor while in mode. (1 (one) through 4 (four)? (1.00)

8.4 Fill in the temperature change rate blanks for the following conditions. (1.50)

a. A maximum heatup rate of _____ per hour with Reactor Coolant System cold leg temperature less than 200°F.

b. A maximum heatup rate of _____ per hour with Reactor Coolant System cold leg temperature greater than 200°F and less than or equal to 345°F.

c. A maximum heatup rate of _____ per hour with Reactor Coolant System cold leg temperature greater than 345°F.

d. A maximum cooldown rate of _____ per hour with Reactor Coolant System cold leg temperature less than 135 °F.

e. A maximum cooldown rate of _____ per hour with Reactor Coolant System cold leg temperature greater than or equal to 135°F and less than or equal to 180°F.

f. A maximum cooldown rate of _____ per hour with Reactor Coolant System cold leg temperature greater than 180°F.

8.5 List the limits of reactor coolant system leakage. (5 answers, 4 required for full credit.) (2.00)

8.6 List the requirements that must be met for issuing a temporary change to a procedure. (2 answers, 1 point each.) (2.00)

8.7 List the actions that must be taken if a safety limit is violated. (4 answers, 1 answer has 3 parts worth 1.0, other 3 at 0.50) (2.50)

8.8 List the maximum power for 1, 2, and 3, and 4 inoperable safety valves on any operationg steam generator per the technical specification. (4 answers, 0.25 each) (2.00)

8.9 Fill in the blanks in the following three (3) questions relating to the Safety Limits at Wateford.

a. The ______________ of the reactor core shall be maintained greater than or equal to ____________. (1.0)

b. The ______________ of the fuel shall be maintained less than or equal to ______________ KW/ft. (1.0)

c. The ______________ pressure shall not exceed ____________ psia. (1.0)

8.10 Fill in the blanks in the following three (3) questions relating to Reactor Coolant Pump (RCP) operation at Waterford.

a. The maximum RCP operating time without CCW flow is ______________ minutes. (0.5)

b. If CCW flow can be restored within ____________ minutes, then the pump may be restarted. (0.5)

c. With the motor at operating temperature, do not attempt more than ____________ start(s) at __________ minute intervals. (1.0)

8.11 Explain the two Technical Specification BASES for the control rod transient insertion limit. (2.0)

8.12 List four (4) parameters which must be manually calculated when Core Operating Limit Supervisory System (COLSS) is out of service. (2.0)

$$SUR = \frac{26.06}{\tau}$$

$$CR_2 = \frac{S}{1-Keff_2}$$

$$\frac{CR_1}{CR_2} = \frac{1-Keff_1}{1-Keff_2}$$

$\ell^* \approx 1X10^{-5}$ sec

$$C_1\,(1-K_1) = C_2\,(1-K_2)$$

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

$$\Delta\rho = \frac{K_2 - K_1}{K_2\;K_1}$$

1 gal = 3.785 ℓ

1 in = 2.54 cm

1 Kg = 2.2 lbs

1 gm/cm^3 = 62.4 lbs/Ft^3

latent Heat of Vaporization 970 BTU/lb

latent Heat of Fusion 144 BTU/lb

3.413 BTU/hr = 1 Watt/hr

1 HP - 746 Watts

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

Cp = BTU/lb - ^oF

h = BTU/lb

$$SCR = \frac{S}{1-Keff}$$

$$l_1 d_2^2 = l_2 d_1^2$$

$$\tau = \frac{\ell^* + (\beta - \partial K)\bar{\tau}}{\alpha K}$$

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

Facility: Waterford 3

Reactor Type: CE-PWR

Date Administered: 6-26-84

Examiner: R. SMITH

Candidate: ____________

INSTRUCTIONS TO CANDIDATE:

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25	25			5. Theory of Nuclear Power Plant Operations, Fluids, and Thermodynamics
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25	25			8. Administrative Procedures Conditions, and Limitations
100				TOTALS

Final Grade ________ %

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

____
Candidate's Signature

THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS

5.1 Give the method and explain how a gamma photon interacts with matter to transfer energy. (3 methods) (3.00)

ANS: (1.00 ea)

Photoelectric effect: interaction with an atom by imparting the total energy to eject an orbital electron.

Compton effect: interaction with an orbital electron where part of the energy ejects an electron from the atom. The remaining energy is a photon of less energy.

Pair production: a photon of equal or greater energy than 1.02 Mev. passes near a nucleus of an atom causing emission of a positron and a negatron. The additional energy greater than 1.02 Mev. is transferred to the kinetic energy of the two particles.

REF: General Reactor Theory

0.25 name, 0.75 explanation - 0.25 ea. for:
what it interacts with, how much, and effects.

5.2 Describe the basic mechanisms of natural circulation. (1.50)

ANS: (0.50 ea)

The basic mechanism of natural circulation is due to the elevation of the reactor core and the steam generators, and the different water densities in the hot and cold legs. The different density is caused by different temperatures and the elevation of the steam generators and the reactor core is the main driving force for natural circulation.

REF: Basic Thermodynamics

5.3 The relative worth of a CEA is dependent on the neutron flux reaching it. List and explain four (4) factors or conditions which affect the flux reaching a particular CEA. (3.00)

ANS:

(Any 4)

A. Temperature - Neutrons travel longer at higher temperature, therefore, higher probability of capture. Worth higher.

B. Boron concentration - As concentration goes up fewer neutrons reach the rods. Worth lower.

C. Fission product poisons - As poison concentration increase fewer neutrons reach the rods. Worth lower.

D. Rod shadowing - When adjacent rods are inserted, the existing rod sees a lower flux. Worth lower.

E. Radial position - Rods near the center of the core see higher flux than peripheral rods. Worth higher.

F. Axial position - The farther a rod is inserted into the core the greater the flux it is exposed to. Worth higher.

REF: Std reactor theory and core characteristics.

Question value is 3 pts., 0.25 for factor/condition, and 0.5 for explanation.

5.4 What happens to the moderator temperature coefficient as boron concentration is increased? Explain why. (2.50)

ANS:

The MTC become less negative and can eventually become positive. (1.0)

As the moderator heats up its expansion removes boron from (1.5) the core. This introduces positive reactivity because neutron absorption has been reduced. The expansion of the moderator alone introduces negative reactivity because there is less neutron thermalization in the core. As boron concentration increases, it's positive reactivity effect increases until it may become greater than the negative contribution due to reduced thermalization.

ACCEPT curve if complete and correct with appropriate explanation.

REF: Nuclear Fundamentals

5.5 Explain the importance of delayed neutrons in light water reactor technology. (1.00)

ANS:

Greatly extend average neutron generation time - allow stable control by increasing period.

REF: Nuclear Reactor Theory

5.6 For a constant power level, why does fuel rod centerline temperature decrease over core life? (1.00)

ANS:

As fuel is depleted in the rods fission product gases build up causing the fuel pellets to swell. The swelling of the pellets reduces the voids between the fuel and the cladding thereby improving heat transfer.

Also, clad creed further reduces gas gap.

(2 ans. @ 0.5 ea.)

REF: Reactor Theory

5.7 Describe how and why centrifugal pump discharge flow (gpm) changes when: (2.00)

A. Throttling the suction valve (0.50)

B. Decreasing the fluid temperature (0.50)

C. Throttling the discharge valve, and (0.50)

D. Increasing pump speed. (0.50)

ANS:

A. Flow decreases due to decreased pump inlet pressure.

B. No change in volumetric flow rate (mass flow and pump work increase) because the pump is a volumetric device.

C. Flow decreases as the system resistance (discharge pressure) increases.

D. Flow increases with speed (until onset of cavitation). (0.2 change; 0.3 explanation)

REF: Basic Thermo

5.8 The value of the core delayed neutron fraction changes between the beginning of core life and the end of core life.

a. Does this value increase or decrease? (0.50)

b. Briefly explain why the two values are different. (1.50)

ANS:

a. <u>Decrease</u> (0.50)

b. About 30%-45% of the fissions are caused by Pu-239 at EOL. Since the delayed neutron fraction for Pu-239 is considerably less than that for U-235, the core delayed neutron fraction decreases with an increase in burnup. (accept similar) (1.50)

REF: Basic LWR Theory

5.9 During a routine startup the reactor is subcritical with a stable count rate of 200 CPS on all source range instruments and the shutdown groups are fully withdrawn with a Keff of 0.95. The operator withdraws C.E.A.S until the count rate is 400 CPS then stops CEA motion.

a. What will be the new Keff? Show all work. (1.00)

b. What will happen if the same amount of +reactivity is added again? Show all work. (1.00)

ANS:

a. Since the count rate has doubled, the margin to criticality has been halved. Keff = 0.975. Can also be shown by using the formula C1 (1-K1) = C2 (1-K2)

b. If the same amount of reactivity is added again the margin to criticality = 0 and Keff = 1.0. The reactor is critical.

Accept slightly supercritical (K=1.03).

REF: Reactor Theory

5.10 The core is operating in the nucleate boiling region. Reactor coolant pressure is increased. What effect does this have on heat transfer at the clad/coolant interface? Explain. (1.00)

ANS:

Heat transfer at the clad/coolant interface <u>decreases</u>. (0.50)

When the pressure is increased <u>nucleate boiling</u> rate is <u>decreased</u> due to saturation temperature increasing. This reduction in nucleate boiling <u>reduced heat transfer</u>. (0.50)

REF: Heat transfer and Thermodynamics.

5.11 After operating in natural circulation for 2 hours, a complete loss of natural circulation flow occurs. How will the following parameters change (increase, decrease, or remain the same)? Briefly explain your answer. (assume no further operator action) (3.00)

a. Core delta T

b. Core thermocouple temperature

c. Steam generator pressure

d. Reactor coolant system pressure

ANS:

a. Increase (0.25). Th (0.25) will increase while Tc (0.25) remains relatively constant.

b. Increase (0.25). Less means available to remove the heat (0.5).

c. Decrease (0.25) less primary to secondary heat transfer (0.5).

d. Increase (0.25). Decay heat heats water (0.5).

REF: Heat transfer, Thermodynamics, and Fluid Flow, General Physics

5.12 Provide FIVE physical factors, that change over core life, which affect the heat transfer capability of the fuel and the full power center line temperature. Indicate whether each factor will tend to INCREASE or DECREASE center line temperature over the life of the core. (3.00)

ANS:

(FIVE from the following)

a. Fuel densification--increase
b. Fuel pellet swelling--decrease
c. Clad creep--decrease
d. Clad corrosion--increase
e. Crud buildup--increase
f. Nature of the gas in the gap--increase
g. Concentration of the gas in the gap--decrease

[0.4 each factor + 0.2 each direction] (3.00)

REF: General Physics - Heat Transfer, Thermodynamics & Fluid Flow

WSES 3-FSAR 4.2.1.2.4.5 and 4.2.1.2.5, p. 4.2-17

Plant Systems Design

6.1 List nine (9) of the twelve (12) functions that are provided by the Chemical and Volume Control System. (3.00)

ANS: (0.33 ea)

a. Maintain reactor coolant inventory
b. Control boron concentration in the RCS
c. Continuous measurement of the RCS boron concentration
d. Maintain the chemistry and purity of the reactor coolant during preoperational phases, startup, normal operation, and shutdowns
e. Collects bleedoff from RCP seals
f. Provide auxiliary pressurizer spray
g. Testing the SIS/RCS isolation check valves
h. Leak test the RCS
i. Provides a controlled path for discharging reactor coolant to the boron management system
j. Provides for (automatic) emergency boration
k. Provides an alternative means for filling the RCS
l. Hydro testing the RCS
m. Continuous measurement of fission product activity RCS gross gamma activity

REF: SDD Vol 1 Chem & Vol Cont System, pages 1-4

6.2 Describe the eight (8) channels of excore nuc?ear instrumentation. Include number of detectors, type of detectors, and the range of each detector. Also explain the principal of power level detection for each detector. (12 answers) (4.00)

ANS: (0.25 ea)

4 safety channels $2x10^{-8}\%$ - 200%, 3 vertically stacked detectors for each channel also 0 (of 1) to 125% stacked fission chambers (log channel uses middel detector only)

Fission of ^{235}U for ionization

2 control channels, 0 (or 1) to 125% - one each uncompensated ion chamber.

Boron lined reaction $^{10}_{5}B + ^{1}_{0}N \rightarrow ^{11}_{5}B \rightarrow ^{4}_{2}He + ^{7}_{3}Li$ for ionization

2 startup channels 1 to 10^5 cps, 2 Bf3 preoperational counters

Boron triflouride gas detector $^{10}_{5}B + ^{1}_{0}N \rightarrow ^{11}_{5}B \rightarrow ^{4}_{2}He + ^{7}_{3}Li$ for ionization

REF: SDD Volume II Excore Nuclear Inst Sup, pages 1 - 10

6.3 List six (6) systems which are physically connected to the Component Cooling Water System and cooled by it. (2.00)

ANS:

Containment Spray, Safety Injection, Reactor Coolant (Pumps), Chemical and Volume Control, Fuel Pool Cooling, Gaseous Waste Management, Emergency Diesel Generators, Process Sampling, CEDM Cooling, RAB HVAC, Cont. Fan Coolers, Boron Mgmt. Sys., and LWMS (Any 6 @ 0.333 ea)

REF: WSES Sys Des SD-6-0, pages 6-40/41, fig. 2

6.4 Explain how the Containment Spray System can limit offsite dose or radiation post-accident. (2.00)

ANS: (1.00 ea)

a. Spray reduces containment pressure which reduces out-leakage.
b. Spray scavenges atmospheric iodine.

REF: WSES Sys Des SD-15c-0, pages 15c-1

. . .

6.5 Give three (3) of the four (4) subsystems of the Containment Cooling and Ventilation System and explain the principal function of each. (3.00)

ANS:

1. Airborne Radioactivity Removal Subsystem (ARRS) - limit airborne rad. inside containment during normal operation (to allow personnel access)
2. Containment Fan Cooling Subsystem (CCS) - cool the containment atmosphere during both normal and post-accident conditions
3. Reactor Cavity Cooling Subsystem (RCC) - cool the reactor cavity area to limit surface temperature of the concrete and thermal growth of the vessel supporting steelwork
4. Control Element Drive Mechanism Cooling Subsystem (CDC) - cool magnetic drive coils (so temperature stays below 350°F)
(Any 3 ans - name @ 0.25; function @ 0.75)

REF: WSES Sys Des SD-17-0, p. 17-1/2

6.6 Give two (2) methods of removing or limiting the post-accident hydrogen concentration inside containment. (1.50)

ANS:

1. Hydrogen Recombiners
2. Containment Atmosphere Release System
   (2 ans @ 0.75 ea.)

REF: WSES Sys Des SD-19-0, p. 19-2

6.7 Name two (2) of the three (3) off-normal operating modes of the Control Room HVAC system, explain their major differences from normal, and state one signal which would initiate each. (3.00)

ANS:

1. SIAS/High Radiation Mode - isolates intake and exhaust, starts recirc/bypass mode, starts emergency filtration units - aligns on high radiation @ intake or SIAS
2. Toxic Gas Mode - same as (1) except emergency intakes interlocked closed - aligns on input from Chemical Detection Systems (chlorine, ammonia, and broad range gas)
3. Smoke Purge Mode - isolates supply/return ducts to computer room - aligns on halon suppression system actuation

(Any 2 ans - name @ 0.50, differences @ 0.50, signal @ 0.50)

REF: WSES Sys Des SD-21-0, p. 21-14/16, CWD 1153-S

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6.8 Give the maximum <u>and</u> normal blowdown rates in percentage of the maximum steaming rate for the Steam Generator Blowdown System. (1.00)

ANS:

Max: 1%/SG
2% of MSR
320 gpm
Normal: 0.2%/SG
0.4% of MSR
64 gpm

Any 2 ans. @ 0.5 ea.

REF: WSES Sys Des SD-36-0, p. 36-1

6.9 Place the list below in the proper sequence along a feedwater line starting at the main condenser and continuing downstream to a steam generator. (2.00)

a. Main Feedwater Flow Control Valve
b. High Pressure Feedwater Heaters
c. Main Feedwater Isolation Valve
d. Low Pressure Feedwater Heaters
e. Emergency Feedwater Tee
f. Condensate Pump
g. Steam Generator Feedwater Pump
h. Gland Steam Condenser

ANS:

f, h, d, g, b, a, c, e
(8 ans @ 0.25 ea.)

REF: WSES Sys Des SD-45-0, Figure 45-1; SD-46-J, Figure 46.1

6.10 Explain how the presence of steam in a post-accident containment environment can affect air-hydrogen combustion. (2.00)

ANS:

Steam introduces 2 competing effects: (1) steam tends to retard the combustion process which reduces the pressure spike and (2) steam implies pre-existing steam pressure which increases the total pressure spike (2 effects @ 1.00 ea.)

REF: WSES Sys Des SD-19-0, Figure 19-6

6.11 List three (3) Engineered Safety Features Sytems which provide direct support to the containment. (1.50)

ANS:

Safety Injection System, Shield Building Ventilation System, Contaiment Isolation System, Combustible Gas Control System, and Controlled Ventilation Area System - Containment Spray, Contaiment Cooling.
(Any 3 ans @ 0.50 ea.)

REF: WSES Sys Des SD-16-0, p. 16-1/2

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PROCEDURES - NORMAL - ABNORMAL EMERGENCY AND RADIOLOGICAL

7.1 List conditions or events that require emergency boration. (4 answers required for full credit) (2.00)

ANS: (0.50ea)

1. One or more CEA's not fully inserted after a reactor trip
2. Shut down margin less than 5.15% ΔK/K and T are greater than 200°F/Mode 1-4
3. Shut down margin less than 2% ΔK/K and T are less than or equivalent to 200°F/Mode 5
4. Unexplained positive reactivity addition
5. Uncontrolled cooldown of the R.C.S. by steam flow/feed flow
6. Keff greater than .95. with Boron concentration less than 1720 ppm head removed or head bolts not tensioned/Mode 6

REF: OP-901-013 page 3.

1. Rx tripped outside CR
2. More than 1 CEA stuck out

REF: OP-901-004

1. More than 1 CEA stuck out

REF: OP-902-00

7.2 List the immediate actions that must be taken for a reactor trip. (16 actions total-12 required for full credit) (4.00)

ANS: (0.33ea)

1. check for reactor trip
2. check for turbine trip
3. check for generator trip
4. check containment temperature less than 120°F
5. check that Train A Safety 4KV Bus is powered by startup transformer A or Emergency Diesel A
6. check that Train B Safety 4 KV Bus is powered by startup transformer B or Emergency Diesel B
7. reset moisture separator reheater controls, then check temperature control valves closed
8. check steam bypass control system is operating in automatic
9. check feedwater control in reactor trip override
10. stop the RCPs if RCS pressure is equal or less than 1621 psia
11. if RCS pressure is greater than 1684 psia verify that pressurizer level and pressure control are operating in automatic
12. if pressurizer pressure is equal or less than 1684 psia on 2 of 4 channels <u>or</u> containment pressure is equal or greater than 17.4/17.1 psia on two of four channels, verify SIAS has actuated
13. check containment area monitors are normal
14. if containment pressure is equal or greater than 17.7 psia on 2 of 4 channels verify that CSAS has occurred
15. either steam generator level equal to or less than 27.4% narrow range on 2 of 4 channels verify EFAS has actuated
16. if containment pressure is equal to or greater than 17.4/17.1 psia on 2 of 4 channels or either steam generator pressure is equal to or less than 764 psia on 2 of 4 channels. Verify MSIS by checking MSIVs and steam generator feed valves closed.

REF: OP-902-000 page 2 to 10

7.3 List the symptoms of a loss of coolant accident.
(10 answers - 8 for full credit) (4.00)

ANS: (0.50ea)

1. Decreasing pressurizer pressure
2. Decreasing pressurizer level
3. Increasing containment pressure
4. Increasing containment sump and Safety Injection sump levels
5. Increasing containment temperature and humidity
6. High radiation levels in the containment
7. Mismatch in charging header flow versus letdown flow
8. Decreasing volume control tank level
9. Decreasing subcooled margin
10. A loss of coolant from the pressurizer steam space as indicated by increasing pressurizer level, increasing Quench tank pressure, level and temperature.

Accept alarms for partial credit.

REF: OP-902-002 page 2.

7.4 List the precautions and controls that must be provided per 10 CFR 20 at each entrance or access point to a HIGH RADIATION AREA. (3 answers required) (1.50)

ANS: (0.50ea)

1. Equipped with a control device which shall cause the level of radiation to be reduced below that at which an individual might receive a dose of 100 millirems in 1 hour upon entry into the area; or
2. Equipped with a control device which shall energize a conspicuous visible or audible alarm signal in such a manner that the individual entering the high radiation area and the licensee or a supervisor of the activity are made aware of the entry; or
3. Maintained locked except during periods when access to the area is required, with positive control over each individual entry
4. Posted as a "High Radiation Area."

REF: 10 CFR 20 Section 20.203, Page 242.

7.5 Provide immediate action that must be taken if two or more seals fail on a reactor coolant pump. (2 actions) (1.00)

ANS:

Trip the reactor secure the affected RCP

REF: OP-901-010 Page 8 Att. 6.1

7.6 List the actions that must be taken immediately to initiate emergency boration. (7 answers, 5 required for full credit) (2.50)

ANS:

1. Place the makeup mode selector switch in the "MANUAL" position
2. Open Emergency Borate Valve BAM-133
3. Start one Boric Acid Pump
4. Close the running Boric Acid Pump's recirc valve BAM-126A(B)
5. If a Boric Acid Pump cannot be started, then open Boric Acid Makeup Tank Gravity Feed Valves BAM-113A(B)
6. Close Volume Control Tank Discharge Valve CVC-183
7. Verify at least one Charging Pump is running and charging flow $\geq$40 gpm

Valve Nos. not required for credit.

REF: OP-901-013 Page 5

7.7 Provide the automatic actions that occur when there is a loss of charging flow. (4 answers) (2.00)

ANS:

1. Pressurizer Level Control System will decrease letdown flow
2. Letdown Isolation Valve CVC-101 will auto-isolate at 470 degrees outlet temperature of the Regenerative Heat Exchangers
3. The Process Radiation Monitor, Boronometer and Ion Exchangers will isolate on a high temperature of 140 degrees
4. Letdown will auto-divert from the VCT to the Flash Tank at the 76% level in the VCT

Valve No. and set pts. not required for full credit.

Accept other answers based on pzr press/level response if correct.

REF: 09-901-D14 page 6 att 6.1

7.8. List the immediate actions that must be taken following a turbine trip. (8 steps, 6 required) (3.00)

ANS: (0.50ea)

1. Verify that all Tubine Throttle, Governor, Reheat and Intercept Valves close and turbine speed is decreasing
2. Verify that the Generator Output and Exciter Field breakers open. If not, manually open
3. Ensure that the Reactor Regulating System and/or the Reactor Power Cutback System initiate(s) rod movement to reduce power to the AMI threshold level in an effort to balance power
4. Verify that the unit loads shift to designated startup transformers
5. Ensure that Steam Generator pressure is being maintained at 1000 psia by the Turbine Bypass Valves and/or 1050 psia by the Atmospheric Dump Valves
6. Ensure that Steam Generator levels are being maintained at or near 70% NR by the Main Feedwater or 70% WR by the Emergency Feedwater System
7. Notify the Nuclear Operations Supervisor of the turbine trip and actions taken to rectify the problem

   Announce "Turbine Trip" on the PA System.

8. If a Reactor Cutback occurs, perform OP-901-003, concurrently with this procedure

REF. OP-901-034 page 6.

. .

7.9 List the immediate actions that must be taken on an inadvertant safety injection actuation. (2 steps required) (1.00)

ANS: (0.50ea)

1. Trip the Reactor Carryout Reactor Trip Procedure
2. After all CEAs have been inserted for at least five seconds trip the RCPs

REF: OP-901-042, Rev. 0, page 5.

Or

1. Verify SI termination criteria
   a. Pzr level GT 28%.
   b. Pzr water temp. GT 28°F higher than highest Th and Tsat for highest press. SG (subcooled margin GT 28°F).
   c. 1 SG level GT/EQ 50% WR and not decr.

REF: OP-901-042, Rev. 1, Page 5.

7.10 (a) What is the purpose of a danger tag? (1.00)
(b) Who may authorize the installation and removal of danger tags? (1.00)
(c) What is the purpose of a caution tag? (1.00)

ANS:

(a) Prohibits the operations of equipment or systems which could jeopardize personnel safety or endanger equipment

(b) CRS and SS

REF: VNT-5-003 page 4 and 5

ANS:

(c) A tag to provide precautionary information or to provide temporary special instructions

REF: OP-100-003 page 2.

7.11 What is the reason reactor coolant pump 2A must not be operated at with one of more pumps at temperatures below 340°F. (1.00)

ANS:

RCP 2A was manufactured with a higher lift than the other RCPs, the precautions must be followed to prevent exceeding core uplift limitations

REF: OP-001-002 page 4

Or

Keep RCP motor within name plate hp/amp data.

REF: CE letter C-CE-7050, dated March 19, 1981.

ADMINISTRATIVE PROCEDURES, CONTROLS, AND LIMITATIONS

8.1 List the conditions that must be met to ensure containment integrity per the technical specifications. (5 answers required for full credit) (3.0)

ANS:

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

1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. All Equipment hatches are closed and sealed.

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

REF: Technical Specifications 1.7 page 1-2

8.2 List the position, number of individuals required for minimum shift crew composition while in modes 1 (one) through 4 (four). (1 point for positions, 1 point for numbers). (2.00)

ANS:

POSITION	NUMBER OF INDIVIDUALS REQUIRED
	MODE 1, 2, 3, OR 4
SS	1
SRO	1
RO	2
AO	2
STA	1

Accept additional information or answers (e.g., fire brigade) @ no penalty.

REF: Technical Specification Table 6.2-1, page 6-5

8.3 Who may be designated to assume the control room command function in the absence of the shift supervisor while in mode. (1 (one) through 4 (four). (1.00)

ANS:

An individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function

REF: Technical Specification Section 6.2, page 6-5

8.4 Fill in the temperature change rate blanks for the following conditions. (1.50)

a. A maximum heatup rate of _____ per hour with Reactor Coolant System cold leg temperature less than 200°F.

b. A maximum heatup rate of _____ per hour with Reactor Coolant System cold leg temperature greater than 200°F and less than or equal to 345°F.

c. A maximum heatup rate of _____ per hour with Reactor Coolant System cold leg temperature greater than 345°F.

d. A maximum cooldown rate of _____ per hour with Reactor Coolant System cold leg temperature less than 135 °F.

e. A maximum cooldown rate of _____ per hour with Reactor Coolant System cold leg temperature greater than or equal to 135°F and less than or equal to 180°F.

f. A maximum cooldown rate of _____ per hour with Reactor Coolant System cold leg temperature greater than 180°F.

ANS: (0.25ea)

a. 30°F
b. 50°F
c. 100°F
d. 10°F
e. 30°F
f. 100°F

REF: Technical Specification Section 3.4.8.1, page 3/4 4-28

8.5 List the limits of reactor coolant system leakage. (5 answers, 4 required fcr full credit.) (2.00)

ANS: (0.50ea)

a. No PRESSURE BOUNDARY LEAKAGE

b. 1 gpm UNIDENTIFIED LEAKAGE

c. 1 gpm total primary-to-secondary leakage through all steam generators and 720 gallons per day through any one steam generator (0.5 gpm)

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 = 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1

REF: Technical Specification Section 3.4.5.2, page 3/4 4-18

8.6 List the requirements that must be met for issuing a temporary change to a procedure. (2 answers) (i.e., when and how per T.S.) (2.00)

ANS: (1.00ea)

a. The intent of the original procedure is not altered

b. The change is approved by two members of the plant management staff and one must hold a senior operator license

c. Not required for credit. The change is documented, reviewed by the PORC and approved by the Plant Manager-Nuclear within 14 days of implementation

REF: Technical Specification Section 6.8.3, page 6-15

. .

8.7 List the actions that must be taken if a safety limit is violated. (4 answers)

(2.50)

ANS:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President-Nuclear Operations and the SRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be operated. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Senior Vice President-Nuclear Opeartions within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

e. S/D within 1 hr. (T.S. 2.1, p. 2-1)

Accept any 4 @ 0.625 ea.

REF: Technical Specification Section 6.7.1, page 6-14

8.8 List the maximum power for 1, 2, and 3, and 4 inoperable safety valves on any operationg steam generator per the technical specification. (4 answers) (2.00)

ANS: +/- 5% (0. 5ea)

1. 86.8 percent of Rated Thermal Power

2. 69.4 percent of Rated Thermal Power

3. 52.1 percent of Rated Thermal Power

4. 34.7 percent of Rated Thermal Power

REF: Technical Specification Table 3.7-2, page 3/4 7-3

8.9 Fill in the blanks in the following three (3) questions relating to the Safety Limits at Waterford.

a. The ________________ of the reactor core shall be maintained greater than or equal to _____________. (1.0)

b. The ________________ of the fuel shall be maintained less than or equal to ________________ KW/ft. (1.0)

c. The __ _____________ pressure shall not exceed _____________ psia. (1.0)

ANS: (0.50ea)

a. DNBR, 1.20 (0.5ea)

b. peak LHGR/LPD/LHR, 21.0

c. RCS, 2750

REF: Technical Specification 2.1, p. 2-1

8.10 Fill in the blanks in the following three (3) questions relating to Reactor Coolant Pump (RCP) operation at Waterford.

a. The maximum RCP operating time without CCW flow is ________________ minutes. (0.5)

b. If CCW flow can be restored within ______________ minutes, then the pump may be restarted. (0.5)

c. With the motor at operating temperature, do not attempt more than _____________ start(s) at __________ minute intervals. (1.0)

ANS:

a. 3 (0.5ea)

b. 10

c. 1 60

REF: OP-1-002, p. 3

8.11 Explain the two Technical Specification BASES for the control rod transient insertion limit. (2.0)

ANS:

1. assure S/D margin (1.0)

2. assure ejection accident within design limit (1.0)

REF: Technical Specification, page B 3/4 1-5

8.12 List four (4) parameters which must be manually calculated when Core Operating Limit Supervisory System (COLSS) is out of service. (2.0)

ANS: DNBR Margin Verification (12 hrs)

Axial Shape Index

Linear Heat Rate (2 hrs)

Aximuthal Power Tilt (12 hrs)

RCS Total Flow (12 hrs) (4/5 @ 0.5ea)

Secondary Calorimetric

REF: COLSS Inoperable OP-901-011, p. 7 and 8