U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Examination Report No. 50-397/OL-86-01

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~~.
2/4-6/86

Chief Examiner: Robert J Pate 4/22/86
R. J. Pate, Chief Date Signed
Reactor Safety Branch

Approved: J. O. Elin 4/22/86
J. O. Elin, Chief Date Signed
Operations Section

Summary:

Examinations on ~~November 6-8, 1984~~ 2/4-6/86

Written and operating (oral and simulator) examinations were administered to four SRO and seven RO candidates. All RO and SRO candidates passed the examinations.

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REPORT DETAILS

1. Examiners

   *Lenord Wiens, NRC
   L. Miller, NRC
   R. Pate, NRC
   M. King, EG&G
   J. Sherman, EG&G

   *Chief Examiner

2. Examination Review Meeting

   An exam review meeting was held after the written exam was administered on February 6, 1986. The facility comments and subsequent Region V responses are attached.

3. Exit Meeting

   At the conclusion of the site visit February 6, 1986, the examiners met with representatives of the plant staff to discuss the results of the examinations. Those individuals who clearly passed the oral and simulator examination were identified in this meeting.

   The current status of the plant simulator was discussed. The simulator was found to be very limited in the number of malfunctions that could be simulated. The examiners noced that the WNP-2 simulator was marginally acceptable and the problem appeared to need prompt and effective management attention.

   a. Attendees were:

      NRC

      Robert Pate, Chief, Reactor Safety Branch
      Lenord Wiens, Senior Reactor Engineer, OLB HQ
      Lee Miller, Training and Assessment Specialist
      Mike King, Examiner, INEL
      Jeff Sherman, Examiner, INEL

      Utility

      John Wyrick, Licensed Training Manager
      Jack Baker, Assistant Plant Manager, WNP-2
      Lou Frank, Principle Training Specialist, WNP-2
      Bob Beardsly, Assistant Operations Manager, WNP-2

   b. The examiner reported that there were four 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.

WNP-2 FACILITY COMMENTS AND RESOLUTION
REACTOR OPERATOR EXAMINATION GIVEN ON 2/4/86

1. **Facility Comment on Question 1.02**

   Also give credit for using doubling count rate the new Keff is half the distance to one

i.e.	100	200 cps	Keff	.95	Keff	.975
	200	250 cps	Keff	.975	Keff	.981

   Examiner Resolution

   Comment rejected, because method only works for a one-step doubling, not for this situation.

2. **Facility Comment on Question 1.03**

   Stating half life of the longest lived precursor was not asked in the question, and should not be required for full credit.

   Examiner Resolution

   Agree with comment and answer key changed accordingly.

3. **Facility Comment on Question 1.04**

   Section d. Also accept increases - due to less flow losses see attached G.E. HTTFF pages 7-94, 7-95.

   Examiner Resolution

   Correct answer is changed on the answer key to **INCREASES** for part 'd'.

4. **Facility Comment on Question 1.05**

   Also accept the following answer from G.E. THTFF. (See attached page 9-51)

   Examiner Resolution

   Comment rejected, because the question asks **WHY** is flow orificing necessary, not **HOW** it is accomplished. For full credit, answer must include the flow starvation effect of increased voiding on higher power fuel bundles.

5. **Facility Comment on Question 1.07(b)**

   Also accept "resonance absorber build-up causing more resonance capture." Question did not state "list the isotope."

Examiner Resolution

The facility comment is correct, but the question asks for the primary effect. PU-240 build-up is the correct answer.

6. **Facility Comment on Question 1.08**

Accept for part b "any number less than 2%". To memorize values for a table to 1/10 of a percent is unrealistic. Also there is no direct "decay heat" meter or indicator in the control room.

Examiner Resolution

The answer key was changed to accept 0.5% to 2.0% for part b; this increases the range of acceptable answers.

7. **Facility Comment on Question 1.10**

Accept 57°F ±1°F question did not ask to determine cooldown rate to nearest 1/00 ±°F. Also in changing PSIG to PSIA accept use of 15# vice 14.7#.

Examiner Resolution

Answer key changed to accept a wider more realistic range and 15 psi.

8. **Facility Comment on Question 1.12**

a. The "why" section of the question does not ask the student to state two reasons why. "Tripping off line" should not be required for full credit. Should accept any one of the three.

Ref. Examiners Stand ES 202 #18 open ended question's should be avoided.

Examiner Resolution

Will accept over-heating, electrical damage, or tripping off-line for full credit.

9. **Facility Comment on Question 1.13**

Comment - delete the question. Question not covered by learning objectives and can not find the answer in the stated reference.

Examiner Resolution

Comment accepted, question and point value (2.00) deleted. Answer can't be referenced in WNP-2 documents. Section 1 becomes 22.00.

SAMPLE PROBLEM:
(Continued)

1.4

Solution:

The total NPSH on the recirculation pump is calculated by first determining the inlet pressure P_i in Equation 7-43.

$$P_i = P_{dome} + P_{H_2O} - P_{losses}$$

where:

P_{dome} = pressure as measured in the steam dome (lbf/ft^2)

P_{H_2O} = pressure due to the water column (lbf/ft^2)

P_{losses} = pressure loss due to irreversible flow losses (lbf/ft^2)

The dome pressure is 1000 psia. The pressure due to the height of the water column is the density of water times the height of the column (plus a change in units to lbf/ft^2). The density of the water is taken as the saturation density of 20-Btu/lb-subcooled water ($47.3\ lb/ft^3$) which can be found from a table of subcooled water properties. The irreversible losses are a function of the square of the fluid velocity and the effects of elbows, pipe fittings, and suction valve in the recirculation pump suction line. It is normally 20 psia at rated conditions and decreases as the square of the flow rate. Equation 7-43 becomes:

$$P_i = 1000\ lbf/in^2 \times 144\ in^2/ft^2 + \frac{[illegible]\ ft \times 47.3\ lbm/ft^3 \times 32.2\ ft/sec^2}{32.2\ \frac{ft\text{-}lbm}{lbf\text{-}sec^2}}$$

$$- 20\ lbf/in^2 \times 144\ in^2/ft^2$$

$$P_i = 144{,}000\ lbf/ft^2$$

SAMPLE PROBLEM:
(Continued)

The saturation pressure is found in the following way. First, find the saturated liquid enthalpy at 1000 psia. This turns out to be 542.6 Btu/lbm. But since the water in the downcomer is subcooled 20 Btu/lbm by the feedwater, the actual enthalpy at the eye of the pump is 522.6 Btu/lbm. The saturation pump which corresponds to this liquid enthalpy if approximately 875 psia.

$$P_s = 875 \frac{\text{lbf}}{\text{in}^2} \times 144 \frac{\text{in}^2}{\text{ft}^2} = 126{,}000 \frac{\text{lbf}}{\text{ft}^2}$$

The NPSH is then:

(7-43) $$\text{NPSH} = (P_i - P_s) \times \frac{g_c}{\rho\ g}$$

$$\text{NPSH} = \frac{(144{,}000 - 126{,}000) \frac{\text{lbf}}{\text{ft}^2} \times 32.2 \frac{\text{lbm-ft}}{\text{lbf-sec}^2}}{32.2 \frac{\text{ft}}{\text{sec}^2} \times 62.4 \frac{\text{lbm}}{\text{ft}^3}}$$

$$\text{NPSH} = \frac{18{,}000\ \text{lbf/ft}^2}{62.4\ \text{lbf/ft}^3}$$

$$\text{NPSH} = 288 \text{ ft of } H_2O$$

The contribution of the water column above the pump can be calculated separately by first assuming no subcooling and neglecting the suction line head loss.

From G.E reference: Thermodynamics Heat Transfer and Fluid Flow

CORE ORIFICING

We mentioned that for single or two-phase flow, the constant term k represented a resistance due the inlet orifices which are placed in the fuel support pieces in the core support plate. One might ask why would we artificially and intentionally create a flow restriction?

1.5

To obtain a qualitative picture of the effect of core inlet orifices, first consider the BWR core without inlet orifices. The pressure drop across all the fuel assemblies is the same since, as we said, they share a common inlet and outlet plenum. Assume further that all the bundles have the same flow resistance characteristics so that, at zero power and minimum recirculation system flow, all the fuel bundles have the same flow.

Now increase core power as in a normal startup where there are some high powered bundles and some low powered bundles. As bundle power reaches the point of increasing water temerature in the channel, the bundle flow will increase. This occurs because the hotter water in the channel is less dense than the water in the downcomer region and gravity will cause an increase in flow in the warmer bundles. In addition, as boiling begins, the buoyant force of the steam bubbles will cause a further increase in bundle flow.

As power continues to increase, however, the channel quality in the highest powered bundle increases as does the two-phase flow friction multiplier $\phi^2_{2\phi}$ (See Figure 9-20). The result is a large increase in flow resistance as quality increases. Since the channel pressure drop is controlled by the inlet and outlet plenums (i.e. for constant ΔP), equation 9-20 indicates that the flow through the fuel bundle will decrease as R increases. The result is that flow which should go to the highest powered bundle is being diverted to lower powered bundles. That is, the flow seeks the path of least resistance. This is, of course, undesireable.

Flow orifices are provided at the bundle inlet to minimize the undesireable effect of a quality increase on bundle flow. The inlet orifice has the effect of providing a larger resistance to flow so that any additional flow resistance caused by two-phase flow is acceptably small in comparison.

There is a classic analogy to this effect which can be illustrated by the simple electrical example shown in Figure 9-22.

1.8

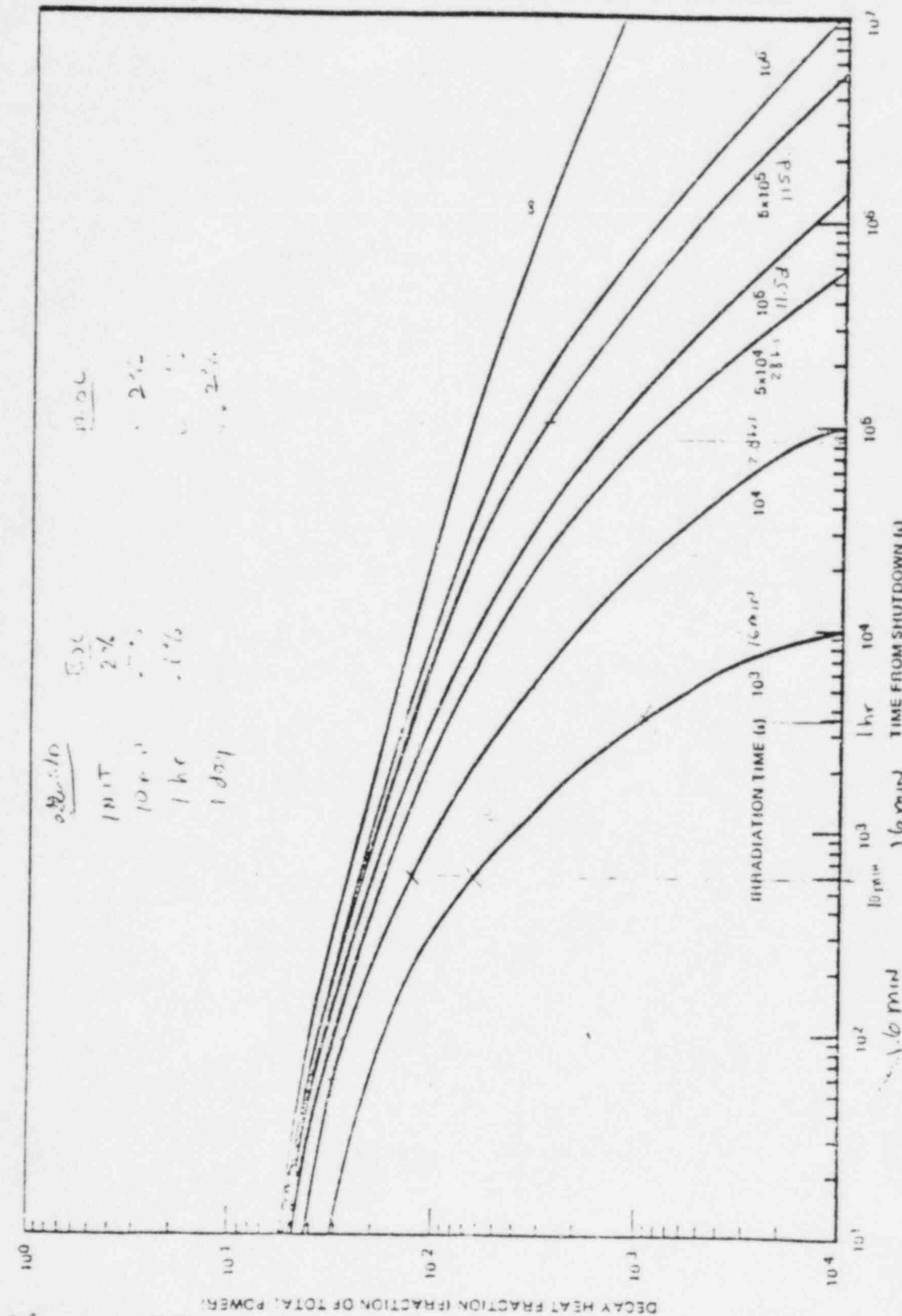

Figure 2.3 Decay Heat Fraction as a Function of Length of Irradiation and Time from Shutdown

3. The examination should include questions to determine a candidate's understanding of his responsibilities related to the administrative procedures, precautions, environmental and radiation release requirements, and pressure/temperature limits.

4. Questions on health physics and chemistry procedures should be determined on the basis of the facilities' type of health physics coverage.

5. Extended definitions questions (e.g., 6-factor formula) should be avoided.

6. Questions on detailed system characteristics or instrumentation, such as annunciator logic or setpoints, should be avoided unless required for safety system operations.

7. Questions should be based on

   a. a review by the examiner of material provided by the facility
   b. a review of past examinations given at the facility
   c. content validity study results, when available

8. Other sources of questions are

   a. standard questions and answers
   b. Examination Question Bank
   c. examinations on similar facilities
   d. personal file of questions and answers

9. A rule of thumb is

   a. approximately 55 to 70 responses for a 6-hour examination
   b. a response that requires about 3 to 4 minutes to write

10. Examinations shall be 6 hours long.

11. Examination questions should consist of short word sentences using the terminology of the facility as much as practicable.

12. "Discuss"-type questions should be avoided; questions should be specific to elicit short precise answers.

13. Practical realistic questions that relate to operator knowledge and required operating practice should be used.

14. Multipart questions should be broken down into logical sequential parts. The answer sheet should show points assigned for subparts of answers.

10. **Facility Comment on Question 2.03**

b. This requires the operator to memorize switch positions - which are labeled and would definitely be referred to where any switch manipulation is required. This switch is covered in our training material but memorization of each switch position is not required by our learning objectives.

Examiner Resolution

Agree with comment and the switch position portion of answer is deleted for full credit. Interpretation of meter reading is still required.

11. **Facility Comment on Question 2.05**

a. Should also accept - prevent exceeding design external to internal containment P (2 psid) (for any reason - wouldn't have to be restricted to "condensing steam").

Examiner Resolution

Disagree with comment. The design purpose, as stated in the reference, is to prevent a vacuum in the primary containment which would occur while condensing steam.

12. **Facility Comment on Question 2.07**

b. RCIC should also be accepted as a system redundant to HPCS see attached T.S. page B 3/4 5-2. Operators are trained to utilize RCIC as a backup to HPCS. Also recognized in T.S. 3.5.1 in Div. 3 ECCS.

Examiner Resolution

Agree with comment and the answer key is changed to accept either ADS or RCIC.

13. **Facility Comment on Question 2.08**

Answer #1 "RCIC equipment area and/or pipe routing area high temp" should be accepted as 2 separate signals if so listed.

Examiner Resolution

Agree with comment and will give credit for two separate signals, if so listed.

14. **Facility Comment on Question 2.09**

b. The stop control for HPCS in the control room is a switch, not a push button.

ECCS - OPERATING and SHUTDOWN (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 516/1550/6350 gpm at differential pressures of 1160/1130/200 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

2.7

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety/relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected safety/relief valves although the safety analysis only takes credit for six valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS, and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITION 1, 2, or 3 is required by Specification 3.6.2.1.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

(2.7)

3.5.1 ECCS divisions 1, 2, and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

3. Seven OPERABLE ADS valves.

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

2. Seven OPERABLE ADS valves.

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*#, and 3*.

*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 128 psig.

#See Special Test Exception 3.10.6.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:

1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.

2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. For ECCS division 2, provided that ECCS divisions 1 and 3 are OPERABLE:

1. With either LPCI subsystem "B" or "C" inoperable, restore the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 7 days.

2. With both LPCI subsystems "B" and "C" inoperable, restore at least the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

c. For ECCS division 3, provided that ECCS divisions 1 and 2 and the RCIC system are OPERABLE:

1) With ECCS division 3 inoperable, restore the inoperable division to OPERABLE status within 14 days.

2) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE:

1) With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

Examiner Resolution

Comment noted, but this does not change the answer to the question.

15. **Facility Comment on Question 2.10**

a. Answer #1 also accept "undervoltage" on associated bus.

Examiner Resolution

Facility comment is correct and answer annotated to accept undervoltage.

16. **Facility Comment on Question 2.11**

RWCU pump no longer trip on high RCC temp., they would trip on V-4 not being full open. Also FCV-33 auto closes when V-4 (or V-1) goes closed. This response should also be accepted.

Examiner Resolution

Comment was verified to be correct. The RWCU pumps will not trip directly on high RCCW temperature, but they will trip indirectly as a result of V-4 closing due to high temperature at NRHX outlet. FCV-33 auto closes when V-4 closes.

Answer is changed to:

1. Affected components will be non-regenerative heat exchangers [0.25], V-4 [0.25], reactor water cleanup pumps [0.25], and FCV-33 [0.25].

2. High temperature at NRHX outlet will cause isolation valve V-4 to close. [0.5] V-4 closure causes RWCU pump trip [0.25] and FCV-33 closure [0.25].

Due to facility comments, reference is changed to: WNP-2 Systems, Volume I, Tab. 9, pp. 7-8.

17. **Facility Comment on Question 2.12**

Answer #1 "Feedflow < 30% w/ 15 sec. T.D." Time delay should not be required for full credit. Question asks for setpoints only.

Answer #4 ">142 #turbine press." is when the trip is, available, not the setpoint at which the trip occurs Setpoint is: when the throttle valves are not full open or upon low EH fluid pressure <1250#. Ref. T.S. 3/4 3-44.

Examiner Resolution

Answer #1: The time delay is part of the condition and is required for full credit.

2. If a high pressure develops in the piping <u>downstream</u> of the FCV, a high pressure setting on PS-14 will close the FCV at 140 psig.

3. Closure of either V-1 or V-4 will cause FCV-33 to close. FCV-33 closes to prevent system depressurization which causes the hot water in the system to flash to steam (the water flashing to steam and resulting water hammer when flow is re-established could possibly damage the system piping or heat exchangers).

   When V-1 and V-4 are both open, FCV-33 will automatically reopen to the position determined by its manual controller on P602.

2.11

C. The following will cause a pump trip:

   1. Inlet isolation valves (V-1 or V-4) not full open

   2. Low system flow as sensed by FE-35 - 70 gpm

D. Filter Demineralizers

   1. The filter-demineralizers each have a flow control valve (air operated) which senses effluent flow and maintains a constant flow rate through the vessel for varying pressure drops across the filter. The vessel dP should range from a low of 1 psid to a high of 20-25 psid (Figure 3).

   2. The flow recorder control stations 74A and 74B start and stop their respective vessel hold pump. The hold pump will start at less than or equal to 100 gpm and stop at greater than 100 gpm. (FRCS 74A and B are located on Panel 26 in the radwaste control room).

Agree with comment concerning answer #4. Answer #4 deleted and replaced with the following:

4. Turbine throttle valve-closure $\leq$5% closed.

5. Turbine Governor valve - fast closure $\geq$1250 psig.

Add to existing reference:

WNP-2 Technical Specifications 3/4 3-44.

18. **Facility Comment on Question 2.13**

b. Should also accept - possible RCIC overspeed due attempting max. flow. Also, the minimum flow valve does not receive its flow signal from the same F transmitter and therefore is not affected by this failure.

Ref. RCIC GE Elec.

Examiner Resolution

Some of the facility's comment is correct, therefore, will accept possible RCIC overspeed due to attempting maximum flow. However, the RCIC flow control transmitter (FT-3) and RCIC min flow valve flow switch (FIS-2) are separate, but are arranged in parallel such that a break in the D/P cell on FT-3 will cause a zero D/P to be seen on both instruments. The min flow valve will remain open during this failure. Additional reference: WNP-2 Drawing M519 (RCIC System)

General Comment: This section (03) was very well written. Questions were clear and concise, answers were brief but complete. Every question was something an operator should know!

19. **Facility Comment on Question 3.04**

b. Placing the master controllers to manual does not reset the "setpoint setdown", it would take manual control of the feed pump, but taking manual control of a feed pump controller (601A and/or B) would also give you manual control of the feed pump speed. I would not expect 2 actions or for the second action I would accept takng manual control of any of the feed pump controls.

Examiner Resolution

The facility comment would achieve the desired effect. Will accept manual control of individual feed pump in lieu of master controller placed in manual in part b.

2.12

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Throttle Valve-Closure	$\leq$ 5% closed	$\leq$ 7% closed
2. Turbine Governor Valve-Fast Closure	$\geq$ 1250 psig	$\geq$ 1000 psig

20. **Facility Comment on Question 3.05**

c. When level error matches steam-feed flow error part of answer was not asked, nor should be required for full credit.

Examiner Resolution

The comment is noted, however, this part of answer required to complete the description as to why level stops decreasing.

21. **Facility Comment on Question 3.08**

Also accept word description instead of valve numbers.

V-123 = under piston or insert drive wtr valve
V-121 = over piston or insert exhaust wtr valve
V-122 = over piston or withdraw drive wtr valve
V-120 = under piston or withdraw exhaust wtr valve

Examiner Resolution

Will accept either word description or valve numbers.

22. **Facility Comment on Question 3.10**

b. We don't have a "thumbwheel mode selector switch" for bypassing LPRM's. We have a small toggle switch inside the associated APRM cabinet. Also, for answer number 3 in this part you should accept "bypass lite indication of the full core display (P603)". This is the correct terminology.)

Examiner Resolution

Comment on bypass switch noted. Comment on bypass light indication rejected. The light indication on P603 is the four rod display (same as answer 2).

23. **Facility Comment on Question 4.03**

b. Correct answer = None RCIC auto shift on low CST level (Ref. Volume III, Tab 3 P 19) however, this is a misleading question.

Examiner Resolution

Agree with facility comment. The examiner was in error. Part b of question was deleted and section 4 point value reduced to 25.0.

24. **Facility Comment on Question 4.04**

Should accept any 5 of the 11 steps of this PPM. These 11 steps are generally treated as immediate actions - however none of these steps are defined as Immediate Actions since this is not an abnormal procedure.

Examiner Resolution

Agree with facility comment and 5 of the 11 steps required for full credit.

25. **Facility Comment on Question 4.05**

Should give full credit even if... "including the public.." is not included, since injury to personnel is all i.clusive - includes any persons public or employees.

Examiner Resolution

Agree with comment. The words "including the public" not required for full credit.

26. **Facility Comment on Question 4.06**

a. Also accept (P&RTS bases, 3/4 4.1), "ensures adequate core flow coastdown following a LOCA."

b. Any answer that refers to positive reactivity addition should be accepted since no reason is given in the precaution section.

Examine Resolution

Agree with facility comments.

a. Full credit will be given for "ensures adequate core flow coast-down following a LOCA".

b. Comment Accepted.

27. **Facility Comment on Question 4.07**

This answer requires memorization of a normal operating procedure - which is not required for ES202 page B.4. Also, if the "Water Leg Pump Discharge Press. Low" alarm is lit - the HPCS pump should not be started.

Ref.: PPM 2.4.4 Pre req. F and Limit. C (rev. 2)

Examiner Resolution

ES 202 part B.4 stated "The candidate is not expected to have normal procedures committed to memory but should be able to explain reasons, cautions, and limitations of normal operating procedures." The question refers to a limitation on operation of the HPCS.

SAFETY RELATED CONTROLLED COPY

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

PLANT PROCEDURES MANUAL

WNP-2

PROCEDURE NUMBER	APPROVED	DATE
*4.601.A1-6.7	J. Martin	8/22/84

VOLUME NAME	
4	ABNORMAL CONDITION PROCEDURES
SECTION	
4.601.A1	ANNUNCIATOR RESPONSE, P601 ANNUNCIATOR A1
TITLE	
*4.601.A1-6.7	HPCS WATER LEG PUMP DISCHARGE PRESSURE LOW

4.3

6-7 WINDOW	SOURCE	AUTOMATIC ACTIONS
HPCS WATER LEG PUMP DISCH PRESS LOW	HPCS-PIS-13 (≤ 53 PSIG)	NONE

1. Verify HPCS Water Leg Pump (HPCS-P-3) discharge pressure less than or equal to 53 PSIG as read on HPCS-PI-13(P601).

2. Verify HPCS Water Leg Pump (HPCS-P-3) running (P601).

NOTE: The following step is designed to prevent pump run-out and possible loss if LOCA should occur when HPCS-P-1 is running in the test mode.

3. If the Water Leg Pump fails when HPCS is required to be operable, start and run HPCS-P-1 in a test mode while maintaining approximately 1,200 psig discharge pressure; maintain these conditions until the Water Leg Pump is made operational.

4. If during a power interruption, the HPCS Water Leg Pump Discharge Pressure Low alarm is received and adequate core cooling is assured, hold HPCS-P-1 control switch in the STOP position or otherwise prevent pump start until the system can be filled and vented.

Mark,

Some people put this as an answer for the HPCS question. (RO test) ~~ME~~ This should also ~~also~~ be considered for at least partial credit.

Dan

B&R: M520

GE ELEM: 807E172TC, sheet 5

C. Vendors Manual

GEK-71334

D. FSAR

Sections 6.3, 7.3, 8.3

2.4.4.3 Prerequisites

A. The Reactor Building Heating, Ventilation and Air Conditioning System in operation to support HPCS System Operation.

B. HPCS service water system available to support HPCS diesel and HPCS system operations.

4.3

C. The condensate storage tanks have the required amount of water to support HPCS operation (7'7" each tank or 13'3" in one tank minimum per Technical Specification 3.5.3).

D. Must have at least minimum fuel supply (30,000 gallons) on site for HPCS diesel.

E. The suppression pool level normal (31 ft. 2 in. to 30 ft. 10 in.).

F. The HPCS pump should not be started when the "HPCS WATER LEG PUMP DISCH PRESS LOW" alarm is lit. The water leg pump is designed to remain in service throughout system operation and during standby status. Refer to Technical Specification 3.5.1.

2.4.4.4 Limitations

A. Observe RWP requirements per PPM 11.2.8.1.

B. The HPCS System shall not be removed from service anytime it is required to be operable by Technical Specifications. See Technical Specification 3.5.1.

3/50

C. If during or following a power interruption the HPCS WATER LEG PUMP DISCH PRESS LOW alarm is received and adequate core cooling is assured, hold the pump control switch in the STOP or otherwise prevent pump start until the system can be filled and vented.

4.7

D. The HPCS System shall be maintained full anytime the system is required to be operable. If water leg pump HPCS-P-3 fails, start and operate HPCS-P-1 on recirculation to the CST until corrective action is completed.

E. The condensate storage and supply system may only be used to flush the HPCS System. It shall never be used to keep the system charged or lined up to an unattended system.

Since the question did not state the cause of the water leg pump failure, if the candidate assumes (and states) the failure is due to power loss, half credit will be given for answering; the HPCS pump switch should be held in STOP or otherwise prevent pump start until the system can be filled and vented.

28. **Facility Comment on Question 4.08**

This response based upon PPM 4.2.1.2 Rev. 2 should be:
A Notify CRS
B Verify Auto Actions
C Take Manual Control of FWLC and reduce RPV level

This is the current PPM in use and given to the licensee candidates prior to the exam.

Examiner Resolution

Comment response will be accepted for full credit and the answer key changed, per reference stated by facility.

29. **Facility Comment on Question 4.10**

4.10 a. No reason is given in PPM 2.4.4 (HPCS) for this limitation - Also our Supp. Pool Chem. results show that it is well within the Chem. Specs. for the RPV.

b. Again - no reason given, however, to prevent overheating of motor windings should be given full credit.

Examiner Resolution

Per ES 202 candidates are expected to be able to explain reasons for limitations in normal operating procedures, whether they are stated in the procedure or not. Facility offers no alternative answers, and therefore the answer key was not changed.

30. **Facility Comment on Question 4.12**

The operators are not required to memorize normal operating procedures. (ES 202 B.9.) These checklists are referred to during each turnover by the RO and memory does not nor should not be relied upon to complete these checklists.

Also this is not required per our Volume 1 PM Learning Objectives. Invalid question. Also the point value is excessive on this question - 3.5 pts 14%. If an operator didn't memorize the turnover checklist he is down to an 86% on Section 4.

Examiner Resolution

Comments noted. ES 202 part B.4. states that administrative procedures may be included to the extent they are directly applicable to an operator. The RO initials the Shift Turnover Checklist each time he/she takes the watch, so it is reasonable to expect the candidate to know what is on the list. Point value is 20% of section, thus meeting requirements of ES-107. Will accept other implicit items included as part of the items listed.

WNP-2 ~~FACILITY COMMENTS~~

5.03 Point total for the answer does not add up to the point value for the question.

5.04 Question did not ask for a discussion of boiling boundary only voids, and core net reactivity. Discussion of boiling boundary should not be required for full credit.

5.06 Tech Specs and our procedure (7.4.1.1.2) define shutdown margin as the amount sub critical with a "cold" "clean" core with highest reactivity rod withdrawn. If the candidate assumes that the stated shutdown margin is cold and clean then he should be given credit for answering that the SDM is acceptable and that there would be no consequences over the subsequent 20 hours. (Note: per PPM 7.4.1.1.2 we do not measure SDM until shutdown for 50 hours and it is corrected for temperature).

5.07 Unable to find answer to part "b" in the stated reference. The statement "the increase frictional resistance lowers the total flow to less than twice the original flow" should not be required for full credit.

Part B - delete [0.5] following "less than double the original flow", otherwise point value is 2.0 for total question.

5.08 Also accept per the abnormal procedure 4.4.4.2 (see attached copies) reactor power and pressure perturbation, reactor vessel level perturbation or the following explanation:

Turbine load - may initially decrease due to spraying cold water on steam exiting the core. If reactor power then increased (without causing a scram due to APRMs) because of colder water being returned to the down comer and flowing into the bottom of the core pressure will increase and turbine load will increase.

Reactor water level indication - may initially decrease due HPCS spray causing a pressure drop (steam condensing, void collapse) and water from the downcomer will flow into the core because of less back pressure. As HPCS continues to inject this effect is overcome by the amount of water injected and reactor water then increases until it is compensate by the FWLC system.

Feedwater flow - may initially increase a small amount due to the indicate water level decrease but as HPCS continues to inject and reactor water level increases the FWLC will then act to lower the feed rate from the feedwater pumps.

As of date, we do not have data for this transcient so it is difficult to predict actual plant response.

In Section 6.0 question #5 is the same as 5.8b, this double jeopardy is not allowed by the examiners standard.

5.11 Unable to find answer for shape of curves drawn on Figure 1 in stated reference for the answer key. Also, this curve is not a standard curve used by the operating unit. And the discuss answer given in the key requires that students have the bases for Tech Spec memorized verbatim which is unrealistic.

Give full credit for either drawing the curves or discussion of effects.

5.12 Section "b" also accept for full credit: "Increasing the flow rate increased the heat removal capability and places the bundle farther from OTB."

6.02 No comment on answer, however - question stipulates a mode (flex auto) of RFC that we do not use at present.

6.03 a. should give full credit for 1) loading in U234 "breeder" material 2) reduction of the "sputtering" effect <u>also</u> this was <u>not</u> required knowledge item as part of the LPRM learning objectives
b. no comment

6.04 "The drive piston"... should not be required for full credit. A similar response such as the "control rod" moves past the "00" position should deserve full credit as well.

6.05 Part ii asks for FWLC response with a HPCS initial at 90% power - this item is not covered in our "Systems training mat'l and the plant response would be dependant upon how fast the FWLC system & RFPs could respond to the increase in level part d could also be acceptable (RPV level until turbine trip). Also this question is also asked in question 5.8c (FW response due to HPCS initial) Double Jeopardy

6.06 Answered Part b the flow elements are located in the pump suction rather than the discharge ref systems manual Vol. 1 Tab 6 Fig. 2.

6.07 Answer Part d should also accept RHR pump 2b will not start* due to suction valve interclock - no suction path, no injection because the pump did not start.

* won't start due to bkr cycle close then open

6.08 a. should also accept - provide a more stable flux signal to minimize FCV ball valve wear due to hunting

6.09 a. also accept - increase to 100% recirc flow

general comment - again- RFC <u>not</u> used in Loop Auto

6.10 "Anticipatory" scrams is very vague all scrams are due "anticipating" further plant problems degraded conditions. We do not classify scrams under this type of category. This questions assumes all other scrams do not anticipate other problems. Should accept any scram signal - with justification.

Question Assumes only 3, but gives 4 as answer

6.11 General comment

question #5: 6.0? (2 pts) 6.08 (3 pts) and 6.09 (3 pts) all referred to the Recirc Flow Control System and its components or operation in auto modes we currently do not use. Also, the total point value of these questions 8 pts accurate for 32% of this section of the exam on the topic of the Recirc Flow Control system which is not in agreement with ES-107 C.5 "no topic is worth more than 20% of that category.

7.01 A. The answer does not match the question. The question asks "What do Tech Spec's say?" Per Tech Spec, the required action is "to reduce suppression pool temp to 90°F within 24 hours," (3.6.2.1, action 6). In addition, this question requires memorization of a 24 hour Tech Spec Action Statement.

B. It requires memorization of a long term (24 hour) action statement.

C. Same comment as "B" above.

7.04 General Comment Just because both loops of S/D cooling are INOP., per 5.3.5. Tech Spec 3.4.9.1 or 3.4.9.2 does not require that alt. S/D Cooling be performed.

c. Per PPM 5.3.5, Alternate Shutdown Cooling. Alt. Shutdown cooling can be accomplished by any low pressure ECCS pump, thus LPCS, RHR "C", or either RHR "B" or "C" may be used to inject into the Reactor Vessel. Should accept any of these as correct or should not "count-off" for not listing RHR A or B as injecting into RPV.

d. PPM 5.3.5 is used in accident conditions where cooldown is required and normal S/D cooling cannot be accomplished. It is not reasonable to expect an operator to "memorize" this procedure or the numbers in it. This RPV pressure response is only a "rough estimate" to provide indirect indication of core flow. Step 6.3 is used to supercede this to ensure a cooldown rate of less than 100°F/hr. Should accept "less than 120psig (to allow low press. ECCS pumps to inject)."

7.06 a. This question requires memorization of a limitation not an immediate action on a procedure which is not an abnormal or an emergency procedure.

7.08 The question states that a reactor shutdown was performed. This would make part 1 of the answer "not applicable." Should accept "items performed during the outage were easily tracked and controlled by shift staff." (PPM 3.1.4, discussion A, paragraph 1).

7.09 Answer key does not give all possible answers per PPM 4.4.2.1. Should also accept the other answers shown included in Section 4.4..2.1.4.C (RWCU system heat exchangers or CRD & reject vra RWCU). Note: the answer key does not refer to the latest revision of this. PPM. we have attached a copy of the latest revision.

7.10 B.

1. You cannot make the assumption that because a control rod is "untrippable" it is immovable. Thus, part B does not "jive" with part A.

2. Should also accept step 7 of PPM 5.1.3 as a full credit answer." If the reactor cannot be shutdown before suppression pool temperature reaches 110°F."

8.01 This question requires memorization of a normal operating procedure, does not agree with our learning objectives.

8.03 A. Should accept any three of the six listed on PPM 1.11.3, Health Physics Program under (radiological) " conditions that require an RWP" (page 9 of 25)

8.04 This answer requires memorization of an administrative procedure. This also not included in our learning objectives.

8.05 b. should accept either safety related or

8.06 This question is very vague! It assumes that the shift manager either does not or cannot (for whatever reason) remove the clearance order, then perform his test. If you want the SRO to know the restrictions placed on "Temporarily lifting danger tags," then ask it that way! The PPM 1.3.8 states the S. M. authorizes the temporary lifting at tags, not system checkout.

8.08 Answer #7 contains 3 distinct guidelines to be accomplished - should consider these separately.

8.09 PPM 1.3.1, Standing orders has been updated to be consistent with our Emergency Operating Procedures. Please see latest rev. (attached) of PPM 1.3.1 for correct answer.

8.11 This question is not valid because the justification for use or non-use has not been documented or clarified in normally "testable" information. In addition, the lates revision of PPM 1.3.1, no longer includes this instrument on the "unqualified Instrument List." Recommend deletion from exam. (see attached PPM 1.3.1).

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

MASTER COPY

FACILITY: WNP-2

REACTOR TYPE: BWR-GE5

DATE ADMINISTERED: 86/02/04

EXAMINER: SHERMAN, J.

APPLICANT: ______________________

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answers 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
22.00 ~~24.00~~	22.74 ~~23.04~~	________	________	1.	PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
24.75 ~~25.25~~	25.58 ~~25.19~~	________	________	2.	PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.00	25.84 ~~24.04~~	________	________	3.	INSTRUMENTS AND CONTROLS
25.00 ~~26.00~~	25.84 ~~26.04~~	________	________	4.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
96.75 ~~100.25~~	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 1.01 (1.50)

INCREASING recirculation flow causes movement of the boiling boundary and a power increase. Name the DOMINANT reactivity coefficient(s) which are acting during the power increase when:

a. The boiling boundary moves upward (0.5)

b. The boiling boundary moves downward (1.0)

QUESTION 1.02 (1.50)

During a reactor startup, Keff is .95 when the SRM channels read 100 cps. What will the new Keff be when SRM channels read 250 cps?

STATE ANY ASSUMPTIONS YOU MAKE.
SHOW ALL YOUR WORK.

QUESTION 1.03 (2.00)

Several minutes after a reactor scram you notice indicated power decreasing on an 80 second period. What causes this behavior and why is the period 80 seconds?

QUESTION 1.04 (2.00)

State whether the following changes will directly affect AVAILABLE recirculation pump net positive suction head (NPSH): (Limit answer to: INCREASE, DECREASE, or NO AFFECT)

a. Feedwater temperature increases (0.5)

b. Reactor pressure decreases (0.5)

c. Reactor water level increases (0.5)

d. Recirculation pump speed decreases (0.5)

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.05 (2.00)

Flow orifices serve to provide the majority of pressure drop across the core. WHY is flow orificing necessary in a BWR?

QUESTION 1.06 (2.00)

When called upon for an emergency shutdown, the Standby Liquid Control System must inject at a rate neither too slow or too fast. DESCRIBE why this is the case.

QUESTION 1.07 (1.50)

The fuel temperature (Doppler) coefficient of reactivity changes over core life.

a. Does it become MORE NEGATIVE OR LESS NEGATIVE? (0.5)

b. DESCRIBE the primary effect that causes it to change over core life. (1.0)

QUESTION 1.08 (1.00)

WHAT percentage of full power is produced by decay heat at the following times after a reactor shutdown from 100 % power? (Assume equilibrium reached prior to shutdown.)

a. One second after shutdown.

b. One hour after shutdown.

QUESTION 1.09 (1.50)

During a reactor startup, the IRM readings go from 30 to 65 on the same range in 2 minutes with no operator actions. What PERIOD is the reactor on? SHOW YOUR WORK.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.10 (2.00)

A reactor vessel is depressurized from 1000 psig to 600 psig in a period of one hour. WHAT is the cooldown rate? Assume saturated conditions. SHOW YOUR WORK.

QUESTION 1.11 (3.00)

Following a normal reduction in power from 90% to 70% with recirculation flow, HOW will the following change (increase, decrease, or remain the same) AND WHY:

a. The pressure difference between the reactor and the turbine steam chest.

b. Condensate depression at the exit of the condenser.

c. Final Feedwater temperature.

QUESTION 1.12 (2.00)

a. WHAT is "pump runout" and WHY is it an undesirable condition? (1.0)

b. Define the term cavitation, and GIVE TWO (2) examples of detrimental effects. (1.0)

QUESTION 1.13 (2.00)

What are the two considerations that determine the maximum and and minimum control rod speed limit? (Consider normal movement, not scram)

(***** END OF CATEGORY 01 *****)

QUESTION 2.Ø1 (2.ØØ)

STATE TWO purposes for the RHR water leg pumps.

QUESTION 2.Ø2 (1.5Ø)

An "IRM Inoperative" annunciator is received. What THREE conditions could be the cause of this alarm?

QUESTION 2.Ø3 (2.ØØ)

Regarding the APRM system:

a. WHAT is the MINIMUM number of LPRM inputs a channel must have to be considered operable? (Ø.5)

b. STATE the meter switch position you would use to determine the number of LPRMS in service on an APRM channel. Include required interpretation of the meter reading. (1.5)

QUESTION 2.Ø4 (2.5Ø)

NAME five of the six physical barriers which will limit the release of fission products from the fuel to the environment.

QUESTION 2.Ø5 (2.ØØ)

Concerning the reactor building-to-wetwell relief lines:

a. WHAT purpose do they serve? (1.Ø)

b. Under WHAT condition will they relieve (setpoint required)? (1.Ø)

(***** CATEGORY Ø2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.Ø6 (1.ØØ)

"Souping", a light load phenomenon in a diesel generator, occurs when: (CHOOSE ONE)

a. Excessive oil enters the combustion chamber and accumulates in the exhaust system.
b. Electrical loads are added too quickly immediately after the engine reaches rated speed.
c. Diesel fuel excessively contaminates the lubricating oil system
d. The water jacket cooling system becomes fouled with biological contaminants.

QUESTION 2.Ø7 (1.5Ø)

Regarding the High Pressure Core Spray System:

a. List the two auto initiation signals (setpoints not required). (1.Ø)

b. STATE the system which is redundant to HPCS for small break protection. (Ø.5)

QUESTION 2.Ø8 (2.5Ø)

LIST five of the six conditions that will cause automatic isolation of the RCIC system (setpoints not required). (2.5)

QUESTION 2.Ø9 (1.ØØ)

Are the statements below TRUE or FALSE concerning the High Pressure Core Spray Diesel Generator System?

1. The bus SM-4 is powered by the diesel generator due to loss of preferred power. After the preferred power source is made available, the operable preferred power source breaker is closed. The HPCS-DG output breaker automatically OPENS. (Ø.5)

2. There are two NORMAL engine stop pushbuttons: 1. in the Control Room. 2. on the local panel. BOTH pushbuttons are ALWAYS active. (Ø.5)

(***** CATEGORY Ø2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.10 (3.00)

a. Regarding the Plant Electrical System, the standby diesel generators start automatically upon receiving what signals? (three required) (1.5)

b. When a LOCA signal is present to the Standby Diesel Generator System, what three auto shutdown signals or functions are NOT bypassed? (1.5)

QUESTION 2.11 (2.00)

WHAT will happen to the Reactor Water Cleanup System if Reactor Closed Cooling Water is lost to all components RCCW serves in the RWCU system?
Your answer should include the affected components, and any trips or interlocks associated with this condition. (Setpoints not required)

QUESTION 2.12 (2.25)

Regarding the recirculation pumps, WHAT are THREE conditions that will cause a fast speed trip AND an LFMG auto start? (Setpoints ARE Required.)

QUESTION 2.13 (2.00)

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.

b. The D/P cell, for the RCIC flow control element, has a perforated diaphram.

(***** END OF CATEGORY 02 *****)

QUESTION 3.01 (2.50)

STATE five signals which will automatically close the main steam isolation valves (MSIV's). (Setpoints not required.)

QUESTION 3.02 (1.00)

ALL 125 VDC power has been lost.
Which of the following is correct:

a. SRV's will be operable in the pressure relief, safety relief, and ADS modes.
b. SRV's will be operable in the pressure relief ~~and ADS~~ mode~~s ONLY~~.
c. SRV's will be operable in the safety relief ~~and ADS~~ mode~~s ONLY~~.
d. SRV's will be operable in the ADS mode ~~ONLY~~.

QUESTION 3.03 (2.00)

Each condensate filter demineralizer (demin) has a hold pump associated with it. Regarding hold pumps:

a. WHAT is their purpose? (0.5)
b. STATE the conditions under which the pump will start or stop if in the auto mode. (1.0)
c. WHAT problem may exist with the demin if the hold pump failed to start in auto when it is required to? (0.5)

QUESTION 3.04 (2.00)

Regarding the setpoint setdown feature of the feedwater level control system:

a. WHAT TWO changes take place automatically in the control circuitry following a reactor scram?
b. The feedwater level control system will remain in the setpoint setdown condition following a scram until one of two actions is performed by the operator. STATE these TWO actions.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.05 (3.00)

DESCRIBE the changes in the following parameters in the FIRST FEW MINUTES following the low failure of one steam flow transmitter. (Assume 100% reactor power, three element FWLC, and no operator action)

a. Reactor power
b. Feedwater flowrate
c. Reactor water level

QUESTION 3.06 (2.00)

For each of the following state whether a ROD BLOCK, HALF-SCRAM, FULL-SCRAM, or NO PROTECTIVE ACTION is generated for that condition.
NOTE: If two or more actions are generated, i.e. rod block and a half-scram, state the most severe, i.e. half-scram. Assume NO operator actions.

a. APRM downscale, Mode Switch in Run.
b. 12 LPRM inputs to APRM B, Mode Switch in Startup.
c. Both Flow Conversion Units Upscale (>108% flow), Mode Switch in Run.
d. APRM A and D >20%, Mode Switch in STARTUP. (4 @ 0.5 ea.)

QUESTION 3.07 (2.00)

For each of the IRM (Intermediate Range Monitoring) range changes below, provide the following (Mode Switch in STARTUP):
1. The indicated level on the NEW RANGE.
2. All automatic actions initiated as a result of the indicated level on the NEW RANGE.

a. Switch from Range 5, Reading 25, to Range 7.

b. Switch from Range 6, Reading 39, to Range 5.

QUESTION 3.08 (2.50)

LIST the sequence of valve movement that occurs when a single notch rod withdrawal is demanded from the Reactor Manual Control System. (Specific times are not required; correct sequence IS required; FIVE events for full credit.)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.09 (2.50)

Both the SRM and IRM compensate their detector signals with a unique type of discrimination process.

a. WHAT type of radiation does the discriminator eliminate? (0.5)

b. STATE the method each system, IRM and SRM, uses to accomplish this task. (1.0)

c. WHY is there a difference between the two discrimination processes? (1.0)

QUESTION 3.10 (2.50)

Answer the following questions concerning the Local Power Range Monitors.

a. What is used in the detector to extend the neutronic lifetime? (0.5)

b. Where are the three (3) indications that the thumbwheel mode selector switch is in the bypass position? (1.0)

c. Where is the signal of the flux amplifier fed, when the mode selector switch is in operate? (Four required for full credit.) (1.0)

QUESTION 3.11 (3.00)

Concerning the Area Radiation Monitors(ARMS):

1. What PURPOSE do these instruments serve? (1.0)

2. What is INDICATED by the following lights on the front face of the indicator and trip unit:
   a. WHITE light[0.5]
   b. AMBER light[0.5] (1.0)

3. Assume several ARM's in the reactor building area are alarming. HOW would the control room operator determine the specific locations of the alarming ARM's? (1.0)

(***** END OF CATEGORY 03 *****)

QUESTION 4.01 (1.00)

According to Emergency Procedure 5.1.3 (Reactor Power Control), two systems OTHER THAN Standby Liquid Control could be used to inject boron into the reactor vessel. Name the two systems.

QUESTION 4.02 (2.00)

WHAT are the entry conditions for Emergency Procedure 5.4.1 -- Station Blackout? (FOUR items)

QUESTION 4.03 ~~(2.00)~~ e (1.00)

Emergency Procedure General Precautions, Caution #7, requires specific operator action if "HPCS Suction Switchover Suppression Pool Level High or HPCS/RCIC Suction Switchover CST Level Low alarms occur".

a. WHAT operator action is required?
~~b. Which system must be switched manually in a CST low level condition?~~ e

QUESTION 4.04 (2.50)

According to General Operating Procedure 3.3.1 (Reactor Scram), WHAT are the first FIVE steps required to be performed? (Each STEP may contain more than one item.)

QUESTION 4.05 (1.00)

According to the Standing Operating Orders (Procedure 1.3.1), under what conditions may operating personnel depart from approved operating procedures?

(***** CATEGORY Ø4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.06 (3.00)

Procedure 2.2.1 (Reactor Recirculation System), contains a number of restrictions for operation of the recirculation system. STATE the reason(s) for the following limitations:

a. For two loop operation, maintain loop-to-loop flow mismatch less than 10% when less than 70% rated core flow and maintain loop-to-loop flow mismatch less than 5% when greater than 70% rated core flow.

b. Do not start an idle reactor recirculation pump during the approach to criticality or when critical below the power range.

c. Following a recirculation pump trip, close the discharge valve. After 5 minutes (maximum), open the discharge valve.

QUESTION 4.07 (1.00)

WHAT action must you take if the water leg pump for High Pressure Core Spray fails when HPCS is required to be operable, according to Procedure 2.4.4 (High Pressure Core Spray System)?

QUESTION 4.08 (3.00)

LIST THREE IMMEDIATE ACTIONS you would take upon a reactor HIGH water level condition according to Abnormal Condition Procedure 4.2.1.2 (Reactor Vessel High Water Level).

QUESTION 4.09 (2.00)

DESCRIBE TWO problems, other than CRD accumulator problems, which will occur upon a complete loss of CRD flow, according to Abnormal Conditions Procedure 4.1.1.2 (Complete Loss of CRD Drive Flow).

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.10 (2.00)

STATE the reason(s) for each of the following limitations in the High Pressure Core Spray System Operating Procedure.

a. Whenever the HPCS system is discharging into the vessel under any conditions other than an actual loss of coolant accident, make sure that pump suction is being taken from the condensate storage tank.

b. Pump Start Limitation - Two starts in succession from ambient temperature or one start from rated temperature.

QUESTION 4.11 (3.00)

What are SIX of the NINE immediate operator actions you would perform if a control room evacuation were ordered by the Shift Manager? (Assuming you have time.). (3.0)

QUESTION 4.12 (3.50)

The Control Room Operator Shift Turnover Checklist requires the oncoming CRO to take certain actions.

a. What are FOUR items to be reviewed DURING turnover? (2.0)

b. What are THREE items to be reviewed SHORTLY AFTER shift turnover? (1.5)

(***** END OF CATEGORY 04 *****)
(************* END OF EXAMINATION ***************)

ANSWER 1.01 (1.50)

a. Void coefficient [0.5] (0.5)
b. Void coefficient [0.5], fuel temperature (Doppler) coefficient [0.5] (1.0)

REFERENCE
WNP-2 Reactor Theory Student Text, Pg. 21

ANSWER 1.02 (1.50)

CR1(1-Keff1) = CR2(1-Keff2) [1.0]

CR1/CR2(1-Keff1) = (1-Keff2)

100/250(1-.95) = (1-Keff2)

.02 = (1-Keff2)

Keff2 = .98 [0.5]

REFERENCE
WNP-2 Reactor Theory Student Text, Pg. 8

ANSWER 1.03 (2.00)

Delayed neutron precursors releasing delayed neutrons.[1.0] Longest lived delayed neutron precursor (has half life of 56 seconds) produces period of -80 seconds. [1.0]

REFERENCE
WNP-2 Reactor Theory Student Text, Pg. 24

ANSWER 1.04 (2.00)

a. DECREASES (0.5)
b. DECREASES (0.5)
c. INCREASES (0.5)
d. ~~NOT AFFECT~~ INCREASES (0.5)

REFERENCE
G.E. Thermodynamic, Heat Transfer, and Fluid Flow, Pg. 7-92 +

ANSWER 1.05 (2.00)

Higher bundle power causes increased voiding and therefore increased resistance to coolant flow [1.0]. If no orificing, high power (central) bundles would be starved of cooling, while more coolant flow would be diverted to lower power (peripheral) bundles [1.0]. (2.0)

REFERENCE
WNP-2 Systems Vol. I, Tab I, Pg. 14

ANSWER 1.06 (2.00)

1. If injection is too fast, uneven mixing could occur and power chugging could result. (1.0)

2. If injection is too slow, the initial shutdown margin would be reduced due to lower concentration of fission product poisons. (1.0)

REFERENCE
WNP-2 Systems Vol. III, Tab 8, Pg. 8

ANSWER 1.07 (1.50)

a. MORE NEGATIVE (0.5)

b. Pu-240 builds up with exposure. (1.0)
(Pu-240 resonance absorption is stronger than V-238)

REFERENCE
WNP-2 Reactor Theory Student Text, Pg. 19

ANSWER 1.08 (1.00)

a. 6.0% (Range 5.0 - 7.0%) (0.5)
b. 1.6% (Range ~~1.4 - 1.8%~~) (0.5)
0.5 -- 2.0%)

REFERENCE
WNP-2 Reactor Theory Student Text, Pg. 9

ANSWER 1.09 (1.50)

P = Po e t/T

P/Po = e t/T

ln(P/Po) = t/T

T = t/ln(P/Po) (1.0)

= 120 sec./ln(65/30)

= 155 sec. (0.5)

REFERENCE
WNP-2 Reactor Theory Student Text, Section 6, p. 14

ANSWER 1.10 (2.00)

From Steam Tables
1000 psig + 14.7 = 1014.7 psia (Acceptable if used 15#
600 psig + 14.7 = 614.7 psia instead of 14.7 #) (0.5)

Tsat1 = [(14.7/50)(550.53 - 544.58)] + 544.58
= 546.33 F (0.5)

Tsat2 = [(14.7/50)(494.89 - 486.20)] + 486.20
= 488.75 F (0.5)

Cooldown rate = 546.33 - 488.75 = 57.57 F (0.5)
(Acceptable range 57 ± 1 °F)

REFERENCE
Steam Tables

ANSWER 1.11 (3.ØØ)

a. Decreases [Ø.25]. There is less steam flow, therefore, less pressure drop through the main steam lines [Ø.75]. (1.Ø)

b. Increases [Ø.25]. With the same amount of cooling water through the condenser [Ø.25] and less of a heat load [Ø.5]. (1.Ø)

c. Decreases [Ø.25]. Less extraction steam from the turbine to heat the feedwater [Ø.75]. (1.Ø)

REFERENCE
EIH Heat Transfer Lesson Plan, pp. 75 & 78, and
EIH Nuclear Training, p. 1Ø.4-11.

WNP2 G.E. Thermodynamics, Heat Transfer, and Fluid Flow, pg 6-68

ANSWER 1.12 (2.ØØ)

a. Running a centrifugal pump at minimun head and maximum capacity ~~(Ø.5)~~. Runout causes 1 electrical over heating, 3 possible electrical damage, and 2 likely tripping off line ~~(Ø.5)~~. (1.Ø)
(only one required for full credit)

b. Cavitation is the flashing to vapor of liquid in the pump suction or a low pressure area.[Ø.5] Cavitation results in any or all of the following: excess vibration and noise, reduced pump efficincy, pitting and corrosion of pump impeller. (not limited to these 5) (2 required at Ø.25 each) (1.Ø)

REFERENCE
WNP2 G.E. Thermodynamics, Heat Transfer and Fluid Flow, pg 7-124 & 7-91

ANSWER 1.13 (2.00) Question and point value (2.0) deleted

1. Maximum control rod speed is intended to limit the rate at which control rods can be withdrawn during reactor startup. Fast reactivity insertion rate results in short periods and could cause reactor core to rapidly overheat and become damaged. (1.0)

2. The rate of control rod speed must be sufficient to overcome xenon reactivity decrease during burnout. (1.0)

REFERENCE
WNP2 Systems Vol II, Tab 8

ANSWERS -- WNP-2 -86/02/04-SHERMAN, J.

ANSWER 2.01 (2.00)

1. Time required for water to reach reactor vessel upon system initiation is minimized. (1.0)

2. Possibility of water-hammer damage is minimized. (1.0)

REFERENCE
WNP-2 Systems Vol. III, Tab 6, Pg. 7

ANSWER 2.02 (1.50)

1. Detector high voltage low (0.5)
2. Module unplugged (0.5)
3. IRM mode switch not in "operate" (0.5)

REFERENCE
WNP-2 Systems Vol. II, Tab 2, Pg. 23

ANSWER 2.03 ~~(2.00)~~ (1.500)

a. 14 (0.5)

b. "Count" switch position. 2 (0.5)
Meter reading (in percent) divided by five yields number of LPRM inputs. (1.0)

REFERENCE
WNP-2 Systems, Vol. II, Tab 4, Pg. 12

ANSWER 2.04 (2.50)

1. Fuel pellet itself
2. Fuel cladding
3. Reactor coolant
4. Reactor pressure vessel
5. Primary containment
6. Secondary containment [5 required, 0.5 each] (2.5)

ANSWERS -- WNP-2 -86/02/04-SHERMAN, J.

REFERENCE
WNP-2 Systems, Vol. III, Tab. 11, Pg. 4

ANSWER 2.05 (2.00)

a. Equalize pressure differential that [0.5] may occur between secondary containment and suppression chamber, thus preventing a vacuum from developing in the primary containment [0.25] due to condensing steam. [0.25] (1.0)

b. When pressure in secondary containment exceeds pressure in suppression chamber [0.5] by 0.5 psig [0.5]. (1.0)

REFERENCE
WNP-2 Systems, Vol. II, Tab 11, Pg. 11

ANSWER 2.06 (1.00)

a (1.0)

REFERENCE
WNP-2 Systems, Vol. IV, Tab 19, Pg. 70

ANSWER 2.07 (1.50)

a. Low reactor water level (0.5)
High drywell pressure (0.5)
b. ADS or RCIC (one required for full credit) (0.5)

REFERENCE
WNP-2 Systems, Vol. III, Tab 1, Pg. 5

ANSWER 2.08 (2.50)

1. RCIC equipment area ~~and/or~~ pipe routing area high temperature (credit for 2 answers if listed separately)
2. RCIC equipment area high differential temperature
3. Low steam supply pressure
4. Exhaust diaphragm high pressure
5. RCIC high steam flow or instrument line break
6. Combined RCIC and RHR high steam flow

[5 required, 0.5 each] (2.5)

REFERENCE
WNP-2 Systems, Vol. III, Tab 3, Pg. 14

ANSWER 2.09 (1.00)

1. FALSE (0.5)

2. False { The LOCAL/REMOTE switch will select the active button } (0.5)

REFERENCE
WNP2 - Systems, Vol IV, Tab 9, pg 43

ANSWER 2.10 (3.00)

a. 1. loss of voltage (or undervoltage) on associated class 1E bus
   2. high drywell pressure
   3. low reactor vessel water level (3 @ 0.5 ea)

b. 1. Engine overspeed
   2. Generator differential current
   3. Incomplete sequence (3 @ 0.5 ea)

REFERENCE
WNP2 - Systems, Vol IV, Tab 9, pg 43

ANSWERS -- WNP-2 -86/02/04-SHERMAN, J.

1. **Affected components will be non-regenerative heat exchangers [0.25], V-4 [0.25], reactor water cleanup pumps [0.25], and FCV-33 [0.25].**

2. High temperature at NRHX outlet will cause isolation valve V-4 to close. [0.5] V-4 closure causes **RWCU pump trip** [0.25] and FCV-33 **closure** [0.25].

ANSWER 2.11

1. Affected components will be RWCU pumps [0.5] and non-regenerative heat exchangers [0.5]. (1.0)
2. Operating RWCU ~~pump will trip on high temperature [0.5].~~ High temperature ~~at~~ NRHX outlet will cause outboard isolation valve ~~(V-4)~~ to close [0.5]. (1.0)

REFERENCE
WNP2 - Systems, Vol I, Tab 9, ~~pg 26~~ pp 7-8

ANSWER 2.12 (2.25) [Three required of the following 5, @ .75 each]

1. Feedflow <30% with 15 sec. T.D. (cavitation Limit)
2. Main Steamline/Pump suction temperature differential <9.9F
3. Reactor vessel low level 3
4. ~~RPT - Turbine trip or load reject >142 psig turbine first stage press (Trips @ 0.5 ea) (Setpoints @ 0.25 ea)~~

See ~~attachment~~ below

REFERENCE
WNP2 - Systems, Vol I, Tab 6, pg 38, WNP-2, T.S. 3/4, 3-44.

ANSWER 2.13 (2.00)

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]. (and may overspeed) (1.0)
b. RCIC will inject at maximum rate and the min flow valve will not respond, (will remain open) [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

4. **Turbine Throttle valve-closure ≤ 5% closed.**

5. **Turbine Governor valve - fast closure ≥ 1250 psig.**

ANSWER 3.01 (2.50)

1. Reactor water low level (Level 2)
2. Main steam line high radiation
3. Main steam line high steam flow
4. Main steam line low pressure with mode switch in run
5. Main steam line tunnel high temperature, or high ventilation system differential temperature
6. Main condenser low vacuum.

(5 @ 0.5 ea.)

REFERENCE
WNP-2 Systems Vol. V, Tab 1, Pg. 16

ANSWER 3.02 (1.00)

c

REFERENCE
WNP-2 Systems, Vol. V, Tab. 1, Pg. 33

ANSWER 3.03 (2.00)

a. Hold pumps maintain enough flow to maintain precoat on the filter elements. (0.5)
b. Starts in auto upon low flow (<1,600 gpm) through associated demin unit. Stops when condensate flow re-established. (1.0)
c. Precoat may have been lost. (0.5)

REFERENCE
WNP-2 Systems, Vol. V, Tab 11, Pg. 7

ANSWER 3.04 (2.00)

a. 1. Transfer from three element to single element control. (0.5)
   2. Reactor vessel level setpoint transferred to +18 inches. (0.5)
b. 1. Master controller or individual controller is placed in manual. (0.5)
   2. Setpoint setdown reset pushbutton is depressed. (0.5)

REFERENCE
WNP-2 Systems, Vol. V, Tab. 14, Pg. 6

ANSWER 3.05 (3.00)

a. No change (1.0)
b. Decreases initially due to false low steam flow indication, then returns to the same as initially when level equilibrates at lower point. (1.0)
c. Lowers due to decreased feed flow initially, then stabilizes at some new lower point when level error matches steam-feed flow error. (1.0)

REFERENCE
WNP-2 Systems, Vol. V, Tab 14, Pg. 15

ANSWER 3.06 (2.00)

a. Rod block
b. Half-scram
c. Rod block
d. Full scram

REFERENCE
WNP-2 Systems, Vol. II, Tab 4, Pg. 20

ANSWER 3.07 (2.00)

a. New reading on Range 7 is 2.5. No auto actions. (1.0)
b. New reading on Range 5 is 39. IRM high rod block and HI-HI half scram will be in. (1.0)

REFERENCE
WNP-2 Systems, Vol. II, Tab. 2, Pg. 14

ANSWERS -- WNP-2 -86/Ø2/Ø4-SHERMAN, J.

ANSWER 3.Ø8 (2.5Ø)

1. Insert valves (121 and 123) activate to lift rod off collect fingers.
2. Insert valves (121 and 123) close.
3. Withdraw valves (12Ø and 122) activate to withdraw rod.
4. Valve 122 (drive pressure) shuts.
5. Valve 12Ø (settle valve) shuts. (5 @ Ø.5 ea.)

Also accept word description instead of valve numbers

REFERENCE
WNP-2 Systems, Vol. II, Tab 6, Pg. 13

ANSWER 3.Ø9 (2.5Ø)

a. Gamma (Ø.5)
b. SRM - pulse height discrimination.
   IRM - Cambelling OR mean square voltage. (1.Ø)
c. Due to the low number of events and greater sensitivity, the SRM deals with individual counts (pulses), while the IRM signal is a voltage level due to pulse overlap. (1.Ø)

REFERENCE
WNP-2 Systems, Vol. II, Tab 1, Pg. 14
Tab 2, Pg. 7

ANSWER 3.1Ø (2.5Ø)

a. U234 is added to the coating (Ø.5)
b. 1. Bypass light on panel 6Ø8
   2. The four-rod group display (P603)
   3. The LPRM Bypass indicator on the APRM front panel (3 @ Ø.33 ea)
c. 1. LPRM upscale trip circuit.
   2. LPRM downscal trip scircuit.
   3. The associated APRM channel.
   4. Plant Process computer (analog input).
   5. Rod Block Monitor.
   6. Four Rod display on CØ5 [4 @ Ø.25 ea] (1.Ø)

REFERENCE
WNP2 - Systems, Vol II, tab. 3, pg 7,13,15

ANSWER 3.11 (3.00)

1. Warn of abnormal radiation levels in areas where radioactive material may be present, stored, handled, or inadvertently introduced. (1.0)

2. a. Downscale trip [0.5]
   b. Upscale trip [0.5] (1.0)

3. Only one annunciator window is provided for reactor building area, so operator would have to check back panels to see which ARM's are alarming [0.5]. Specific locations of those ARM's would be determined from plant drawings (labels accepted if actually installed in plant)[0.5]. (1.0)

REFERENCE
WNP-2 Systems, Volume III, Tab. 10, pp.1,6.

ANSWERS -- WNP-2 -86/02/04-SHERMAN, J.

ANSWER 4.01 (1.00)

1. RCIC (0.5)
2. RWCU (0.5)

REFERENCE
WNP-2 Procedure 5.1.3, Pg. 5

ANSWER 4.02 (2.00)

Loss of ALL of the following:

1. Offsite 230 KV startup power
2. Offsite 115 KV backup power
3. Diesel Generator #1
4. Diesel Generator #2 (4 @ 0.5 ea.)

REFERENCE
WNP-2 Procedure 5.4.1, Pg. 1

ANSWER 4.03 ~~(2.00)~~ (1,00)

a. Confirm automatic transfer of or manually transfer HPCS and RCIC suction from the CST to the suppression pool. (1.0)
~~b. RCIC~~ ~~(1.0)~~

REFERENCE
WNP-2 Procedure 5.0.0., Pg. 2

ANSWER 4.04 (2.50)

1. Initiate a manual scram
2. Place mode switch in SHUTDOWN
3. Verify reactor power is decreasing (check APRM's)
4. Verify all control rods have been inserted (various methods)
5. Verify reactor water level is being restored by one or more of Feedwater, RCIC, or HPCS (5 @ 0.5 ea.)

Also accept steps 6-11 on attached

4.04 Answer Continued

6. **Verify 500KV Breakers 4885 and 4888 have opened.** If not, depress <u>either</u> emergency trip push button on Panel C.

7. Verify the Main Turbine has tripped. If not, depress <u>both</u> emergency trip pushbuttons on Panel B.

8. Press the red "RESET" pushbutton on the reheat temperature controller until it backlights red.

9. Verify the 4160V and 6900V buses have auto transferred to the startup transformer.

10. Verify both Reactor recirculation pumps have auto transferred to 15 Hz.

11. Verify scram discharge volume vent and drain valves indicate closed at Panel P603.

ANSWERS -- WNP-2 -86/02/04-SHERMAN, J.

REFERENCE
WNP-2 Procedure 3.3.1, Pg. 3

ANSWER 4.05 (1.00)

Where necessry to prevent injury to personnel, (including the public) or damage to the facility. (1.0)

REFERENCE
WNP-2 Procedure 1.3.1, Pg. 3

ANSWER 4.06 (3.00)

a. Prevent possible vibration of jet pumps [0.5] and riser braces. [0.5] (1.0)
OR Ensure adequate core flow coastdown following LOCA.
b. Injection of colder water from idle recirc loop could add enough positive reactivity to cause criticality and/or power excursion. (1.0)

c. Close discharge valve to prevent reverse rotation of pump. Open discharge valve to maintain temperature of idle loop. (1.0)

REFERENCE
WNP-2 Procedure 2.2.1, Pg. 3

ANSWER 4.07 (1.00)

Start and operate HPCS on recirculation to the CST until corrective action is completed. (Full credit). (If failure is due to power loss, HPCS pump switch should be held in STOP or otherwise prevent pump start until the system can be filled and vented). [0.5] (1.0)

REFERENCE
WNP-2 Procedure 2.4.4, Pg. 2

1. Notify CRS
2. Verify Auto Actions
3. Take Manual Control of FWLC and reduce RPV level

ANSWER 4.08 (3.00)

1. Take manual control of the controlling feedwater controller and reduce the feed rate to decrease reactor water level to normal .

2. If level can't be returned to normal with controlling controller, adjust applicable RFP governor controlling FW/RPV DP to lower level to normal.

3. If the high level condition occurred due to a failure of level instrumentation, shift to the alternate level instrument.

4. Verify that no ECCS system was inadvertently initiated . Secure them if they are not needed .

(3 ~~of 4~~ @ 1.0 ea.)

REFERENCE
WNP-2 Procedure 4.2.1.2, Pg. 2

ANSWER 4.09 (2.00)

1. Limits operator ability to control reactor
2. Loss of CRD cooling flow will cause control rod drives to be subjected to high temperatures, resulting in reduced seal life.
3. Loss of flow to recirculation pump seals.

(2 @ 1.0 ea.)

REFERENCE
WNP-2 Procedure 4.1.1.2, Pg. 2

ANSWER 4.10 (2.00)

a. CST water is cleaner than suppression pool. Want to minimize unnecessary contaminants in the reactor vessel. (1.0)

b. High current through pump motor windings during start attempt heats them up. Excessive starts could damage windings. (1.0)

ANSWERS -- WNP-2 -86/02/04-SHERMAN, J.

REFERENCE
WNP-2 Procedure 2.4.4, Pg. 3

ANSWER 4.11 (3.00)

1. Manually scram reactor.
2. Place mode switch to shutdown.
3. Manually close all MSIV's, MS-V-16, MS-V-19, and RWCV-FCV-33.
4. Verify APRM downscale lights illuminated.
5. Trip main generator and ensure that auxiliary power is being supplied from TR-S.
6. If recirculation pumps not on LFMG, transfer them to LFMG.
7. Ensure only two condensate booster and two condensate pumps are operating.
8. Ensure not more than two circulating water pumps are running.
9. Verify RFW valve positions.

[6 required @ 0.5 each] (3.0)

REFERENCE
WNP-2 Procedure 4.12.1.1

ANSWER 4.12 (3.50)

a. 1. Control Board Walkdown (0.5)
   2. Active Surveillances/Testing (0.5)
   3. Control Room Operators Log (pertinent to oncoming shift) (0.5)
   4. Offgoing CRO Summary Sheet (pertinent maintenance) (0.5)
   5. unusual occurrences

b. 1. SRV indications (0.5)
   2. Annunciator test (0.5)
   3. Indicating lamp survey (0.5)

4. Required reading
5. Tag logs
6. Charts.
7. upcoming surveillances
8. Log entries.
9. other opr. logs

REFERENCE
WNP-2 Procedure 1.3.6

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

MASTER EXAM & KEY

FACILITY: WNP-2

REACTOR TYPE: BWR-GE5

DATE ADMINISTERED: 86/02/03

EXAMINER: MILLER, L.

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 (1.50)

For each of the following events, STATE WHICH coefficient of reactivity (Power coefficient,void coefficient,moderator coefficient or doppler coefficient) would act FIRST to change reactivity.

a. Control rod drop at power (0.5)

b. SRV opening at power (0.5)

c. One recirc pump trips while at 50% power (0.5)

QUESTION 5.02 (2.00)

Three (3) minutes following a reactor scram from high power, indicated reactor power is 75 on range 4 and decreasing.

a. WHAT will INDICATED power be one (1) minute later? (Show calculations) (1.0)

b. Explain WHY power decreased at this rate. (1.0)

QUESTION 5.03 (1.50)

Concerning control rod worth, compare withdrawing a center control rod at 90% rod density to withdrawing a center control rod at 40% rod density. Which situation is control rod worth greater for the withdrawn control rod? Explain your answer.

QUESTION 5.04 (2.00)

The reactor is operating at 75% power. Recirculation flow is subsequently increased to provide 100% power and 100% flow. Describe HOW the reactor power is increased with the increased recirc flowrate. Continue your description until steady state conditions are reached (include CORE VOID CONTENT AND CORE NET REACTIVITY in your discussion).

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

QUESTION 5.05 (1.00)

What is the 'REVERSE POWER EFFECT' which can occur in the core during control rod movements.(Include how it occurs in your explanation)

QUESTION 5.06 (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.

A. Is the measured SDM acceptable? EXPLAIN (1.0)

B. What are the consequences, if any, to changes in the SDM for the next 20 hours? (1.0)

QUESTION 5.07 (1.50)

Consider a real plant system (NON-IDEAL) with two centrifugal pumps in parallel, one of which is running. The second pump is started.

A. System flow will be: (Choose the correct answer.) ((NOTE BOTH PUMPS OPERATING @ 1800 RPM)) (0.5)

a. double the original flow

b. less than double the original flow

c. greater than double the original flow

d. the same only the discharge head changes

B. EXPLAIN YOUR CHOICE of system flow response in part A. (1.0)

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

QUESTION 5.08 (3.00)

The reactor is operating at 100% power when HPCS inadvertantly initiates. Describe the response of the following parameters during the transient, including why the parameter changes as it does. Assume NO SCRAM occurs. Continue your description until steady state conditions are reached.

a. Turbine Load (1.0)

b. Reactor Water Level (1.0)

c. Feedwater Flow (1.0)

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 the MAPLHGR LIMIT? (0.5)

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

c. If a 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)

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

QUESTION 5.10 (3.00)

Following a normal reduction in power from 90% ' ɔ 70% with recirculation flow, HOW will the following change (increase, decrease, or remain the same) AND WHY:

a. The pressure difference between the reactor and the turbine steam chest. (1.0)

b. Condensate depression at the exit of the condenser. (1.0)

c. Final Feedwater temperature. (1.0)

QUESTION 5.11 (2.00)

Indicate on the attached figure (FIGURE 1) or discuss, HOW the following stresses vary across the reactor vessel wall during a reactor heatup. (2.0)

a. Thermal stress

b. Pressure stress

c. Combined thermal / pressure stress

QUESTION 5.12 (3.00)

Since the parameters which result in fuel damage are not directly observable during reactor operation, CRITICAL POWER is adopted as a convenient limit. For each of the following factors, state HOW critical power will change (increase, decrease or remains the same). EXPLAIN your answer.

a. Inlet subcooling increases (1.0)

b. Core flowrate increases (1.0)

c. Reactor pressure increases (1.0)

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

QUESTION 6.01 (2.00)

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

a. If the physical hardware that provides gamma discrimination for the Intermediate range is not considered, is gamma discrimination necessary over the entire Intermediate range of the neutron instrumentation? EXPLAIN YOUR ANSWER. (1.0)

b. What three conditions will result in an IRM inoperative Rod Block? (1.0)

QUESTION 6.02 (2.00)

Unit 2 is operating at 100% rated thermal power, with recirc in Master Manual(flux automatic). An operator inadvertently INCREASES the "Pressure Set" on EHC by 5 psig.

ASSUME: 1. No Further Operator Actions
2. All other DEH control settings are normal
3. Starting Parameters
- o TCV's - 100% Steam Flow Position
- o BPV's - 0% Steam Flow Position
- o Power - 100% Rated Thermal Power
- o Pressure - 1005 psig

NOTE: FIGURE # 2 IS ATTACHED FOR REFERENCE

Which of the following most accurately describes both the INITIAL RESPONSE and FINAL STATUS of the different parameters and components.

	a	b	c	d
INITIAL RESPONSE				
o TCV's	CLOSE (~83%)	CLOSE (~83%)	CLOSE (~83%)	NO CHANGE
o BPV's	NO CHANGE	OPEN (~17%)	NO CHANGE	OPEN (~25%)
o Power	INCREASE	NO CHANGE	INCREASE	DECREASE
o Pressure	INCREASE	NO CHANGE	INCREASE	DECREASE
FINAL STATUS				
o TCV's	~100 %	~ 83 %	~ 83 %	~ 100 %
o BPV's	0 %	~ 17 %	~ 17 %	0 %
o Power	~100 %	100 %	> 100 %	< 100 %
o Pressure	>1005 psig	1005 psig	>1005 psig	<1005psig

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

QUESTION 6.03 (1.50)

a. The new LPRM detectors (the NA-200 detector) have extended lifetimes, at least 3 times that of the NA-100 detectors for the same end of life output currents. What are the two features of the new LPRM detectors which allow the extended life ? (1.0)

b. A Core Thermal Power and APRM Calibration program (OD-3) is performed and shows APRM A with a Gain Adjustment Factor of 1.03. What does this tell the operator about the relationship between actual and indicated power on APRM channel A? (0.5)

QUESTION 6.04 (1.50)

Following a reactor scram, the four rod display position goes blank, but the green full-in light on the full core display for that control rod is lighted. Is this normal? If so, explain why it occurs. If not, describe the probable cause. (1.5)

QUESTION 6.05 (2.00)

For each of the following situations (i and ii) select the correct Feedwater Control System / plant response from the list (a through e) which follows.(An answer may be used more than once, and no operator actions are taken.)

i. The plant is operating at 70% power, in 3-element control, when one of the Steam Flow Detectors FAILS DOWNSCALE.

ii. The plant is operating at 90% power, in 3-element control, when HPCS inadvertently initiates and injects

FEEDWATER CONTROL SYSTEM / PLANT RESPONSE

a. Reactor water level decreases and stabilizes at a lower level.

b. Reactor water level decreases and initiates a reactor scram.

c. Reactor water level increases and stabilizes at a higher level.

d. Reactor water level increases and initiates a turbine trip.

e. None of the above.

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

QUESTION 6.06 (2.00)

The APRM scram function actually consists of two separate setpoints:

a. FILL IN THE BLANKS:

Flow Biased Scram - __(1)__ * w + __(2)__% (0.25)

Fixed Scram - __(3)__% (0.25)

b. LIST the specific location(s) of the sensor(s) which measure the variable "w". (0.5)

c. While operating at power, one MSIV fails shut resulting in a brief (~ 1 second) flux spike to 121% power. STATE which of the two scram setpoints mentioned above (one or both) should initiate a reactor scram. JUSTIFY your choice. (1.0)

QUESTION 6.07 (4.00)

Describe the operation of the following systems for the following given conditions. Consider each situation separately. State if the system pump will start (and run) and if the system will inject into the vessel and WHY.

GIVEN: The system receives a valid initiation signal.

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

B. The LPCS suction valve is shut. (1.0)

C. The RHR pump 2C suction valve is shut. (1.0)

D. The RHR pump 2B suction valve is shut. (1.0)

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

QUESTION 6.08 (3.00)

With regard to the Recirculation Flow control system, answer the following questions.

a. What is the purpose of the FLUX ESTIMATOR ? (1.0)

b. What two conditions will cause the FLUX ESTIMATOR to "automatically" shift from estimated flux signal to neutron flux as the flux feedback signal to the flux estimator. (1.0)

c. Give 5 of 10 conditions that will cause a FLOW CONTROL VALVE to "LOCK-UP" (1.0)

QUESTION 6.09 (3.00)

Concerning the RECIRCULATION FLOW CONTROL SYSTEM NETWORK, with the network in automatic flow control (flux manual - loop automatic) and 90% core flow:

A. what would be the results on individual loop flows (GIVE APPROMIXATE VALUES) if the Recirculation loop flow A feedback signal DECREASED TO ZERO ? EXPLAIN YOUR ANSWER (RECIRC FLOW CONTROL NETWORK Figure 3 attached) (2.0)

B. What would stop the flow control valve movement with the conditions as stated in part A ? (1.0)

QUESTION 6.10 (3.00)

What are three (3) anticipatory scrams, how is each sensed, and when is each bypassed ? (INCLUDE SETPOINTS)

QUESTION 6.11 (1.00)

The Main Generator is on line at 810 megawatts when a hydrogen leak in the generator reduces hydrogen pressure to 45 psig. Using the attached figure #4 (Estimated Capability Curve), which of the following is the maximum lagging Reactive load allowed on the generator, if a power factor of 0.975 is to be maintained.

A. 890 MVAR

B. 200 MVAR

C. 280 MVAR

D. 430 MVAR

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

QUESTION 7.01 (2.00)

Briefly explain WHY each of the following RECIRCULATION PUMP STARTING LIMITATIONS are necessary. Be specific.

A. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and the operating recirc loop are within 50 degrees F of each other. (1.0)

B. If the temperature of the water in the lower head is more than 145 degrees F below vessel saturation temperature, the recirc pump shall not be started (1.0)

QUESTION 7.02 (2.50)

For the suppression chamber water temperatures listed below, WHAT ACTIONS are required by the Technical Specifications with the unit in operational condition 1 or 2 ?

A. 97 degrees F (0.5)

B. 108 degrees F during RCIC testing (1.0)

C. 121 degrees F following a scram with the MSIV's SHUT. (1.0)

QUESTION 7.03 (2.00)

According to the TECHNICAL SPECIFICATIONS for the REACTIVITY CONTROL SYSTEMS:

A. When must the Rod Worth Minimizer be operable and what actions are necessary to continue operation with the RWM inoperable? (1.0)

B. Define the term "Limiting Control Rod Pattern". (1.0)

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

QUESTION 7.04 (2.50)

A reactor Cooldown is in progress and NEITHER loop of RHR can be placed in the Shutdown Cooling Mode. In accordance with Emergency procedure 5.3.5, Alternate Shutdown Cooling(contingency) has been established to remove decay heat and continue the cooldown. STATE the condition/status of the following components/parameters when operating in this mode of shutdown cooling.

a. MSIV's (0.5)

b. SRV's (0.5)

c. RHR Loops A & B (0.5)

d. Reactor Pressure (Compared to Suppression Chamber Pressure) (0.5)

e. Reactor Level (Provide numerical value or component reference) (0.5)

QUESTION 7.05 (1.50)

A LOCA has occurred and a high temperature steam environment exists in the drywell. EXPLAIN why the drywell sprays must NOT be initiated in the "Unsafe" region of attached Figure #5 "Drywell Spray Initiation Pressure Limit".

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

QUESTION 7.06 (3.00)

A. Procedure 3.3.1 REACTOR SCRAM gives operating requirements for 2 situations for the RWCU system and states that the operator should observe these operating requirements. (CONSIDER each situation seperately and occurring after a reactor scram).

SITUATION #1 - The RFW CAPABILITY is maintained (RFWDT OPERATING) and the RWCU isolates.

SITUATION #2 - The RFW CAPABILITY is lost (RFWDT TRIP OR MSIV CLOSURE) and the RWCU does not isolate.

1. What are the operating requirements for situation #1? (0.75)

2. What are the operating requirements for situation #2? (1.25)

3. What is the purpose of these operating requirements? (0.5)

B. 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. (0.5)

QUESTION 7.07 (1.50)

During a reactor shutdown with the Reactor power at 18% the Rod Sequence Controller becomes inoperative. The STA recommends that he will act as the second knowledgeable individual to allow continued rod insertion. As the Shift Manager would you follow the STA's recommendation ? EXPLAIN your answer.

QUESTION 7.08 (1.00)

As the Shift Manager you have determined that a Reactor shutdown has not been extensively disruptive to the normal alignment of systems, and therefore you can use the Minimum Startup Checklist to ensure preparations are made for a safe and orderly Reactor startup. The Minimum Startup Checklist need not be completed in full if two (2) conditions are met. What are those two (2) conditions.

QUESTION 7.09 (3.00)

The plant is in HOT shutdown(condition 3) with loop 'A' of RHR unavailable. On a loss of the RHR 'B' HX, WHAT are 3 of the 5 preferred alternate methods provided in PPM 4.4.2.1 LOSS OF RHR SHUTDOWN COOLING MODE LOOPS to control reactor temperature.
(For each method, state the system used to provide the reactor cooling water, source of the water used, and the heat sink.)

QUESTION 7.10 (3.00)

A reactor scram has occurred. Four adjacent control rods have failed to insert past position 06.

A. Match the following sets of indications with the appropriate potential problem type.

1. 3 RPS white lights are ON, 1 RPS white light is OFF, 4 blue lights on the full core display, are NOT ON

2. All RPS white lights are OFF, 4 blue lights, on the full core display, are NOT ON

3. All RPS white lights are OFF, all blue lights, on the full core display are ON

a. Air problem

b. Hydraulic problem

c. Electrical problem

(1.0)

B. With a number of control rods immovable, such as above what further criteria needs to be met, per PPM 5.1.3 Reactor Power Control to require initiating Standby Liquid Control? (1.0)

C. If boron injection is required and can not be injected with Standby Liquid Control, what systems per PPM 5.1.3 Reactor Power Control (RPV/Q) are to be used to inject boron into the vessel. (1.0)

QUESTION 7.11 (3.00)

PPM 4.12.1.1 CONTROL ROOM EVACUATION, lists NINE (9) immediate operator actions that should be performed prior to leaving the control room. What are six (6) of the immediate operator actions performed in the control room ?

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

QUESTION 8.01 (2.50)

During plant operation a Drywell entry is desired to locate a possible leak. Give 5 of 8 Prerequisites which must be met prior to personnel entering the Drywell.

QUESTION 8.02 (1.50)

A licensee may take reasonable action that departs from a license condition or a Technical Specifications in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and Technical Specifications that can provide adequate or equivalent protection is immediately apparent.

A. Such an action shall be approved, as a minimum,by WHOM? (0.5)

B. What two (2) notifications should be made prior to the above action if at all possible, but always as soon as possible afterwards ? (1.0)

QUESTION 8.03 (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 resposibilities of the Shift Manager, when reviewing a RWP for approval? (1.5)

QUESTION 8.04 (1.00)

Prior to accepting a clearance order, a maintenance supervisor determines additional clearance tags are required for a safe working condition. How are the additional clearance tags documented as approved on the previously authorized clearance order ?

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

QUESTION 8.05 (2.50)

A. After completing the maintenance on a piece of equipment the responsible maintenance group left the site without obtaining a clearance release. Operations has a need for this equipment. Name 3 individuals by title who can authorize the clearance release if the individual it is issued to can not be contacted. (1.5)

B. When is "Redundant Verification" required by the equipment clearance and tagging procedure ? (1.0)

QUESTION 8.06 (1.00)

After maintenance on a piece of equipment an operational checkout of the equipment is desired. According to the Equipment Clearance and Tagging Procedure, Give 2 of 3 restrictions that would prevent the Shift Manager from authorizing the equipment checkout .

QUESTION 8.07 (1.00)

When is the Plant Manager's permission required before the reactor can be restarted after a reactor scram ?

QUESTION 8.08 (3.00)

During the performance of a surveillance on a safety-related system the need for a TEMPORARY PROCEDURE DEVIATION arises.

A. Give 4 of 7 specific guidelines that must be complied with in the preparation of the temporary procedures deviation. (2.0)

B. Is the procedure deviation required to be documented prior to its implemention ? EXPLAIN YOUR ANSWER. (1.0)

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

QUESTION 8.09 (3.00)

What is the guideline or instruction given for each of the following per the standing operating orders, 1.3.1 attachment 1 ?

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 backseated. (0.75)

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

QUESTION 8.10 (3.00)

What is the minimum facility staffing required for WNP-2 by the Technical Specifications when in condition 2 (mode 2) and indicate what type of license is required for each staffing position, if any ?

QUESTION 8.11 (2.00)

Fuel Zone Indicator MS-LI-610 is classified as "unqualified" in the event of a LOCA per STANDING ORDERS / NIGHT ORDER PROCEDURE 1.3.1.

A. If this instrument appears to be working during a LOCA, can the instrument be used? (1.0)

B. Explain why the instrument can or cannot be used. (DO NOT JUST SAY BECAUSE OF THE PROCEDURE) (1.0)

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

QUESTION 8.12 (1.50)

You as the Shift Manager are reviewing a procedure change to a Technical Specifications surveillance which changes the setpoint of the Reactor Vessel Steam Dome Pressure High Scram from less than or equal to 1057 psig to 1065 psig plus or minus 5 psig.

Does this procedure change involve an unreviewed safety question concern? Explain your answer. (1.5)

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

ANSWERS -- WNP-2 -86/02/03-MILLER, L.

ANSWER 5.01 (1.50)

a. Doppler or fuel temperature
b. Void
c. VOID
(EACH 0.5 pts)

REFERENCE
WNP-2 REACTOR THEORY - STUDENT TEXT pg. 26-31
G.E. REACTOR PHYSICS, REACTOR FUNDAMENTALS TRAINING

ANSWER 5.02 (2.00)

a. Using P = Po e to the t/T then P = 75 e to 60/-80
P = 75 e to -0.75 = 35 on Range 4 [1.0]

b. On down-power transients, the rate of power change is limited by the rate of decay of the longest lived precursors, thus retarding the rate of power decrease.[1.0]

REFERENCE
WNP-2 REACTOR THEORY - STUDENT TEXT pg. 14,24
G.E. REACTOR PHYSICS, REACTOR FUNDAMENTALS TRAINING

ANSWER 5.03 (1.50)

Withdrawal of a center control rod at 90% density has greater worth.(0.5)
The control rod worth is proportional to the (local neutron flux / the core average neutron flux)squared.(0.25)
With 90% rod density the core average neutron flux is very small. Withdrawing a central control rod, increases the local flux in the area of withdrawn rod substantially. Because the rod causes the value of the term (local neutron flux /core average neutron flux)squared to be large its worth for this condition is quite high. Higher than withdrawing the rod at 40% rod density, when core average flux will be higher. (0.75)

REFERENCE
WNP-2 REACTOR THEORY - STUDENT TEXT pg. 26 - 36
G.E. REACTOR PHYSICS, REACTOR FUNDAMENTALS TRAINING

ANSWER 5.04 (2.00)

Voids initially decrease (0.25) as the increased flow moves the boiling boundary higher in the core.(0.25) The decrease in void content initially causes a positive reactivity addition. (0.25) As power increases, the rate of boiling increases (0.25), the increased void formation adds negative reactivity (0.25), and increased fuel temperature adds negative reactivity (0.25). The boiling boundary returns to near its original level (0.25). The net reactivity returns to 0 at steady state conditions (0.25).

REFERENCE
WNP-2 SYSTEMS VOL.I RECIRCULATION FLOW CONTROL pg.26
WNP-2 REACTOR THEORY - STUDENT TEXT pg. 28

ANSWER 5.05 (1.00)

Reverse power effect is a decrease in power with a notch withdrawal of a shallow control rod. The notch withdrawal causes bundle power to increase where the rod is withdrawn. The local power increase will cause increase void content in the bundle. The negative effect of the increase voiding is larger than the positive effect of the notch withdrawal of the shallow rod.

REFERENCE
WNP-2 REACTOR THEORY - STUDENT TEXT pg. 40
G.E. REACTOR PHYSICS, REACTOR FUNDAMENTALS TRAINING

ANSWER 5.06 (2.00)

A. Shutdown margin assumes xenon free for its LCO limit. The shutdown margin is not acceptable if the reactor is only shutdown by 1% dk/k as measured at peak xenon. (1.0)
B. Since the peak xenon reactivity is greater than the 1% SDM a reactor restart would occur. (1.0)

REFERENCE
WNP-2 Reactor Physics Sec.V Fission Product Poisons,part 2 Xenon behavior after Reactor Shutdown
WNP-2 TECHNICAL SPECIFICATIONS 1.39 pg. 1-7

ANSWER 5.07 (1.50)

A. answer b, less than double the original flow (0.5)

B. Less than double the original flow when delivering water into a piping system that offers frictional resistance, 2 pumps operating in parallel will encounter greater resistance to flow. The increased frictional resistance lowers the total flow to less than twice the original flow. [1.0]

REFERENCE
Thermodynamics,Heat Transfer and Fluid Flow pg. 7-121

ANSWER 5.08 (3.00)

a. Turbine load would decrease (0.5) due to the decrease in reactor pressure caused by the cool water spraying into the upper plenum.

b. Reactor level will increase (0.5). A level error must be generated to reestablish steady state conditions in the FWLCS (0.5).

c. Feedwater flow will decrease (0.5). The HPCS injection is providing a portion of the required feed for the reactor and this is not sensed by the Feed Flow detectors (0.5).

REFERENCE
WNP-2 FSAR CH.15.5.1 INADVERTANT HPCS STARTUP

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 = APLHGR/LIMLHGR -or- = APLHGR/MAPLHGR limit (1.0)
-or- = (APLHGR) actual/ (MAPLHGR) LCO max

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

ANSWER 5.10 (3.00)

a. Decreases [0.25]. There is less steam flow, therefore, less pressure drop through the main steam lines [0.75]. (1.0)

b. Increases [0.25]. With the same amount of cooling water through the condenser [0.25] and less of a heat load [0.5]. (1.0)

c. Decreases [0.25]. Less extraction steam from the turbine to heat the feedwater [0.75]. (1.0)

REFERENCE
WNP-2 SYSTEMS VOL.V DEH Fig. 13A
WNP-2 SYSTEMS VOL.V EXTRACTION STEAM
WNP-2 SYSTEMS VOL.V CIRCULATING WATER

ANSWER 5.11 (2.00)

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensil at the outer wall. These thermal induced compressive stresses tend to alleviate the tensil stresses induced by the internal pressure. (Drawing attached Figure 1)

REFERENCE
WNP-2 TECHNICAL SPECIFICATIONS BASES 3/4.4.6

ANSWER 5.12 (3.00)

a. Critical power increases (0.25) due to the increase in enthalpy rise that is required to bring the coolant to saturated conditions (0.75).

b. Critical power increases (0.25) due to increased power required to bring the coolant to saturation conditions or increasing the flowrate increases the heat removal capability and places the bundle farther from OTB. (0.75)

c. Critical power decreases (0.25) due to it requiring a smaller enthalpy rise to change a liquid into a vapor at higher pressures than at lower pressures (0.75).

REFERENCE
WNP-2 THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW 9-85 TO 9-87

ANSWER 6.01 (2.00)

a. NO (0.25), the fission gamma and neutron signal produced for the upper portion of the IRM range is greater than the signal that is contributed by the decay gammas, and fission gammas are proportional to the number of fissions taking place. (0.75)

b. 1. Detector High Voltage Low [0.33]
   2. Module Unplugged [0.33]
   3. IRM Mode Switch not in operate [0.33]

REFERENCE
WNP-2 Systems Vol. II, IRM pg. 23
WNP-2 SYSTEMS VOL. II, SRM figure 10

ANSWER 6.02 (2.00)

a

REFERENCE
WNP-2 SYSTEMS VOL V DEH pg. 62

ANSWER 6.03 (1.50)

a. 1. The U234 loading (79%) allows the detector to regenerate the fissile coating.
   2. The sputtering effect is reduced.

b. The APRM reading is lower than actual power (0.5).

REFERENCE
WNP-2 SYSTEMS VOL.II LPRM pg.7
WNP-2 SYSTEMS VOL.I PROCESS COMPUTER pg.14

ANSWER 6.04 (1.50)

Yes, it is normal (0.5). The drive piston moves the RPIS magnet past the "00" reed switch (or the control rod moves past "00" position) and actuates only the green full-in light, "overtravel in" reed switch (1.0).

REFERENCE
WNP-2 SYSTEMS REACTOR MANUAL CONTROL, pg 19

ANSWER 6.05 (2.00)

i. a

ii. c

REFERENCE
WNP-2 SYSTEMS VOL V. FWLC pg 15
WNP-2 FSAR 15.5.1 INADVERTANT HPCS STARTUP

ANSWER 6.06 (2.00)

a. .66 * w + 51% (0.25)
118% (0.25)

b. Recirc Loop flow elements (pump suction) (0.5)

c. Only the 118% fixed scram (0.5) This is because the flow biased scram incorporates a time delay into its actuation (~ 6 seconds, representative of the fuel thermal time constant) (0.5)

REFERENCE
WNP-2 SYSTEMS VOL II APRM pg. 9,13 AND 20

ANSWER 6.07 (4.00)

A. The RCIC pump will not start (0.25) and will not inject. (0.25)

(RCIC-V-45) will not open if the exhaust valve (RCIC-V-68) is not full open (0.5)

B. The LPCS pump will start and run (0.25) but will not inject (0.25) because there is no flowpath available.(0.5)

C. The RHR pump 2C will start and run (0.25) but will not inject (0.25) because there is no flowpath available.(0.5)

D. The RHR pump 2B will start and trip (0.25) and will not inject (0.25) because of the short duration of the start. (0.5) (also accept RHR 2B will not start due to the suction valve interlocks,the supply breaker will shut and then reopen, there will be no injection due to no suction path and no pump running.)

REFERENCE
WNP-2 System Vol III RCIC pg. 21
WNP-2 System Vol III RHR pg 19 and 20
WNP-2 System Vol III LPCS pg. 8

ANSWER 6.08 (3.00)

A. The flux estimator is used to provide a more stable flux feedback signal to the recirc flow control system than the actual APRM can provide to minimize the valve hunting.(also accept - to provide a more stable flux signal to minimize FCV wear due to valve hunting) (1.0)

B. 1. APRM flux signal greater than 110% of rated neutron flux (0.5)
2. The difference between the APRM and the estimated flux signals exceed +/- 5% of rated neutron flux. (0.5)

C. 1. An actual High Drywell Pressure signal. (1.68)
2. A High Drywell / Valve Motion Inhibit Relay Logic test switch signal.
3. Loss of any DC power supply.
4. Loss of 120 VAC power
5. Loss of control signal.
6. Pump motor overload / undervoltage
7. Oil temperature hot (150 degrees F)
8. Reservoir oil level low-low (20 gallons per segment)
9. Pump discharge pressure loss (less than 1300 psig).
10. Servo valve control power loss.
(Any 5 of 10 0.2 pts each)

REFERENCE
WNP-2 SYSTEMS VOL. I RECIRC FLOW CONTROL pg. 16,18

ANSWER 6.09 (3.00)

A. Individual loop 'A' flow would increase 10% (0.5)
(also accept increase loop 'A' flow to 100%)
individual loop 'B' flow would remain the same (0.5)
The error signal from the flow reference signal and the flux controller signal summer would be limited to 10% by the error limiter.(1.0)

B. The flow control valve motion would stop when the valve opened the initial 10% . The position feedback would null the signal from the loop flow controller. (1.0)

REFERENCE
WNP-2 SYSTEMS VOL.I RECIRCULATION FLOW CONTROL pg. 25 AND FIG.9

ANSWER 6.10 (3.00)

1. MISV CLOSURE VALVE POSITION(0.3), greater than 6%(0.1) closed as sensed by limit switches on the valve(0.3), bypassed when the MODE switch is not in RUN and RPV pressure is less than 1037 psig.(0.3)

2. TURBINE THROTTLE VALVE CLOSURE(0.3), greater than 5% closed(0.1) as sensed by fast acting switch on any throttle valve(0.3), bypassed when power is less than 30% (as sensed by first stage pressure).(0.3)

3. TURBINE GOVERNOR VALVE FAST CLOSURE(0.3), less than 1250 psig on the EH oil(0.1) as sensed by a pressure switch(0.3), bypassed when power is less than 30% (as sensed by first stage pressure)(0.3).

REFERENCE
WNP-2 TECHNICAL SPECIFICATIONS BASES RPS LSSS
WNP-2 SYSTEMS VOL II RPS pg. 15-25

ANSWER 6.11 (1.00)

B. 200 MVAR

REFERENCE
WNP-2 SYSTEMS VOL.IV MAIN GENERATOR FIG 9.

ANSWERS -- WNP-2 -86/02/03-MILLER, L.

ANSWER 7.01 (2.00)

A. Prevents an undue stress on the vessel nozzles and bottom head region. (1.0)

B. Limits undue thermal stress on vessel (1.0)

REFERENCE
WNP-2 TECHNICAL SPECIFICATIONS B3/4.4.1 pg B3/4 4-1

ANSWER 7.02 (2.50)

A. Initiate suppression pool cooling or reduce suppression pool temperature to less than 90 degrees F.(0.5)

B. Stop RCIC testing (0.5) and initiate and restore average temperature to less than 90 degrees F (0.5)(within 24 hrs or be in HOT SHUTDOWN in the next 12 hrs and COLD SHUTDOWN within the following 24 hrs. NOT REQUIRED FOR FULL CREDIT)

C. Depressurize the reactor pressure vessel (0.5) to less than 200 psig (0.5)(within 12 hrs. NOT REQUIRED FOR FULL CREDIT)

REFERENCE
WNP-2 TECHNICAL SPECIFICATIONS, 3.6.2.1 pg 3/4 6-13

ANSWER 7.03 (2.00)

A. Whenever the reactor is in the startup or run mode and less than or equal to 20% rated thermal power.(0.25) Verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor console.(0.75)

B. A pattern which results in the core being on a thermal hydraulic limit (i.e.,operating on a limiting value for APLHGR,LHGR or MCPR). (1.0)

REFERENCE
WNP-2 TECHNICAL SPECIFICATIONS 3.1.4.1
WNP-2 TECHNICAL SPECIFICATIONS 1.19

ANSWER 7.04 (2.50)

a. Closed (0.5)

b. 2 (or 3) Open (0.5)

c. 1 RHR loop injecting to the Reactor (0.25)
1 RHR loop in suppression pool cooling (0.25)
(also acceptable RHR A and B in suppression pool cooling)

d. 76 - 120 psig > Suppression Chamber Pressure (0.5)
(any value between 76 and 120 psig is acceptable)

e. Reactor level > +648 inches above vessel zero (Main Steam Line elevation) (0.5)

REFERENCE
WNP-2 EMERGENCY PROCEDURES VOL.V , 5.3.5 ALTERNATE SHUTDOWN COOLING (CONTINGENCY)

ANSWER 7.05 (1.50)

Because spraying the drywell may result in a depressurization rate in the containment (drywell and supression chamber) which is beyond the capacity of the Reactor Building-to-Suppression Chamber Vacuum Breakers,(0.5) resulting in negative containment pressures in excess of design, leading to failure of the primary containment.(1.0)

REFERENCE
WNP-2 EPGs SEC. 08 pg.8.4-8

ANSWERS -- WNP-2 -86/02/03-MILLER, L.

ANSWER 7.06 (3.00)

A.1. RWCU should be diverted to the condenser (0.5) before the differential between the return to the RPV and feedwater exceeds 165 degrees F. (0.25)

2. Cooldown and depressurize the RPV (0.5) and do not divert RWCU flow away from the RPV (0.5) until the differential temperature is 165 degrees F or less (0.25).

3. The RWCU system operating requirements are to minimize feedwater piping thermal stress. (0.5)

B. The flow is returned to the RPV to repressurize the feed water piping to prevent excessive water hammer. (0.5)

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

ANSWER 7.07 (1.50)

NO (0.5), Control rod movement is not permitted below 20% rated thermal power except by scram, if the RSCS is not operable (1.0).

REFERENCE
WNNP-2 VOL III GENERAL OPERATING PROCEDURES 3.2.1 NORMAL SHUTDOWN TO COLD SHUTDOWN DEVIATION 85-466

ANSWER 7.08 (1.00)

1. The Reactor Trip and Recovery Report requires no corrective action prior to restart. (0.5)
2. Verbal concurrence is obtained from the Operations Manager or his designee to omit specified portions of the checklist. (0.5)

REFERENCE
WNP-2 VOL.III GENERAL OPERATING PROCEDURES MINIMUM STARTUP CHECKLIST 3.1.4, pg.1

ANSWER 7.09 (3.00)

1. RCIC injection with steam rejected to the main condenser or suppression pool.
2. Condensate pump injection via feedwater lines with steam rejected to the condenser (or suppression pool) and then to CST's via filter demins (F.D.'s).
3. HPCS (feed and bleed) suction from CST's with RPV steam rejected to the condenser (or suppression pool) and then to CST's via filter demins (F.D.'s).
4. LPCS (feed and bleed) suction on the suppression pool; steam to condenser.
5. RHR-C (feed and bleed) suction on suppression pool; steam to condenser.
6. Increase cooling water flow to RWCU system heat exchangers.
7. Operate a CRD pump and reject via RWCU.
   (Any 3 of the 7 Each method 1.0 pts.)

REFERENCE
WNP-2 VOL IV ABNORMAL CONDITION PROCEDURES 4.4.2.1 LOSS OF RHR SHUTDOWN COOLING MODE LOOPS pg 2

ANSWER 7.10 (3.00)

A. 1. c
   2. a
   3. b (3 @ 0.33ea) (1.0)

B. Reactor power is > 5% (0.5) or can not be determined and Suppression pool temperature > 110 deg F (0.5) (1.0)

C. 1. RCIC (0.5)
   2. RWCU (0.5)

REFERENCE
WNP-2 SYSTEMS VOL II RPS
WNP-2 VOL V EMERGENCY PROCEDURES REACTOR POWER CONTROL 5.1.3 pg 2
WNP-2 SYSTEMS VOL I CRDH

ANSWER 7.11 (3.00)

1. Manually scram the reactor.
2. Place the Reactor Mode Switch to Shutdown.
3. Manually close all MSIV's, MS-V-16,MS-V-19,and RWCU FCV-33
4. Verify that the APRM downscale lights are illuminated.
5. Trip the main generator and ensure that auxiliary power is being supplied from TR-S.
6. If the reactor recirc pumps are not being supplied from the LFMG, then transfer both recirc pumps to the LFMG.
7. Ensure only two condensate booster and two condensate pumps are operating.
8. Ensure not more than two circulating water pumps are running.
9. Verify valve positions RFW-10 CONTROLLER IN AUTO,RFW-V-118, RFW-V-117A,RFW-V-117B OPEN. RFW-V-112A,AND RFW-V-112B CLOSED.

(ANY 6 OF 9 0.5 pts.EACH)

REFERENCE
WNP-2 VOL.IV ABNORMAL CONDITIONS PROCEDURES PPM.4.12.1.1

ANSWER 8.01 (2.50)

A. Authorization for personnel to enter the drywell must be obtained from the Shift Manager and the Health Physics/Chemistry Manager prior to initiating this procedure. (Plant Manager's permission is required if the reactor is critical when personnel enter the drywell.)

B. The drywell must be deinerted prior to entry. (ensure oxygen readings > 19.5% on containment oxygen monitors)

C. Prior to entry into the Primary Containment with the reactor critical, reactor power level must be less than or equal to 5% and stable.

D. The Transverse in Core Probes (TIPS) must be withdrawn from the core and tagged out of service.

E. The nitrogen inerting system must be tagged out of service.

F. An RWP must be initiated (by the Shift Manager).

G. The CAS or SAS shall be notified immediately prior to opening any hatch into the drywell and a security officer posted.

H. The Plant Manager, or his designee, shall provide a list of personnel authorized access to the Primary Containment.

(ANY 5 OF THE 8 0.5 PTS. EACH)

REFERENCE
WNP-2 ADMINISTRATIVE PROCEDURES 1.9.3 PERSONNEL ENTRY TO PRIMARY CONTAINMENT PG. 1 AND 2

ANSWER 8.02 (1.50)

A. The action shall be approved by a licensed Senior Operator as a minimum. (0.5)

B. Operations Manager (0.5)
NRC (0.5) Operations Center

REFERENCE
10 CFR 50.54 X AND Y ,ADMINISTRATIVE PROCEDURES 1.2.3 pg.2;1.1.3 pg.3; 1.3.1 pg.2

ANSWER 8.03 (3.00)

A. DIRECT RADIATION >/= 2.5mREM/HR (0.5)
AIRBORNE RADIOACTIVITY >/= 25% OF MPC (0.5)
CONTAMINATION >/= 1000 DPM/100CM2 BETA GAMMA OR
>/= 100 DPM/100CM2 ALPHA (0.5)

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 PPM.1.11.3 pg.9

ANSWER 8.04 (1.00)

The changes need to be initialed by the shift manager to signify his approval.

REFERENCE
WNP-2 ADMINISTRATION PROCEDURE 1.3.8 PROCEDURE DEVIATION 84-1058

ANSWER 8.05 (2.50)

A. 1. The worker's supervisor
2. The worker's department manager
3. The shift manager
(3 @ 0.33 pts. EACH)

B. When the component is Safety related or Fire Protection related. (0.5)

REFERENCE
WNP-2 EQUIPMENT CLEARANCE AND TAGGING PROCEDURE 1.3.8 pg 6

ANSWER 8.06 (1.00)

1. If the clearance order is a multiple clearance order, temporary lifting of the tags is not permitted.

2. Only checkouts or tests that should take a short time (less than one hour) are allowed.

3. The checkout or test may not run through a shift change.

(ANY 2 OF 3 0.5 PTS. EACH)

REFERENCE
WNP-2 EQUIPMENT CLEARANCE AND TAGGING PROCEDURE 1.3.8 pg 8

ANSWER 8.07 (1.00)

1. The cause of the reactor trip has not been determined and corrected. (0.5)

2. There are reportable occurrences, other than the reactor trip itself, associated with the reactor trip. (0.5)

REFERENCE
WNP-2 REACTOR TRIP AND RECOVERY,pg 2

ANSWER 8.08 (3.00)

A. 1. The procedure deviation shall not change the intent of an approved procedure.
2. The deviation should be marked-up in the appropriate text sections of the affected procedure.
3. If a deviation is too extensive to be easily understood or cannot be markedup on the applicable pages, a procedure revision is mandatory.
4. If more than one procedure is affected by a procedure deviation, each procedure shall have a seperate deviation.
5. Shall not alter any of the criteria items listed 1 through 5 on the Procedure Revision form.
6. Approved by two members of Plant Management / Supervisory Staff, at least one of whom holds a Senior Reactor Operator's license for the plant; both of whom are knowledgable in the areas affected.
7. Deviation is documented with the "Procedure Deviation form"
8. Reviewed by the POC
9. Approved by the Plant Manager within 14 calendar days of implementation.

(ANY 4 OF THE 9 0.5 PTS. EACH)

B. No, Providing it has been approved verbally by two members of Plant Management / Supervisory Staff (1.0)

REFERENCE
WNP-2 USE OF PLANT PROCEDURES 1.2.3 pg 2 AND 3
WNP-2 TECHNICAL SPECIFICATIONS 6.8.3

ANSWER 8.09 (3.00)

A. Operators are not to override the automatic actions of ECCS and other safety features, unless confirmed by at least two independent indications of a misoperation in the automatic mode or adequate core cooling is assured.

B. An operator may place a controller in the manual mode from the automatic mode whenever, in the judgement of the operator, continued automatic operation is undesirable.

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

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 EACH)

REFERENCE
WNP-2 STANDING ORDERS / NIGHT ORDERS, 1.3.1 ATT 1 pg 2-5

ANSWER 8.10 (3.00)

1 - Shift Manager - SRO	(0.6)
1 - Control Room Supervisor - SRO	(0.6)
2 - Reactor Operators - RO	(0.6)
2 - Equipment Operators - none	(0.6)
1 - Shift Technical Advisor - none	(0.6)

REFERENCE
WNP-2 TECHNICAL SPECIFICATIONS TABLE 6.2.2-1 pg 6-6

ANSWER 8.11 (2.00)

A. No, the instrument can not be relied on. (1.0)

b. The instrument is unquailified due to enviromental conditions in the containment. (1.0)

REFERENCE
WNP-2 PPM 1.3.1 pg. 2,7

ANSWER 8.12 (1.50)

Yes,(0.5) the procedure change does involve an unreviewed safety concern. The change decreases the margin of safety as defined in the basis for the Technical Specifications. (1.0)

REFERENCE
10CFR50.59 (A)(2)
WNP-2 TECHNICAL SPECIFICATIONS 2.2.1 pg. 2-4

QUESTION	VALUE	REFERENCE
05.01	1.50	LZM0000001
05.02	2.00	LZM0000002
05.03	1.50	LZM0000003
05.04	2.00	LZM0000004
05.05	1.00	LZM0000005
05.06	2.00	LZM0000006
05.07	1.50	LZM0000007
05.08	3.00	LZM0000008
05.09	2.50	LZM0000009
05.10	3.00	LZM0000010
05.11	2.00	LZM0000011
05.12	3.00	LZM0000012

	25.00	
06.01	2.00	LZM0000034
06.02	2.00	LZM0000035
06.03	1.50	LZM0000036
06.04	1.50	LZM0000037
06.05	2.00	LZM0000038
06.06	2.00	LZM0000039
06.07	4.00	LZM0000040
06.08	3.00	LZM0000041
06.09	3.00	LZM0000042
06.10	3.00	LZM0000043
06.11	1.00	LZM0000044

	25.00	
07.01	2.00	LZM0000013
07.02	2.50	LZM0000014
07.03	2.00	LZM0000015
07.04	2.50	LZM0000016
07.05	1.50	LZM0000017
07.06	3.00	LZM0000018
07.07	1.50	LZM0000019
07.08	1.00	LZM0000020
07.09	3.00	LZM0000021
07.10	3.00	LZM0000022
07.11	3.00	LZM0000023

	25.00	
08.01	2.50	LZM0000024
08.02	1.50	LZM0000025
08.03	3.00	LZM0000026
08.04	1.00	LZM0000027
08.05	2.50	LZM0000028
08.06	1.00	LZM0000029
08.07	1.00	LZM0000030

E 2

QUESTION	VALUE	REFERENCE
08.08	3.00	LZM0000031
08.09	3.00	LZM0000032
08.10	3.00	LZM0000033
08.11	2.00	LZM0000045
08.12	1.50	LZM0000046

	25.00	

	100.00	