U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Examination Report No. 50-397/OL-85-01

Facility: Washington Nuclear Plant No. 2

Docket No. 50-397

Examinations administered at Washington Nuclear Plant No. 2, Richland, Washington from May 29 to May 30, 1985.

Chief Examiner: ______________________ 8/26/85
R.J. Pate, Chief — Date Signed
Reactor Safety Branch (Acting)

Approved: ______________________ 8/26/85
R.J. Pate, Chief — Date Signed
Operations Section

Summary:

Examinations on November 6-8, 1984

Written examinations were administered to four SRO candidates. An operating examination (oral) was administered to one SRO candidate. Three SRO candidates passed the examinations.

REPORT DETAILS

1. Persons Examined

   Examinations were administered to four senior operator candidates.

2. Examiner:

   R.J. Pate

3. Examination Review Meeting

   An exam review meeting was held immediately after the written exam was administered, on May 29, 1985. The following utility representatives were in attendance:

   John Wyrick
   Mark Westergren
   Tim Messersmith
   Sam McKay

   Additionally, the following NRC representative was present:

   Robert Pate

   The responses to the comments provided by the utility representatives are included as enclosure (1). Additional comments were provided by letter from G.C. Sorensen to J.B. Martin, dated June 10, 1985. The responses to these comments are included as enclosure (2). Where applicable the examination keys have been changed.

4. Exit Meeting

   An exit meeting was held with the facility on May 30, 1985. The attendees were:

   NRC:

   Robert Pate - Chief, Reactor Safety Branch

   Utility:

   Jack Shannon - Deputy, Managing Director
   Jerry Martin - Assistant Manager Director for Operations
   Chris Powers - Plant Manager, WNP-2
   John Wyrick - Nuclear Technical Training
   Rick Stickney - Manager Technical Training
   Jack Baker - Assistant Plant Manager, WNP-2

The examiner reported that there were no candidates that were a clear pass on the Operating Examination (Oral). The criteria used for determining whether a candidate passed the oral examination was discussed.

The current status of the plant simulator was discussed. The facility staff stated that the repair of the simulator should be complete prior to the next NRC exam.

The small number of candidates (one) taking the oral examination made it possible to identify any weaknesses in the facility training program.

The examiner noted that many of the comments on the examination review resulted from incomplete or out-of-date reference material. The examiner accepted the facility comments when based on additional information. However, the effort of making the comments and NRC having to respond to the comment could be saved with better reference material.

Enclosure (1)

SRO EXAM REVIEW COMMENTS AND RESOLUTIONS

Comments on the following questions were accepted and the master answer key suitably modified:

SRO EXAM

Section 5: No comments accepted without examiner comments.

Section 6: 6.01 ; 6.02; 6.07.

Section 7: 7.03; 7.06; 7.09; 7.11.

Section 8: 8.06; 8.07.

Comments on the following questions were not accepted as explained below:

SRO EXAM

Section 5
Question 5.01c:

Facility comment: The discharge head will change only by a little with increasing temperature. Should accept little or no change.

Response: Will accept little change, but not no change.

Question 5.02b:

Facility Comment: Question does not require the calculations be shown.

Response: The calculation in the key does not have to be shown A correct answer will be accepted for full credit, but no partial credit will be given without the calculations being shown.

Question 5.04

Facility Comment: Reviewer provided suggestions on assignment of partial credit.

Response: Distributions of partial credit is at the discretion of the examiner.

Question 5.05

Facility Comment: Reviewer request that full credit be given if candidate says how the power changes locally and overall without describing the effects discussed in the answer key.

Resolution: Full credit will be given if the candidate describes what is happening in the core. No change to key.

Section 6
Question 6.04

Facility Comment: Manual should not be one of the required responses. The low FW flow is initiated at 30% flow.

Response: Manual will be required for full credit. The 30% flow will be included in the answer key.

Section 8
Question 8.02

Facility Comment: Memorization of tech. specs. should not be required.

Response: Overtime limits have a direct affect on the operator and senior operator and the limits should be known. No change to answer key.

Enclosure (2)

Response to Facility Comments Provided in Sorensen letter.

Question 7.09

The answer key has been changed to accept other items listed as indications of lack of containment integrity as requested by the facility.

MASTER

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

FACILITY: WNP-2

REACTOR TYPE: BWR-GE5

DATE ADMINISTERED: 85/05/29

EXAMINER: R. J. Pate

APPLICANT: ____________________

INSTRUCTIONS TO APPLICANT:

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	APPLICANT'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	100.00	________	________	TOTALS

FINAL GRADE ______________%

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

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (3.00)

Describe HOW and WHY centrifugal pump discharge head is affected for each of the following (consider each condition separately):

a. Suction pressure increases. (1.0)

b. The discharge valve is throttled closed. (1.0)

c. The temperature of the fluid being pumped increases. [NPSH is not lost.] (1.0)

QUESTION 5.02 (3.00)

The reactor is subcritical with a Keff of .95 a SRM countrate of 200 cps. The control rods are withdrawn and the new countrate is 400 cps.

a. How much reactivity was added? (2.0)

b. What would be the status of the reactor if the same amount of reactivity, determined in a., was added again? (1.0)

QUESTION 5.03 (3.00)

Explain what happens in the core and why, when recirculation flow is decreased, while at power and with no control rod movement.

QUESTION 5.04 (3.50)

a. The end of core life or end of cycle is usually define as what? (0.5)

b. Describe the three methods of extending operations beyond end of cycle. (3.0)

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

QUESTION 5.05 (3.00)

Describe the core response, both local and overall for each of the following. (Assume power level is greater than 75%.)

a. The withdrawal of a Deep Control Rod. (From a deep position to another deep position.) [1.5]

b. The withdrawal of a Shallow Control Rod. [1.5]

QUESTION 5.06 (3.00)

Below is listed some of the data for a recirculation pumps at slow and fast speed. Using the data provided, determine the values for the four (4) missing parameters.

Slow Speed	Fast Speed
15 HZ	60 HZ
450 RPM	--a-- RPM
--b-- GPM	47,200 GPM
--c-- ft of head	805 ft of head
150 HP	--d-- HP
(Show all work.)	(4 @ 0.75 ea)

(3.0)

QUESTION 5.07 (2.00)

A reactor has just scrammed from extended full power operation. Ten (10) hours later cooldown is complete, and the SDM is measured at that time to be 1% dk/k. Describe the changes, if any, to the SDM for the next 20 hours. (Include in your answer any adverse conditions.)

QUESTION 5.08 (2.00)

Explain WHY core orificing is necessary and HOW orificing accomplishes this purpose. (2.0)

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

QUESTION 5.09 (2.50)

With regard to the MAPLHGR thermal limit

a. Briefly, WHAT is the reason, or bases for having a MAPLHGR thermal limit? (1.0)

b. WHICH TWO of the following four parameters affect the MAPLHGR LIMIT? (0.5)

1. Moderator Temperature
2. Type of fuel
3. Fuel exposure
4. Reactor pressure

c. If an P-1 is selected on the Process Computer, the program provides, among other things, MAPRAT. WHAT is the relationship between MAPRAT and MAPLHGR? (1.0)

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

QUESTION 6.01 (3.00)

With regard to the Reactor Vessel Level Instrumentation, describe the five (5) types of levels used. (Include range, reference point for zero and, YES or NO if any automatic functions are initiated.) (3.0)

QUESTION 6.02 (3.50)

The Leakage Detection System can be divided into two general groups: abnormal leakage WITHIN the primary containment and abnormal leakage OUTSIDE the primary containment. The Leakage Detection System provides INDICATION, ALRAM, and ISOLATION SIGNALS for what system components? (Include both Inside and Outside Primary Containment.) and/or

QUESTION 6.03 (3.00)

With regard to the Intermediate Range Monitoring System (IRM), answer the following questions:

a. What is the purpose of the range switches? (0.5)

b. The IRM is reading 24 on range 5.

1. What would be the reading if range 6 was selected? [0.33]

2. What would be the reading if range 7 was selected? [0.33]

3. Would there be any alarms resulting from the selection of range 7? [0.34] (1.0)

c. What three conditions will result in an IRM inoperative Rod Block? [3 @ 0.5 ea] (1.5)

QUESTION 6.04 (3.00)

The recirculation pump will be downshifted from fast speed to slow speed for any of five reasons.

a) What are the five (5) reasons? (1.5)

b) What are the sequence of events that the recirculation system undergoes (include incomplete transfer). (1.5)

QUESTION 6.05 (3.00)

Delete
use substitute 6.05 attached.

a. When the DEH is in Auto Turbine Control:

1. The A&C programs are in control of what three items when in speed control? [0.6]

2. What two items are the A&C programs in control of when in load control? [0.4]

b. Briefly describe the four modes of turbine operation used at WNP-2. [2.0] (3.0)

QUESTION 6.06 (3.00)

When the HPCS system logic is initiated, what components receive signals and what type of signals are they? (i.e., open, close, start or stop) (3.0)

QUESTION 6.07 (3.50)

Concerning the Reactor Water Cleanup (RWCU) system:

a. List two conditions that will cause the blowdown flow control valve (FCV-33) to auto close. (1.0)

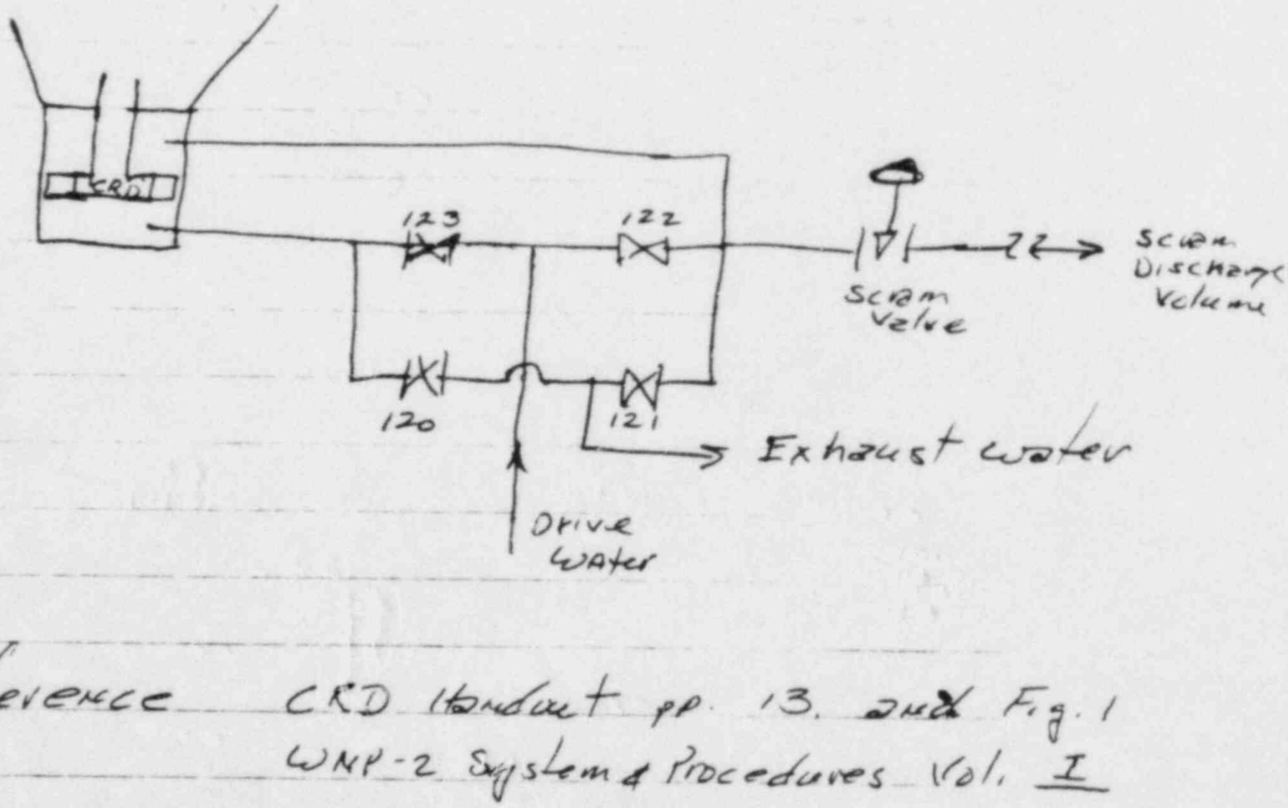
b. Should the blowdown restricting orifice bypass valve (V-31) be open with reactor pressure greater than 125 psig? Explain your answer. (1.0)

c. What conditions (list four) will close the inboard isolation ~~vlave~~ valve (V-1)? (1.0)

d. How would a loss of service air affect system operation? (0.5)

605 a)

In the CRD system ~~valves~~ four directional control valves are energized by the rod drive control system (RDCS) to direct driving pressure and exhaust water through the HCU to and from the under and over-piston ports in the associated CRD for normal control rod in and out positioning. Which ~~two~~ of these valves include speed control elements? (0.5) ~~What~~ why do these valves have ~~adjustable~~ speed control elements. (0.5)



Reference CRD Handout pp. 13. and Fig. 1
WNP-2 System & Procedures Vol. I

ANS: a) Valves 120 and 123

value not req'd for full credit → b) To adjust and maintain normal CRD speed of (3 in./sec.)

QUESTION 6.08 (3.00)

Assume the RCIC system receives an initiation signal, all system components function properly, except the items listed below. Each failure is present prior to the initiation signal being received.

Describe the RCIC systems response for each of the following and justify your answer. Consider each item seperately.

a. The turbine exhaust valve (RCIC-V-68) is stuck shut. (1.0)

b. The Ramp Generator portion of the RGSC (Ramp Generator Signal Converter) has failed producing a large signal. (1.0)

c. The D/P cell, for the RCIC flow control element, has a perferated diaphram. (1.0)

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

QUESTION 7.01 (3.00)

Using WNP-2 Figures 7 & 8 answer the following questions concerning thermal stresses:

a. When considering thermal stresses, the turbine rotor is the most critical element because of it's large diameter. Why then is the 1st stage turbine temperature used vise turbine rotor temperature to determine the turbine starting procedure? (1.0)

b. The turbine 1st stage temperature is 175 F. How long would it take to bring the turbine generator to 100% power? (Consider turbine limits only.) (1.0)

c. With the unit at 10% rated load, WHAT is the recommended load changing rate (%/min) to increase power to 95%. (Assume 10,000 cycle fatique index.) (1.0)

QUESTION 7.02 (1.50)

During Core Alterations Procedure 2.1.2, Neutron Monitoring System requires at least 2 SRMs in specific locations. WHAT are the specified locations?

QUESTION 7.03 (2.00)

With the unit operating at 35% power:
One recirculation pump is secured and it's discharge valve closed by procedure. After 4 minutes the operator attempts to open the discharge valve. The discharge valve fails to open. WHAT two (2) steps are taken to prevent excessive cooldown of this recirculation loop?

QUESTION 7.04 (2.00)

In what way could changing RWCU system flow adversly affect the recirculation pumps (per procedure 2.2.1, Reactor Recirculation System)?

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

QUESTION 7.05 (1.00)

How are the control rods verified inserted after a reactor scram (4 ways required for full credit) [4 @ 0.25 ea] (1.0)

QUESTION 7.06 (1.00)

After a reactor scram with the loss of condensate booster pumps, WHY should the RWCU be lined up to return to the RPV before restarting a booster pump, per procedure 3.3.1, reactor scram recovery?

QUESTION 7.07 (3.00)

During power operation the operator receives a rod drift alarm. The rod is selected and is found to be still drifting out. No automatic scram setpoints have been reached. What are the immediate operator (control room) actions? (Actions involving operations/communication at the HCU are not required.)

QUESTION 7.08 (2.00)

On a complete loss of CRD drive flow when are you required to scram the reactor and how is it accomplished?

QUESTION 7.09 (2.50)

There are eight (8) "Indications" listed in procedure, 4.3.1.1, "Primary Containment Not Operable", that indicate a loss of Primary Containment integrity. WHAT are five (5) of these "Indications"?

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

QUESTION 7.10 (4.00)

In reference to the Loss of Condenser Vacuum procedure:

a. List 4 of the automatic actions associated with a decreasing condenser vacuum. Include setpoints. (2.0)

b. What are the four immediate operator actions assuming vacuum has not yet decreased to the point where the automatic actions of part "a" have occurred? (2.0)

QUESTION 7.11 (3.00)

During power operation at 100%:

a. A loss of SM-7 occurs, WHAT are 4 of the 5 immediate operator action step required? [An action step may consist of more than one action item.] (2.0)

b. One of the indications of the loss of SM-7 is a scram from the loss of RPS-MG-1. HOW does the loss of RPS-MG-1 cause the reactor to scram? [Give the chain of events.] (1.0)

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

QUESTION 8.01 (3.00)

According to WNP-2 Technical Specifications, WHAT is the minimum staffing requirements for WNP-2 when in condition 3 [mode 3] AND indicate what type of license is required for each staffing position, if any?

QUESTION 8.02 (3.00)

What are the guidelines, per Tech Spec, used when assigning overtime for unit staff members who perform safety-related functions at WNP-2?

QUESTION 8.03 (2.50)

It is discovered today, May 14 day shift (08-1600), that a monthly surveillance item due on Tuesday May 7 mid shift (12-0800) was not performed. This item has been performed on time for the past six months. Has the specified time interval for this surveillance item been violated? (Yes or No) EXPLAIN your answer.

QUESTION 8.04 (2.50)

a. Describe how a Safety Limit and a Limiting Safety System Setting relate to each other. (1.5)

b. What does the term "Limiting Condition for Operation" mean? (1.0)

QUESTION 8.05 (3.00)

Answer the following questions with regard to the issuance of a Radiation Work Permit (RWP).

a. What are the radiological limits that require the use of a RWP? (1.5)

b. What are the responsibilities of the Shift Manager, when reviewing a RWP for approval? (1.5)

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

QUESTION 8.06 (3.00)

With regard to equipment clearance and tagging procedures answer the following:

a. If the person the clearance is issued to cannot be contacted for a clearance release, how is the clearance release accomplished? (1.0)

b. How is an addition and/or deletion of tags made to a previously authorized clearance order prior to the clearance order being acceptanced by the authorized individual? (1.0)

c. What is meant by the term "Redundant Verification" ? (1.0)

QUESTION 8.07 (3.00)

What is the guideline or instruction given for each of the following per the standing operating orders, 1.3.1 attachment I?

a. Overriding the automatic action of an ECCS system. (0.75)

b. Placing a controller in the manual mode from the automatic mode. (0.75)

c. A safety related motor operated valve has been manually seated. (0.75)

d. The instructions for aligning more than 2 valves or circuit breakers. (0.75)

QUESTION 8.08 (3.00)

According to WNP-2 Technical Specifications 4.1.3.6 which states each affected control rod shall be demonstrated to be coupled to its drive mechanism...:

a. When is the coupling to be verified? (1.5)

b. How is the coupling verified? (1.5)

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

QUESTION 8.09 (2.00)

In reference to the Emergency Plan Implementing Procedures:

a. Who has the sole responsibility for timely classification and declaration of any emergency situation? (0.4)

b. What is the normal line of succession for the position responsible for classification and declaration of an emergency situation? (1.6)

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

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Network out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

$W = v\ \Delta P$

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

$\Delta E = 931\ \Delta m$

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

$\dot{Q} = \dot{m} Cp \Delta t$

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

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - k_{eff2})$

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

$T = \ell/(\rho - \beta)$

$T = (\beta - \rho)/(\lambda\rho)$

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

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

$M = (1 - K_{effo})/(1 - K_{eff1})$

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

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

$\bar{\lambda} = 0.1\ \text{seconds}^{-1}$

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

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

$\Sigma = \sigma N$

$I_1 d_1 = I_2 d_2$

$I_1 d_1{}^2 = I_2 d_2{}^2$

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

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

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 $lbf/in.^2$

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

FIGURE 3

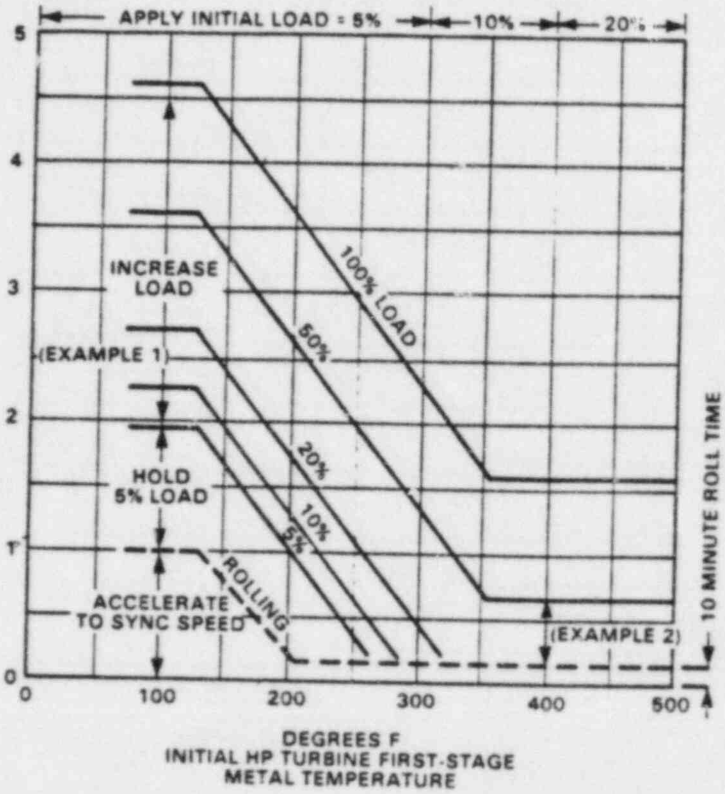


FIGURE 7. START-UP RECOMMENDATIONS
700—1200 MW NUCLEAR STEAM SYSTEM UNITS

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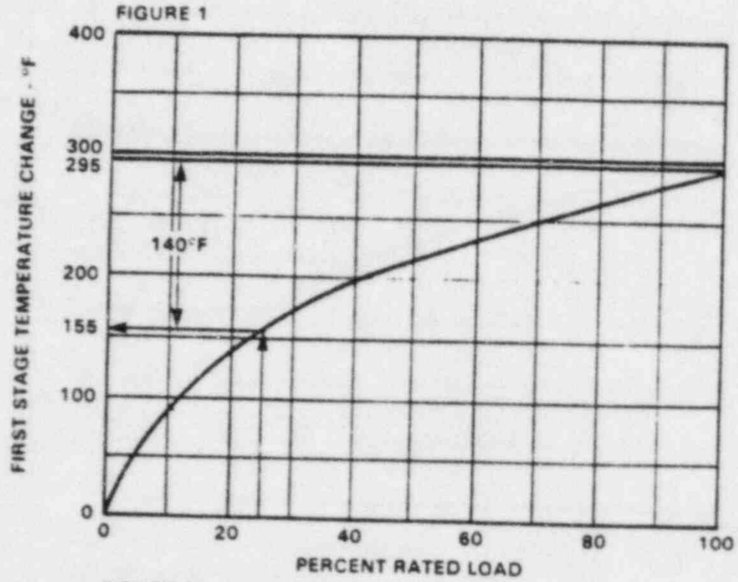

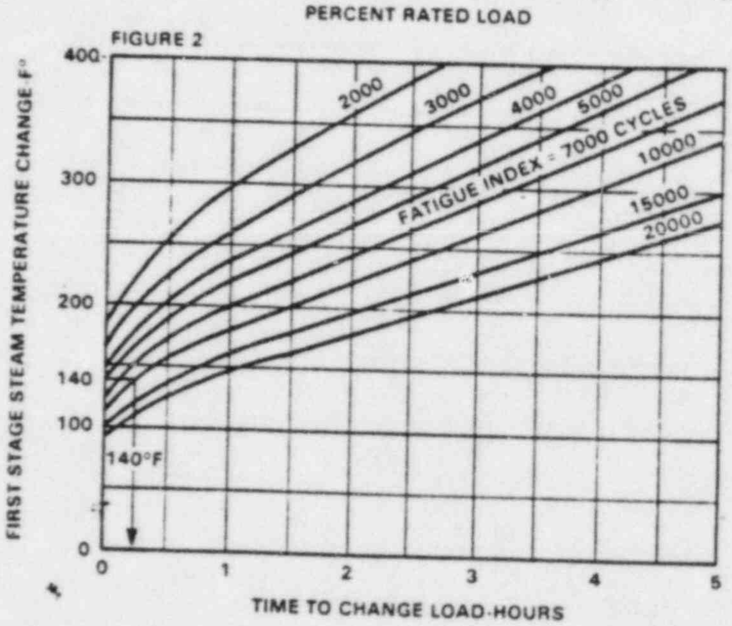

FIGURE 8. LOAD CHANGING RECOMMENDATIONS
700—1200 MW NUCLEAR STEAM SYSTEM UNITS

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APRIL 1984

MASTER

ANSWER 5.01 (3.00)

a. Head increases [0.5] pump is still putting the same amount of work into the fluid, therefore the same delta pressure increase across the pump, so as suction pressure increases so will the discharge head. [0.5] (1.0)

b. Head increases [0.5] as system resistance to flow increases pump head increases. [0.5] (1.0)

c. Head decreases [0.5] as temperature increases system resistance to flow decreases (lower viscosity), therefore pump head decreases. [0.5] (1.0)

REFERENCE
C. E. Thermodynamics Heat Transfer and Fluid Flow Chapter 7, Fluid Statics, Dynamics, and Delivery, Pg 7-104 through 7-124.

ANSWER 5.02 (3.00)

a. CR1 (1-Keff1) = CR2 (1-Keff2) [0.75]
200 (1-.95) = 400 (1-Keff2)
200 (1-.95)/400 -1 = -Keff2
-.975 = -Keff2 [0.25]

delta p = Keff2-1/Keff2 - Keff1-1/Keff1 [0.75]
delta p = .975-1/.975 - .95-1/.95
delta p = (-.0256) - (-.0526)
delta p = .027 [0.25] (2.0)

b. Part b. will be graded independently of part a.

delta p = Keff3-1/Keff3 - Keff2-1/Keff2 [0.75]
.027 = Keff3-1/Keff3 - .975-1/.975
.027 = 1-1/Keff3 - (-.0256)
.0014 = 1-1/Keff3
-.9986 = -1/Keff3
-.9986 Keff3 = -1 [0.25]
Keff3 = 1.0014 super critical (will accept critical) (1.0)

.9500
.0281
.9767
.0256
1.0029

REFERENCE
WNP-2 Reactor Physics Section VI Reactor Operations Part A. Subcritical Reactor (no page numbers available), subsection e. Determine Criticality.

Also see pp 21 of Section III for reactivity

ANSWER 5.03 (3.00)

When recirculation flow decreases the boiling boundary moves down so there are more voids in the core. [0.5] This adds negative reactivity and power level starts to decrease. [0.5] As power level drops,the fuel cools down which cools the water and the boiling boundary starts to move up again. [0.5] Power will continue to decrease and boiling boundary will continue to move up increasing reactivity until the reactivity balance is zero and the reactor returns to steady state at a new lower power level. [0.5] The boiling boundary does not return to the same level as before the power decrease. It remains somewhat lower than before recirculation flow was decreased. The reason is that although the negative reactivity comes from only one source (boiling boundary moving down) the positive reactivity comes from two sources. The first is the boiling boundary moving back up due to the decreased heating caused by the power level decrease. The other is the positive reactivity from the doppler coefficient resulting from the fuel cooling down. [1.0] (The final results is an increase in void reactivity to off set the doppler reactivity decrease.) (3.0)

REFERENCE
WNP-2 III Neutron Life Cycle, Criticality, and Reactivity
Reactor Theory pp 34-35.

ANSWER 5.04 (3.50)

a. Defined as the last day, the core can produce rated power at rated conditions with all the control rods removed from the core. (0.5)

b. Coastdown - a mode of operation in which the reactor power level may drift downward with exposure after end of cycle, this is a reduction in negative coefficient effects with the reduced reactor power compensating for the fissionable istope depletion (power coefficient). [1.0]

Final Feedwater Temperature Reduction (FFWTR) - a mode of operation where the feed temperature is reduced by reducing the extraction steam to the feedwater heaters, which increases the core inlet subcooling. The increased subcooling reduces the core void fraction, which results in a core reactivity increase. [1.0]

Increased Core Flow - a mode of operation where core flow rates greater than the referenced 100% rate value is used. The core reactivity is increased by reducing the core void content, which results from recirculatng more water through the core. [1.0] (3.5)

REFERENCE

Reference, WNP-2 Reactor Physics VII Core Aging, VII, E.1.a-b-c, pg 8, 9, 10, 11 & 12.

ANSWER 5.05 (3.00)

a. The core response to the withdrawal of a deep control rod is to raise core power in the upper region of the core, especially in the areas where the rod was withdrawn. Since the void content of the upper portion of the core is high at operating conditions, the effects of deep control rod withdrawal, although axially dampened, will be substantial radially. [1.5]

b. The core response to withdrawal of a shallow control rod is to raise the power locally in the region where the control rod is withdrawn. The local power increase at the core bottom will pull the boiling boundary down, increasing the void content above the withdrawn control rod. The negative effects of voids may be stronger than the positive effects of the rod and power may decrease. The shallow rod strongly affects the axial power shape and not the overall core power. [1.5] (3.0)

REFERENCE
WNP-2 Reactor Physics VIII Operatng Characteristics, VIII.A.6.C.1) & 2), pg 38 & 39.

ANSWER 5.06 (3.00)

a. 60/15 = x/450
(4)(450) = x
x = 1800 rpm

b. 1800/450 = 47,200/x
4x = 47,200
x = 11,800 gpm

c. (1800/450) squared = 805/x
(4) squared x = 805
x = 50.3 ft of head

d. (1800/450) cubed = x/150
(4) cubed x 150 = x
x = 9,600 hp

(4 @ 0.75 ea) (3.0)

REFERENCE
WNP-2 Requalification Program, Reactor Recirculation System and G.E. Thermodynamics Heat Transfer and Fluid Flow.

ANSWER 5.07 (2.00)

The xenon peak following a shutdown (scram) can have important effects on reactor operations throughout core life. If the reactor was shut down by 1% dk/k as measured at the time of peak xenon, then the SDM will decrease as xenon decays. Since xenon (peak) is greater than the 1% dk/k a reactor restart would occur. (2.0)

REFERENCE
WNP-2 Reactor Physics Sec. V Fission Product Poisons, part 2 Xenon behavior after Reactor Shutdown (no page numbers).

ANSWER 5.08 (2.00)

As the boiling rate increases, two-phase flow resistance increases. this would tend to divert coolant flow from the higher powered center fuel bundles where it is needed the most [1.0]. Orificing has the effect of providing a large resistance to flow so that any additional resistance caused by two-phase flow is acceptably small [1.0]. (2.0)

REFERENCE
WNP-2 G.E. Thermodynamics, Chapter 9 BWR Thermal Limits, Pg 9-51

ANSWER 5.09 (2.50)

a. Minimize fuel damage during a DBA LOCA by limiting the peak clad temperature (to < 2200 F) -OR- limiting bundle stored energy. (1.0)

b. 2 and 3. (0.5)

c. MAPRAT = ~~M~~APLHGR/LIMLHGR -or- = ~~M~~APLHGR/MAPLHGR limit -or- = (~~M~~APLHGR) actual/ (MAPLHGR) LCO max (1.0)

REFERENCE
WNP-2 G.E. Thermodynamics, Chapter 9, BWR Thermal Limits pg 9-68
9-71
9-74

ANSWER 6.01 (3.00)

Narrow Range, 0-60 inches above instrument zero [0.3], bottom edge of steam dryer skirt (527.5 inches), yes [0.3]

Wide Range 150 inches below to 60 inches above instrument [0.3] zero, (527.5 inches), yes. [0.3]

Fuel Zone, -110 ~~150~~ inches ~~below~~ to -310 ~~50~~ inches below ~~above TAF~~ [0.3], (527.5 ~~366.3~~ inches), no [0.3]

Upset range, 0 to 180 inches above instrument zero [0.3], (527.5) no. [0.3]

Shutdown, 0 to 400 inches above instrument zero [0.3], (527.5) no. [0.3] (3.0)

REFERENCE
WNP-2 System and Procedure Vol. 1, Nuclear Boiler Instrumentation, Data Sheet 1, pg 55, 56, & 57

ANSWER 6.02 (3.50)

Inside
1. Containment Area Temperatures
2. RPV Head Seal Leakoff
3. Valve Stem Leakage RRC-V-60 A&B
4. Drywell Equipment and Floor Drain Sumps (4 @ 0.5 ea) [2.0]

Outside
1. Reactor Building Equipment and Floor Drain Sumps
2. Area Leak Detectors - Various Reactor Bldg, Steam Tunnel and Turbine Bldg area temperatures
3. System Isolation - RHR, RWCU, RCIC and MSIV's (3 @ 0.5 ea) [1.5] (3.5)

REFERENCE
System and Procedure Vol. I, Leak Detection Systems, pg 2.

systems INSIDE	systems OUTSIDE
RPV Head seal	RHR
RRC (valve leakage)	RWCU
RCC	RCIC
RWCU	MSIV.

ANSWER 6.03 (3.00)

a. The different position of the IRM range switches provide the attenuation necessary for a particular IRM channel to cover six decades of reactor power indication. (0.5)

b. 1. 24 [0.33]
   2. 2.4 [0.33]
   3. none [0.34] or Yes (white light on Panel 603) (1.0)

c. 1. Detector High Voltage Low [0.5]
   2. Module Unplugged [0.5]
   3. IRM Mode Switch not in operation [0.5] (1.5)

REFERENCE
WNP Systems and Procedures Vol. II, IRM pg 14, & 23, and figure 10.

ANSWER 6.04 (3.00)

a. 1. Manual initiated downshift because of low reactor power operations.
   2. Low total feedwater flow (<30%) with a 15 sec time delay.
   3. Low reactor vessel water level (level 3).
   4. Main steam line/pump suction line differential temperature high. (<9.9°F)
   5. Turbine trip or governor valve fast closure (with power greater than ~~20%~~ 30%). (> 142 psig Turbine 1st stage Press.) [5 @ 0.3 ea] (1.5)

b. Any of the downshift signals will trip RPT-3A(B), allowing the pump to start coasting down. As the pump coasts down, the LFMG comes up to rated speed and voltage [0.4]. Breakers 2A(B) for each pump will close when pump speed is between 20-26% [0.4] and the other close permissives are met. (Any time the low speed transfer is initiated, the recirculatoin loop flow controller automatically shifts to the manual mode ~~[0.5]~~.) When the low speed transfer sequence is activated, an incomplete transfer sequence timer starts [0.3]. If the pump is not between 20-26% speed, or breaker 2A(B) is not closed after 40 seconds, the incomplete transfer relay trips breaker 1A(B) [0.4]. ~~[5 @ 0.3 ea]~~ (1.5)

REFERENCE
WNP-2, System and Procedures, Vol. I, Reactor Recirculation System, Fast to Slow Speed Transfer Sequence, pg 34.
Vol. V Feedwater System, Sec VII Interlock/Control Actions "M" pg 23

ANSWER 6.05 (3.00)

a. 1. Setting speed demand
Setting acceleration values
Generating speed holds

2. Setting Loading rates
Generating load hold signals [5 @ 0.2 ea] [1.0]

b. Mode 1 - Reactor Start - the DEH has little control other than maintaining the setpoint for BPV operation, or maintain plant pressure if the startup is placed in a hold.

Mode 2 - Turbine Start - the turbine is latched rolled off the turning gear, accelerated to synchronous speed, and turbine control is transferred from the throttle valves to the governor valves.

Mode 3 - Turbine Load Control - starts when the main generator output breakers close. Load is pick-up until BPV just closed.

Mode 4 - Turbine Follow Reactor Manual - the 10% bias to the load control signal. A 3% closing bias is applied to the bypass valves. The turbine is slaved to the reactor. [4 @ 0.5 ea] [2.0] (3.0)

REFERENCE
WNP-2, System and Procedure, Vol V DEH L.P., pg 22, 23, and 57-62.

ANSWER 6.06 (3.00)

1. HPCS pump - start signal.
2. CST Suction Valve (HPCS-V-1) - open signal, if Suppression pool suction valve (HPCS-V-15) is not fully open.
3. The injection valve (HPCS-V-4) - open signal.
4. The CST flow test valve (HPCS-V-10 & 11) - close signal.
5. The suppression pool test flow valve - close signal.
6. Division 3 diesel generator - start signal

[6 @ 0.5 ea] (3.0)

REFERENCE
WNP-2 System & Procedure HPCS LP, pg 7.

ANSWER 6.07 (3.50)

a. 1. Low pressure in piping upstream of the FCV. (0.5)
   2. High pressure in piping downstream of the FCV. (0.5)

b. No, the orifice bypass valve is opened when the influent pressure is low. This would increase the flow rate. The purpose of the orifice is to limit flow rates to the condenser or radwaste when the influent pressure is high. — Press. ok, if candidate associates press. & flow (1.0)
(Also acceptable: 1. Because it violates plant operating procedures and 2. Because it will "starve" the RHX and increase F/D influent temperature).

c. 1. Low Reactor Water-Level 2
   2. RWCUS Inlet and Outlet High Flow Differential
   3. High Ambient Temperature in RWCU Equipment Room
   4. High dT across Equipment Room ventilation ducts
   5 RWCU Pipe Area High Temp (4 required @ 0.25 each) (1.0)
   6. RWCU/RCIC Pipe Area High Temp.

d. Service air is used in the filter-demineralizer backwash operation of the precoat evolution. Could not back wash (0.5)

REFERENCE
WNP2 Systems and Procedures Training
RWCU L.P., pg. 6,7,14,16, & 22, 24

ANSWER 6.08 (3.00)

a. RCIC will not initiate, [0.25] the RCIC steam stop valve (RCIC-V-45) will not open if the exhaust valve (RCIC-V-68) is not full open [0.75]. (1.0)

b. The turbine will trip on overspeed, [0.25] the ramp generator is usually the low signal which controls the turbine on quick starts with this signal high the turbine is up to speed before sufficient oil pressure is available to the governor valve to close it [0.75]. (will remain open) (1.0)

c. RCIC will inject at maximum rate and the min flow valve will not respond, [0.25] the flow signal is at minimum due to the zero d/p sensed therefore demanding Max. flow from the RCIC system [0.75]. (1.0)

REFERENCE
WNP-2 System & Procedures Vol III RCIC L.P. pg 11, 12 & 13

ANSWER 7.01 (3.00)

a. Rotor temp is not directly measured. 1st stage metal temp is used as a good approximation. (1.0)
b. 3.9 hours (3 hours 55 minutes) [3.8-4.0 hrs acceptable] (1.0)
c. 10% = 90 F 1st stage temp
   95% = 285 F 1st stage temp
   285 F - 90 F = 195 F
   195 F from figure 2 (of fig 8) 1.3 hours
   power change 95% - 10% = 85%
   85% / 78 min = 1.09 or 1.1% / min (Full credit for answer without showing work. No part credit unless work is shown)

REFERENCE
WNP-2 Training Handout, Vol. V DEH, pg 53, fig 7 & 8

ANSWER 7.02 (1.50)

One SRM in the core quadrant in which the core alteration is taking place, [0.75] and another in an adjacent quadrant. [0.75] (1.5)

REFERENCE
WNP-2 Plant Procedure Manual, 2.1.2 Neutron Monitoring System pg 2

ANSWER 7.03 (2.00)

1. Slowly reduce seal purge injection flow (maintaining seal temp per procedure). Also accept - stop purge to the pump (1.0)
2. Maintain or establish RWCU flow from the loop. (1.0)

REFERENCE
WNP-2 PPM 2.2.1 Reactor Recirculation System pg 6

ANSWER 7.04 (2.00)

Any operation that changes the RWCU non-regen hx heat load would affect RCCW temperature and may cause a undesirable thermal transient to the recirc pump seals. (2.0)

ANSWERS -- WNP-2 -85/05/29-MORGAN, T.

REFERENCE
WNP-2 PPM 2.2.1 Reactor Recirc Sys pg 4

ANSWER 7.05 (1.00)

1. RSCS display
2. GDS display Graphic Display
3. CRD position printout (computer)
4. Full core display [4 @ 0.25 ea] (1.0)

REFERENCE
WNP-2 PPM 3.3.1 Scram Recovery pg 3

ANSWER 7.06 (1.00)

The flow is returned to the RPV to repressurize the feed water piping. to prevent excessive thermal stress and/or water hammer (1.0)

REFERENCE
WNP-2 PPM 3.3.1 Reactor Scram Recovery, pg 2

ANSWER 7.07 (3.00)

1. Depress continuous in button
2. If Rod motion stops then reduce power to ~65% with recirc flow
3. Insert deep rods to maintain power below flow-biased APRM scram
4. If rod motion continues mannally scram the reactor before heat flux exceeds 105% or an APRM scram set point
5. Refer to T.S. 3.1.3.1

[4 @ 0.75 ea] (3.0)

REFERENCE
WNP-2 PPM 4.1.1.1 Rod Drift pg 2 & 3

ANSWERS -- WNP-2 -85/05/29-MORGAN, T.

ANSWER 7.08 (2.00)

(If neither CRD pump starts and) When a second accumulator trouble alarm is received [1.0] scram the reactor by placing the mode switch in shutdown. [1.0] (2.0)

REFERENCE
WNP-2 PPM 4.1.1.2 Complete Loss of CRD Drive Flow, pg 2

ANSWER 7.09 (2.50)

1. One or more of the suppression chambers drywell vacuum breakers are not operable and closed.
2. One or more of the Reactor Building suppression chamber vacuum breakers are not operable and closed.
3. The drywell or suppression chamber oxygen concentration greater than 3.5% by volume during operational condition 1 after 24 hours from the time the reactor power exceeded 15% and not within 24 hours of reducing reactor power to less than 15% preliminary to a scheduled reactor shutdown as indicated on containment oxygen recorders CMS-O2R-1 and CMS-O2R-2 at Panel K1 and K2.
4. The primary containment air lock or associated doors are not operable.
5. One or more of the primary containment equipment hatches are not closed and sealed.
6. One or more of the sealing mechanisms associated with the primary containment penatration is found to be inoperable.
7. Primary containment leak rates are too high as defined in Technical Specifications 3.6.1.2, 3.6.2.1.e or 3.6.3.
8. Primary containment automatic isolation valve not operable.
9. other items listed in loss of containment integrity in Tech. Specs.

REFERENCE
WNP-2 PPM 4.3.1.1 Primary Containment Not Operable, pg 1

ANSWERS -- WNP-2 ~~85/05/29~~ MORGAN, T.

ANSWER 7.10 (4.00)

a. 1. Main turbine trip @ vacuum <19" Hg
2. MSIV Closure @ vacuum <10"Hg
3. BPV Closure @ <7" Hg.
4. RFPT Trip @ 0" Hg.
5. Low Vacuum alarm @ 25" Hg.
(4 required @ 0.5 each) (2.0)

b. 1. Rapidly reduce reactor power with recirculation flow
2. Verify normal gland steam seal header pressure of 200 psig
3. Place standby set of air ejectors in service
4. If reactor power <5%, start mechanical vacuum pump
(4 required @ 0.5 each) (2.0)

REFERENCE
WNP-2 PPM AP 4.6.5.1 Loss of Condenser Vacuum pg 1 & 2

ANSWER 7.11 (3.00)

a. 1. Transfer RPS bus A to the Alternate A power supply and reset scram and isolation signals.
2. If all cooling (CRD and RCCW) was lost (and cannot be restored within 60 seconds) to the Reactor Recirculation Pumps (RRC-P-1A and 1B)
o Place the Reactor Mode Switch in Shutdown.
o Trip both Reactor Recirculation Pumps.
o Refer to PPM 3.3.1, "Reactor Scram"
3. If CRD cooling is still available to the Reactor Recirculation Pumps THEN shift both Reactor Recirculation Pumps to the 15 Hz MG sets and balance recirculation loop flows if necessary.
4. If the reactor has scrammed THEN carry out PPM 3.3.1, "Reactor Scram".
5. Notify plant personnel and the Dittmar Load Dispatcher.
[~~5~~ 4 @ ~~0.4~~ 0.5 ea] (2.0)

b. 1. The loss of RPS bus 1 deenergizes NSSSS
2. NSSSS deenergizing trips CW pump 'C'
3. The loss of CW pump 'C' causes a loss of condenser vacuum
4. The loss of condenser vacuum causes Reactor scram~~s~~ ~~the reactor~~ ~~(not a direct scram)~~
• ON high pressure.
[4 @ 0.25 ea] (1.0)

REFERENCE
WNP-2 PPM AP4.7.1.8 Loss of Power to SM-7 pg 1, 2 & 3

ANSWER 8.01 (3.00)

1 - Shift Manager - SRO
1 - Control Room Supervisor - SRO
2 - Reactor Operators - RO
2 - Equipment Operators - None
1 - Shift Technical Advisor - None
(number [0.2] title [0.2] license [0.2]) (3.0)

REFERENCE
WNP-2 Tech Specs Table 6.2.2-1 pg 6-6

ANSWER 8.02 (3.00)

1. An individual should not be permitted to work more than 16 hours straight, [0.5] excluding shift turnovers. (0.5)
2. An individual should not be permitted to work more than 16 hours in any 24-hour period [0.5], nor more than 24 hours in any 48-hour period [0.5], nor more than 72 hours in any 7-day period [0.5], all excluding shift turnover time. (1.5)
3. A break of at least 8 hours should be allowed between work periods [0.5], including shift turnover time. (0.5)
4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift. (0.5)

REFERENCE
WNP-2 Tech Specs Sec 6.2.2.f pg 6-2

ANSWER 8.03 (2.50)

No, [0.5] A maximum allowable extension is not to exceed 25% of the surveillance interval or 7.75 days [1.0] AND a total maximum combined interval for any three consecutive tests not to exceed 3.25 times the specified surveillance interval. [1.0] (2.5)

REFERENCE
WNP-2 Tech Spec 4.0.2 Surveillance Requirements pg 3/4 0-2

ANSWER 8.04 (2.50)

a. Safety Limits are limits upon important process variables below which the reasonable maintenance of the cladding and primary systems are assured. LSSS's are settings on instrumentation which initiate automatic protective action at a level such that the Safety Limit will not be exceeded. (1.5)

b. The LCO's specify the minimum acceptable levels of system performance necessary to assure safe startup and operation. When these conditions are met, abnormal situations can be safely controlled. (1.0)

REFERENCE
WNP-2 Tech Spec SL & LSSS Bases pg B2-1 & 3/4 0-1

ANSWER 8.05 (3.00)

a. o Direct Radiation >/= 2.5 mrem/hr [0.5]
o Airborne Radioactivity >/= 25% of MPC [0.5]
o Contamination >/= 1000 dpm/100 cm2 Beta Gamma or
>/= 100 dpm/100 cm2 alpha [0.5] (1.5)

b. To determine the effects of the task on the overall plant and the effect of other plant parameters on the task to be performed. (1.5)

REFERENCE
WNP-2 Health Physics Program Discription HPD 3.1.8 Radiation work permit pg 2

ANSWER 8.06 (3.00)

a. The worker's supervisor or Department Manager can initiate the release after satisfying himself that equipment and personnel safety will not be jeopardized. As a last resort, the Shift Manager has the authority to initiate a clearance release. (1.0)

b. The change needs to be initiated by the Shift Manager to signify his approval. (1.0)

c. Redundent Verification is that requirement directing a second knowledgeable individual to make an independent verification of correct equipment status (for safety related and fire protection equipment.) (1.0)

REFERENCE
WNP-2 Equipment Clearance and Tagging Procedure 1.3.8 pg 2, 5 and change # 84-1058

ANSWER 8.07 (3.00)

a. Operators are not to override the automatic actions of ECCS and other safety features, unless continued operation of such will result in unsafe plant conditions. or two independent indications of a misoperation and adequate core cooling is assured. (0.75)

b. An operator may place a controller in the manual mode from the automatic mode whenever, in the operator's judgement, continued automatic operation is undesirable. (0.75)

c. When a safety related motor operated valve has been manually seated or back seated, the valve shall be declared inoperable until motor operation can be demonstrated. (0.75)

d. Instructions for aligning more than 2 valves or circuit breakers should be written on a Component Status Change Order and carried by the operator performing the change unless the operation is performed using the procedure or checklist. (0.75)

REFERENCE
WNP-2 Standing Orders/Night Orders, 1.3.1 Att 1 pg 2-5

ANSWER 8.08 (3.00)

a. 1. Prior to reactor criticality after completing Core Alterations that could have affected the control rod drive coupled integrity,
   2. Anytime the control rod is withdrawn to the "Full Out" position in subsequent operation,
   3. Following maintance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

[3 @ 0.5 ea] (1.5)

b. Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position [1.0] and then verifying that the control rod drive does not go to the overtravel position [0.5]. (1.5)

REFERENCE
WNP-2 Tech Spec 4.1.3.6 Reactivity Control Systems, Surveillance requirements pg 3/4 1-12

ANSWER 8.09 (2.00)

a. Plant Emergency Director (0.4)

b. Plant Manager (0.4)
Assistant Plant Manager (0.4)
Operations Manager (0.4)
Shift Manager (0.4)

REFERENCE
EPIP 13.1.1, pg 1 & 2; 13.1.2 pg 1