

Examination Report No. 50-397/OL-84-03

Facility: Washington Nuclear Plant No. 2

Docket No. 50-397

Examinations administered at Washington Nuclear Plant No. 2, Richland, Washington from November 6 to November 8, 1984.

Chief Examiner:

R. Pate for:
F. C. Gage
Operator License Examiner

3/9/85
Date Signed

Approved:

R. Pate for:
J. O. Elin, Chief
Operations Section (Acting)

3/9/85
Date Signed

Summary:

Examinations on November 6-8, 1984

Written examinations were administered to eight SRO and one RO candidates. Operating examinations (Oral and simulator) were administered to eight SRO and one RO candidates. Four SRO candidates passed the examinations.

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REPORT DETAILS

1. Persons Examined

Examinations were administered to nine candidates; one Reactor Operator candidate, and eight Senior Operator candidates.

2. Examiners

Paul Gage
Ira Levy
Gary Sly

3. Examination Review Meeting

An exam review meeting was held immediately after the written exams were administered, on November 6, 1984. The following utility representatives were in attendance:

John Wyrick
Mike Kappl
Bob Beardsly
Andy Langdon
Mark Westergren
Tim Messersmith
Steve Redniak
Ed Wright

Additionally, the following NRC representatives were present:

Robert Pate
Paul Gage
Ira Levy
Gary Sly
Gary Johnston (observer)

The responses to the comments provided by the utility representatives are included as enclosure (1). Additional comments were provided by letter from G. C. Sorensen to J. B Martin, dated November 13, 1984. The responses to these comments are included as enclosure (2). Where applicable the examination keys have been changed.

4. Exit Meeting

An exit meeting was held with the facility on November 9, 1984. The attendees were:

NRC:

Robert Pate - Chief, Reactor Safety Branch
Paul Gage - NRC Examiner
Gary Johnston - NRC Examiner (Observer)
Al Toth - Sr. Resident Inspector

Utility:

Jack Shannon - Director, Power Generation
Jerry Martin - Plant Manager, WNP-2
Roger Corcoran - WNP-2 Operations Manager
Clu Powers - Assistant Plant Manager, WNP-2
John Wyrick - Nuclear License Training Manager
Rich Stickney - Manager Technical Training
John Little - Plant Training Coordinator

The candidates that were a clear pass on the Operating Examination (Oral and Simulator) were identified.

A discussion of how Job Task Analysis (JTA) would be used in future examinations was held. It was explained to the facility personnel that the written examinations for this examination were not based on JTA.

The current status of the plant simulator was discussed. The NRC position was that the simulator was very limited in its present state and was barely satisfactory for use in examinations. The facility staff stated that there were plans to improve the condition of the simulator.

RO/SRO EXAM REVIEW COMMENTS AND RESOLUTIONS

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

RO Exam

Section 1: 1.2a; 1.6;

Section 2: 2.2b; 2.2d; 2.3a(2); 2.4b(3); 2.7a, 2.7d; 2.9b

Section 3: 3.5a; 3.5b; 3.6c; 3.8b; 3.8c

Section 4: 4.3b(4); 4.4b; 4.14

SRO Exam

Section 5: 5.7b; 5.8; 5.10.

Section 6: 6.1b (2); 6.2a; 6.3a; 6.5b; 6.6; 6.8; 6.9a.

Section 7: 7.2; 7.5b.

Section 8: 8.2b; 8.3; 8.8a; 8.11.

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

KO Exam

Section 1

Question 1.1

Facility Comment: Request partial credit for using period of 40 second.

Response: Knowledge of "doubling time" is important part of question. Anyone can plug equation if period is given in seconds. No change to answer key.

Question 1.13.b

Facility Comment: Assuming no change in core average neutron flux.

Response: Question said "taken separately" and core average flux is part of question 1.13.c. No change to answer key.

Question 1.13.c

Facility Comment: Assuming no change in local flux. Usually local flux will change and could result in no change in core average flux.

Response: Even if an increase in core average flux did cause an increase in local flux, the change would be small relative to that of the core. The ratio of the square of the local over the average would still decrease and, therefore, so would differential rod worth.

Section 2

Question 2.2.a

Facility Comment: Facility requests low level alarm be acceptable as a safety function. Also comment that level 4 may not necessarily be assumed to be coincident with loss of a RFP.

Response: A low level alarm is not an auto-initiated safety function. No credit for that answer. Recirculation pump run back will only occur, if there is a low level and a coincident loss of one reactor feed pump.

Question 2.4.a

Facility Comment: Other acceptable answers are:

(1) SRM, IRM, APRM slide links for bypassing non coincident scrams.

(2) Manual bypass of IRM.

(3) Manual bypass of SRM.

Request full credit for 3 or 4 correct answers

Response: Additional answers added to key. All correct answers required for full credit.

Question 2.5. a&b.

Facility Comment: The question is unfair as it requires the candidate to memorize relay numbers.

Response: Required key response is too detailed. Parts a and b of this question have been deleted.

Question 2.5c

Facility Comment: Operators are not required to memorize 480V distribution.

Response: Operators should recognize which bus supplies the shutdown cooling system. If asked the operator should be able to draw a one line diagram of the 480V distribution system that serves the engineered safety features equipment.

Question 2.7.b

Facility Comment: Operators should not be required to memorize steps in an operating procedure. Also, first part of answer should be "yes".

Response: See response to Sorensen letter Attachment 2. The key for the first part of the answer was changed to "yes".

Question 2.8b

Facility Comment: Time delay should not be required. Also, TSW pump selected for auto start will come on after the diesel has powered the bus on loss of offsite power.

Response: Key changed to remove time delay and add additional correct answer. All three answers required for full credit.

Section 3

Question 3.1b

Facility Comment: Answer in key is true only for a "hardware failure". Otherwise, during its normal function, no alarms are operable.

Response: Alarms are for abnormal conditions like hardware failure. It's true they will not be in the alarmed condition during the normal function of the RWM. No change to key.

Question 3.3a and b

Facility Comment: Question unfair, MOB's are covered in requalification training not as part of hot license training.

Response: Question 3.3a has been deleted. The part b is a systems question. The candidates should know why the temperatures are monitored even if they had not read the MOB's.

Question 3.6.a

Facility Comment: This question is not realistic. Operators do not memorize FIS switch numbers.

Response: The question stated that it was the flow indicating switch for the RCIC pump. The number was provided, but was not necessary. However, since this question asks for more detail than normally asked, it has been deleted.

Question 3.6b

Facility Comment: Part b, should also accept "all located on flow elbows inside drywell"

Response: Partial credit will be given for "all located on flow elbows inside drywell".

Section 4

Question 4.1

Facility Comment: Question weighted too heavily for a single paragraph out of one operating procedure.

Response: Examiner disagrees with facility comment. No change to answer key.

Question 4.2

Facility Comment: This procedure is not used for operation of the RWM. The volume 7 surv. procedure is used to verify RWM operability and to operate that piece of equipment. Does not assess the operators knowledge of system operation. It checks which lights come on or off. Rote memorization of procedure not required.

Response: See response to Sorensen letter.

Question 4.5a

Facility Comment: Only answer 1 if found in the abnormal procedures listed in the reference. #2 is not ever applicable, #3 and #4 are the same thing and they result from #1.

Response: The key has been changed to delete #3. #4 is acceptable only if there is no turbine trip.

Question 4.5b

Facility Comment: This is okay if #1 only is accepted. On a turbine trip, you don't worry about this because a loss of F.W. heating occurs with every turbine trip.

Response: With the changes to 4.5a, no change to 4.5b is necessary.

Question 4.6

Facility Comment: This question requires total recall of procedure.

Response: See response to Sorenson letter.

Question 4.12b

Facility Comment: This can easily be confused with the licensed operator requirement of T.S. 6.2.2.d which requires an SRO for core alterations. Should give credit for a true answer as well.

Response: A "true" answer is acceptable, if the T.S. is referenced.

SRO EXAM

Section 5.

No additional facility comments.

Section 6.

Question 6.2.b

Facility Comment: Throw out-not relevant

Response: See response to Sorensen letter.

Question 6.3.b

Facility Comment: Too detailed, throw out.

Response: Key answer too detailed. Changed to accept a description that it is a centrifugal device used to actuate the trip lever or some similar statement.

Question 6.4.a

Facility Comment: Add "all 7 ADS valves" to key.

Response: Since there are 14 switches, answering "all 7 ADS valves" is not specific enough.

Question 6.4.b

Facility Comment: Use of the word "normal" is misleading - would never push ADS pushbuttons under normal conditions - may have confused some people.

Response: If it appears the use of the word "normal" confused a candidate, consideration will be given in the grading.

Question 6.5.c

Facility Comment: Throw out-too much detail-candidate not required to know all power supplies by memory.

Response: ES-402-A.2 states "Candidate should be able to reproduce from memory...electrical distribution system." Also see response to Quesiton 2.5c.

Question 6.7.a

Facility Comment: Also withdraw block, RWM Block (possibly-not always).

Response: The answer key will be changed to add withdrawal block, but not RWM block.

Question 6.7b

Facility Comment: "Or bypassing on RSCS." Also may say to bypass RWM.

Response: Answer key changed to include "bypassing on RSCS".

Question 7.3.a

Facility Comment: Answer should be generalized to simply say strip bus, energize bus, reload bus. The present answer requires memorization of operating procedures.

Response: The answer key has been changed to remove panel and breaker numbers as required for full credit. However, more description is required than proposed by the facility. See response to Sorensen letter.

Question 7.4

Facility Comment: There are more than 1400 Annunciator procedures. Two points is too great of importance to place on one of them. Please consider reducing point value.

Response: The correct answer is a logical response to any annunciator that has upscale and down scale alarms and is applicable to many annunciators, especially alarms for monitors that measure radiation and could indicate a possible excessive radiation level or effluent release. Memorization of 1400 annunciator procedures is not required. Also see response to Sorensen letter. Answer key revised to remove numbers from the required answer.

Question 7.5.b

Facility Comment: Requires total recall of operating procedures. Also the evolution is only for startup after complete drain down and is done very seldom.

Response: This question requires system knowledge expected of an SRO candidate. However, since this is done very seldom, this part of question was deleted.

Question 7.6

Facility Comment: Procedure is wrong - Tech. Specs. require RWM-RSCS to be operable prior to decreasing below 20%.

Response: Examiner disagrees with facility comment. Surveillance is required to be performed within one hour after RWM auto-initiation when reducing thermal power (WNP-2, T.S; 4.1.4.1.c, 3/4 1-16). No change to answer key.

Question 7.8

Facility Comment: Question requires total recall; doubtful anyone will get right. Please consider point values.

Response: The question includes information that an SRO is expected to know, but to confirm by checking the Tech. Specs. Question has been deleted.

Question 8.2.a

Facility Comment: Answer "a" gives only the action statement for the LCO. Credit should also be given for explanation of why the candidate believes his answer to be as he indicated. If you wanted the action statement, it should have been asked for.

Comment: Credit will be given for reasonable explanations as to why the Tech. Specs. require the candidate to take action.

Question 8.9

Facility Comment: This is an unfair question!

Initial training does not necessarily incorporate MOB training into it. Without the benefit of reading the MOB, the chances of candidates getting the correct answer is minimal! Additionally, part a and c are more of a system oriented question than admin., etc., and should not be part of category 8.

Response: Part b has been deleted. Parts a and c require systems knowledge. The candidates should be able to answer the questions without reading the MOB. No change to parts a and c.

Question 8.10

Facility Comment: Memorization of EPIP 13.1.1 is not feasible. While the majority of the symptomatic conditions which call for event classification are well known, the situational based events are much more vague. I cannot imagine any operator declaring an emergency events without reference to EOP's or EPIP's for guidance.

Response: SRO candidates are expected to have knowledge of both symptomatic-based condition and situation-based conditions which call for event classification. However, we agree that they would be expected to check the EOP's or EPIP's prior to making an emergency event declaration.

Response to Facility Comments Provided in Sorensen Letter

RO Exam

Question 2.5

Parts a) and b) are deleted, however, the operator should recognize which bus supplies the shutdown cooling system. If asked, the operator should be able to draw a one line diagram of the electrical distribution system for all engineered safety features equipment.

Question 2.7

The question asks about a specific valve by functional name and number. This is an important component in an important system. An operator should know the location relative to the pump (upstream or downstream). The point of the questions is whether the operator knows to close the valve to prevent a possible water hammer due to voids in the line. This is a fair question for several systems.

Memorization of a specific valve number or specific steps in the normal operating procedure is not required. Most of the credit will be for knowing about preventing water hammer.

Question 3.9

NUREG-1021, ES-202 B.3 states in part, "The candidate should have sufficient knowledge of the nuclear..., the process..., and radiological instrument (e.g., ionization, G-M, and scintillation), to answer questions concerning principles of detector operations, location and setpoints..." The question asked for types of detectors for radiological instruments (e.g., scintillation, ion chamber, fission chamber or Geiger Mueller) for specific processes. This is clearly within the scope of ES-202 B.3.

Question 4.2

The question asks for the operator to have knowledge of how the test/select and inop/reset push buttons work. He does not have to have the procedure memorized. Since these are controls available for the operator he should have knowledge of what happens when he operates them.

Question 4.6

Step by step memorization of the procedure is not required. The information requested is only part of the steps of the procedure. The intent of the question is to determine the level of awareness of the operator to control room indications. The operator should know that there are alarms associated with energizing an important safety bus (the exact name of the alarm is not required). Also, the operator should be aware of what voltage readings are available and the availability and meaning of control board indicating lights. These things are expected to be within the knowledge of the operator.

Question 4.12

NUREG-1021, ES-202 B.4 states that administrative procedures, including operating restrictions, limitations in the facility license and technical specifications may be included to the extent they are directly applicable to an operator. The number of operators required in the control room and on the refueling floor is directly applicable to the operator. He is expected to know the administrative requirements applicable to his job position.

SRO ExamQuestion 6.2

As stated in response to Question 3.9, types of detectors are expected knowledge for RO and SRO candidates. Also, logic systems that control radiation releases are very important to know so the SRO can understand and diagnose abnormal system behavior. If this information is not covered in the candidate training program, there is a serious gap in the program which can not only lead to failures on the NRC examination, but much worse, the candidate may pass the NRC examination and become a licensed operator not fully trained to perform the job and thereby, become a potential part of an excessive radiological release.

Question 7.3

All that is required for the answer is that the candidate know in general what needs to be done to accomplish the operation of tying the 250V battery B2-1 to the DC distribution bus S2-1. A step by step response is not required for full credit.

Question 7.4

The responses asked for do not require the procedure to be memorized. The question specifies the things the operator must do (i.e., verification and check). The candidate should have knowledge of which system and components are associated with an alarm, their general location (i.e., control room, local) and the safety significance.

Question 7.8

This question asks for more detail than normally required. The question has been deleted.

Question 8.2

The candidate should have thorough knowledge of what is addressed in the Technical Specification and should know whether actions are required with the Division 1 250V battery discharged.

The part b of the question requires the candidate to memorize the surveillance requirements for the 250 volt battery, thus beyond the scope of the examination as defined by ES-402, A.4. Part b of the question has been deleted.

Question on placing RCIC Controller in Manual (Question 8.6 b)

The question refers to the limitations in SOP 2.4.6, Reactor Core Isolation System. In this context, only the additional answer "When, in the operator's judgement, continued automatic operation is undesirable (Ref: PPM 1.3.1, Att 1, Item 3)" is applicable. This additional answer will be allowed if the standing order is referenced. The additional answer has been added to the answer key.

Simulator Exam Scenarios

In order to provide each candidate with one or more malfunction to respond to during each scenario, unrelated malfunctions were selected. This, however, is not that much different from some operating events. We have had plant events that have had several unrelated malfunctions. (e.g., Trojan had a diesel driven auxiliary feedwater pump, a turbine driven auxiliary feed, an emergency diesel generator and the main steam isolation valves all malfunction in one event.)

There were several events in each scenario, but in most cases the operator was allowed to deal with each event prior to the initiation of the next event. It is not rare to have 6 to 10 malfunction during the operation of a power plant. Operators must learn to be aware of which systems are operating and thereby, be able to perform there normal evolutions efficiently and safely.

General Comments

1. Need specific examples of which questions were unclear and how prior exam review could have made them more clear. Also which questions used unfamiliar terminology and what terminology should have been used.
2. Sometimes the examiner must answer in this manner when a direct response to the candidate's question would provide an answer to the exam question. In these cases, we generally request the candidate to provide as much information as he can and qualify the response with any assumptions that were made. This will give the candidate the best opportunity to get full credit for the response.
3. There are no questions that request an open ended discussion. All questions can be answered with a short response. Typically the question asks for a response and then asks "why" or "explain choice" or the question will set the conditions and then state "briefly explain what happens." The questions that are vague and open ended should be specified.
4. Except for the questions that were deleted, both examinations were within the scope of topics listed in the 10 CFR 55 and Exam standard NUREG-1021. Presently 10 CFR 55 and NUREG-1021 are the standard used to define the scope of the examinations. If the training objectives for the WNP operator training program are not consistent with these standards, the training objectives should be redefined. We have not written our examinations to fall within the WNP training program training objectives nor have we written our examination to be consistent with the Job Task Analysis (JTA) for RO/SRO personnel. The NRC has a very active program which includes industry participation to evaluate exam questions. Our

goal is to be able to use JTA to evaluate the importance and applicability of examination questions to RO/SRO job performance. We hope that in the near future all of our examinations will contain questions that have been evaluated based on JTA. We do not expect this to result in any significant changes in the scope of the examination, but will result in assigning question values that are more consistent with the importance of the required knowledge to the performance of the job.

MASTER

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

Facility: WNP-2
Reactor Type: BWR-5
Date Administered: 11/6/84
Examiner: I. S. Levy
Candidate: _____

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheet. Points for each question are indicated in parenthesis 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.

<u>Category Value</u>	<u>% of Total</u>	<u>Candidate's Score</u>	<u>% of Cat. Value</u>	<u>Category</u>
<u>25</u>	<u>25</u>	_____	_____	1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer, and Fluid Flow
<u>25</u>	<u>25</u>	_____	_____	2. Plant Design Including Safety and Emergency Systems
<u>22</u> 25	<u>22</u> 25	_____	_____	3. Instruments and Controls
<u>25</u>	<u>25</u>	_____	_____	4. Procedures: Normal, Abnormal, Emergency, and Radiological Control
<u>97</u> 100		_____		TOTALS
		Final Grade	_____ %	

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

Candidate's Signature

Questions and Answers to WNP-2 RO Exam - 11/6/84

1.0 Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow (25.0)

1.1 Regarding a Reactor Startup:

a. Does the magnitude of the initial level of source range counts affect the Estimated Critical Position? Why? (1.25)

b. How long will it take to reach 0.08% power, if the reactor is just critical at 0.002% power and on a steady period with a "doubling time" of 40 seconds? (Show all work.) (1.0)

1.1 a. No (0.5). Initial count rate does not affect the amount of reactivity required to go critical, which determines ECP (0.75). [The higher the initial count rate, the higher the count rate when criticality is reached.] (1.25)

Ref: WNP-2 Reactor Theory, pg. 29 and 43.

b. $P = P_0 e^{(t/T)}$
 $T = 40 / (\ln 2) = 57.71 \text{ sec}^{-1} \quad (\text{0.5})$
 $t = 57.71 (\ln(P/P_0)) = \underline{212.88} \text{ sec} \quad (\text{0.5})$ (1.0)

Ref: WNP-2 Reactor Theory, pg. 61.

1.2 With regard to some aspects of Fission Product Poisons:

- a. Of the two fission product poisons Xe and Sm, give two (2) reasons why xenon is more troublesome. (1.5)
- b. What is the mechanism(s) for removal of Samarium-149 once it is produced in the core? (1.0)

- any 2 (0.75 each)*
- 1.2 a. (1) Because of its exceptionally large thermal neutron absorption cross-section (0.75); (2) its concentration varies with reactor power level and/or time (0.75); (3) Xe produced as result of rel. large fraction of fission. (1.5)
- b. Sm-149 is removed only by burnout (1.0)

Ref: WNP-2 Reactor Theory, pp. 83 and 87, respectively.

1.3 When control rod density in the core decreases at higher burnups (from pulling rods) the void coefficient of reactivity becomes more or less (choose one) negative? Why? (1.5)

1.3 Less (0.5). Since local steam voids cause an increase in thermal diffusion length (0.5), control rods, which absorb thermal neutrons, make the steam void reactivity coefficient more negative (0.5) [Therefore, reduced control rod density causes the void coefficient to be less negative.] (1.5)

Ref: WNP-2 Reactor Theory, pp. 98-99.

1.4 The effective decay constant for up power transients will be higher or lower (choose one) compared to its value for down power transients. Give the reason for your answer. (1.5)

1.4 Higher (0.5). For up power transients the short lived precursors are dominant due to the addition of power (0.5), while for down power transients the long lived precursors dominate the decay constant (0.5) (1.5)

Ref: WNP-2 Reactor Theory, pg. 54.

1.5 Following a scram from high power, answer the following:

- a. What are the most reactive regions of the core? (1.0)
- b. Why are these regions more reactive? (1.25)
- c. What problem does this cause for the operator during a subsequent start up. Why? (1.25)

- 1.5 a. Near the edges and at the top and extreme bottom (1.0)
- b. Xe concentrates, during power operation, where power is highest, i.e., in the center and near the bottom of the core (0.75), where it acts as a poison, adding negative reactivity (0.50) (1.25)
- c. Operator must be extremely cautious while pulling edge and top rods (0.5) since normally low worth rods now have excessively high incremental worths (0.75) (1.25)

Ref: WNP-2 Lesson Plan, Figure 4-12.

1.6 Give four (4) inputs or outputs for a reactor heat balance, stating whether it is an input or output and a brief description as to why it is.

(3.0)

1.6 (any 4 @ 0.75 each)

(3.0)

- a. Feedwater flow - heat input (0.25) going into the vessel (or system) with positive enthalpy (0.5).
- b. Steam flow - heat output (0.25) due to steam removing energy from the core (0.5).
- c. Recirc pump - heat input (0.25) due to energy added to the fluid in the core (or system) by the pumps (0.5)
- d. CRD flow - heat input (0.25) due to fluid flowing into the core (or system) with a positive enthalpy (0.5).
- e. Rx core thermal energy - heat input (0.25) due to being primary source of heat input (0.5).
- f. any other reasonable answer such as RWCU inputs/outputs, etc.
ambient losses OK.

Ref: MTC; Thermo/HT/FF (3/83), pp. 8-50.

- 1.7 Following initial criticality (MSIVs closed, moderator $T > 212^{\circ}\text{F}$), a constant positive period is established. Briefly explain what happens over the next several hours to pressure, temperature and power if no rod movement occurs. (1.5)

- 1.7 Power initially increases but levels off due to negative reactivity insertion resulting from increasing moderator temperature (0.5). Pressure and temperature initially increase but level off when power levels off then reduces due to ambient losses (0.5). The reduced T causes the cycle to start again so that long term power, pressure and temperature will oscillate around point of adding heat (0.5). (1.5)

Ref: Standard Reactor Theory.

1.8 What are two (2) reasons a centrifugal pump should be started with the discharge piping filled and the discharge valve shut?

(1.0)

1.8 Water hammer and excessive starting current.

(1.0)

Ref: Morris T.C.; Thermo/HT/Fluid Flow (3/83), pg. 7-123.

- 1.9 a. Assume the reactor is at 100% power and flow. Explain what happens to core flow, and why, for a reduction in power by driving rods in. (Recirculation pump speed remains constant.) (1.25)
- b. At low power conditions prior to void generation, an increase in reactor power by control rod withdrawal will (increase, decrease, not change) flow through the core. Choose the correct answer and explain your choice. (1.5)

Q- 'low power' could be up to 35% ~~which represents~~ 2 flow regimes.
- What one do you mean? ~~←~~

A. ~~100% low power~~ is $\approx 10-15\%$.

- 1.9 a. Core flow would increase (0.5) due to a reduction in two phase flow condition (and, therefore, in the core less resistance to flow) (0.75). (1.25)
- b. Increase (0.5). Flow resistance in the channels drops due to decreased liquid viscosity with temperature (0.5); and greater density differences between warm channels and cool downcomer will increase flow due to increased thermal driving load causing greater natural circulation (0.5) (1.5)

Ref: Morris T.C.; Thermo/HT/Fluid Flow (3/83), pg. 9-51.

1.10 There are several characteristic transients that would be limiting because of MCPR.

a. List any two (2) of these transients. (0.75)

b. Assuming for each of the transients in (a) they occurred at 100% power, EOC and full rod out conditions, give:

1. the most important reactivity coefficient involved. (0.5)

2. What occurred in the reactor and why it occurred to affect this coefficient. (0.5)

1.10 Any 2 of following. pts: a) 0.375 each; b) (1) 0.25 each; (2) 0.25 each (1.75)

(a)

b(1)

b(2)

- Generator load reject w/o bypass; void coefficient, void collapse from pressure increase
- Turbine trip w/o bypass; void coefficient, void collapse from pressure increase
- Loss of feedwater heating, void coefficient, void collapse from subcooling
- Inadvertent start of HPCS pump; void coefficient, void collapse from subcooling
- Feedwater controlling failure high; void coefficient, void collapse from subcooling

Ref: Morris T.C.; Thermo/HT/Fluid Flow (3/85), pg. 9-94 through 9-96.

1.11 Given:

Rx pressure at time T = 675.3 psig

Rx pressure at time T + 1 hr = 215.3 psig

- a. What is the Rx cool down rate for this hour? Show all calculations. (0.75)
- b. Is this rate acceptable at your plant? (0.5)

1.11 a. (Because the reactor operates at saturated conditions the temperature for time T and T + 1 hr can be found using the steam tables.)

1) Saturation temperature for 690 psia is approximately 502°F.
Saturation temperature for 230 psia is approximately 394°F. (0.375)

2) Cool down rate = $(502^{\circ}\text{F} - 394^{\circ}\text{F})/1 \text{ hr.}$
Cool down rate = 108°F/1 hr. (0.375)

b. No. (The cool down rate limit is 100°F per hour.) (0.5)

Ref: Steam Tables and WNP-2 Tech. Specs.

1.12 Answer TRUE or FALSE for each of the following:

- a. As water flows around a bend in a pipe, the velocity of the water is uniform throughout the diameter of the pipe. (0.5)
- b. The pressure in a static fluid always decreases with increasing elevation of the measurement. (0.5)

1.12 a. False (0.5)

b. True (0.5)

Ref: General Fluid Dynamics Text.

- 1.13 Will an increase in the following factors (taken separately) increase, decrease, or not change differential rod worth? (1.5)
- a. thermal diffusion length
 - b. neutron flux at the rod
 - c. core average neutron flux

- 1.13 a. increase (0.5)
b. increase (0.5)
c. decrease (0.5)

Ref: WNP-2 Reactor Theory, pg. 80.

$$P = P_0 e^{t/\tau}$$

$$1Ci = 3.7 \times 10^{10} \text{Bq}$$

$$\alpha_D = -1 \times 10^{-5} \frac{\Delta K/^{\circ}F}{K}$$

$$\alpha_V = -1 \times 10^{-3} \frac{\Delta K/^{\circ}F \text{ voids}}{K}$$

$$\alpha_H = -4.5 \times 10^{-4} \frac{\Delta K/^{\circ}F}{K}$$

$$\alpha_P = -4.5 \times 10^{-4} \frac{\Delta K/^{\circ}F \text{ power}}{K}$$

$$I(\tau) = I_0 e^{-\lambda \tau}$$

$$T_{1/2} = \ln(2)/\lambda$$

$$C_p = (C_{p\text{base}}) (K_s) (K_A)$$

$$Q = MC_p \Delta t$$

$$\Delta \rho = f \frac{L \rho v^2}{D 2g_c}$$

$$f = 64/Re$$

$$\rho = \frac{k(\text{eff}) - 1}{K(\text{eff})}$$

$$\frac{1}{M} = \frac{CR1}{CR2} = \frac{1 - K(\text{eff})^2}{1 - K(\text{eff})}$$

$$M = \frac{CR2}{CR1} = \frac{1 - K(\text{eff})}{1 - K(\text{eff})^2}$$

$$Q = M \Delta h$$

$$Q = UA \Delta T$$

$$M = 1/(1-k)$$

$$N(\tau) = N_0 e^{-\lambda \tau}$$

$$\alpha_{r'} = (L_f + L_s) \frac{(\phi_{\text{rod}})^2}{(\phi_{\text{avg}})}$$

$$n = v/(1+d)$$

$$P = \Gamma \phi v / (3.7 \times 10^{10})$$

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

$$\bar{\tau} = \bar{I} / \rho + (\beta - \rho) / \lambda \rho$$

$$\tau = 1/(\rho - \beta)$$

$$v = v_f + kv_{fg}$$

$$H = x h_g + (1-x) h_f$$

$$S = x S_g + (1-x) S_f$$

$$1 \text{ in.} = 2.54 \text{ cm}$$

$$1 \text{ gal.} = 3.785 \text{ liters}$$

$$1 \text{ kg} = 2.205 \text{ lb}$$

$$N = \rho A_0 / A$$

$$17.58 \text{ watts} = 1 \text{ BTU/min}$$

$$1 \text{ psi} = 6.895 \text{ Pa}$$

$$1 \text{ psi} = 2.036 \text{ } ^{\circ} \text{H}_2 \text{O} (@ 0C)$$

$$1 \text{ psi} = 27.68 \text{ } ^{\circ} \text{H}_2 \text{O} (@ 4C)$$

$$\bar{\beta} = .0071$$

$$\bar{I} = 2 \times 10^{-5} \text{ sec}$$

Table 1. Saturated Steam: Temperature Table

Temp Fahr t	Abs Press Lb per Sq in o	Specific Volume			Enthalpy			Entropy			Temp Fahr t
		Sat Liquid v _l	Evap v _{lg}	Sat Vapor v _g	Sat Liquid h _l	Evap h _{lg}	Sat Vapor h _g	Sat Liquid s _l	Evap s _{lg}	Sat Vapor s _g	
32.0	0.0859	0.016022	3304.7	3304.7	-0.0179	1075.5	1075.5	0.0000	2.1873	2.1873	32.0
34.0	0.09600	0.016021	3061.9	3061.9	1.996	1074.4	1076.4	0.0041	2.1762	2.1802	34.0
36.0	0.10395	0.016020	2839.0	2839.0	4.008	1073.2	1077.2	0.0081	2.1651	2.1732	36.0
38.0	0.11249	0.016019	2634.1	2634.2	6.018	1072.1	1078.1	0.0122	2.1541	2.1643	38.0
40.0	0.12163	0.016019	2445.8	2445.8	8.027	1071.0	1079.0	0.0162	2.1432	2.1594	40.0
42.0	0.13143	0.016019	2272.4	2272.4	10.035	1069.8	1079.9	0.0202	2.1325	2.1527	42.0
44.0	0.14192	0.016019	2112.8	2112.8	12.041	1068.7	1080.7	0.0242	2.1217	2.1459	44.0
46.0	0.15314	0.016020	1965.7	1965.7	14.047	1067.6	1081.6	0.0282	2.1111	2.1393	46.0
48.0	0.16514	0.016021	1830.0	1830.0	16.051	1066.4	1082.5	0.0321	2.1006	2.1327	48.0
50.0	0.17796	0.016023	1704.8	1704.8	18.054	1065.3	1083.4	0.0361	2.0901	2.1262	50.0
52.0	0.19165	0.016024	1589.2	1589.2	20.057	1064.2	1084.2	0.0400	2.0798	2.1197	52.0
54.0	0.20625	0.016026	1482.4	1482.4	22.058	1063.1	1085.1	0.0439	2.0695	2.1134	54.0
56.0	0.22183	0.016028	1383.6	1383.6	24.059	1061.9	1086.0	0.0478	2.0593	2.1070	56.0
58.0	0.23843	0.016031	1292.2	1292.2	26.060	1060.8	1086.9	0.0516	2.0491	2.1008	58.0
60.0	0.25611	0.016033	1207.8	1207.8	28.060	1059.7	1087.7	0.0555	2.0391	2.0946	60.0
62.0	0.27494	0.016036	1129.2	1129.2	30.059	1058.5	1088.6	0.0593	2.0291	2.0885	62.0
64.0	0.29497	0.016039	1056.5	1056.5	32.058	1057.4	1089.5	0.0632	2.0192	2.0824	64.0
66.0	0.31626	0.016043	989.0	989.1	34.056	1056.3	1090.4	0.0670	2.0094	2.0764	66.0
68.0	0.33889	0.016046	925.5	925.5	36.054	1055.2	1091.2	0.0708	1.9996	2.0704	68.0
70.0	0.36292	0.016050	866.4	866.4	38.052	1054.0	1092.1	0.0745	1.9900	2.0645	70.0
72.0	0.38844	0.016054	811.3	811.3	40.049	1052.9	1093.0	0.0783	1.9804	2.0587	72.0
74.0	0.41550	0.016058	761.1	761.1	42.046	1051.8	1093.8	0.0821	1.9708	2.0529	74.0
76.0	0.44420	0.016063	714.4	714.4	44.043	1050.7	1094.7	0.0858	1.9614	2.0472	76.0
78.0	0.47461	0.016067	673.8	673.8	46.040	1049.5	1095.6	0.0895	1.9520	2.0415	78.0
80.0	0.50682	0.016072	633.3	633.3	48.037	1048.4	1096.4	0.0932	1.9426	2.0359	80.0
82.0	0.54093	0.016077	595.5	595.5	50.033	1047.3	1097.3	0.0969	1.9334	2.0303	82.0
84.0	0.57702	0.016082	560.3	560.3	52.029	1046.1	1098.2	0.1006	1.9242	2.0248	84.0
86.0	0.61518	0.016087	527.5	527.5	54.026	1045.0	1099.0	0.1043	1.9151	2.0193	86.0
88.0	0.65551	0.016093	496.8	496.8	56.022	1043.9	1099.9	0.1079	1.9060	2.0139	88.0
90.0	0.69813	0.016099	468.1	468.1	58.018	1042.7	1100.8	0.1115	1.8970	2.0086	90.0
92.0	0.74313	0.016105	441.3	441.3	60.014	1041.6	1101.6	0.1152	1.8881	2.0033	92.0
94.0	0.79062	0.016111	416.3	416.3	62.010	1040.5	1102.5	0.1188	1.8792	1.9980	94.0
96.0	0.84072	0.016117	392.8	392.8	64.006	1039.3	1103.3	0.1224	1.8704	1.9928	96.0
98.0	0.89356	0.016123	370.9	370.9	66.003	1038.2	1104.2	0.1260	1.8617	1.9876	98.0
100.0	0.94924	0.016130	350.4	350.4	67.999	1037.1	1105.1	0.1295	1.8530	1.9825	100.0
102.0	1.00789	0.016137	331.1	331.1	69.995	1035.9	1105.9	0.1331	1.8444	1.9775	102.0
104.0	1.06965	0.016144	313.1	313.1	71.992	1034.8	1106.8	0.1366	1.8358	1.9725	104.0
106.0	1.1347	0.016151	296.1	296.1	73.989	1033.6	1107.6	0.1402	1.8273	1.9675	106.0
108.0	1.2030	0.016158	280.28	280.30	75.98	1032.5	1108.5	0.1437	1.8188	1.9626	108.0
110.0	1.2750	0.016165	265.37	265.39	77.98	1031.4	1109.3	0.1472	1.8105	1.9577	110.0
112.0	1.3505	0.016173	251.37	251.38	79.98	1030.2	1110.2	0.1507	1.8021	1.9528	112.0
114.0	1.4299	0.016180	238.21	238.22	81.97	1029.1	1111.0	0.1542	1.7938	1.9480	114.0
116.0	1.5133	0.016188	225.84	225.85	83.97	1027.9	1111.9	0.1577	1.7856	1.9433	116.0
118.0	1.6009	0.016196	214.21	214.21	85.97	1026.8	1112.7	0.1611	1.7774	1.9386	118.0
120.0	1.6927	0.016204	203.25	203.26	87.97	1025.6	1113.6	0.1646	1.7693	1.9339	120.0
122.0	1.7891	0.016213	192.94	192.95	89.96	1024.5	1114.4	0.1680	1.7612	1.9293	122.0
124.0	1.8901	0.016221	183.23	183.24	91.96	1023.3	1115.3	0.1715	1.7532	1.9247	124.0
126.0	1.9959	0.016229	174.08	174.09	93.96	1022.2	1116.1	0.1749	1.7453	1.9202	126.0
128.0	2.1068	0.016238	165.45	165.47	95.96	1021.0	1117.0	0.1783	1.7374	1.9157	128.0
130.0	2.2230	0.016247	157.32	157.33	97.96	1019.8	1117.8	0.1817	1.7295	1.9112	130.0
132.0	2.3445	0.016256	149.64	149.66	99.95	1018.7	1118.6	0.1851	1.7217	1.9068	132.0
134.0	2.4717	0.016265	142.40	142.41	101.95	1017.5	1119.5	0.1884	1.7140	1.9024	134.0
136.0	2.6047	0.016274	135.55	135.57	103.95	1016.4	1120.3	0.1918	1.7063	1.8980	136.0
138.0	2.7438	0.016284	129.09	129.11	105.95	1015.2	1121.1	0.1951	1.6986	1.8937	138.0
140.0	2.8892	0.016293	122.98	123.00	107.95	1014.0	1122.0	0.1985	1.6910	1.8895	140.0
142.0	3.0411	0.016303	117.21	117.22	109.95	1012.9	1122.8	0.2018	1.6834	1.8852	142.0
144.0	3.1997	0.016312	111.74	111.76	111.95	1011.7	1123.6	0.2051	1.6759	1.8810	144.0
146.0	3.3653	0.016322	106.58	106.59	113.95	1010.5	1124.5	0.2084	1.6684	1.8769	146.0
148.0	3.5381	0.016332	101.68	101.70	115.95	1009.3	1125.3	0.2117	1.6610	1.8727	148.0
150.0	3.7184	0.016343	97.05	97.07	117.95	1008.2	1126.1	0.2150	1.6536	1.8686	150.0
152.0	3.9065	0.016353	92.66	92.68	119.95	1007.0	1126.9	0.2183	1.6463	1.8646	152.0
154.0	4.1025	0.016363	88.50	88.52	121.95	1005.8	1127.7	0.2216	1.6390	1.8606	154.0
156.0	4.3068	0.016374	84.56	84.57	123.95	1004.6	1128.6	0.2248	1.6318	1.8566	156.0
158.0	4.5197	0.016384	80.82	80.83	125.96	1003.4	1129.4	0.2281	1.6245	1.8526	158.0
160.0	4.7414	0.016395	77.27	77.28	127.96	1002.2	1130.2	0.2313	1.6174	1.8487	160.0
162.0	4.9724	0.016406	73.90	73.92	129.96	1001.0	1131.0	0.2345	1.6103	1.8448	162.0
164.0	5.2124	0.016417	70.70	70.72	131.96	999.8	1131.8	0.2377	1.6032	1.8409	164.0
166.0	5.4623	0.016428	67.67	67.68	133.97	998.6	1132.6	0.2409	1.5961	1.8371	166.0
168.0	5.7223	0.016440	64.78	64.80	135.97	997.4	1133.4	0.2441	1.5892	1.8333	168.0
170.0	5.9926	0.016451	62.04	62.06	137.97	996.2	1134.2	0.2473	1.5822	1.8295	170.0
172.0	6.2736	0.016463	59.43	59.45	139.98	995.0	1135.0	0.2505	1.5753	1.8258	172.0
174.0	6.5656	0.016474	56.95	56.97	141.98	993.8	1135.8	0.2537	1.5684	1.8221	174.0
176.0	6.8690	0.016486	54.59	54.61	143.99	992.6	1136.6	0.2568	1.5616	1.8184	176.0
178.0	7.1840	0.016498	52.35	52.36	145.99	991.4	1137.4	0.2600	1.5548	1.8147	178.0

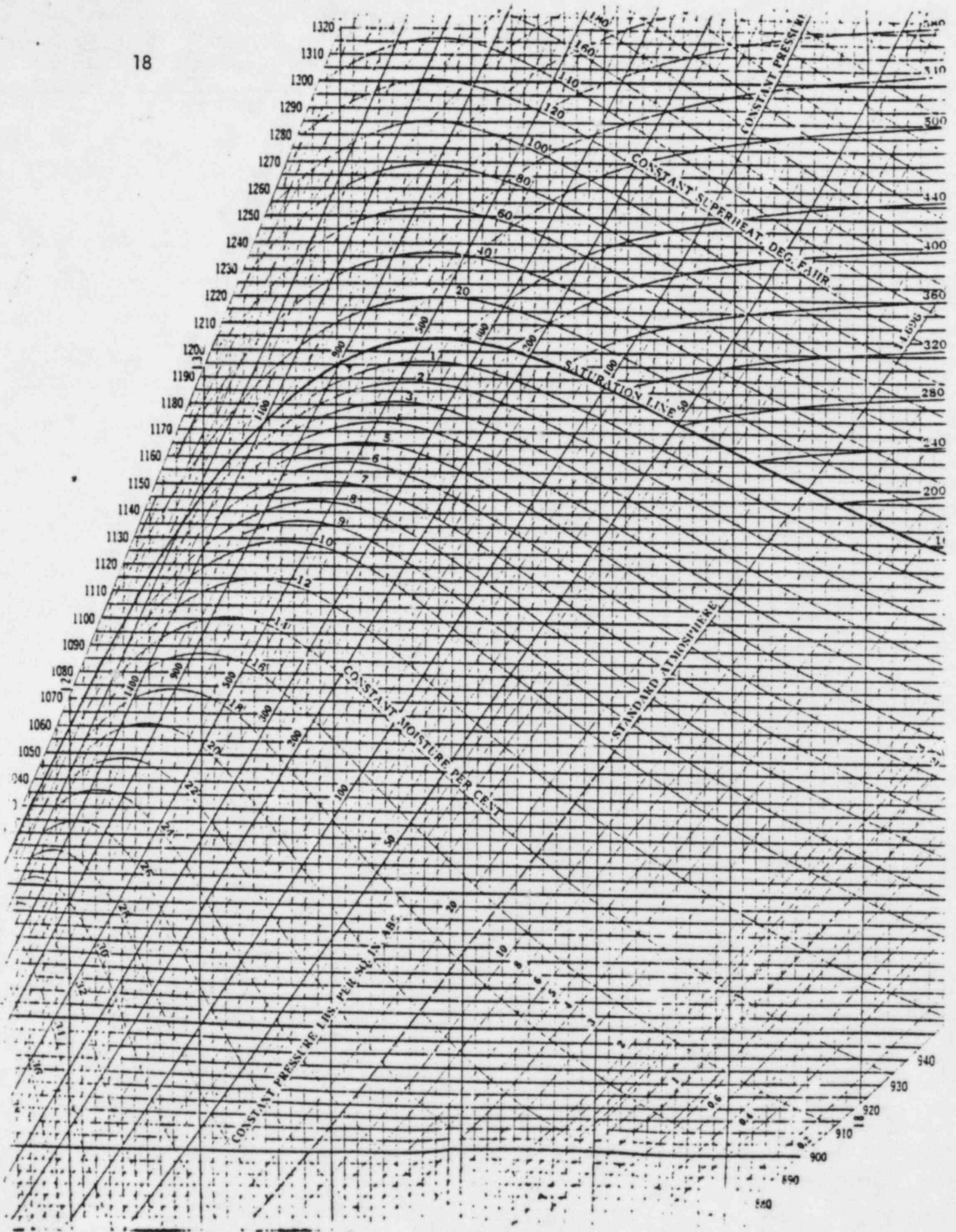
Table 1. Saturated Steam: Temperature Table—Continued

Temp Fahr t	Abs Press Lb per Sq in p	Specific volume			Enthalpy			Entropy			Temp Fahr t
		Sat Liquid v _l	Evap v _{lg}	Sat Vapor v _g	Sat Liquid h _l	Evap h _{lg}	Sat Vapor h _g	Sat Liquid s _l	Evap s _{lg}	Sat Vapor s _g	
180.0	7.5110	0.016510	50.21	50.22	148.00	990.2	1138.2	0.2631	1.5480	1.8111	180.0
182.0	7.850	0.016522	48.172	48.199	150.01	989.0	1139.0	0.2652	1.5413	1.8075	182.0
184.0	8.203	0.016534	46.232	46.249	152.01	987.8	1139.8	0.2694	1.5344	1.8040	184.0
186.0	8.568	0.016547	44.383	44.400	154.02	986.5	1140.5	0.2725	1.5279	1.8004	186.0
188.0	8.947	0.016559	42.621	42.638	156.03	985.3	1141.3	0.2756	1.5213	1.7969	188.0
190.0	9.340	0.016572	40.941	40.957	158.04	984.1	1142.1	0.2787	1.5148	1.7934	190.0
192.0	9.747	0.016585	39.337	39.354	160.05	982.8	1142.9	0.2818	1.5082	1.7900	192.0
194.0	10.168	0.016598	37.808	37.824	162.05	981.6	1143.7	0.2848	1.5017	1.7865	194.0
196.0	10.605	0.016611	36.348	36.364	164.06	980.4	1144.4	0.2879	1.4952	1.7831	196.0
198.0	11.058	0.016624	34.954	34.970	166.08	979.1	1145.2	0.2910	1.4888	1.7798	198.0
200.0	11.526	0.016637	33.622	33.639	168.09	977.9	1146.0	0.2940	1.4824	1.7764	200.0
202.0	12.012	0.016650	31.135	31.151	172.11	975.4	1147.5	0.3001	1.4697	1.7698	202.0
204.0	13.568	0.016661	28.862	28.879	176.14	972.5	1149.0	0.3061	1.4571	1.7632	204.0
212.0	14.694	0.016719	26.782	26.799	180.17	970.3	1150.5	0.3121	1.4447	1.7568	212.0
218.0	15.901	0.016747	24.878	24.894	184.20	967.8	1152.0	0.3181	1.4323	1.7505	218.0
224.0	17.186	0.016775	23.131	23.148	188.23	965.2	1153.4	0.3241	1.4201	1.7442	224.0
228.0	18.556	0.016805	21.529	21.545	192.27	962.6	1154.9	0.3300	1.4081	1.7380	228.0
232.0	20.015	0.016834	20.056	20.073	196.31	960.0	1156.3	0.3359	1.3961	1.7320	232.0
236.0	21.567	0.016864	18.701	18.718	200.35	957.4	1157.8	0.3417	1.3842	1.7260	236.0
240.0	23.216	0.016895	17.454	17.471	204.40	954.8	1159.2	0.3476	1.3725	1.7201	240.0
248.0	24.968	0.016926	16.304	16.321	208.45	952.1	1160.6	0.3533	1.3609	1.7142	248.0
254.0	26.326	0.016958	15.243	15.260	212.50	949.5	1162.0	0.3591	1.3494	1.7085	254.0
258.0	27.796	0.016990	14.264	14.281	216.56	946.8	1163.4	0.3649	1.3379	1.7028	258.0
262.0	30.883	0.017022	13.358	13.375	220.62	944.1	1164.7	0.3706	1.3266	1.6972	262.0
268.0	33.091	0.017055	12.520	12.538	224.69	941.4	1166.1	0.3763	1.3154	1.6917	268.0
280.0	35.427	0.017089	11.745	11.762	228.76	938.6	1167.4	0.3819	1.3043	1.6862	280.0
284.0	37.894	0.017123	11.025	11.042	232.83	935.9	1168.7	0.3875	1.2933	1.6808	284.0
288.0	40.500	0.017157	10.358	10.375	236.91	933.1	1170.0	0.3932	1.2823	1.6755	288.0
292.0	43.249	0.017193	9.738	9.755	240.99	930.3	1171.3	0.3987	1.2715	1.6702	292.0
298.0	46.147	0.017228	9.162	9.180	245.08	927.5	1172.5	0.4043	1.2607	1.6650	298.0
300.0	49.200	0.017264	8.627	8.644	249.17	924.6	1173.8	0.4098	1.2501	1.6599	300.0
304.0	52.414	0.017300	8.1280	8.1453	253.3	921.7	1175.0	0.4154	1.2395	1.6543	304.0
308.0	55.795	0.01734	7.6634	7.6807	257.4	918.8	1176.2	0.4208	1.2290	1.6494	308.0
312.0	59.350	0.01738	7.2301	7.2475	261.5	915.9	1177.4	0.4263	1.2186	1.6449	312.0
318.0	63.084	0.01741	6.8259	6.8433	265.6	913.0	1178.6	0.4317	1.2082	1.6400	318.0
320.0	67.005	0.01745	6.4483	6.4658	269.7	910.0	1179.7	0.4372	1.1979	1.6351	320.0
324.0	71.119	0.01749	6.0955	6.1130	273.8	907.0	1180.9	0.4425	1.1877	1.6303	324.0
328.0	75.433	0.01753	5.7655	5.7830	277.9	904.0	1182.0	0.4479	1.1776	1.6256	328.0
332.0	79.953	0.01757	5.4566	5.4742	282.1	901.0	1183.1	0.4533	1.1674	1.6209	332.0
338.0	84.688	0.01761	5.1673	5.1849	286.3	897.9	1184.1	0.4586	1.1576	1.6162	338.0
340.0	89.643	0.01766	4.8961	4.9138	290.4	894.8	1185.2	0.4640	1.1477	1.6116	340.0
344.0	94.826	0.01770	4.6418	4.6595	294.6	891.6	1186.2	0.4692	1.1378	1.6071	344.0
348.0	100.245	0.01774	4.4030	4.4208	298.7	888.5	1187.2	0.4745	1.1280	1.6025	348.0
352.0	105.907	0.01779	4.1788	4.1966	302.9	885.3	1188.2	0.4798	1.1183	1.5981	352.0
358.0	111.820	0.01783	3.9681	3.9859	307.1	882.1	1189.1	0.4850	1.1086	1.5936	358.0
360.0	117.992	0.01787	3.7699	3.7878	311.3	878.8	1190.1	0.4902	1.0990	1.5892	360.0
364.0	124.430	0.01792	3.5834	3.6013	315.5	875.5	1191.0	0.4954	1.0894	1.5849	364.0
368.0	131.142	0.01797	3.4078	3.4256	319.7	872.2	1191.1	0.5006	1.0799	1.5806	368.0
372.0	138.128	0.01801	3.2423	3.2603	323.9	868.9	1192.1	0.5058	1.0705	1.5763	372.0
378.0	145.424	0.01806	3.0863	3.1044	328.1	865.5	1193.0	0.5110	1.0611	1.5721	378.0
380.0	153.010	0.01811	2.9392	2.9573	332.3	862.1	1194.4	0.5161	1.0517	1.5678	380.0
384.0	160.903	0.01816	2.8002	2.8184	336.5	858.6	1195.2	0.5212	1.0424	1.5637	384.0
388.0	169.113	0.01821	2.6691	2.6873	340.8	855.1	1195.9	0.5263	1.0332	1.5595	388.0
392.0	177.648	0.01826	2.5451	2.5633	345.0	851.6	1196.7	0.5314	1.0240	1.5554	392.0
398.0	186.517	0.01831	2.4279	2.4462	349.3	848.1	1197.4	0.5365	1.0148	1.5513	398.0
400.0	195.729	0.01836	2.3170	2.3353	353.6	844.5	1198.0	0.5416	1.0057	1.5473	400.0
404.0	205.294	0.01842	2.2120	2.2304	357.9	840.8	1198.7	0.5466	0.9966	1.5432	404.0
408.0	215.220	0.01847	2.1126	2.1311	362.2	837.2	1199.3	0.5515	0.9876	1.5392	408.0
412.0	225.516	0.01853	2.0184	2.0369	366.5	833.4	1199.9	0.5567	0.9786	1.5352	412.0
418.0	236.193	0.01858	1.9291	1.9477	370.8	829.7	1200.4	0.5617	0.9696	1.5313	418.0
420.0	247.259	0.01864	1.8444	1.8630	375.1	825.9	1201.0	0.5667	0.9607	1.5274	420.0
424.0	258.725	0.01870	1.7640	1.7827	379.4	822.0	1201.5	0.5717	0.9518	1.5234	424.0
428.0	270.600	0.01875	1.6877	1.7064	383.8	818.2	1201.9	0.5766	0.9429	1.5195	428.0
432.0	282.894	0.01881	1.6152	1.6340	388.1	814.4	1202.4	0.5816	0.9341	1.5157	432.0
438.0	295.617	0.01887	1.5463	1.5651	392.5	810.7	1202.8	0.5866	0.9253	1.5118	438.0
440.0	308.780	0.01894	1.4806	1.4997	396.9	806.2	1203.1	0.5915	0.9165	1.5080	440.0
444.0	322.391	0.01900	1.4184	1.4374	401.3	802.2	1203.5	0.5964	0.9077	1.5042	444.0
448.0	336.463	0.01906	1.3591	1.3782	405.7	798.0	1203.7	0.6014	0.8990	1.5004	448.0
452.0	351.00	0.01913	1.3026	1.3217	410.1	793.9	1204.0	0.6063	0.8903	1.4966	452.0
458.0	366.03	0.01919	1.2487	1.2680	414.6	789.7	1204.2	0.6112	0.8816	1.4928	458.0
460.0	381.54	0.01926	1.1976	1.2167	419.0	785.4	1204.4	0.6161	0.8729	1.4890	460.0
464.0	397.56	0.01933	1.1487	1.1680	423.5	781.1	1204.6	0.6210	0.8643	1.4853	464.0
468.0	414.09	0.01940	1.1021	1.1212	428.0	776.7	1204.7	0.6259	0.8557	1.4815	468.0
472.0	431.14	0.01947	1.0576	1.0771	432.5	772.3	1204.8	0.6308	0.8471	1.4778	472.0
478.0	448.73	0.01954	1.0151	1.0347	437.0	767.8	1204.8	0.6356	0.8385	1.4741	478.0

Table 1. Saturated Steam: Temperature Table—Continued

Temp Fahr t	Abs Press Lb per Sq in p	Specific Volume			Enthalpy			Entropy			Temp Fahr t
		Sat Liquid v _l	Evap v _{lg}	Sat Vapor v _g	Sat Liquid h _f	Evap h _{fg}	Sat Vapor h _g	Sat Liquid s _f	Evap s _{fg}	Sat Vapor s _g	
466.0	466.87	0.01961	0.97463	0.99424	441.5	763.2	1204.8	0.6405	1.199	1.4704	466.0
468.0	485.56	0.01969	0.97586	0.99557	446.1	758.6	1204.7	0.6454	0.8.13	1.4667	468.0
470.0	504.83	0.01976	0.97885	0.91862	450.7	754.0	1204.6	0.6502	0.8127	1.4629	470.0
472.0	524.67	0.01984	0.98345	0.88379	455.2	749.3	1204.5	0.6551	0.8042	1.4592	472.0
474.0	545.11	0.01992	0.98950	0.84950	459.9	744.5	1204.3	0.6599	0.7956	1.4555	474.0
480.0	596.15	0.02000	0.97716	0.81717	464.5	739.6	1204.1	0.6648	0.7871	1.4518	480.0
484.0	587.81	0.02009	0.76613	0.78672	469.1	734.7	1203.8	0.6696	0.7787	1.4481	484.0
488.0	610.10	0.02017	0.73641	0.75658	473.8	729.7	1203.5	0.6745	0.7700	1.4444	488.0
492.0	632.03	0.02026	0.70794	0.72820	478.5	724.6	1203.1	0.6793	0.7614	1.4407	492.0
496.0	656.61	0.02034	0.68065	0.70100	483.2	719.5	1202.7	0.6842	0.7528	1.4370	496.0
500.0	680.86	0.02043	0.65448	0.67492	487.9	714.3	1202.2	0.6890	0.7443	1.4333	500.0
504.0	705.78	0.02053	0.62938	0.64991	492.7	709.0	1201.7	0.6939	0.7357	1.4296	504.0
508.0	731.40	0.02062	0.60530	0.62592	497.5	703.7	1201.1	0.6987	0.7271	1.4258	508.0
512.0	757.72	0.02072	0.58218	0.60289	502.3	698.2	1200.5	0.7036	0.7185	1.4221	512.0
516.0	784.76	0.02081	0.55987	0.58079	507.1	692.7	1199.8	0.7085	0.7099	1.4183	516.0
520.0	812.53	0.02091	0.53864	0.55956	512.0	687.0	1199.0	0.7133	0.7013	1.4146	520.0
524.0	841.04	0.02102	0.51814	0.53916	516.8	681.3	1198.2	0.7182	0.6928	1.4108	524.0
528.0	870.31	0.02112	0.49843	0.51955	521.8	675.5	1197.3	0.7231	0.6843	1.4070	528.0
532.0	900.34	0.02123	0.47943	0.50070	526.8	669.6	1196.4	0.7280	0.6757	1.4032	532.0
536.0	931.17	0.02134	0.46123	0.48257	531.7	663.6	1195.4	0.7329	0.6665	1.3993	536.0
540.0	962.79	0.02146	0.44387	0.46513	536.8	657.5	1194.3	0.7378	0.6577	1.3954	540.0
544.0	995.22	0.02157	0.42757	0.44834	541.8	651.3	1193.1	0.7427	0.6489	1.3915	544.0
548.0	1028.49	0.02169	0.41248	0.43217	546.9	645.0	1191.9	0.7476	0.6400	1.3876	548.0
552.0	1062.59	0.02182	0.39879	0.41660	552.0	638.5	1190.6	0.7525	0.6311	1.3837	552.0
556.0	1097.55	0.02194	0.38594	0.40160	557.2	632.0	1189.2	0.7575	0.6222	1.3797	556.0
560.0	1133.38	0.02207	0.37397	0.38714	562.4	625.3	1187.7	0.7625	0.6132	1.3757	560.0
564.0	1170.10	0.02221	0.36299	0.37320	567.6	618.5	1186.1	0.7674	0.6041	1.3716	564.0
568.0	1207.72	0.02235	0.35241	0.35975	572.9	611.5	1184.5	0.7725	0.5950	1.3675	568.0
572.0	1246.26	0.02249	0.34229	0.34678	578.3	604.5	1182.7	0.7775	0.5859	1.3634	572.0
576.0	1285.74	0.02264	0.33162	0.33426	583.7	597.2	1180.9	0.7825	0.5766	1.3592	576.0
580.0	1326.17	0.02279	0.32137	0.32216	589.1	589.9	1179.0	0.7875	0.5673	1.3550	580.0
584.0	1367.7	0.02295	0.31153	0.31048	594.6	582.4	1176.9	0.7927	0.5580	1.3507	584.0
588.0	1410.0	0.02311	0.30208	0.29919	600.1	574.7	1174.8	0.7978	0.5485	1.3464	588.0
592.0	1453.3	0.02327	0.29299	0.28827	605.7	566.8	1172.6	0.8030	0.5390	1.3420	592.0
596.0	1497.8	0.02345	0.28425	0.27770	611.4	558.8	1170.2	0.8082	0.5293	1.3375	596.0
600.0	1543.2	0.02364	0.27584	0.26747	617.1	550.6	1167.7	0.8134	0.5196	1.3320	600.0
604.0	1589.7	0.02382	0.26774	0.25757	622.9	542.2	1165.1	0.8187	0.5097	1.3274	604.0
608.0	1637.3	0.02402	0.25994	0.24796	628.8	533.6	1162.4	0.8240	0.4997	1.3228	608.0
612.0	1686.1	0.02422	0.25242	0.23865	634.8	524.7	1159.5	0.8294	0.4896	1.3181	612.0
616.0	1735.9	0.02444	0.20516	0.22960	640.8	515.6	1156.4	0.8348	0.4794	1.3141	616.0
620.0	1786.9	0.02466	0.19615	0.22091	646.9	506.3	1153.2	0.8403	0.4689	1.3092	620.0
624.0	1839.0	0.02489	0.18737	0.21225	653.1	496.6	1149.8	0.8458	0.4583	1.3041	624.0
628.0	1892.4	0.02513	0.17880	0.20394	659.5	486.7	1146.1	0.8514	0.4474	1.2988	628.0
632.0	1947.0	0.02539	0.17044	0.19583	665.9	476.4	1142.2	0.8571	0.4364	1.2934	632.0
636.0	2002.8	0.02566	0.16226	0.18792	672.4	465.7	1138.1	0.8628	0.4251	1.2879	636.0
640.0	2059.9	0.02595	0.15427	0.18021	679.1	454.6	1133.7	0.8686	0.4134	1.2821	640.0
644.0	2118.3	0.02625	0.14646	0.17269	685.9	443.1	1129.0	0.8746	0.4015	1.2761	644.0
648.0	2178.1	0.02657	0.13876	0.16534	692.9	431.1	1124.0	0.8806	0.3893	1.2699	648.0
652.0	2239.2	0.02691	0.13124	0.15816	700.0	418.7	1118.7	0.8868	0.3767	1.2634	652.0
656.0	2301.7	0.02728	0.12387	0.15115	707.4	405.7	1113.1	0.8931	0.3637	1.2567	656.0
660.0	2365.7	0.02768	0.11663	0.14431	714.9	392.1	1107.0	0.8995	0.3502	1.2498	660.0
664.0	2431.1	0.02811	0.10947	0.13757	722.9	377.7	1100.6	0.9064	0.3361	1.2425	664.0
668.0	2498.1	0.02858	0.10229	0.13087	731.5	362.1	1093.5	0.9137	0.3210	1.2347	668.0
672.0	2566.6	0.02911	0.09514	0.12424	740.2	345.7	1085.9	0.9212	0.3054	1.2256	672.0
676.0	2636.8	0.02970	0.08799	0.11769	749.2	328.5	1077.6	0.9287	0.2892	1.2179	676.0
680.0	2708.6	0.03037	0.08080	0.11117	758.5	310.1	1068.5	0.9365	0.2729	1.2086	680.0
684.0	2782.1	0.03114	0.07349	0.10463	768.2	290.2	1058.4	0.9447	0.2557	1.1984	684.0
688.0	2857.4	0.03204	0.06595	0.09799	778.8	268.2	1047.0	0.9535	0.2377	1.1872	688.0
692.0	2934.5	0.03313	0.05797	0.09110	790.5	243.1	1033.6	0.9634	0.2110	1.1744	692.0
696.0	3013.4	0.03455	0.04916	0.08371	804.4	212.8	1017.2	0.9749	0.1841	1.1591	696.0
700.0	3094.3	0.03622	0.03857	0.07519	822.4	172.7	999.2	0.9901	0.1490	1.1390	700.0
704.0	3175.5	0.03824	0.02713	0.06497	835.0	144.7	979.7	1.0006	0.1246	1.1252	704.0
708.0	3198.3	0.04108	0.02192	0.05300	854.2	102.0	956.2	1.0169	0.0876	1.1046	708.0
709.41*	3208.2	0.04427	0.01104	0.05730	871.0	61.4	934.4	1.0379	0.0527	1.0856	709.41*
		0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	709.41*

*Critical temperature



2.0 Plant Design, Including Safety and Emergency Systems (25.0)

2.1 With regard to the Diesel Generators:

a. The "Emergency Bypass-Test" selector switch is in "Emergency Bypass" position when a low reactor water level initiation signal is received. Which diesel generator trips would still be operable? (2.0)

b. Which of these trips would be bypassed if the selector switch was in the "Test" position and the same initiation signal was received? (0.5)

2.1 a. (all 4, 0.5 each) (2.0)

1. Engine overspeed
2. Generator differential relay
3. Fail to start (incomplete response)
4. Emergency stop pushbutton

b. None (0.5)

Ref: WNP-2 System and Procedures; DG, pg. 33-35.

2.2 What safety action(s) are auto-initiated at each of the following indications:

- a. Level 4 (31.5")? (0.5)
- b. Level 2 (-50")? (0.5)
- c. 1135 psig? (0.5)
- d. 1076 psig? (0.5)

- 2.2 a. Runback recirc. flow if only 1 reactor feed pump. (0.5)
- b. Initiate RCIC^(0.1) and HPCS^(0.1); trip recirc. pump^(0.2); initiate NSSS iso. groups 1, 2, 3, 4 and 7^(0.1) (Equipment, in lieu of groups, acceptable if all equipment for these groups is given; if not, give only partial credit). (0.5)
- c. ATWS trip of recirc. pumps. (0.5)
- d. ② MSL S/R valves open (relief mode). (0.5)

Ref: WNP-2 System and Procedures; NBI, pg. 55 and 57; Main Steam, pg. 32.

not req'd ("first" or "not all" - OK)

2.3 With respect to the Automatic Depressurization System (ADS):

- a. List (including setpoints) the automatic activation sequence for ADS. (1.5)
- b. Which initiation signal(s) can be cleared by pressing a Seal-In Reset pushbutton(s)? (1.0)
- c. Which of the signal(s) in (b) can be cleared only if the initiating condition no longer exists? (0.5)

- 2.3 a.
 - 1. Hi drywell pressure (>1.65 psig) (0.3)
 - 2. Lo water level (level 3: ~~50"~~ +13") (0.3)
 - 3. Lo water level (level 1: -129") (0.3)
 - 4. 105 second timer timed out (0.3)
 - 5. >1 low pressure ECCS pump (125 psig for RHR/LPCI; 145 psig for LPCS). (0.3)
- b. 0.5 pts each (1.0)
 - 1. ADS A(B) reactor pressure vessel low level logic
 - 2. Hi drywell pressure A(B)
- c. Hi drywell pressure (0.5)

Ref: WNP-2 System and Procedures; AOS, pg. 8-10; 12; and 12, respectively; + NB 1055.

2.4 With regard to the Reactor Protection System:

- a. Which trip(s) can only be bypassed manually? (1.25)
- b. With regard to the backup scram valves:
1. Are they solenoid or air operated? (0.5)
 2. To cause a scram, do they:
(a) energize or (b) deenergize (if solenoid)
(c) pressurize or (d) vent (if air operated)
(choose only 1 answer) (0.5)
 3. What is their function? (0.5)

- 2.4 a ^{all required for full credit}
~~(0.417 each)~~ 0.21 (1.25)
1. APRM Hi-Hi
 2. APRM inop.
 3. Scram discharge volume Hi level trip
 4. ^{Non-coincident scram}
^{F&A (manual bypass)}
- b. 1. solenoid (0.5)
2. energize (a) (0.5)
3. bleed air from scram valves (vent header to atmosphere) (0.5)
(initiate a scram in case of scram pilot valve failure also acceptable)

Ref: WNP-2 System and Procedures; RPS, pg 15, 15, 16;
and 30-31, respectively.

2.5 With regard to the AC Electrical Distribution System:

- a. Which lockout relay will be tripped upon a Transformer Differential Current (87TM)? (0.5)
- b. List three (3) actions which will occur when the lockout relay in (a) is tripped? (1.5)
- c. The loss of which 480V MC bus will deactivate both loops of shutdown cooling? (0.5)
- d. What happens upon the loss of the normal and startup sources for SM-4? (0.75)

Not req'd for credit:

- 2.5 a. Unit lockout (86XU) (0.5)
- b. any 3, 0.5 pts each (1.5)
 - Trips and locks-out all "N" breakers
 - Trips 4F circuit breaker {Generator Exciter Field Breaker}
 - Trips main turbine (20 AST)
 - De-energizes 86XIU
 - Starts oscillograph
 - Starts computer
- c. MC-8-B-A () (0.5)
- d. automatic transfer of SM-4 to the Division 3 EDG (0.75)

all of the following is also acceptable

Ref: WNP-2 System and Procedures; AC Distr.; pg. 15 and 16 for a, b, d; AOP 4.7.1.9, Loss of Power to SM-8, pg. 3 of 4 for c.

radiation, RAE-V-9, not req'd for credit.

*close all 5" breakers
 Trips and locks-out 500+V breakers
 Lock out generator field breaker
 " " 500+V thru supp. circuit
 " " main generator
 Removes gen. ground protection
 Blocks "N" breaker trip annunciation*

Delete 2 & b.

2.6 Concerning the Standby Liquid Control System (SLC):

- a. Give three (3) of the four automatic actions which occur when the SLC System Control Switch is placed in the "Sys A" position? (1.0)
- b. What is the purpose behind the SLC storage tank heater? (1.0)
- c. There is a SLC pump trip on low flow (TRUE or FALSE)? (0.5)

2.6 (any 3, 0.33 each) (1.0)

- a.
 1. Both SLC squib-valves fire.
 2. RWCU-V-4 isolates.
 3. Both SLC storage tank outlet valves open.
 4. SLC-P-1A starts (if at least one suction valve is open).
- b. Maintains solution temperature high enough to prevent precipitation of the sodium pentaborate. (1.0)
- c. False (0.5)

Ref: WNP-2 System and Procedures, SLC; pg. 9, 13, 10, respectively.

2.7 Answer the following questions concerning the Low Pressure Core Spray System (LPCS):

- a. What is the rated flow of the main LPCS pump? (0.75)
- b. In shutting down LPCS to standby readiness, is the injection shut off valve (LPCS-V-5) closed before stopping the LPCS pump (Yes or No)? Why? (1.25)
- c. The check valve located inside the drywell is motor-operated (Yes or No)? (0.5)
- d. What are the interlocks associated with the auto-opening of the LPCS-V-5 injection valve? (0.75)

- not req'd for credit even though "rated" by definition for centrifugal pumps. take for this.*
- 2.7 a. 6350 gpm (at 128 psid reactor to suppression chamber.) (0.75)
 - b. ^{YES} ~~No~~ (0.5); to ensure no voids are left in the discharge line which would cause water hammer upon subsequent pump restart (0.75) (1.25)
 - c. No (0.5)
 - d. (^{not req'd for credit} No undervoltage on SM-7 and) reactor pressure <470 psig. (0.75)
- Ref: WNP-2 System and Procedures, LPCS; pg. 8, 11-12, 5, and 7, respectively.

*Operator should know location of valve relative to pump
Need to know how to prevent water hammer.
Need not know procedure step or valve numbers.*

"pressure interlock" is OK since I didn't specify set point in question.

- 2.8 With regard to the Plant Service Water System (TSW):
- a. What is the purpose of the Chlorine System? (1.0)
 - b. What causes starting of the ^{TSW} TBCCW pumps and opening of discharge valves? (1.5)
 - c. Since TSW provides cooling only to non-essential equipment, why must the plant be shutdown when neither TSW pump can be started? (0.75)

- 2.8 a. To inject chlorine to retard the growth of algae within TSW systems. *"To retard growth of organics" also acceptable* (1.0)
- b. *all 3 required (0.5 each)*
 A Pumps start on low pressure (<80 psig) on alternate *not required* pump or undervoltage on SM 85 (75) for 15 sec (0.75); discharge valve opens when pump starts if control switch in auto (0.75); *TSW pump selected for auto start will come on after diesel has responded to low on loss of offsite power* (1.5)
- c. Because components cooled by TSW are essential for continued operation of the secondary and primary plant. (0.75)

Ref: WNP-2 System and Procedures, TSW; pg. 3, 4, 7, respectively.

2.9 Concerning the CRD Hydraulic System, give the appropriate values for the following:

a. Insert drive water pressure at 400 psig reactor pressure. (0.5)

b. Cooling water pressure at 400 psig reactor pressure. (0.5)

2.9 ^a 1. $Rx + 260 = 660$ psig (0.5)

^b 2. $Rx + 20 = 420$ psig (0.5)

Ref: WNP-2 System and Procedures; CRDH; pg. 2.

2.10 Concerning the Condensate Storage and Transfer System (CSTS):

- a. What is the minimum level that must be maintained in CST tanks at all times? (0.5)
- b. Why is this minimum level required? (1.0)

- 2.10 a. 135,000 gal ^{of} 6'8"± (0.5)
- b. To provide suction for RCIC and HPCS systems to ensure immediate availability of sufficient condensate for ECCS and shutdown (1.0)

Ref: WNP-2 System and Procedures, CSTS; pg. 1 and 13.

3.0 Instruments and Controls (25.0)

3.1 With regard to the Rod Worth Minimizer System (RWM):

- a. Under what two (2) conditions will the Select Error alarm light be lit? (1.0)
- b. Above LPAP, what alarms remain operative? (1.5)
- c. TRUE or FALSE: A rod block is applied upon the second insertion error. (0.5)

- 3.1 a. Whenever a selected control rod is not in the currently latched group (0.5) or one currently positioned so as to cause a withdraw or insert error (0.5) (1.0)
- b. Inop/Reset; Withdraw block; Insert block (1.5)
- c. False (0.5)

Ref: WNP-2, System Description, RWM, pp. 18, 20, 18, respectively.

- 3.2 For the events listed, match the action(s) that will occur in the Recirculation System. Assume that the pumps are running in high speed. (An action may be used more than once)

(2.5)

Events:

1. Suction or discharge valves <90% open
2. Vessel hi pressure (ATWS)
3. Feedflow <30% with FCV <18% open *deleted? yes supervisor*
4. Reactor vessel low level (Level 3)
5. RPT

Actions:

- a. Fast Speed Trip
- b. Slow Speed Trip
- c. LFMG start.

- 3.2 (0.5) each

(2.5)

1. A, B
2. A, B
3. A, C
4. A, C
5. A, C

Ref: WNP-2, RRC, pg. 39.

3.3 According to Monthly Operational Bulletins:

delete

~~a. How did the failure, on two shifts, to check chart movement on wetwell level recorder GMS-LR/RR-4 contribute to loss of wetwell level? (1.0)~~

b. Why is it important to ensure that local temperature indicators at the nitrogen supply shed and in the reactor building are monitored? (1.0)

delete

~~a. The recorder had been, in fact, inadvertently de-energized so that annunciator alarm switches activated by the recorder pen were also O.O.S. (1.0)~~

b. No control room monitors exist. If nitrogen temperature gets too low, nitrogen flow onto a 30 in. dia. containment purge header and onto wetwell and drywell purge liner inside containment could cause failure through nitrogen embrittlement. (1.0)

Ref: WNP-2 Monthly OP Bull: April-May, pg. 1; Feb-Mar, pg. 6, respectively.

3.4 In reference to the Source Range Monitors (SRM):

a. What two (2) types of radiation are separated by the pulse height discriminator (PHD)? Which one causes an output signal from the PHD? (1.0)

b. Indicate (by Yes or No) whether the following trip circuits in the SRM electrical circuitry will generate a signal for use in the RMCS rod block circuitry: (3.0)

1. Downscale
2. Retract Permit
3. Upscale High
4. Upscale High High (shorting links installed)
5. Incp.
6. Reactor period.

3.4 a. Neutron and gamma radiation (0.5); neutrons cause output (0.5) (1.0)

b. (0.5) each (3.0)

1. Yes
2. Yes
3. Yes
4. No
5. Yes
6. No

Ref: WNP-2, System Description, IRM, pg. 14 and 28, respectively.

3.5 With regard to the Reactor Manual Control System (RMCS):

- a. The accumulation light starts flashing: (1.5)
 - 1. What is the cause(s) of this?
 - 2. What causes the light changing to "steady on"?
- b. In mode 5 under what conditions, and in what manner will Select block be indicated? (1.5)
- c. Can an overtravel alarm be received if the control rod is connected to its drive unit (Yes or No)? (0.5)

- 3.5 a. 1. High-level (5 cc) or low N₂ pressure (970 psig) (0.75)
2. Operator acknowledges alarm with "Accumulation Trouble Acknowledge" pushbutton (0.75) (1.5)
- b. Any rod is not fully inserted (0.75); SELECT BLOCK amber light (0.75) (1.5)
not required for credit
↳ also acceptable: If operator withdraws one C.R.
- c. No. (0.5)

Ref: WNP-2, System Description, RMCS, pp. 7, 11, and 18, respectively.

3.6 With regard to the RCIC system:

delete

a. RCIC pump flow indicator RCIC-FIS-2 has two contacts. What is the purpose of each contact?

~~(2.0)~~

b. For monitoring steam flow to the RCIC turbine:

1. How many differential pressure switches (DPS) are used to monitor steamflow?

(0.5)

2. Where are they located?

(0.75)

c. Should the RCIC-V-8 and RCIC-V-63 keylocked control switches be left in OPEN or CLOSED position when resetting any isolation signal?

(0.5)

3.6

delete

a 1. on high flow to send a signal to close the minimum flow bypass valve (RCIC-V-19) (1.0) and

2. on low flow with high discharge pressure (RCIC-PS-20) to open RCIC-V-19 (1.0)

~~(2.0)~~

b. 1. 4

(0.5)

2. 2 - downstream of inboard steam isolation valve (RCIC-V-63) (0.375); 2 - downstream of the branch line to the RHR (0.375) (*give partial credit (0.25) if all located on flow allow valve inside drywell*)

(0.75)

c. Open ~~Closed~~

(0.5)

Ref: WNP-2, System Description, RCIC, pp. 27, 27; and SOP 2.4.6, pg. 2 of 29, respectively.

- 3.7 Concerning vessel instrumentation, state whether the following are TRUE or FALSE:
- a. The Fuel Zone Range level indicators are calibrated cold. (0.5)
 - b. Level 1 (-128") will initiate NSSSS isolation groups: 1, 2, 3, 4, and 7. (0.5)
 - c. The reference leg design of the Level Indicators have been designed to compensate for extreme temperature transients. (0.5)
 - d. Jet pumps 5, 10, 15, and 20 were individually flow calibrated prior to installation. (0.5)
 - e. Pressure measured at the core inlet plenum is also used as input to the CRDH system. (0.5)

3.7 (0.5) each (2.5)

- a. False
- b. False
- c. False
- d. True
- e. False

Ref: WNP-2, System Description, NBI, pp. 57, 56, 5, 7, and 33, respectively.

3.8 With regard to the Power Range Neutron Monitoring System (PRMS):

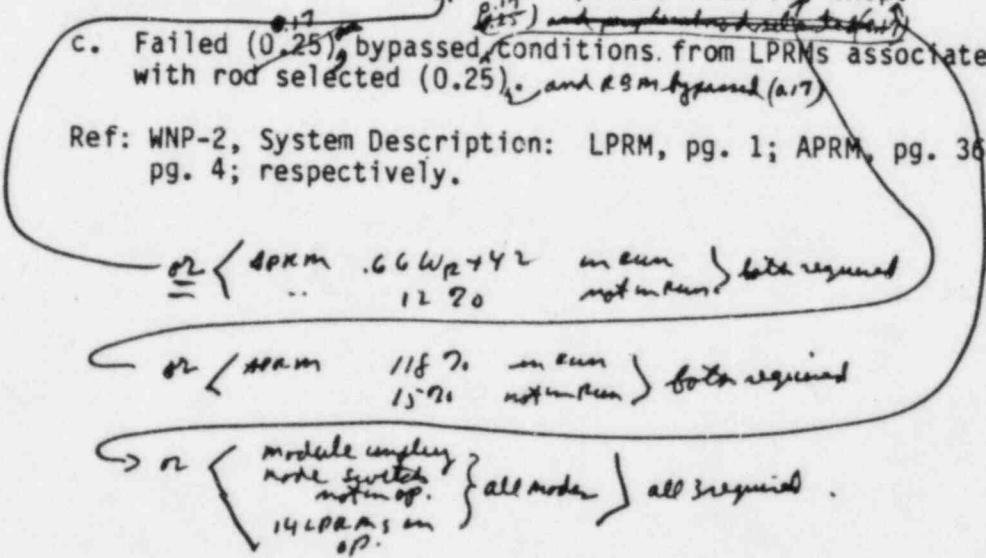
- a. What four (4) subsystems make up the PRMS? (0.75)
- b. Which three (3) trips are input to RPS from the PRMS? (0.75)
- c. For what two (2) conditions and for which components do the white indicators next to the heat flux meters below the full core display become lit? (0.75)

3.8 a. RBM, Flow Unit System, APRM, LPRM (0.75)

b. APRM upscale thermal, APRM upscale neutron, inop. (0.75)

c. Failed (0.25), bypassed, conditions from LPRMs associated with rod selected (0.25) and RSM bypassed (0.17) (0.75)

Ref: WNP-2, System Description: LPRM, pg. 1; APRM, pg. 36; LPRM, pg. 4; respectively.



3.9 Which type of detector (scintillation, ion chamber, fission chamber or Geiger Mueller) is used in the following process radiation measurements?

- a. Main steam line (0.5)
- b. Off-gas post-treatment (0.5)
- c. Reactor building main exhaust (0.5)

- 3.9 a. Ion chamber (0.5)
- b. Geiger-Mueller (0.5)
- c. Geiger-Mueller (0.5)

Ref: WNP-2 Syst. Descript., PRM, pg. 4, 3, 4, respectively.

Section points removed

3.3a	- 1.0
3.6a	- 2.0
	<hr/>
	- 3.0 = 22.0 new total

4.0 Procedures - Normal, Abnormal, Emergency and Radiological Control (25.0)

4.1 With regard to General Operating Procedure 3.1.2, Reactor Plant Cold Startup:

- a. What action(s) should the operator take to prevent RWCU pump trip on low flow. (1.0)
- b. Why should you avoid heat-up rates that demand a high reject temperature . (1.0)
- c. How will RPV water level stability be indicated? (1.0)

- 4.1 a. By adjusting reject valves RWCU-V-31 and V-33 as required. (1.0)
- b. This will cause high RWCU F/D inlet temperatures and RWCU F/D isolation at 140°F. (1.0)
- c. By a small output signal on the RFW-FCU-10 controller. (1.0)

Ref: WNP-2 PPM 3.1.2., pg. 8 of 18.

- 4.2 With regard to performing Rod or Minimizer (RWM) initiation in accordance with the System Operation Procedures for RWM (2.1.4):
- a. How does the operator verify the RWM is not in "rod test"? (1.0)
 - b. What happens when the INOP/RESET pushbutton is depressed before the "System Initialize" pushbutton is depressed? (1.0)
 - c. What happens when the INOP/RESET pushbutton is depressed after the system is initialized? (1.0)

- 4.2 a. By depressing the TEST/SELECT button, observing illumination, depressing again and observing the light goes out. (1.0)
- b. Any previous alarm ("Comp/Program") is reset, the Comp light and the RWM/Program lights are illuminated. (1.0)
- c. The RWM and program lights extinguish. (1.0)

Ref: WNP-2 SOP 2.1.4, pg. 2-3 of 3.

4.3 Relative to the Emergency Operating Procedure for RPV Pressure Control (RPV/P)(5.1.2):

a. List any three (3) of the five entry conditions. (1.5)

b. What are the four (4) systems to be used to augment the main turbine bypass valves for controlling pressure below 1075 psig? Give any limiting condition on the use of these systems. (3.0)

4.3 a. any 3; 0.5 points each (1.5)

- 1) RPV water level below +13.0 in.
- 2) RPV pressure >1037 psig
- 3) Drywell pressure >1.68 psig
- 4) A condition requiring MSIV isolation
- 5) A condition requiring reactor scram and power is above 5% or cannot be determined.

b. (3.0)

- 1) SRV's (0.6), if suppression pool water level >17 ft (0.3)
- 2) RCIC (0.6)
- 3) RWCU (recirculation through heat exchanger and blowdown modes) (0.6); if no boron has been injected into RPV (0.3)
- 4) Main steam line drains (MS-V-16, -19, -21) (0.6)

Ref: WNP-2 EOP 5.1.2, pp. 1 and 6-7 of 8.

not req'd for credit

- 4.4 a. The plant is in the process of starting up (Condition 2) with all systems and components normal except that the "A" IRM has previously failed high and was subsequently bypassed. The "E" IRM now loses power and is declared inoperative. May the plant continue in this condition for an extended period of time without being in violation of Tech. Specs? Also give the appropriate action statement. (1.5)
- b. Could you place the mode switch in run (Condition 1) to bypass the action statement in part "A"? (0.5)

- 4.4 a. Yes (0.5); place the RPS A channel in the tripped position within one hour (1.0). (1.5)
- b. ^{YES}~~NO~~ (0.5). (0.5)

Ref: WNP-2 T/S, pp. 3/4 3-1; 3/4 0-1, respectively.

1 B C D E F

- 4.5 Assuming a loss of feedwater heaters while operating at 100% power, according to Abnormal Operating Procedure 4.2.7.2:
- Give three (3) of the four events that could have caused this? (2.25)
 - What change would you expect to see in the Main Generator MW (increase, decrease)? (0.5)
 - What is the first immediate operator action you should take? (0.75)

- 4.5 a. *Each correct set (Any 3 @ 0.75 each) (1.125) (1.125) (1.125) (1.125) (2.25) (2.25)*
Correct answers are 1 and 2 or 1 and 4 (~~1 and 3~~) (2.25)
- Heater isolation on high water level.
 - Turbine trip.
 - ~~(System malfunction resulting in the) isolation of or more feedwater heaters.~~
 - (System malfunction resulting in the) closure of extraction steam line valves for one or more feedwater heaters. *(OK only if #2 not given)*
- b. Increase (0.5)
- c. Reduce reactor power via recirculation flow control (0.75)

Ref: WNP-2 AOP 4.2.7.2, pg. 1 of 2.

4.6 With regard to the operating procedure for 250V DC Distribution System (SOP 2.7.7), give any three (3) of the four indications that the operator will have if the tie to distribution bus S2-1 has been completed.

(2.25)

4.6 (Any 3 @ 0.75 each) *Exact wording not required*

(2.25)

1. "250 VDC LOSS, BATT B2-1 FAIL" alarm on board "C" in Control Room clears.
2. "250V VDC BATT B2-1 GND" alarm on board "C" remains cleared.
3. Bus S2-1 voltage reads 220 to 250V on board "C".
4. Bus S2-1 ground lamps on board "C" are on.

Ref: WNP-2 LSOP 2.7.7, pg. 2 of 5.

*Procedure memorization
not req'd.
Exact name of alarm
not req'd.
Delete question.
Key by worded.*

4.7 Reactor coolant leakage into the primary containment from unidentified sources shall not exceed (1) gpm and the total coolant leakage shall not exceed (2) gpm. (1.0)

4.7 1. 5 gpm (0.5)

2. 25 gpm (0.5)

Ref: WNP-2 Tech Specs., pg. 3/4 4-9.

4.8 The Reactor Operator reports that "GEN BUS DUCT TEMP HIGH" and "GEN BUS DUCT CLR FLOW LOW" have activated and that bus duct temperatures are increasing. The failure of which component(s) is the most probable cause?

(1.0)

4.8 The TSW solenoid supply valve

(1.0)

Ref: WNP-2 AOP 4.5.6.1, pg. 2 of 2.

4.9 According to AOP 4.8.3.2 "Loss of all RCCW," if no RCCW pumps can be started during power operation, a rapid increase will occur in (Fill in).

(0.5)

4.9 Drywell pressure

(0.5)

Ref: WNP-2 AOP 4.8.3.2, pg. 3 of 3.

4.10 According to the Abnormal Operation Procedures for Fires (4.12.4.1), one indication, other than fire alarm, will be fire header pressure fluctuation (TRUE or FALSE)?

(0.5)

4.10 True

(0.5)

Ref: WNP-2 AOP 4.12.4.1, pg. 1 of 2.

4.11 According to the Limitations stated in the Operating Procedures for the Reactor Core Isolation System (SOP 2.4.6), what must you do if manual isolation is required at any time that system initiation is not sealed in?

(0.75)

4.11 Close the isolation valves using their respective control switches.

(0.75)

Ref: WNP-2 SOP 2.4.6, pg. 2 of 28.

4.12 With regard to Administrative Procedures:

- a. There must be two (2) licensed operators in the Control Room at all times (TRUE or FALSE)? (0.5)
- b. During new fuel handling operations, a licensed operator must be on the refueling floor (TRUE or FALSE)? (0.5)

4.12 a. False (0.5)

b. False or TRUE if answer references Tech Specs. (0.5)

Ref: WNP-2 Admin. Proc: 1.3.2, pg. 3 and 6.2.3, pg. 2, respectively.

T.S. para. 6.2.2.d for "True"

I disagree that "b" can be true. A SRO or SRO restricted to Refueling is required Not ~~an~~ RO "a" is true if ~~control~~ ^{stated} one RO and one SRO.

4.13 According to Standing Order/Night Orders (Admin. Proced. 1.3.1), under what conditions can the reactor operator shut the reactor down without being instructed by the Shift Manager or required by the Emergency Procedures?

(1.0)

4.13 When safety of reactor is in jeopardy or when operating parameters exceed any RPS setpoint and autosutdown does not occur.

(1.0)

Ref: WNP-2 Admin. Proc. 1.3.1, pg. 2.

4.14 With regard to the Health Physics Program, what are the whole body exposure limits for the following:

(1.0)

- a. Administrative exposure limits (day, quarter, year)
- b. Lifesaving actions

1250 mrem/yr also acceptable per ppm 111.3 1950725

- 4.14 1. 300 mrem/day (0.17); 1000 mrem/quarter (0.17);
5000 mrem/year (0.16)
2. 75 rem (0.5)

*(0.5)
(0.5)(1.0)*

Ref: WNP-2 Health Physics Program, 3.1.5, pg. 4 of 5.

Master

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

Facility: WNP-2
Reactor Type: BWR-5
Date Administered: 11/6/84
Examiner: I. S. Levy
Candidate: _____

INSTRUCTIONS TO CANDIDATE:

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

Category Value	% of Total	Candidate's Score	% of Cat. Value	Category
<u>25</u>	<u>25</u>	_____	_____	5. Theory of Nuclear Power Plant Operation, Fluids and Thermodynamics
<u>25</u>	<u>25</u>	_____	_____	6. Plant System Design, Control and Instrumentation
<u>23.5</u> <u>25</u>	<u>24.87</u> <u>25</u>	_____	_____	7. Procedures - Normal, Abnormal, Emergency, and Radiological Control
<u>21.0</u> <u>25</u>	<u>22.22</u> <u>25</u>	_____	_____	8. Administrative Procedures, Conditions, and Limitations
<u>94.5</u> <u>100</u>		_____		TOTALS
		Final Grade	_____ %	

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

Candidate's Signature

Questions and Answers to WNP-2 SRO Exam - 11/6/84

5.0 Theory of Nuclear Power Plant Operations, Fluids and Thermodynamics (25.0)

5.1 Give three (3) reasons why fuel densification is a problem. (2.25)

5.1 (any 3; 0.75 each) (2.25)

1. Local power spikes resulting from axial fuel column gaps.
2. Increased linear heat generation rate due to pellet axial shrinkage.
3. Cladding collapse at the location of axial fuel column gaps.
4. Increased stored energy due to decreased pellet-cladding thermal conductance resulting from increased radial gap size.

Ref: Morris Training Center: Thermo/HT/Fluid Flow (3/83), pg. 9-107

where 4 answers are given, 0.5625 p/answer.

where 5 answers are given, $\frac{2.25}{5} = .45$ each.

5.2 Your latest computer printout of MFLPD and MAPRAT shows the following values for Regions 1 to 3.

Region	1	2	3
MFLPD	0.95	1.0	1.05
MAPRAT	0.92	1.08	1.00

q: T.S. "safety limit" on LCO?

A: I said, "Use on some; give any assumptions you want in your answer"

- Which, if any, of these values are beyond their safety limits? (1.0)
- Why are each of the above limits imposed? (What do they protect against?) (1.5)
- Compared to BOL, would the values for MAPRAT at EOL be larger or smaller? Why? (1.5)

- MFLPD Region 3 (0.5)
MAPRAT Region 2 (0.5)
- MFLPD - Maintains <1% cladding strain, fuel failure. (0.75)
MAPRAT - Maintains <2200°F following LOCA, decay heat removal (0.75)
- Larger (0.5); MAPLHGR limit decreases (0.5) since local peaking factor gets smaller as, with exposure, heat transfer is reduced (0.5). (1.5)

Ref: WNP-2 Systems and Procedures, PC, Pg. 16; and MTC Thermo/HT/FF (3/83), pg. 9-66 through 9-76.

also ok: post-LOCA peak temp heat transfer will decrease since less difference between high power and low power pins in any small region

5.3 In reference to the reactor water cleanup regenerative heat exchanger, assume the following conditions and, then, perform the calculations.

Conditions:

- T inlet from reactor (tube side) = 550°F
- T outlet from H_x (tube side) = 250°F
- T inlet shell side = 120°F
- Tubeside flow rate from reactor = 1300 gal/min
- Shellside flow rate to reactor = 1300 gal/min

Calculate (and show all work):

- a. The amount of heat transfer (Btu/hr) on the tube side. (1.0)
- b. The temperature of the water going back to the reactor. (1.0)
- c. The maximum temperature of water going back to the reactor if the flow back to the reactor were decreased from 1,300 gpm to 800 gpm due to a leak in the tube side of the H_x. (1.0)

*f. Do you want me to calc. new temp. ref. due to leak.
 A. No, it is important to know why decrease occur; only that it did occur.*

5.3 a. $\dot{Q} = \dot{M} \times C_p \times \Delta T$
 $= 1300 \text{ gal/min} \times 60 \text{ min/hr} \times 8.33 \text{ lbm/gal}$
 $\times 1 \text{ Btu/lbm}^\circ\text{F} \times (550-250)$
 $= 1.949 \times 10^8 \text{ Btu/hr}$ (1.0)

b. 420°F (conservation of energy) (1.0)

c. 550°F as follows:

$$\dot{Q} = \dot{M} \times C_p \times \Delta T$$

$$T_{\text{max}} = \dot{Q}/\dot{M} + 120$$

$$= \frac{1.949 \times 10^8 \text{ Btu/hr}}{800 \text{ gal/hr}} \times \frac{1 \text{ hr}}{60 \text{ min}} \times \frac{1 \text{ gal}}{8.33 \text{ lbm}} \times \frac{1}{1 \text{ Btu/lbm}^\circ\text{F}}$$

$$= 607.5^\circ\text{F} \text{ (0.5)} \quad (1.0)$$

but cannot be greater than 550°F (max. T from reactor to inlet) (0.5)

Ref: Morris, T. C.; Thermo/HT/Fluid Flow (3/83), pgs. 8-40,41.

5.4 While at 75% power, the master feedwater controller fails low. Will the NPSH of the recirculation pumps increase, decrease, or remain unchanged? Briefly, explain why. (1.25)

5.4 Decrease (0.5); reduced subcooling (0.75) (1.25)
or
reduced level in downcomer or ~~down~~

Ref: Morris, T. C.; Thermo/HT/Fluid flow (3/83), pg. 7-96.

5.5 With regard to excess reactivity:

- a. The excess reactivity for a cold, clean, critical reactor is greater than that for the hot, clean, zero power condition (TRUE or FALSE). (0.5)
- b. The excess reactivity for the hot, clean, zero power condition is smaller than that for the hot, full power, equilibrium Xe and Sm condition (TRUE or FALSE). (0.5)
- c. The excess reactivity at EOL is greater than that at BOL (TRUE or FALSE). Why? (1.25)

- 5.5 a. True (0.5)
- b. False (0.5)
- c. False (0.5); fuel depletion and fission products increase (0.75) (1.25)

Ref: WNP-2, Reactor Theory Rev. pg. 39.

5.6 The WNP-2 reactor is taken to criticality from a cold condition and then placed on an 80 second positive period.

a. From control room nuclear instrumentation, how can the operator tell when the heating range has been reached? (Rod position and recirculation are held constant). (0.75)

b. In which of the following intervals was the heating range entered? Explain the reason for your answer. (Show all work.) (1.5)

Interval 1 - reactor power increased by a factor of 6 in 143.3 seconds.

Interval 2 - reactor power increased by a factor of 3 in 99.0 seconds

Interval 3 - reactor power increased by a factor of 5 in 128.8 seconds.

(Note: the intervals may not be in sequence.)

5.6 a. Operator can notice that period has become longer^(0.391) and that power change on IRMs is leveling off^(0.375). (0.75)

b. (From $P = P_0 e^{t/T} + T = \frac{t}{\ln P/P_0}$) Interval 2 (0.5); the period has lengthened from 80 seconds. The other intervals have 80 second periods (1.0). (1.5)

Ref: General control room indications; WNP-2, Reactor Theory, pg. 58.

5.7 With regard to Reactivity Coefficients:

a. Which reactivity coefficient is the most dominant under the following conditions: (2.0)

- 1) During rod drop accident at 15% power
- 2) Pulling rods at 1% power
- 3) MSIV closure at 100% power
- 4) Feedwater controller fails high at 100% power?

b. For "feedwater controller fails high at 100% power," a.4) above:

1) Give the reason for your answer to a.4) above. (1.0)

2) What will happen to power (increase, decrease, stay the same)? (0.5)

3) What is the approximate value at BOL of the coefficient you gave as your answer to a.4) above? (0.5)

5.7 a. (0.5 for each) (2.0)

- 1) Doppler coefficient
- 2) Moderator coefficient
- 3) Void coefficient
- 4) Void coefficient

b. 1) Increase in core subcooling^(0.33) which reduces void fraction^(0.33) whose coefficient has greatest effect on reactivity^{0.34/} (1.0)

2) Increase (0.5)

3) $-1 \times 10^{-3} \Delta K/K/1\%$ void change (give credit for correct value for answer to a.4 even if a.4 is incorrect) (0.5)

Ref: Standard Reactor Theory.

5.8 Near the end-of-cycle (EOC), will differential control rod worths near the bottom of the core be lower or higher than those near the top of the core? Why?

(1.75)

5.8 Higher (0.5). As fuel burns, control rods must be withdrawn causing flux to peak lower in the core (0.75); and rod worth is proportional to flux² (0.5).

(1.75)

Ref: WNP-2, Reactor Theory, pg. 81.

$\left(\frac{\phi_{\text{local}}}{\phi_{\text{avg}}}\right)^2$ OK.

5.9 With regard to the effects of equilibrium Xenon and Samarium (the reactor has been operated at a constant power for many days):

a. If reactor power is then doubled, will the new equilibrium Samarium concentration be exactly twice as great (YES or NO)? Explain. (1.5)

b. If the reactor is shut down, initially by a 1% $\Delta K/K$, will the initial effect of Xenon be to increase or decrease the shutdown margin? (0.5)

5.9 a. No (0.5); the equilibrium value of samarium does not depend on flux, and, therefore, it does not depend on power level (1.0). (1.5)

Ref: WNP-2 Reactor Theory, pg. 87.

b. Increase. (0.5)

Ref: Standard Reactor Theory

5.10 With regard to Delayed Neutrons:

- a. In causing fissions, what is the major difference between delayed neutrons and prompt neutrons? (0.75)
- b. Explain how and why the value of the delayed neutron fraction, Beta, changes from the beginning of core life to the end of core life. (0.75)
- c. Explain the effect on reactor control of the change in Beta with core life. (0.75)

- 5.10 a. Delayed neutrons have a lower probability of causing fast fission (the "importance" factor is less than 1). (0.75)

Ref: Standard Reactor Theory.

- b. Beta will decrease from about 0.007 at BOL to 0.0054 (0.375) at EOL due to buildup of Pu-239 and depletion of U-235 (0.375) (0.75)

0.0056 for WSP-2 Reactor Theory Review

Ref: Standard Reactor Theory.

- c. As beta decreases with core age, reactor period decreases and, therefore, for the same reactivity addition rate, a shorter period and less easy control is obtained at EOL. (0.75)

Ref: Standard Reactor Theory

$$P = P_0 e^{t/\tau}$$

$$1Ci = 3.7 \times 10^{10} Bq$$

$$\alpha_D = -1 \times 10^{-5} \frac{\Delta K / Z^{\circ} F}{K}$$

$$\alpha_V = -1 \times 10^{-3} \frac{\Delta K / Z^{\circ} F}{K} \text{ voids}$$

$$\alpha_H = -4.5 \times 10^{-4} \frac{\Delta K / Z^{\circ} F}{K}$$

$$\alpha_P = -4.5 \times 10^{-4} \frac{\Delta K / Z^{\circ} F}{K} \text{ power}$$

$$I(t) = I_0 e^{-\lambda t}$$

$$T_{1/2} = \ln(2)/\lambda$$

$$C_p = (C_{p_{base}}) (K_s) (K_A)$$

$$Q = MC_p \Delta t$$

$$\Delta p = f \frac{L \rho v^2}{D 2g_c}$$

$$f = 64/Re$$

$$\rho = \frac{k(\text{eff}) - 1}{K(\text{eff})}$$

$$\frac{1}{M} = \frac{CR1}{CR2} = \frac{1 - K(\text{eff})^2}{1 - K(\text{eff})}$$

$$M = \frac{CR2}{CR1} = \frac{1 - K(\text{eff})}{1 - K(\text{eff})^2}$$

$$\dot{Q} = MAh$$

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

$$M = 1/(1-k)$$

$$N(t) = N_0 e^{-\lambda t}$$

$$\alpha_{rod} = (L_2 + L_3) \frac{(\phi_{rod})^2}{(\phi_{avg})}$$

$$n = v/(1+d)$$

$$P = I \phi v / (3.7 \times 10^{10})$$

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

$$\bar{\tau} = \frac{1}{\rho} + (\beta - \rho) / \lambda \rho$$

$$\tau = L / (\rho - \beta)$$

$$v = v_f + x v_{fg}$$

$$H = x h_g + (1-x) h_f$$

$$S = x S_g + (1-x) S_f$$

$$1 \text{ in.} = 2.54 \text{ cm}$$

$$1 \text{ gal.} = 3.785 \text{ liters}$$

$$1 \text{ kg} = 2.205 \text{ lb}$$

$$N = \rho A Q / A$$

$$17.58 \text{ watts} = 1 \text{ BTU/min}$$

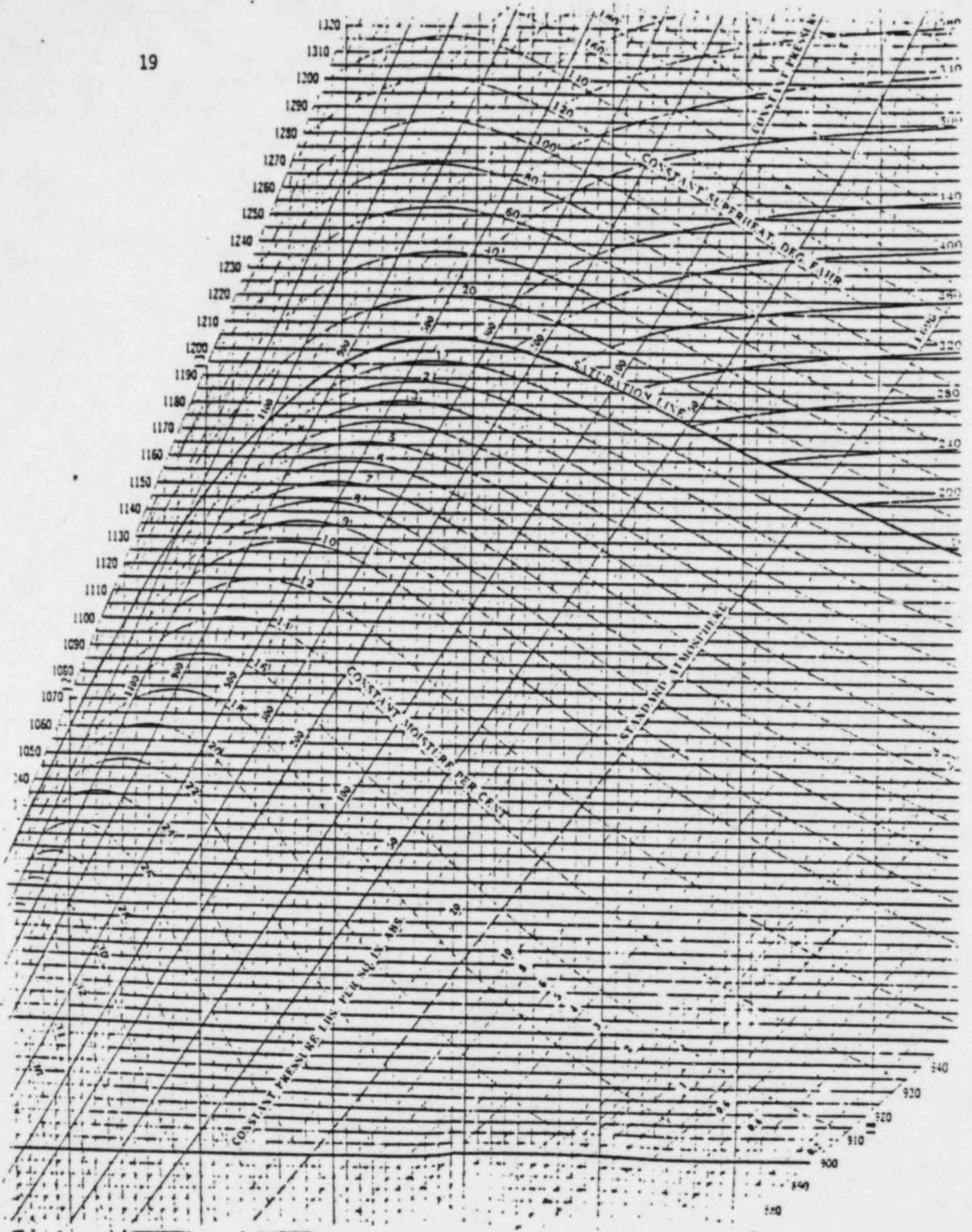
$$1 \text{ psi} = 6.895 \text{ Pa}$$

$$1 \text{ psi} = 2.036 \text{ } \frac{\text{H}_2\text{O}}{\text{H}_2\text{O}} \text{ (@ } 0\text{C)}$$

$$1 \text{ psi} = 27.68 \text{ } \frac{\text{H}_2\text{O}}{\text{H}_2\text{O}} \text{ (@ } 4\text{C)}$$

$$\bar{\beta} = .0071$$

$$\bar{I} = 2 \times 10^{-5} \text{ sec}$$



6.0 Plant System Design, Control and Instrumentation (25.0)

6.1 With regard to the Vessel Instrumentation:

- a. For what specific condition(s) is the Fuel Zone Range level indicators calibrated? (1.0)
- b. For the Narrow Range Level Indicators, MS-LIS-24A,B,C & D: (2.0)
- 1) Where are they located in the plant?
 - 2) To what systems/components do they send signals?
 - 3) At what levels are these signals initiated?
 - 4) What system/component action is initiated?
- c. How is the signal for jet pump flow generated in an individual (non-calibrated) jet pump? (1.0)

6.1 a. $\left. \begin{array}{l} 212^\circ\text{F at } 0 \text{ psig OR} \\ \text{Saturated steam conditions} \end{array} \right\}$ at 0 psig in the vessel and drywell with no jet pump flow. (1.0)

b. 1) Located in Reactor Building, 522' level. (0.5)

2), 3), and 4):

- RCIC/RCIC turbine; at level 8; shut steam supply. (0.75)
- RPS; level 3; scram and NSSS Group (5) and (6) isolations. (0.75)

c. Pressure, at pressure tap on pump throat, is compared to a common jet pump discharge pressure sensed at the core inlet plenum area by the SLCS injection line to develop a ΔP signal. (1.0)

Ref: WNP-2, Systems and Procedures: NBI, pgs. 5, 14 and 9, respectively. NS4, pgs 9 + 10

where extra way around given, divide value for appropriate area, (i.e. RPS + RCIC = $\frac{.50}{7} = .0714$ / ans. To give value/ans.

6.2 With regard to Process Radiation Monitoring:

- a. The main steam line radiation monitoring system will initiate condenser isolation (via AR-V-1) when appropriate channels and their trips are activated. What are these channels and trips? (1.0)
- b. What type of detector (G-M, etc.) is used in the following radiation monitors: (1.5)
- 1) R_x Building stack extended range
 - 2) Off-gas post treatment
 - 3) Circulating water effluent?
- c. Which of the monitors, in b. above, will initiate system isolation or trips? (1.0)

- 6.2 a. Channel A or C; HI HI or Inop. (1.0) *← 3x Normal OK.*
- b. 1) Beta scintillation (0.5)
- 2) GM (0.5)
- 3) Gamma scintillation (0.5)
- c. (0.5 each): 2,3 (1.0)

Ref: WNP-2, Systems and Procedures: PRM, pg. 7; 5, 3, 3; and 3, respectively.

6.3 With regard to the Emergency Diesel Generators:

- a. What are all the start signals for DG #2? (1.5)
- b. Describe the overspeed trip mechanism (including its location, its major components, and how it initiates an engine trip). (2.0)
- c. What is the purpose of the soak back pump? (0.75)

- deleted in Sept 84 rev. of S+P - no credit to be given*
- 6.3 a. 1) Bus SM-3 dead (0.3) or TR-S undervoltage (0.3)
 2) TR-B undervoltage (0.3)
 3) SM-8 undervoltage (0.3) ~~(0.3)~~ (0.5)
 4) (K98B) containment high pressure or reactor low water level (2.3) Level 1 ~~(0.3)~~ (0.3)
 5) *also manual from C.N. and local (0.3)* (1.5)
- b. Located in the camshaft counterweight housing (0.5); it consists of a flyweight held by a tension spring. When engine exceeds set limit, spring tension is overcome by centrifugal force acting on the flyweight, permitting an outward movement of the flyweight which actuates the trip lever (0.2). This causes latching the injector rocker arm in the depressed position, preventing fuel injection into the engine (0.5). (2.0)
- c. To provide oil to the turbocharger for prelubrication. (0.75)

Ref: WNP-2, Systems and Procedures: DG, pg. 62; Fig. 14; and Fig 8c and pg. 22, respectively.

if extra program divided 1.5 by total # of answers for individual pt values. i.e. 1.5 = 3/2 each.

*Part 1.
It is a significant device used to trip lever*

6.4 With respect to the Automatic Depressurization System (ADS):

- a. What ADS controls are located on control room backpanels P628 and 631, and which components do they control? (1.0)
- b. Under what normal condition(s) will pushing the four ADS main initiation pushbuttons not result in ADS initiation? What is the reason for this? (1.5)

- 6.4 a. (14) ^{OK: Manual Keyed} AUTO/OPEN control switches; ~~SRVs "A" and "B"~~ ^{required} solenoids. (1.0)
- b. If a low pressure ECCS pump is not operating (0.75); so that core will not be uncovered (0.75) (1.5)

Ref: WNP-2 System and Procedures, ADS, pg. 13 and 8, respectively.

(b) case of
"normal" condition?
If so make adjust-
ment to circuit.

6.5 With regard to the AC Electrical Distribution System:

- a. Which lockout relay will be tripped upon a Transformer Differential Current (87TM)? (0.5)
- b. List three (3) actions which will occur when the lockout relay in (a) is tripped? (1.5)
- c. The loss of which 480V MC bus will deactivate both loops of shutdown cooling? (0.5)
(9 - 20 bus - continuously separated / only 1 person - the other must be present)
- d. What happens upon the loss of the normal and startup sources for SM-4? (0.75)

question during exam.

- 6.5 a. Unit lockout (86XU) (0.5)
- b. [any 3, 0.5 pts each] (1.5)
 - Trips and locks-out all "N" breakers
 - Trips (4F circuit breaker) {Generator Exciter Field Breaker}
 - Trips main turbine (20 AST)
 - De-energizes 86XIU
 - Starts oscillograph
 - Starts computer
- c. MC-8-B-A (via deenergizing N.A.A.-V.9) (0.5)
∴ answer to question above
- d. Automatic transfer of SM-4 to the Division 3 EDG (0.75)
reposition of SM-4 by Div 3 EDG after it has started - also OK

Ref: WNP-2 System and Procedures; AC Distr.; pg. 15 and 16 for a, b, d; AOP 4.7.1.9, Loss of Power to SM-8, pg. 3 of 4 for c.

*clear all "S" breakers
 Trips and locks - out 500KV breaker
 locks at exciter field breaker
 " " 500KV/hr sync. circuit
 " " main generator
 removes 9m. ground protection.
 blocks "u" breaker trip annunciate*

6.6 For the Reactor Core Isolation Cooling System, what position should the RCIC-V-8 and RCIC-V-63 keylocked control switches be placed in prior to resetting any isolation signal? Why? (1.5)

6.6 CLOSED or STOP (0.5); otherwise (i.e. in OPEN), when isolation signal is ^{reset} received the valves would immediately open (0.5) which could cause extensive damage to system piping and components (0.5). (1.5)

Ref: WNP-2, SOP 2.4.6, pg. 2 of 28.

6.7 If a selected control rod is in the selected 4 rod group and too many reed switch closures occur:

- a. What indication(s) would the operator observe? (1.0)
b. How could the control rod be moved through this position? (0.75)

(
Q - RSCS or RLM rod group
A - answer for both if no other way to answer.
(front 1 guy = 2 on rt.)
- stop in late wire value ~~to~~ the terminal ^{selected} (4 rod group)

6.7 a. RPIS Data Fault, ~~Insert Block would light~~ [?] ~~with~~ ~~block~~. (1.0)

b. By using Substitute Data on RSCS (0.375) or bypassing on RSC (0.375) (0.75)

Ref: WNP-2 Systems and Procedures, RMCS, pp. 15, 16.

RSCS, p. 8.

if additional wrong answers
are given divide 1.0 by # of answers
to get value per answer.

6.8 What design provisions are made to assure air flow through Standby Gas Treatment System filters. What is the basis for this?

(1.75)

6.8 1) Each SGT train has two full-capacity fans powered from separate emergency buses (0.75); to prevent filters from igniting on loss of air flow due to the decay heat generated by entrained radioactive materials (1.0)^(0.5) (1.75)
2) heaters on inlet ~~to keep moisture from causing~~ filter blockage (0.5)
Ref: WNP-2 Systems and Procedures, SGT pg. 24, 3, respectively
(0.325)

(if more than 2 correct answers reduce value appropriateness of any incorrect answer.)

$\frac{.75}{3} = .25$ and $\frac{1.0}{3} = .33$ }

6.9 With regard to the Average Power Range Monitor System (APRM):

- a. What three (3) things could happen if the back panel APRM Mode Switch was placed into the standby position during normal operation? (1.0)
- b. What is the AC power source for APRM Channel F? (0.5)
- c. Why is the Thermal Power Monitor necessary? (1.0)

note: now only 3 things to happen - inop trip + 1/2 scram + red flag

- 6.9 a. ~~APRM operates normally if bypassed (0.33); gives an INOP trip (to warn operator if channel has not been bypassed) (0.34). Will give 1/2 scram (if channel not bypassed) (0.33).~~ *not req'd for credit* (1.0)
- b. (Bus B supplied by) RPS MG set B. *will give red flag (0.5) also OK.* (0.5)
- c. To avoid unnecessary scrams during power increase transients (due to flux leading thermal power). (1.0)

Ref: WNP-2 Systems and Procedures, APRM, pp. 26, 28 and 32-33, respectively.

*1 pt 4 ans. awards 1.0 by 4
for each answer = .25*

7.0 Procedures - Normal, Abnormal, Emergency, and Radiological Control (25.0)

7.1 According to procedures for Emergency RPV Depressurization (Contingency) (5.3.2):

- a. What is the primary system used to cause depressurization? (0.5)
- b. If the system in (a) is partially or totally unavailable, what is the next system to be used? (0.75)
- c. Under what conditions would systems other than those in a) and b) above be used and what are these systems? (2.5)

- 7.1 a. ADS (0.5)
- b. Other SRV's^(.50) until 7 are open (.25) (0.75)
- c. If <3 SRV's are open (0.5); main condenser, main steam line drains, RCIC, head vents (2.0) (2.5)

Ref: WNP-2 EOP 5.3.2, pp. 1, 1, and 2 of 8, respectively.

7.2 What changes to the reactor power, MCPR, and MAPLHGR operating limits are required before one (1) recirculation loop operation is permitted? (2.25)

7.2 ~~(0.75 each)~~ *NONE.* (2.25)

Reactor power: reduce it to <50% rated thermal power

MCPR: increase it by 0.01

MAPLHGR: reduce limit to 0.84 times the two loop op. limit.

Ref: WNP-2 AOP 4.2.1.10, pg. 2 of 3, and T.S. 3.4.1.1, pg. 3/4 4-1.

7.3 With regard to the operating procedure for 250V DC Distribution System (SOP 2.7.7):

- a. What should the operator do and where would he do it, to tie the 250V DC battery B2-1 to the 250V DC distribution bus S2-1? (2.0)
- b. When bus S2-1 is energized:
 - 1) What indication(s) does the operator have that no ground is present? (0.75)
 - 2) What indication(s) does the operator have when a ground develops? (0.75)

- 7.3 a. 1) Verify open, on DP-S2-1, (0.34)
 (a) breaker S2-1/4B (0.33)
 (b) breaker S2-1/4C (0.33)
 (c) breaker S2-1/4D (0.33)
- 2) Close, on DP-S2-1, (0.34) breaker S2-1/3C and verify (0.33). (2.0)
- b. 1) Two ground lamps on board "C" are lit. (0.75)
 2) The lamp connected to the grounded polarity goes out, and the appropriate bus ground annunciator comes on. (0.75)

Ref: WNP-2 SOP 2.7.7, pg. 2 of 5.

required if DC doesn't show knowledge of 2 lamps

2) Close breaker S2-1/3C (0.34) on DP-S2-1 and verify (0.33)

Question on applicability of question - response starting with "no".

not req'd but should know general location
not req'd but should know of breaker involving S2-1

words reflecting substance of answer all that is required - which is all SOP anyway

7.4 According to WNP-2 Annunciator Procedures, if the "OFF GAS VAULT RAD MONITOR DNSCL" annunciator alarmed, the operator should make a "verification" and a "check". Describe the verification and check to be made and at what location in the plant.

(2.0)

7.4 Verify: ^(0.25) downscale condition of OG-RIS-11 ^(0.25) ~~(0.5)~~; at P606 (0.5) (1.0)
Check: ^(0.25) radiation level of OG-RI-11 ^(0.25) ~~(0.5)~~; locally (0.5) (1.0)
Ref: WNP-2 Annunciator Procedures, 4.602.A5-4.2.^{5.7}

not req'd.
but location → required; i.e. r.

step by step or
Memorized Response
not req'd.

7.5 According to the Operating Instructions for the Control Rod Drive System (SOP 2.1.1):

a. During system ^{initial valve lineup} startup, what should be the position (open or closed) for the following Hydraulic Control Unit (HCU) valves: (2.0)

- 1) CRD-V-111 (Cartridge Valve Nitrogen Inlet)
- 2) CRD-V-107 (Accumulator Water Drain)
- 3) CRD-V-102 (Withdraw Water Isolation)
- 4) Withdraw Line "Dragon" valve.

delete b. During system startup, in what position should the operator place the charging water header vent valve (CRD-V-65) that is located between the charging water header isolation valve (CRD-V-34) and the HCU charging water inlet isolation valves (CRD-V-113/HCU)? Why? (1.6)

7.5 a. (0.5 ea.) ^{operational} (note: This is normal ^{operational} procedure after system filled, vented and HCU is on charge) (2.0)

- 1) Open
- 2) Closed
- 3) Open
- 4) Closed

delete b. Open (0.5); to provide leak off (0.5) so that HCU accumulators will not pressurize due to valve leakage (0.5) (1.5)

Ref: WNP-2 SOP 2.1.1, pp. 4 and 6 of 41, respectively.

7.6 According to the procedures for Normal Shutdown To Cold Shutdown (G.O.P. 3.2.1), at what percent power (approximate) should the following actions be performed? (2.0)

- a. Transfer the 6900 volt switchgear from the normal auxiliary transformer to the startup transformer.
- b. Remove one feedwater pump, one condensate booster pump and one condensate pump from service.
- c. Verify operability of RWM and RSCS.
- d. Transfer recirculation pump to 15 Hz.
- e. Unload and shut down main turbine.

7.6 (0.4 ea.) (2.0)

- a. <20% >15%
- b. <40% >35%
- c. <20% >15%
- d. 35%
- e. <5%

(reference "Below LPSR" site illumination step, after reducing power to 20% step 17)

Ref: WNP-2 GOP 3.2.1, pp. 4, 3, 4, 3 and 11, respectively.

- 7.7 According to the procedure AOP 4.7.4.1 for Loss of Inverter-1(IN-1):
- a. What are two (2) of the three annunciator alarms the operator should see? (1.0)
 - b. What two (2) automatic actions will occur? (1.0)
 - c. If voltage is not normal on US-PP, list the actions that should be taken and in their preferred sequence. (1.5)

- 7.7 a. Annunciator alarm (any 2, 0.5 ea.): (1.0)
- 1) "250 VDC Inverter ALT Source Loss"
 - 2) "250 VDC Inverter ON Alt. Source" ~~LOSS~~
 - 3) "250 VDC Inverter TROUBLE" ~~Alt source loss~~
- b. (0.5 ea.) (1.0)
- 1) Static switch transfers to ALT AC input (MC-7F)
 - 2) DEH system will auto transfer to alternate AC.
- c. (0.4 ea. action; 0.1 ea. sequence) (1.5)
- 1) Attempt to restore IN-1 to service, or
 - 2) Switch IN-1 to "Maintenance" position, or
 - 3) Shift US-PP to bypass source via "KIRK KEY INTERLOCK" (MC-7A).

Ref: WNP-2 AOP 4.7.4.1, pp. 1, 1-2, and 2, respectively.

7.8 With regard to Accident Monitoring Instrumentation, according to Tech. Specs.:

a. Under what operational conditions (use number only) are the following instruments to be operable? (2.0)

- 1) Post-accident Sampling Primary Coolant radiation monitor
- 2) Standby Service Water Flow
- 3) Neutron Flux - IRM
- 4) Safety/Relief Valve Position Indicators.

b. For the instruments in (a) above, what are the required minimum number of operable channels? (1.0)

- 7.8 a. 1) 1, 2, 3 (0.68)
2) 1, 2 (0.44)
3) 1, 2 (0.44)
4) 1, 2 (0.44)

*extra increments on
if ~~the~~ given dividers
by no. of channels
of valve/ans.)*

(2.0)

b. (0.25 ea.)

(1.0)

- 1) 1
- 2) 1/loop
- 3) 1
- 4) 1/valve

Ref: WNP-2, Tech. specs., pp. 3/4 3-71 and 72, 71 and 72, respectively.

*Delete
Questions.*

7.9 According to the WNP-2 Health Physics Program:

- a. For Emergency Exposure Guides, two (2) emergency situations are given with their exposure guideline values. What are these situations and values? (1.0)
- b. An RWP is required when work is to be performed in an area that is posted for airborne radioactivity 5% of MPC (TRUE or FALSE). (0.5)

- 7.9 a. [situation (0.35), value (0.15) ea.] (1.0)
- 1. Life saving - 75 rems; whole body ^(.10) (.05)
 - 2. Protection of public health or property - 25 rems ^(.10) whole body ^(.05) ^(.05) ^(.10)
- b. False (0.5)

Ref: WNP-2, Health Physics Program; 3.1.5 pg.3 and 3.1.8 and pg.2, respectively.

deleted
7.5 b
7.8

kt
- 1.5 = 23.5 new total
- 3.0 = 20.5 new total
RJP

8.0 Administrative Procedures, Conditions, and Limitations (25.0)

- 8.1 a. What are the two (2) LCOs with regard to leakages from unidentified sources? (1.25)
- b. What is the basis behind the unidentified leakage rates? (0.75)

- 8.1 a. 1. 5 gpm total (0.5)
2. 2 gpm increase within any 4-hr period. (0.75)
- b. The crack associated with such leakage would not grow rapidly (would be less than the critical size for rapid propagation). (0.75)

Ref: WNP-2 Tech. Spec., pg. 3/4 4-9 and B 3/4 4-2, respectively.

8.2 During plant shutdown, the maintenance supervisor informs you that on routine checking he found the Division 1 250 volt battery B2-1 discharged, the reason unknown:

a. Do Tech. Specs. require action (YES or NO)? Explain. (1.5)

b. What three battery "parameters" are checked at least once every seven days to verify they meet Category A surveillance requirements? (1.5)

delete

~~(1.5)~~

See p 1, 2, 3, 1
heavy Hot shutdown
question are in typical as cold shutdown
answer is NO (0.5)
applicable to condition 1, 2, 3 (1.0)
if de-energized then cond. 4 or 5 require Div 1 or Div 2
but original answer OK if say condition 3

8.2 a. Yes, (0.5); with less than Div. 1 and/or Div. 2 above required battery or chargers operable, suspend core alterations, handling of irradiated fuel in sec. containment and operations with potential of draining vessel (1.0) (1.5)

see alternate answer

- b. (0.5) each:
- Electrolyte level
 - Float voltage
 - Specific gravity

delete

delete part (b)

Ref: WNP-2 T/S, pg. 3/4 8-15 (and 8-14 respectively)

Correct answer for conditions 4 or 5
NO (0.25) assuming Div 2 operable (1.5) (Why req. Div 1 or Div 2 (1.0))
Yes (0.25) if assume Div 2 inoperable (1.5), then suspend core alterations else (1.0)
Correct answer for condition 3 (Hot shutdown)
Yes (assuming condition 3) (0.5); restore inop. battery to operable status within 2 hrs or less cold shutdown in 24 hrs. (1.0)
Must state "Condition" in answer

8.3 Which of the following occurrences require 1 hour reports to the NRC:

(2.0)

- a. reactor water level -50 inches
- b. reactor water level < -129 inches
- c. site boundary dose > 50 MR/hr whole body
- d. stuck open main steam relief valve

() *not required for each*

8.3 All of the 4 occurrences (since each requires a declaration of an emergency event and this is a category of reportable events).

(2.0)

Ref: CAF.

8.4 With regard to the Fire Brigade:

- a. What is the minimum number of personnel required? (0.5)
- b. Who are specifically excluded from the Fire Brigade? (0.75)
- c. Where and when is the Fire Brigade to be maintained? (0.75)
located - (stated to be at candidates)

8.4 a. 5 (0.5)

b. The Shift Supervisor ^(.1875) the STA ^(.1875) and the 3 members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency. ^(.1875) (0.75)

c. Onsite (0.375) at all times (0.375). (0.75)

Ref: WNP-2 Tech. Spec., pg. 6-1.

to account additional crew take 42-83 weight value

also OT: Sewer Building Lunderoom (.375) up notification by radio or PA (.375)

8.5 According to Tech. Specs., in order for the Fire Suppression Water System to be considered operable, three (3) conditions must be met. State these conditions. (3.0)

- 8.5 a. At least two of the three OPERABLE fire suppression pumps pumping from the circulating water basin, or one diesel-driven pump pumping from the secondary water supply tank, with their discharge aligned to the fire suppression header. (1.0)
- b. Two separate fire water supplies, the recirculating water pump house inlet basin and the secondary water supply tank. (1.0)
- c. An OPERABLE flow path ^(a.s.) capable of taking suction from the circulating water pump house inlet basin and the secondary water supply tank and transferring the water, through distribution piping with OPERABLE sectionalizing, control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device, on each sprinkler or hose standpipe and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE (a.s.) (1.0)

Ref: WNP-2 T.S. pg. 3/4 7-18.

8.6 According to the Limitations stated in the Operating Procedures for the Reactor Core Isolation System (RCIC) (SOP 2.4.6):

- a. What must you do to manually isolate RCIC when system initiation is not sealed in? (0.75)
- b. Under what condition(s) can the auto flow controller (RCIC-FIC-600) be removed from automatic? (0.75)

for extra credit answer divide 375 by 600 & 17 answer to get value/ans.

*(1.25) RCIC turbine steam supply - V-8
 steam ISO & RCIC turbine - V-63
 RCIC steam & RCIC-P-1 - V-91
 at lowest end - V-68*

- 8.6 a. Close the isolation valves using their respective control switches. (0.75)
- b. Only with permission of the Shift Supervisor. (0.75)

Ref: WNP-2 SOP 2.4.6, pg. 2 and 3 of 28, respectively

(a375) will be allowed if answer says "in operator's judgement" if adequate refer to standing orders.

Must Ref. Standing order to use operator judgement.

8.7 With regard to certain shift personnel and their functions:

- Told to state assumptions (ie. Condition 1, 2 & no. of operators) (0.5)
- a. There must be two (2) licensed reactor operators in the Control Room at all times (TRUE or FALSE)? (0.5)
 - b. During new fuel handling operations, a licensed reactor operator must be on the refueling floor (TRUE or FALSE)? (0.5)
 - c. During what modes of operation shall the Shift Technical Advisor be on shift? (0.5)
 - d. If, while at power, the Shift Manager is incapacitated, what action(s) should be taken? (0.75)

- 8.7 a. False (0.5)
- b. False (0.5)
- c. Modes 1, 2 and 3. (0.5)
- d. The CRS^(2.5) or licensed SRO (other than STA)^(2.9) shall assume his duties and immediately advise the Operating Manager^(2.1). (0.75)

Ref: WNP-2 Admin. Proc: 1.3.2, pp. 3 (for a and c) and 2 for (d); 6.2.3, pg. 2 for (b).

8.8 According to Standing Order/Night Orders (Admin. Proc. 1.3.1):

- a. A break occurs in a RCIC line. Would the following control room instruments provide valid indications (Yes or No)? If "NO", how could you verify the necessary information? ~~(1.5)~~

- delete*
- 1) MS-LI 610
 - 2) RHR-FI-603B
 - 3) SLC-LI-601

- b. Following each refueling outage, independent verification of the operable status is required for what types of equipment? (1.0)

- 8.8 a. 1) No (0.2); Use alternate instrument MS-LR-615 (0.3)
2) No (0.2); locally (0.3)
3) No (0.2); locally (0.3) ~~(1.5)~~

- b. Safety related (0.5) and fire protection equipment (0.5) (1.0)

Ref: WNP-2 Admin. Proc. 1.3.1, pp. 7 and 3, respectively.

8.9 According to Monthly Operational Bulletins:

a. What would occur if RHR valves V-8, 9, 6^A and RHR-27A were opened? (1.0) *Called to attention of all SMOs.*

delete b. How did the failure, on two shifts, to check chart movement on wetwell level recorder CMS-LR/RR-4 contribute to loss of wetwell level? ~~(1.0)~~

c. Why is it important to ensure that local temperature indicators at the nitrogen supply shed and in the reactor building are monitored? (1.0)

8.9 a. The reactor vessel would be drained to the suppression pool. (1.0)

delete b. The recorder had been, in fact, inadvertently de-energized so that annunciator alarm switches activated by the recorder pen were also O.O.S. ~~(1.0)~~

c. No control room monitors exist. ^{(0.5) 0.2} If nitrogen temperature gets too low, nitrogen flow onto a 30 in. dia. containment purge header and onto wetwell and drywell purge liner inside containment could cause failure through nitrogen embrittlement. (0.5) (1.0)

Ref: WNP-2 Monthly OP Bull: January 1984, pg. 4; April-May, pg. 1; Feb-Mar, pg. 6, respectively.

8.10 With regard to the Emergency Plan Implementation Procedures, which of the four emergency classes would you place the following:

(2.0)

- 1) An ATWS
- 2) HCTL exceeded
- 3) Volcanic ash fallout severe enough to warrant plant shutdown
- 4) Transport of a contaminated individual offsite

8.10 (0.5 ea.)

(2.0)

- 1) Alert
- 2) Site Area Emergency
- 3) Alert
- 4) Unusual Event

5, para 2 a): any plant condition requiring plant shutdown or a normal controlled shutdown when plant is in a normal controlled shutdown state.

Ref: WNP-2 EPIP: 13.1.1, pp. 6, 20, 7 of 21, respectively.

Facility Review Copy
 Reviewed by: *Stephy J Rejinal*
Stephy J Rejinal

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

Facility: WNP-2
 Reactor Type: BWR-5
 Date Administered: 11/6/84
 Examiner: I. S. Levy
 Candidate: _____

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheet. Points for each question are indicated in parenthesis 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.

<u>Category Value</u>	<u>% of Total</u>	<u>Candidate's Score</u>	<u>% of Cat. Value</u>	<u>Category</u>
<u>25</u>	<u>25</u>	_____	_____	5. Theory of Nuclear Power Plant Operation, Fluids and Thermodynamics
<u>25</u>	<u>25</u>	_____	_____	6. Plant System Design, Control and Instrumentation
<u>25</u>	<u>25</u>	_____	_____	7. Procedures - Normal, Abnormal, Emergency, and Radiological Control
<u>25</u>	<u>25</u>	_____	_____	8. Administrative Procedures, Conditions, and Limitations
<u>100</u>		_____		TOTALS
		Final Grade	<u> </u>	

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

 Candidate's Signature

Revised by:
Steph J Rejnial
STEPHAN J REJNIAL
11-6-84

Questions and Answers to WNP-2 SRO Exam - 11/6/84

5.0 Theory of Nuclear Power Plant Operations, Fluids and Thermodynamics (25.0)

5.1 Give three (3) reasons why fuel densification is a problem. (2.25)

5.1 (any 3; 0.75 each) (2.25)

1. Local power spikes resulting from axial fuel column gaps.
2. Increased linear heat generation rate due to pellet axial shrinkage.
3. Cladding collapse at the location of axial fuel column gaps.
4. Increased stored energy due to decreased pellet-cladding thermal conductance resulting from increased radial gap size.

Ref: Morris Training Center: Thermo/HT/Fluid Flow (3/83), pg. 9-107

5.2 Your latest computer printout of MFLPD and MAPRAT shows the following values for Regions 1 to 3.

Region	1	2	3
MFLPD	0.95	1.0	1.05
MAPRAT	0.92	1.08	1.00

- a. Which, if any, of these values are beyond their safety limits? (1.0)
- b. Why are each of the above limits imposed? (What do they protect against?) (1.5)
- c. Compared to BOL, would the values for MAPRAT at EOL be larger or smaller? Why? (1.5)

- 5.2 a. MFLPD Region 3 (0.5)
MAPRAT Region 2 (0.5)
- b. MFLPD - Maintains <1% cladding strain, fuel failure. (0.75)
MAPRAT - Maintains <2200°F following LOCA, decay heat removal (0.75)
- c. Larger (0.5); MAPLHGR limit decreases (0.5) since local peaking factor gets smaller as, with exposure, heat transfer is reduced (0.5). (1.5)

Ref: WNP-2 Systems and Procedures, PC, Pg. 16; and MTC Thermo/HT/FF (3/83), pg. 9-66 through 9-76.

- 5.3 In reference to the reactor water cleanup regenerative heat exchanger, assume the following conditions and, then, perform the calculations.

Conditions:

T inlet from reactor (tube side) = 550°F
 T outlet from H_x (tube side) = 250°F
 T inlet shell side = 120°F
 Tubeside flow rate from reactor = 1300 gal/min
 Shellside flow rate to reactor = 1300 gal/min

Calculate (and show all work):

- a. The amount of heat transfer (Btu/hr) on the tube side. (1.0)
 b. The temperature of the water going back to the reactor. (1.0)
 c. The maximum temperature of water going back to the reactor if the flow back to the reactor were decreased from 1,300 gpm to 800 gpm due to a leak in the tube side of the H_x. (1.0)

- 5.3 a. $\dot{Q} = \dot{M} \times C_p \times \Delta T$
 $= 1300 \text{ gal/min} \times 60 \text{ min/hr} \times 8.33 \text{ lbm/gal}$
 $\times 1 \text{ Btu/lbm}^\circ\text{F} \times (550-250)$
 $= 1.949 \times 10^8 \text{ Btu/hr.}$ (1.0)
- b. 420° (conservation of energy) (1.0)
- c. 550°F as follows:
 $\dot{Q} = \dot{M} \times C_p \Delta T$
 $T_{\text{max}} = \dot{Q}/\dot{M} + 120$
 $= \frac{1.949 \times 10^8 \text{ Btu/hr}}{800 \text{ gal/hr}} \times \frac{1 \text{ hr}}{60 \text{ min}} \times \frac{1 \text{ gal}}{8.33 \text{ lbm}} \times \frac{1 \text{ }^\circ\text{F}}{1 \text{ Btu/lbm}}$
 $= 607.5^\circ\text{F}$ (1.0)

but cannot be greater than 550°F (max. T from reactor to inlet)

Ref: Morris, T. C.; Thermo/HT/Fluid Flow (3/83), pgs. 8-40,41.

5.4 While at 75% power, the master feedwater controller fails low. Will the NPSH of the recirculation pumps increase, decrease, or remain unchanged? Briefly, explain why. (1.25)

5.4 Decrease (0.5); reduced subcooling (0.75) (1.25)

Ref: Morris, T. C.; Thermo/HT/Fluid flow (3/83), pg. 7-96.

Reviewed by
Steph Reynolds

5.5 With regard to excess reactivity:

- a. The excess reactivity for a cold, clean, critical reactor is greater than that for the hot, clean, zero power condition (TRUE or FALSE). (0.5)
- b. The excess reactivity for the hot, clean, zero power condition is smaller than that for the hot, full power, equilibrium Xe and Sm condition (TRUE or FALSE). (0.5)
- c. The excess reactivity at EOL is greater than that at BOL (TRUE or FALSE). Why? (1.25)

- 5.5 a. True (0.5)
- b. False (0.5)
- c. False (0.5); fuel depletion and fission products increase (0.75) (1.25)

Ref: WNP-2, Reactor Theory Rev. pg. 39.

Revised by:
JFK (Ramp)
at

5.6 The WNP-2 reactor is taken to criticality from a cold condition and then placed on an 80 second positive period.

a. From control room nuclear instrumentation, how can the operator tell when the heating range has been reached? (Rod position and recirculation are held constant). (0.75)

b. In which of the following intervals was the heating range entered? Explain the reason for your answer. (Show all work.) (1.5)

Interval 1 - reactor power increased by a factor of 6 in 143.3 seconds.

Interval 2 - reactor power increased by a factor of 3 in 99.0 seconds

Interval 3 - reactor power increased by a factor of 5 in 128.8 seconds.

(Note: the intervals may not be in sequence.)

5.6 a. Operator can notice that period has become longer and that power change on IRMs is leveling off.. (0.75)

b. (From $P = P_0 e^{t/T} \rightarrow T = \frac{t}{\ln P/P_0}$) Interval 2 (0.5); the period has lengthened from 80 seconds. The other intervals have 80 second periods (1.0). (1.5)

Ref: General control room indications; WNP-2, Reactor Theory, pg. 58.

Reviewed by:
Step 1/2/3/4/5/6/7/8/9/10

5.7 With regard to Reactivity Coefficients:

- a. Which reactivity coefficient is the most dominant under the following conditions: (2.0)
- 1) During rod drop accident at 15% power
 - 2) Pulling rods at 1% power
 - 3) MSIV closure at 100% power
 - 4) Feedwater controller fails high at 100% power?
- b. For "feedwater controller fails high at 100% power," a.4) above:
- 1) Give the reason for your answer to a.4) above. (1.0)
 - 2) What will happen to power (increase, decrease, stay the same)? (0.5)
 - 3) What is the approximate value at BOL of the coefficient you gave as your answer to a.4) above? (0.5)

5.7 a. (0.5 for each) (2.0)

- 1) Doppler coefficient
 - 2) Moderator coefficient
 - 3) Void coefficient
 - 4) Void coefficient
- b. 1) Increase in core subcooling which reduces void fraction whose coefficient has greatest effect on reactivity. (1.0)
- 2) Increase (0.5)
 - 3) $-1 \times 10^{-3} \Delta K/K/1\%$ void change (0.5)

Ref: Standard Reactor Theory.

• if the candidate gives the wrong coeff. in a.4) but gives correct value credit should be given (Error carried forward)
i.e. a.4) Moderator coeff
b.2) $-1 \times 10^{-4} \Delta K/K$

Reviewed by
Sgt. K. J. [unclear]

5.8 Near the end-of-cycle (EOC), will differential control rod worths near the bottom of the core be lower or higher than those near the top of the core? Why?

(1.75)

5.8 Higher (0.5). As fuel burns, control rods must be withdrawn causing flux to peak lower in the core (0.75); and rod worth is proportional to flux² (0.5). (MINOR COMMENT) should be written $\left(\frac{\phi_{local}}{\phi_{ave}}\right)^2$

(1.75)

Ref: WNP-2, Reactor Theory, pg. 81.

*Revised by
Steph H. Quinn*

5.9 With regard to the effects of equilibrium Xenon and Samarium (the reactor has been operated at a constant power for many days):

a. If reactor power is then doubled, will the new equilibrium Samarium concentration be exactly twice as great (YES or NO)? Explain. (1.5)

b. If the reactor is shut down, initially by a 1% $\Delta K/K$, will the initial effect of Xenon be to increase or decrease the shutdown margin? (0.5)

5.9 a. No (0.5); the equilibrium value of samarium does not depend on flux, and, therefore, it does not depend on power level (1.0). (1.5)

Ref: WNP-2 Reactor Theory, pg. 87.

b. Increase. (0.5)

Ref: Standard Reactor Theory

5.10 With regard to Delayed Neutrons:

- Revised: Dept. Manual*
- a. In causing fissions, what is the major difference between delayed neutrons and prompt neutrons? (0.75)
 - b. Explain how and why the value of the delayed neutron fraction, Beta, changes from the beginning of core life to the end of core life. (0.75)
 - c. Explain the effect on reactor control of the change in Beta with core life. (0.75)

5.10 a. Delayed neutrons have a lower probability of causing fast fission (the "importance" factor is less than 1). (0.75)

Ref: Standard Reactor Theory.

b. Beta will decrease from about 0.007 at BOL to 0.0054 at EOL due to buildup of Pu-239 and depletion of U-235. (0.75)

0.0056 Ref. Revised LWR-2 React Theory Review

Ref: Standard Reactor Theory.

c. As beta decreases with core age, reactor period decreases and, therefore, for the same reactivity addition rate, a shorter period and less easy control is obtained at EOL. (0.75)

Ref: Standard Reactor Theory

$$P = P_0 e^{\tau/\tau}$$

$$1Ci = 3.7 \times 10^{10} Bq$$

$$\alpha_D = -1 \times 10^{-5} \frac{\Delta K/Z \cdot F}{K}$$

$$\alpha_V = -1 \times 10^{-3} \frac{\Delta K/Z}{K} \text{ voids}$$

$$\alpha_{T1} = -4.5 \times 10^{-4} \frac{\Delta K/Z \cdot F}{K}$$

$$\alpha_P = -4.5 \times 10^{-4} \frac{\Delta K/Z \text{ power}}{K}$$

$$I(\tau) = I_0 e^{-\lambda \tau}$$

$$T_{1/2} = \ln(2)/\lambda$$

$$C_p = (C_{p_{base}}) (K_s) (K_A)$$

$$Q = MC_p \Delta t$$

$$\Delta \rho = \frac{f L \rho v^2}{D 2g_c}$$

$$f = 64/Re$$

$$\rho = \frac{k(\text{eff}) - 1}{K(\text{eff})}$$

$$\frac{1}{M} = \frac{CR1}{CR2} = \frac{1 - K(\text{eff})^2}{1 - K(\text{eff})}$$

$$M = \frac{CR2}{CR1} = \frac{1 - K(\text{eff})}{1 - K(\text{eff})^2}$$

$$Q = M \Delta h$$

$$Q = U A \Delta T$$

$$M = 1/(1-k)$$

$$N(\tau) = N_0 e^{-\lambda \tau}$$

$$\alpha_{r-} = (L_f + L_s) \frac{(\phi_{rod})^2}{(\phi_{avg})^2}$$

$$n = v/(1+d)$$

$$P = \Sigma \phi v / (3.7 \times 10^{10})$$

$$\tau = (\beta - \rho) / \lambda_0$$

$$\dot{\tau} = \bar{L}/\rho + (\beta - \rho) / \lambda_0$$

$$\tau = L/(\rho - \beta)$$

$$v = v_f + x v_{fg}$$

$$H = x h_g + (1-x) h_f$$

$$S = x S_g + (1-x) S_f$$

$$1 \text{ in} = 2.54 \text{ cm}$$

$$1 \text{ gal.} = 3.785 \text{ liters}$$

$$1 \text{ kg} = 2.205 \text{ lb}$$

$$N = \rho A_0 / A$$

$$17.58 \text{ watts} = 1 \text{ BTU/min}$$

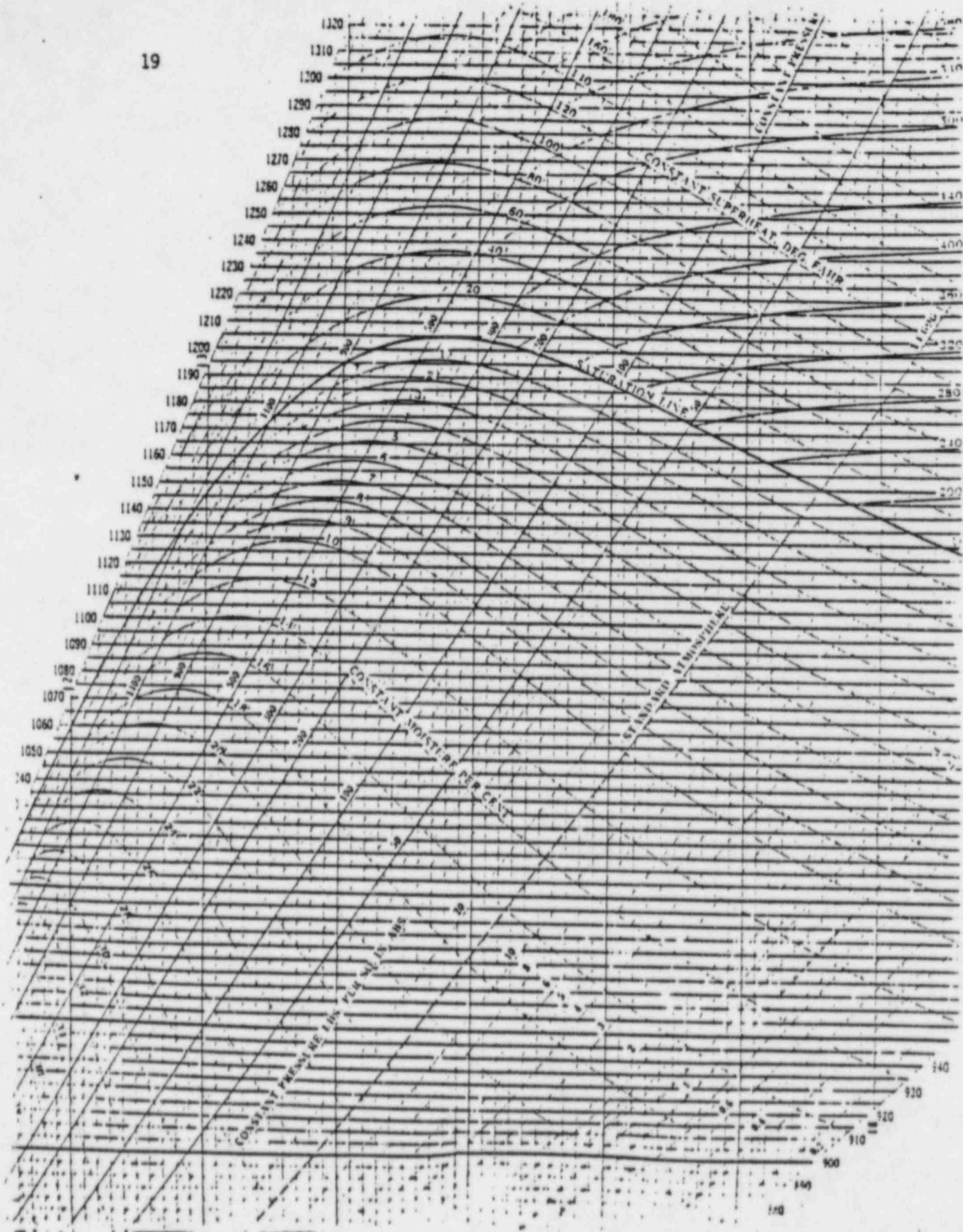
$$1 \text{ psi} = 6.895 \text{ Pa}$$

$$1 \text{ psi} = 2.036 \text{ " H}_2\text{O (}@ 0\text{C)}$$

$$1 \text{ psi} = 27.68 \text{ " H}_2\text{O (}@ 4\text{C)}$$

$$\bar{\beta} = .0071$$

$$\bar{L} = 2 \times 10^{-5} \text{ sec}$$



LANGSDON - COMMENTS

6.0 Plant System Design, Control and Instrumentation (25.0)

6.1 With regard to the Vessel Instrumentation:

- ✓ a. For what specific condition(s) is the Fuel Zone Range level indicators calibrated? (1.0)
- ✓ b. For the Narrow Range Level Indicators, MS-LIS-24A,B,C & D: (2.0)
- 1) Where are they located in the plant?
 - 2) To what systems/components do they send signals?
 - 3) At what levels are these signals initiated?
 - 4) What system/component action is initiated?
- ✓ c. How is the signal for jet pump flow generated in an individual (non-calibrated) jet pump? (1.0)
- 6.1 ✓ a. Saturated steam conditions at 0 psig in the vessel and drywell with no jet pump flow. (1.0)
- ✓ b. 1) Located in Reactor Building, 522' level. (0.5)
- 2), 3), and 4):
- RCIC/RCIC turbine; at level 8; shut steam supply (0.75)
 - RPS; level 3; scram and NSSSS Group 5 and 6 isolations (0.75)
- ✓ c. Pressure, at pressure tap on pump throat, is compared to a common jet pump discharge pressure sensed at the core inlet plenum area by the SLCS injection line to develop a ΔP signal. *or will description of same* (1.0)
- Ref: WNP-2, Systems and Procedures: NBI, pgs. 5, 14 and 9, respectively.

6.2 With regard to Process Radiation Monitoring:

- ✓ a. The main stream line radiation monitoring system will initiate condenser isolation (via AR-V-1) when appropriate channels and their trips are activated. What are these channels and trips? (1.0)
- ✓ b. What type of detector (G-M, etc.) is used in the following radiation monitors: (1.5)
- 1) Ry Building stack extended range
 - 2) Off-gas post treatment
 - 3) Circulating water effluent?
- ✓ c. Which of the monitors, in b. above, will initiate system isolation or trips? (1.0)

throw out - not relevant

- 6.2 ✓ a. Channel A or C; HI HI or Inop. (1.0)
- ✓ b. 1) Beta scintillation (0.5)
2) GM (0.5)
3) Gamma scintillation (0.5)
- ✓ c. (0.5 each): 2,3 (1.0)

see above

Ref: WNP-2, Systems and Procedures: PRM, pg. 7; 5, 3, 3; and 3, respectively.

6.3 With regard to the Emergency Diesel Generators:

- ✓ a. What are all the start signals for DG #2? (1.5)
- too detailed
throw out* b. Describe the overspeed trip mechanism (including its location, its major components, and how it initiates an engine trip). (2.0)
- ✓ c. What is the purpose of the soak back pump? (0.75)

- 6.3 ✓ a. 1) Bus SM-3 dead (0.3) or TR-S undervoltage (0.3)
2) TR-B undervoltage (0.3)
3) SM-8 undervoltage (0.3)
4) K98B (containment high pressure or reactor low water level - Level 1) (0.3) (1.5)
- not refer* b. Located in the camshaft counterweight housing (0.5).; it consists of a flyweight held by a tension spring. When engine exceeds set limit, spring tension is overcome by centrifugal force acting on the flyweight, permitting an outward movement of the flyweight which actuates the trip lever. This causes latching the injector rocker arm in the depressed position, preventing fuel injection into the engine (1.5). (2.0)
- see above* ✓ c. To provide oil to the turbocharger for prelubrication. (0.75)
- This has been deleted SFP Sept 89 revision*

Ref: WNP-2, Systems and Procedures: DG, pg. 62; Fig. 14; and Fig 8c and pg. 22, respectively.

6.4 With respect to the Automatic Depressurization System (ADS):

- ✓ a. What ADS controls are located on control room backpanels P628 and 631, and which components do they control? (1.0)
- ✓ b. Under what normal condition(s) will pushing the four ADS main initiation pushbuttons not result in ADS initiation? What is the reason for this? (1.5)

this word is misleading - would never push ADS pushbuttons under normal conditions - may have confused some people.

- 6.4 va. (14) AUTO/OPEN control switches; SRVs "A" and "B" solenoids. *⇒ All 7 ADS valves* (1.0)
- ✓ b. If a low pressure ECCS pump is not operating (0.75); so that core will not be uncovered (0.75) (1.5)

Ref: WNP-2 System and Procedures, ADS, pg. 13 and 8, respectively.

6.5 With regard to the AC Electrical Distribution System:

- ✓ a. Which lockout relay will be tripped upon a Transformer Differential Current (87TM)? (0.5)
- ✓ b. List three (3) actions which will occur when the lockout relay in (a) is tripped? (1.5)
- ✓ c. The loss of which 480V MC bus will deactivate both loops of shutdown cooling? (0.5)
- ✓ d. What happens upon the loss of the normal and startup sources for SM-4? (0.75)

Throw out - too much detail - candidate not required to know all power supplies by memory

6.5 ✓ a. Unit lockout (86XU) ~~also accept actions initiated by this (same reference) S&P p. 15~~ (0.5)

✓ b. [any 3, 0.5 pts each] (1.5)

- Trips and locks-out all "N" breakers
- Trips 4F circuit breaker (Generator Exciter Field Breaker)
- Trips main turbine (20 AST)
- De-energizes 86XIU → also accept actions initiated by this (same reference) S&P p. 15
- Starts oscillograph
- Starts computer

HIF

c. MC-8-B-A *see above* (0.5)

✓ d. Automatic transfer of SM-4 to the Division 3 EDG (0.75)

Ref: WNP-2 System and Procedures; AC Distr.; pg. 15 and 16 for a, b, d; AOP 4.7.1.9, Loss of Power to SM-8, pg. 3 of 4 for c.

Also accept those activated by 86XIU since they will also occur. Same ref.

- ✓ 6.6 For the Reactor Core Isolation Cooling System, what position should the RCIC-V-8 and RCIC-V-63 keylocked control switches be placed in prior to resetting any isolation signal? Why? (1.5)

reset

- ✓ 6.6 CLOSED or STOP (0.5); otherwise (i.e. in OPEN), when isolation signal is received the valves would immediately open (0.5) which could cause extensive damage to system piping and components (0.5). (1.5)

Ref: WNP-2, SOP 2.4.6, pg. 2 of 28.

6.7 If a selected control rod is in the selected 4 rod group and too many reed switch closures occur:

- ✓ a. What indication(s) would the operator observe? (1.0)
- ✓ b. How could the control rod be moved through this position? (0.75)

- 6.7 a. RPIS Data Fault, Insert Block would light. (1.0)
- ✓ b. By using Substitute Data on RSCS or *bypassing on RSCS* (0.75)

Ref: WNP-2 Systems and Procedures, RMCS, pp. 15, 16.

||

RSCS p. 3

↳ also may say to Bypass Rwm - see Rwm p. 21

↳ also withdraw Block, Rwm Block (possibly - not analyzed)
same ref. as above

6.8 What design provisions ^{implies > 1} are made to assure air flow through Standby Gas Treatment System filters. What is the basis for this?

(1.75)

6.8 Each SGT train has two full-capacity fans powered from separate emergency buses (0.75); to prevent filters from igniting on loss of air flow due to the decay heat generated by entrained radioactive materials (1.0)

(1.75)

Ref: WNP-2 Systems and Procedures, SGT pg. 24.

Would also likely get answer of heaters on inlet to keep moisture from causing filter blocking, and demister. See S&P pg 3

6.9 With regard to the Average Power Range Monitor System (APRM):

- ✓ a. What three (3) things could happen if the back panel APRM Mode Switch was placed into the standby position during normal operation? (1.0)
- ✓ b. What is the AC power source for APRM Channel F? (0.5)
- ✓ c. Why is the Thermal Power Monitor necessary? (1.0)

not true by definition, not a valid answer; accept only 2 other answers
Tech spec. of operability

- 6.9 ✓ a. APRM operates normally if bypassed (0.33); gives an INOP trip to warn operator if channel has not been bypassed (0.34). Will give 1/2 scram if channel not bypassed (0.33). *or Inop Inhibit switch depressed* (1.0)
59P APRM P. 29
- ✓ b. (Bus B supplied by) RPS MG set B. (0.5)
- ✓ c. To avoid unnecessary scrams during power increase transients (due to flux leading thermal power). (1.0)

Ref: WNP-2 Systems and Procedures, APRM, pp. 26, 28 and 32-33, respectively.

7.0 Procedures - Normal, Abnormal, Emergency, and Radiological Control (25.0)

7.1 According to procedures for Emergency RPV Depressurization (Contingency) (5.3.2):

- a. What is the primary system used to cause depressurization? (0.5)
- b. If the system in (a) is partially or totally unavailable, what is the next system to be used? (0.75)
- c. Under what conditions would systems other than those in a) and b) above be used and what are these systems? (2.5)

- 7.1 a. ADS (0.5)
- b. Other SRV's until 7 are open (0.75)
- c. If <3 SRV's are open (0.5); main condenser, main steam line drains, RCIC, head vents (2.0) (2.5)

Ref: WNP-2 EOP 5.3.2, pp. 1, 1, and 2 of 8, respectively.

7.2 What changes to the reactor power, MCPR, and MAPLHGR operating limits are required before one (1) recirculation loop operation is permitted? (2.25)

7.2 (0.75 each) (2.25)

Reactor power: reduce it to <50% rated thermal power

MCPR: increase it by 0.01

MAPLHGR: reduce limit to 0.84 times the two loop op. limit.

Ref: WNP-2 AOP 4.2.1.10, pg. 2 of 3, and T.S. 3.4.1.1, pg. 3/4 4-1.

ANSWER SHOULD BE NONE. ALL ACTIONS
ARE REQUIRED AFTER THE PLANT IS IN
SINGLE LOOP

7.3 With regard to the operating procedure for 250V DC Distribution System (SOP 2.7.7):

- a. What should the operator do and where would he do it, to tie the 250V DC battery B2-1 to the 250V DC distribution bus S2-1? (2.0)
- b. When bus S2-1 is energized:
- 1) What indication(s) does the operator have that no ground is present? (0.75)
- 2) What indication(s) does the operator have when a ground develops? (0.75)

- 7.3 a. 1) Verify open, on DP-S2-1, (0.34)
(a) breaker S2-1/4B (0.33)
(b) breaker S2-1/4C (0.33)
(c) breaker S2-1/4D. (0.33)
- 2) Close, on DP-S2-1, (0.34) breaker S2-1/3C and verify (0.33). (2.0)
- b. 1) Two ground lamps on board "C" are lit. (0.75)
- 2) The lamp connected to the grounded polarity goes out and the appropriate bus ground annunciator comes on. (0.75)

Ref: WNP-2 SOP 2.7.7, pg. 2 of 5.

ANSWER SHOULD BE GENERALIZED TO
SIMPLY SAY 1. STRIP BUS, ENERGIZE
BUS, RELOAD BUS. THE PRESENT
ANSWER REQUIRES MEMORIZATION OF
OPERATING PROCEDURES.

7.4 According to WNP-2 Annunciator Procedures, if the "OFF GAS VAULT RAD MONITOR DNSCL" annunciator alarmed, the operator should make a "verification" and a "check". Describe the verification and check to be made and at what location in the plant. (2.0)

7.4 Verify: downscale condition of OG-RIS-11 (0.5); at P606 (0.5) (1.0)
Check: radiation level of OG-RI-11 (0.5); locally (0.5) (1.0)
Ref: WNP-2 Annunciator Procedures, 4.02.A5-4.2.

THERE ARE MORE THAN 1400 ANNUNCIATOR PROCEDURES. 2 POINTS IS TOO GREAT OF IMPORTANCE TO PLACE ON ONE OF THEM. PLEASE CONSIDER REDUCING POINT VALUE.

7.5 According to the Operating Instructions for the Control Rod Drive System (SOP 2.1.1):

a. During system startup, what should be the position (open or closed) for the following Hydraulic Control Unit (HCU) valves: (2.0)

- 1) CRD-V-111 (Cartridge Valve Nitrogen Inlet)
- 2) CRD-V-107 (Accumulator Water Drain)
- 3) CRD-V-102 (Withdraw Water Isolation)
- 4) Withdraw Line "Dragon" valve.

b. During system startup, in what position should the operator place the charging water header vent valve (CRD-V-65) that is located between the charging water header isolation valve (CRD-V-34) and the HCU charging water inlet isolation valves (CRD-V-113/HCU)? Why? (1.5)

→ REQUIRES TOTAL RECALL OF OPERATING PROCEDURES. ALSO THIS EVOLUTION IS ONLY FOR STARTUP AFTER COMPLETE DRAIN DOWN, AND DONE VERY SELDOM

7.5 a. (0.5 ea.) (2.0)

- 1) Open
- 2) Closed
- 3) Open
- 4) Closed

b. Open (0.5); to provide leak off (0.5) so that HCU accumulators will not pressurize due to valve leakage (0.5) (1.5)

Ref: WNP-2 SOP 2.1.1, pp. 4 and 6 of 41, respectively.

7.6 According to the procedures for Normal Shutdown To Cold Shutdown (G.O.P. 3.2.1), at what percent power (approximate) should the following actions be performed? (2.0)

- a. Transfer the 6900 volt switchgear from the normal auxiliary transformer to the startup transformer.
- b. Remove one feedwater pump, one condensate booster pump and one condensate pump from service.
- c. Verify operability of RWM and RSCS.
- d. Transfer recirculation pump to 15 Hz.
- e. Unload and shut down main turbine.

→ PROCEDURE IS WRONG - TECH SPECS
REQUIRES RWM - RSCS TO BE OPERABLE
PRIOR TO ~~STOPPING~~ DECREASING BELOW
20%

7.6 (0.4 ea.) (2.0)

- a. <20% >15%
- b. <40% >35%
- c. <20% >15%
- d. 35%
- e. <5%

Ref: WNP-2 GOP 3.2.1, pp. 4, 3, 4, 3 and 11, respectively.

7.7 According to the procedure AOP 4.7.4.1 for Loss of Inverter-1(IN-1):

- a. What are two (2) of the three annunciator alarms the operator should see? (1.0)
- b. What two (2) automatic actions will occur? (1.0)
- c. If voltage is not normal on US-PP, list the actions that should be taken and in their preferred sequence. (1.5)

- 7.7 a. Annunciator alarm (any 2, 0.5 ea.): (1.0)
 - 1) "250 VDC Inverter ALT Source Loss"
 - 2) "250 VDC Inverter ON Alt Source Loss"
 - 3) "250 VDC Inverter TROUBLE Alt source Loss"
- b. (0.5 ea.) (1.0)
 - 1) Static switch transfers to ALT AC input (MC-7F)
 - 2) DEH system will auto transfer to alternate AC.
- c. (0.4 ea. action; 0.1 ea. sequence) (1.5)
 - 1) Attempt to restore IN-1 to service, or
 - 2) Switch IN-1 to "Maintenance" position, or
 - 3) Shift US-PP to bypass source via "KIRK KEY INTERLOCK" (MC-7A).

Ref: WNP-2 AOP 4.7.4.1, pp. 1, 1-2, and 2, respectively.

7.8 With regard to Accident Monitoring Instrumentation, according to Tech. Specs.:

a. Under what operational conditions (use number only) are the following instruments to be operable? (2.0)

- 1) Post-accident Sampling Primary Coolant radiation monitor
- 2) Standby Service Water Flow
- 3) Neutron Flux - IRM
- 4) Safety/Relief Valve Position Indicators.

b. For the instruments in (a) above, what are the required minimum number of operable channels? (1.0)

7.8 a. 1) 1, 2, 3 (0.68)
2) 1, 2 (0.44)
3) 1, 2 (0.44)
4) 1, 2 (0.44) (2.0)

b. (0.25 ea.) (1.0)

- 1) 1
- 2) 1/loop
- 3) 1
- 4) 1/valve

Ref: WNP-2, Tech. specs., pp. 3/4 3-71 and 72, 71 and 72, respectively.

QUESTION REQUIRES TOTAL RECALL
DOUBTFUL ANYONE WILL GET RIGHT
PLEASE RECONSIDER POINT VALUE.

7.9 According to the WNP-2 Health Physics Program:

a. For Emergency Exposure Guides, two (2) emergency situations are given with their exposure guideline values. What are these situations and values? (1.0)

b. An RWP is required when work is to be performed in an area that is posted for airborne radioactivity 5% of MPC (TRUE or FALSE). (0.5)

7.9 a. [situation (0.35), value (0.15) ea.] (1.0)

1. Life saving - 75 rems whole body
2. Protection of public health or property - 25 rems whole body.

b. False (0.5)

Ref: WNP-2, Health Physics Program; 3.1.5 pg.3 and 3.1.8 and pg.2, respectively.

8.0 Administrative Procedures, Conditions, and Limitations (25.0)

- 8.1 a. What are the two (2) LCOs with regard to leakages from unidentified sources? (1.25)
- b. What is the basis behind the unidentified leakage rates? (0.75)

- 8.1 a. 1. 5 gpm total (0.5)
2. 2 gpm increase within any 4-hr period. (0.75)
- b. The crack associated with such leakage would not grow rapidly (would be less than the critical size for rapid propagation). (0.75)

Ref: WNP-2 Tech. Spec., pg. 3/4 4-9 and B 3/4 4-2, respectively.

8.2 During plant shutdown, the maintenance supervisor informs you that on routine checking he found the Division 1 250 volt battery B2-1 discharged, the reason unknown:

a. Do Tech. Specs. required action (YES or NO)? Explain. (1.5)

b. What three battery "parameters" are checked at least once every seven days to verify they meet Category A surveillance requirements? (1.5)

• We have never held operators responsible for ^{specific} content of surveillances which they do not actively participate in. This question should be deleted or much flexibility allowed for in the candidates' answer.

8.2 a. Yes, (0.5); with less than Div. 1 and/or Div. 2 above required battery or chargers operable, suspend core alterations, handling of irradiated fuel in sec. containment and operations with potential of draining vessel (1.0). (1.5)

b. (0.5) each:

- Electrolyte level
- Float voltage
- Specific gravity

(1.5)

Ref: WNP-2 T/S, pg. 3/4 8-11 and 8-14 respectively.

ANSWER "a" gives only the ACTION STATEMENT for the LCO. Credit should also be given for an explanation of why the candidate believes his ANSWER to be AS he indicated. If you wanted the ACTION STATEMENT, it should have been ASKED for.³⁰

8.3 Which of the following occurrences require 1 hour reports to the NRC:

(2.0)

- a. reactor water level -50 inches
- b. reactor water level < -129 inches
- c. site boundary dose > 50 MR/hr whole body
- d. stuck open main steam relief valve

NOT READ AS PART OF ANSWER!

8.3 All of the 4 occurrences (since each requires a declaration of an emergency event and this is a category of reportable events).

(2.0)

Ref: CAF.

declaration of an emergency may or may not occur depending upon the amount of time that each item exists (ie, if a SRV can be shut quickly, why declare a UE immediately?).

W

8.4 With regard to the Fire Brigade:

- a. What is the minimum number of personnel required? (0.5)
- b. Who are specifically excluded from the Fire Brigade? (0.75)
- c. Where and when is the Fire Brigade to be maintained? (0.75)

8.4 a. 5 (0.5)

b. The Shift Supervisor, the STA and the 3 members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency. (0.75)

c. Onsite (0.375) at all times (0.375). (0.75)

Ref: WNP-2 Tech. Spec., pg. 6-1.

8.5 According to Tech. Specs., in order for the Fire Suppression Water System to be considered operable, three (3) conditions must be met. State these conditions. (3.0)

- 8.5 a. At least two of the three OPERABLE fire suppression pumps pumping from the circulating water basin, or one diesel-driven pump pumping from the secondary water supply tank, with their discharge aligned to the fire suppression header. (1.0)
- b. Two separate fire water supplies, the recirculating water pump house inlet basin and the secondary water supply tank. (1.0)
- c. An OPERABLE flow path capable of taking suction from the circulating water pump house inlet basin and the secondary water supply tank and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE. (1.0)

Ref: WNP-2 T.S. pg. 3/4 7-18.

8.6 According to the Limitations stated in the Operating Procedures for the Reactor Core Isolation System (RCIC) (SOP 2.4.6):

- a. What must you do to manually isolate RCIC when system initiation is not sealed in? (0.75)
- b. Under what condition(s) can the auto flow controller (RCIC-FIC-600) be removed from automatic? (0.75)

- 8.6 a. Close the isolation valves using their respective control switches. (0.75)
- b. Only with permission of the Shift Supervisor. (0.75)

Ref: WNP-2 SOP 2.4.6, pg. 2 and 3 of 28, respectively

0.375 will be allowed if answer given says "in or judgment" ~~in or judgment~~ in any reference standing orders. (entry should be to master) ISL

- 8.7 With regard to certain shift personnel and their functions:
- a. There must be two (2) licensed reactor operators in the Control Room at all times (TRUE or FALSE)? (0.5)
 - b. During new fuel handling operations, a licensed reactor operator must be on the refueling floor (TRUE or FALSE)? (0.5)
 - c. During what modes of operation shall the Shift Technical Advisor be on shift? (0.5)
 - d. If, while at power, the Shift Manager is incapacitated, what action(s) should be taken? (0.75)

- 8.7 a. False (0.5)
- b. False (0.5)
- c. Modes 1, 2 and 3. (0.5)
- d. The CRS or licensed SRO (other than STA) shall assume his duties and immediately advise the Operating Manager. (0.75)

Ref: WNP-2 Admin. Proc: 1.3.2, pp. 3 (for a and c) and 2 for (d); 6.2.3, pg. 2 for (b).

8.8 According to Standing Order/Night Orders (Admin. Proc. 1.3.1):

a. A break occurs in a RCIC line. Would the following control room instruments provide valid indications (Yes or No)? If "NO", how could you verify the necessary information? (1.5)

- 1) MS-LI 610
- 2) RHR-FI-603B
- 3) SLC-LI-601

b. Following each refueling outage, independent verification of the operable status is required for what types of equipment? (1.0)

→ PERSONNEL ARE NOT EXPECTED TO MEMORIZE THE LIST OF UNQUALIFIED ^(OR ALTERNATE) INSTRUMENTS AS REFERENCED IN APP. 1.3.1. THE WHOLE PURPOSE OF PUTTING THE APPENDIX IN THE PPM WAS TO ALLOW IT TO BE REFERENCED, IF REQUIRED! INSTRUMENTS ARE LABELED TO REFLECT THIS STATUS.

Also the instruments may give a false reading, depends on the nature of the accident.

8.8 a. 1) No (0.2); Use alternate instrument MS-LR-615 (0.3)
2) No (0.2); locally (0.3)
3) No (0.2); locally (0.3) (1.5)

b. Safety related (0.5) and fire protection equipment (0.5) (1.0)

Ref: WNP-2 Admin. Proc. 1.3.1, pp. 7 and 3, respectively.

8.9 According to Monthly Operational Bulletins:

- a. What would occur if RHR valves V-8, 9, ~~6B~~^A and RHR-27A were opened? (1.0)
- b. How did the failure, on two shifts, to check chart movement on wetwell level recorder CMS-LR/RR-4 contribute to loss of wetwell level? (1.0)
- c. Why is it important to ensure that local temperature indicators at the nitrogen supply shed and in the reactor building are monitored? (1.0)

- 8.9
- a. The reactor vessel would be drained to the suppression pool. (1.0)
 - b. The recorder had been, in fact, inadvertently de-energized so that annunciator alarm switches activated by the recorder pen were also O.O.S. (1.0)
 - c. No control room monitors exist. If nitrogen temperature gets too low, nitrogen flow onto a 30 in. dia. containment purge header and onto wetwell and drywell purge liner inside containment could cause failure through nitrogen embrittlement. (1.0)

Ref: WNP-2 Monthly OP Bull: January 1984, pg. 4; April-May, pg. 1; Feb-Mar, pg. 6, respectively.

This is AN UNFAIR QUESTION! INITIAL TRAINING does NOT NECESSARILY INCORPORATE MOB TRAINING INTO IT. WITHOUT THE BENEFIT OF READING THE MOB THE CHANCES OF CANDIDATES GETTING THE CORRECT ANSWER IS MINIMAL! Additionally, PART a & c ARE MORE OF A SYSTEM ORIENTED QUESTION THAN ADMIN, ETC., AND SHOULD NOT BE PART OF CATEGORY 8.

8.10 With regard to the Emergency Plan Implementation Procedures, which of the four emergency classes would you place the following:

(2.0)

- 1) An ATWS
- 2) HCTL exceeded
- 3) Volcanic ash fallout severe enough to warrant plant shutdown
- 4) Transport of a contaminated individual offsite

8.10 (0.5 ea.)

(2.0)

- 1) Alert
- 2) Site Area Emergency
- 3) Alert
- 4) Unusual Event

Ref: WNP-2 EPIP: 13.1.1, pp. 6, 20, 7 of 21, respectively.

Memorization of EPIP 13.1.1 is NOT feasible. While the majority of the symptomatic conditions which call for event classification are well known, the situational based events are much more vague. I cannot imagine any ~~operator~~ operator declaring an emergency event without reference to EOP's or EPIP's for guidance.

8.11 With regard to Control Of Plant Operating Keys:

- a. What is the limitation(s) with regard to issuing of bypass and interlock keys not required for normal operation? (0.75)
- b. Besides the bypass and interlock keys, there are two (2) other kinds of keys: (1.0)
- 1) What are they?
 - 2) Where are they each stored?
 - 3) Who is responsible for them?

- 8.11 a. Issued to requesting party only with permission of Shift Manager/CRS. (0.75)
- b. (0.17 each for key type, location, responsible person.) (1.0)
- Control room panel keys - key cabinet in CR - Shift Manager/CRS.
 - Miscellaneous keys - key cabinet in Radwaste CR - Shift Support ~~Manager~~ ^{SUPERVISOR}

Ref: WNP-2, admin. Proc. 1.3.23, pp. 1 and 2 of 21, respectively.

Facility Review Copy

Reviewed by
Edwin L. Wright
E2Wright

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

Facility: WNP-2
Reactor Type: BWR-5
Date Administered: 11/6/84
Examiner: I. S. Levy
Candidate: _____

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheet. Points for each question are indicated in parenthesis 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.

<u>Category Value</u>	<u>% of Total</u>	<u>Candidate's Score</u>	<u>% of Cat. Value</u>	<u>Category</u>
<u>25</u>	<u>25</u>	_____	_____	1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
<u>25</u>	<u>25</u>	_____	_____	2. Plant Design Including Safety and Emergency Systems
<u>25</u>	<u>25</u>	_____	_____	3. Instruments and Controls
<u>25</u>	<u>25</u>	_____	_____	4. Procedures: Normal, Abnormal, Emergency, and Radiological Control
<u>100</u>		_____		TOTALS
		Final Grade	_____ %	

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

Candidate's Signature

Reviewed by
E. L. Wright
ELWright

Questions and Answers to WNP-2 RO Exam - 11/6/84

1.0 Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow (25.0)

N/A. question autoclear - ~~ask~~
re: 'doubling time' not used,
or 4 sec.

Revised by
Sep 11/84

1.1 Re

a gives $(.5)^{1/4}$ for T $T =$ (0.5)
 $(.5)^{1/4}$ for t $t =$ (0.5)
check T is sec^{-1}
not sec th (1.0)

1.1 a. ... initial count rate does not affect the amount of reactivity required to go critical, which determines ECP (0.75). [The higher the initial count rate, the higher the count rate when criticality is reached.] (1.25)

Ref: WNP-2 Reactor Theory, pg. 29 and 43.

b. $P = P_0 e^{(t/T)}$
 $T = 40 / (\ln 2) = 57.71 \text{ sec}^{-1}$
 $t = 57.71 (\ln(P/P_0)) = 212.88 \text{ sec}$ (1.0)

Ref: WNP-2 Reactor Theory, pg. 61.

Consider giving partial credit on b. if candidate used period of 40sec since the question asks for the most part the candidate to demonstrate the ability to manipulate the equation $P = P_0 e^{t/T}$.

*Review to
After Report*

1.2 With regard to some aspects of Fission Product Poisons:

- a. Of the two fission product poisons Xe and Sm, give two (2) reasons why xenon is more troublesome. (1.5)
- b. What is the mechanism(s) for removal of Samarium-149 once it is produced in the core? (1.0)

- 1.2 a. (1) Because of its exceptionally large thermal neutron absorption cross-section (0.75); (2) its concentration varies with reactor power level and/or time (0.75) (2) Xe produced as result of rel. large fraction of fissions Ref (1.5)
- b. Sm-149 is removed only by burnout (1.0)

Ref: WNP-2 Reactor Theory, pp. 83 and 87, respectively.

Control rod
density
OK

1.3 When control rod density in the core decreases at higher burnups (from pulling rods) the void coefficient of reactivity becomes more or less (choose one) negative? Why? (1.5)

1.3 Less (0.5). Since local steam voids cause an increase in thermal diffusion length (0.5), control rods, which absorb thermal neutrons, make the steam void reactivity coefficient more negative (0.5) [Therefore, reduced control rod density causes the void coefficient to be less negative.] (1.5)

Ref: WNP-2 Reactor Theory, pp. 98-99.

Revised by
Steph Legend
or

1.4 The effective decay constant for up power transients will be higher or lower (choose one) compared to its value for down power transients. Give the reason for your answer.

(1.5)

1.4 Higher (0.5). For up power transients the short lived precursors are dominant due to the addition of power (0.5), while for down power transients the long lived precursors dominate the decay constant (0.5)

(1.5)

Ref: WNP-2 Reactor Theory, pg. 54.

1.5
Revised by:
Steph M. Young
a

Following a scram from high power, answer the following:

- a. What are the most reactive regions of the core? (1.0)
- b. Why are these regions more reactive? (1.25)
- c. What problem does this cause for the operator during a subsequent start up. Why? (1.25)

- 1.5 a. Near the edges and at the top and extreme bottom (1.0)
- b. Xe concentrates, during power operation, where power is highest, i.e., in the center and near the bottom of the core (0.75), where it acts as a poison, adding negative reactivity (0.50) (1.25)
- c. Operator must be extremely cautious while pulling edge and top rods (0.5) since normally low worth rods now have excessively high incremental worths (0.75) (1.25)

Ref: WNP-2 Lesson Plan, Figure 4-12.

1.6 Give four (4) inputs or outputs for a reactor heat balance, stating whether it is an input or output and a brief description as to why it is.

(3.0)

1.6 (any 4 @ 0.75 each)

(3.0)

- a. Feedwater flow - heat input (0.25) going into the vessel (or system) with positive enthalpy (0.5).
- b. Steam flow - heat output (0.25) due to steam removing energy from the core (0.5).
- c. Recirc pump - heat input (0.25) due to energy added to the fluid in the core (or system) by the pumps (0.5)
- d. CRD flow - heat input (0.25) due to fluid flowing into the core (or system) with a positive enthalpy (0.5).
- e. Rx core thermal energy - heat input (0.25) due to being primary source of heat input (0.5).
- f. any other reasonable answer such as RWCU inputs/outputs, etc.

Ref: MTC; Thermo/HT/FF (3/83), pp. 8-50.

ambient losses

- 1.7 Following initial criticality (MSIVs closed, moderator $T > 212^{\circ}\text{F}$), a constant positive period is established. Briefly explain what happens over the next several hours to pressure, temperature and power if no rod movement occurs. (1.5)

- 1.7 Power initially increases but levels off due to negative reactivity insertion resulting from increasing moderator temperature (0.5). Pressure and temperature initially increase but level off when power levels off then reduces due to ambient losses (0.5). The reduced T causes the cycle to start again so that long term power, pressure and temperature will oscillate around point of adding heat (0.5). (1.5)

Ref: Standard Reactor Theory.

1.8 What are two (2) reasons a centrifugal pump should be started with the discharge piping filled and the discharge valve shut?

(1.0)

1.8 Water hammer and excessive starting current.

(1.0)

Ref: Morris T.C.; Thermo/HT/Fluid Flow (3/83), pg. 7-123.

- 1.9 a. Assume the reactor is at 100% power and flow. Explain what happens to core flow, and why, for a reduction in power by driving rods in. (Recirculation pump speed remains constant.) (1.25)
- b. At low power conditions prior to void generation, an increase in reactor power by control rod withdrawal will (increase, decrease, not change) flow through the core. Choose the correct answer and explain your choice. (1.5)

- 1.9 a. Core flow would increase (0.5) due to a reduction in two phase flow condition (and, therefore, in the core less resistance to flow) (0.75). (1.25)
- b. Increase (0.5). Flow resistance in the channels drops due to decreased liquid viscosity with temperature (0.5); and greater density differences between warm channels and cool downcomer will increase flow due to increased thermal driving load causing greater natural circulation (0.5) (1.5)

Ref: Morris T.C.; Thermo/HT/Fluid Flow (3/83), pg. 9-51.

1.10 There are several characteristic transients that would be limiting because of MCPR.

a. List any two (2) of these transients. (0.75)

b. Assuming for each of the transients in (a) they occurred at 100% power, EOC and full rod out conditions, give:

1. the most important reactivity coefficient involved. (0.5)

2. What occurred in the reactor and why it occurred to affect this coefficient. (0.5)

1.10 Any 2 of following. pts: a) 0.375 each; b) (1) 0.25 each; (2) 0.25 each (1.75)

(a)

b(1)

b(2)

- Generator load reject w/o bypass; void coefficient, void collapse from pressure increase
- Turbine trip w/o bypass; void coefficient, void collapse from pressure increase
- Loss of feedwater heating, void coefficient, void collapse from subcooling
- Inadvertent start of HPCS pump; void coefficient, void collapse from subcooling
- Feedwater controlling failure high; void coefficient, void collapse from subcooling

Ref: Morris T.C.; Thermo/HT/Fluid Flow (3/85), pg. 9-94 through 9-96.

1.11 Given:

Rx pressure at time T = 675.3 psig

Rx pressure at time T + 1 hr = 215.3 psig

- a. What is the Rx cool down rate for this hour? Show all calculations. (0.75)
- b. Is this rate acceptable at your plant? (0.5)

1.11 a. (Because the reactor operates at saturated conditions the temperature for time T and T + 1 hr can be found using the steam tables.)

1) Saturation temperature for 690 psia is approximately 502°F.
Saturation temperature for 230 psia is approximately 394°F. (0.375)

2) Cool down rate = $(502^{\circ}\text{F} - 394^{\circ}\text{F})/1 \text{ hr.}$
Cool down rate = 108°F/1 hr. (0.375)

b. No. (The cool down rate limit is 100°F per hour.) (0.5)

Ref: Steam Tables and WNP-2 Tech. Specs.

1.12 Answer TRUE or FALSE for each of the following:

- a. As water flows around a bend in a pipe, the velocity of the water is uniform throughout the diameter of the pipe. (0.5)
- b. The pressure in a static fluid always decreases with increasing elevation of the measurement. (0.5)

- 1.12 a. False (0.5)
- b. True (0.5)

Ref: General Fluid Dynamics Text.

- 1.13 Will an increase in the following factors (taken separately) increase, decrease, or not change differential rod worth? (1.5)
- a. thermal diffusion length
 - b. neutron flux at the rod
 - c. core average neutron flux

- 1.13 a. increase (0.5)
 b. increase - assuming no change in core average flux (0.5)
 c. decrease - assuming no change in local flux. Usually an increase in core average flux would cause some change in local (0.5)

Ref: WNP-2 Reactor Theory, pg. 80.

flux - it may be that this part ^{can} be interpreted so that there is no change in DRW; No change should be acceptable.

$$P = P_0 e^{t/\tau}$$

$$1Ci = 3.7 \times 10^{10} Bq$$

$$\alpha_D = -1 \times 10^{-5} \frac{\Delta K / ^\circ F}{K}$$

$$\alpha_V = -1 \times 10^{-3} \frac{\Delta K / Z \text{ voids}}{K}$$

$$\alpha_H = -4.5 \times 10^{-4} \frac{\Delta K / Z ^\circ F}{K}$$

$$\alpha_P = -4.5 \times 10^{-4} \frac{\Delta K / Z \text{ power}}{K}$$

$$I(t) = I_0 e^{-\lambda t}$$

$$T_{1/2} = \ln(2)/\lambda$$

$$C_p = (C_{p\text{base}}) (K_s) (K_A)$$

$$Q = MC_p \Delta t$$

$$\Delta p = f \frac{L \rho v^2}{D 2g_c}$$

$$f = 64/Re$$

$$\rho = \frac{k(\text{eff}) - 1}{K(\text{eff})}$$

$$\frac{1}{M} = \frac{CR1}{CR2} = \frac{1-K(\text{eff})^2}{1-K(\text{eff})}$$

$$M = \frac{CR2}{CR1} = \frac{1-K(\text{eff})}{1-K(\text{eff})^2}$$

$$\dot{Q} = M \Delta h$$

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

$$M = 1/(1-k)$$

$$N(t) = N_0 e^{-\lambda t}$$

$$\alpha_{gr} = (L_f + L_g) \frac{(\phi_{rod})^2}{(\phi_{avg})}$$

$$n = v/(1+d)$$

$$P = \Gamma \phi v / (3.7 \times 10^{10})$$

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

$$\dot{\tau} = \bar{L} / \rho + (\beta - \rho) / \lambda \rho$$

$$\tau = L / (\rho - \beta)$$

$$v = v_f + x v_{fg}$$

$$H = x h_g + (1-x) h_f$$

$$S = x S_g + (1-x) S_f$$

$$1 \text{ in.} = 2.54 \text{ cm}$$

$$1 \text{ gal.} = 3.785 \text{ liters}$$

$$1 \text{ kg} = 2.205 \text{ lb}$$

$$N = \rho A_0 / A$$

$$17.58 \text{ watts} = 1 \text{ BTU/min}$$

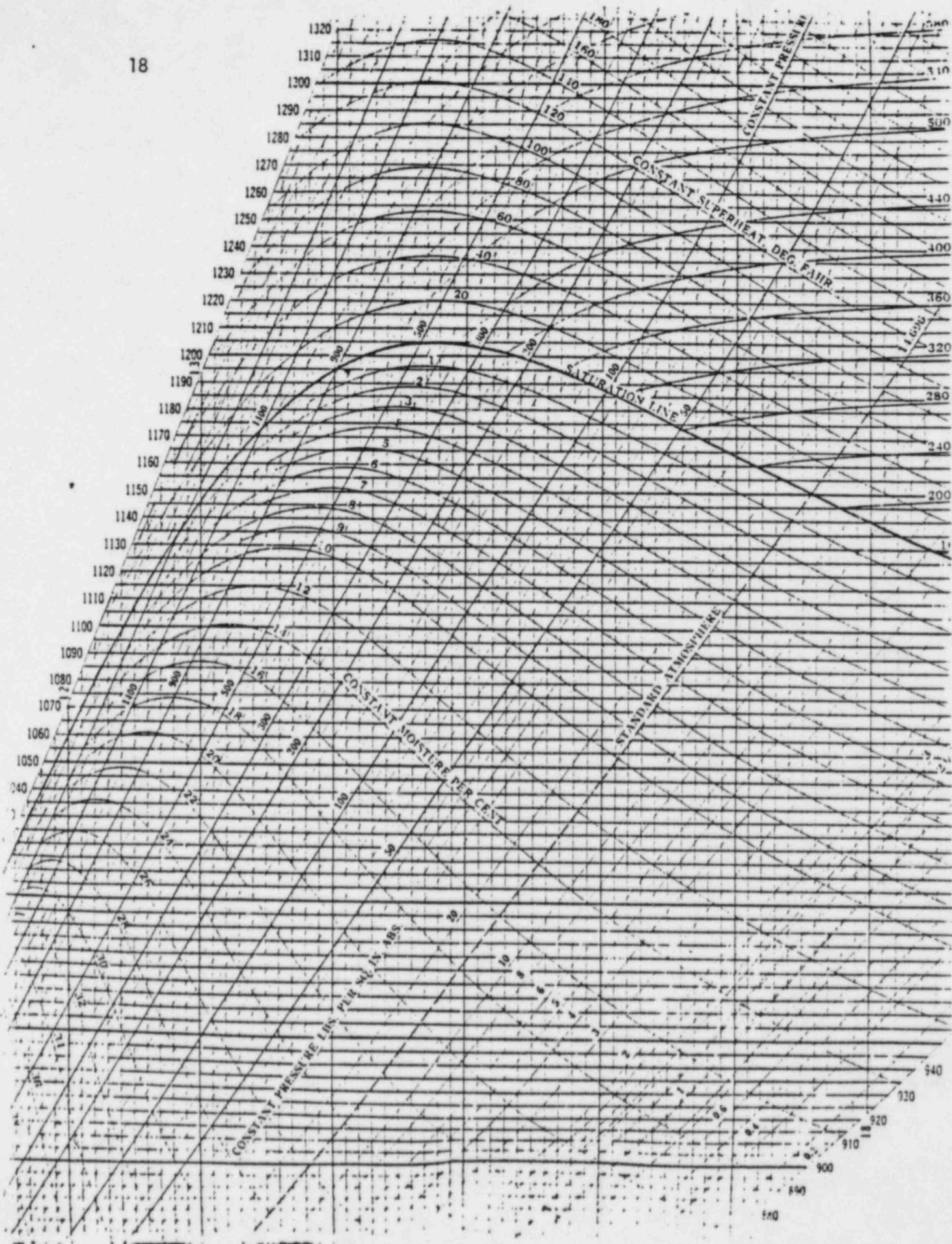
$$1 \text{ psi} = 6.895 \text{ Pa}$$

$$1 \text{ psi} = 2.036 \text{ } ^\circ \text{H}_g \text{ (@ } 0\text{C)}$$

$$1 \text{ psi} = 27.68 \text{ } ^\circ \text{H}_0 \text{ (@ } 4\text{C)}$$

$$\bar{\beta} = .0071$$

$$\bar{L} = 2 \times 10^{-5} \text{ sec}$$



2.0 Plant Design, Including Safety and Emergency Systems (25.0)

2.1 With regard to the Diesel Generators:

a. The "Emergency Bypass-Test" selector switch is in "Emergency Bypass" position when a low reactor water level initiation signal is received. Which diesel generator trips would still be operable? (2.0)

b. Which of these trips would be bypassed if the selector switch was in the "Test" position and the same initiation signal was received? (0.5)

2.1 a. (all 4, 0.5 each) (2.0)

1. Engine overspeed
2. Generator differential relay
3. Fail to start (incomplete response)
4. Emergency stop pushbutton

b. None (0.5)

Ref: WNP-2 System and Procedures; DG, pg. 33-35.

No Comment

2.2 What safety action(s) are auto-initiated at each of the following indications:

- a. Level 4 (31.5")? (0.5)
- b. Level 2 (-50")? (0.5)
- c. 1135 psig? (0.5)
- d. 1076 psig? (0.5)

- 2.2 a. Runback recirc. flow if only 1 reactor feed pump. (0.5)
↳ Lo level alarm has safety implication & should be accepted
- b. Initiate RCIC and HPCS; trip recirc. pump; initiate NSSS iso. groups 1, 2, 3, 4 and 7. (0.5)
 - c. ATWS trip of recirc. pumps. (0.5)
 - d. 2 MSL S/R valves open (relief mode). (0.5)

Ref: WNP-2 System and Procedures; NBI, pg. 55 and 57;
Main Steam, pg. 32.

- a. level 4 may not necessarily be assumed to be co-incident with a loss of a RFP.
- b. level 2 initiates N₂ isolation - groups are not commonly used at this site
equipment listed, i.e. RWCU, MSIV's etc should be acceptable.
- c. no comment
- d. or, first MSRV's open as relief valves

2.3 With respect to the Automatic Depressurization System (ADS):

- confusing* →
- a. List (including setpoints) the automatic activation sequence for ADS. (1.5)
 - b. Which initiation signal(s) can be cleared by pressing a Seal-In Reset pushbutton(s)? (1.0)
 - c. Which of the signal(s) in (b) can be cleared only if the initiating condition no longer exists? (0.5)

- S&P RPS scram table* →
- 2.3 a.
 - 1. Hi drywell pressure (>1.65 psig) (0.3)
 - 2. Lo water level (level 3: -50") *This is +13"* (0.3)
 - 3. Lo water level (level 1: -129") (0.3)
 - 4. 105 second timer timed out (0.3)
 - 5. >1 low pressure ECCS pump (125 psig for RHR/LPCI; 145 psig for LPCS). (0.3)
 - b. 0.5 pts each (1.0)
 - 1. ADS A(B) reactor pressure vessel low level logic
 - 2. Hi drywell pressure A(B)
 - c. Hi drywell pressure (0.5)

Ref: WNP-2 System and Procedures; AOS, pg. 8-10; 12; and 12, respectively.

2.4 With regard to the Reactor Protection System:

- a. Which trip(s) can only be bypassed manually? (1.25)
- b. With regard to the backup scram valves:
1. Are they solenoid or air operated? (0.5)
 2. To cause a scram, do they:
(a) energize or (b) deenergize (if solenoid)
(c) pressurize or (d) vent (if air operated)
(choose only 1 answer) (0.5)
 3. What is their function? (0.5)

2.4 a (0.417 each) (1.25)

1. APRM Hi-Hi
 2. APRM inop.
 3. Scram discharge volume Hi level trip
- b.
1. solenoid (0.5)
 2. energize (a) (0.5)
 3. bleed air from scram valves (vent header to atmosphere) (0.5)

Ref: WNP-2 System and Procedures; RPS, pg 15, 15, 16;
and 30-31, respectively.

a- this question seems to be confusing, other acceptable answers should include the slide links for bypassing non coincident scrams (SRM, IRM, APRM) manual bypass of IRM, SRM. The question should have given a limit (ie 3,4) on answers expected.

b.3 initiate a scram in case of scram pilot valve failure

2.5 With regard to the AC Electrical Distribution System:

- a. Which lockout relay will be tripped upon a Transformer Differential Current (87TM)? (0.5)
- b. List three (3) actions which will occur when the lockout relay in (a) is tripped? (1.5)
- c. The loss of which 480V MC bus will deactivate both loops of shutdown cooling? (0.5)
- d. What happens upon the loss of the normal and startup sources for SM-4? (0.75)

2.5 a. Unit lockout (86XU) (0.5)

b. any 3, 0.5 pts each (1.5)

- Trips and locks-out all "N" breakers
- Trips 4F circuit breaker (Generator Exciter Field Breaker)
- Trips main turbine (20 AST)
- De-energizes 86XIU
- Starts oscillograph
- Starts computer

c. MC-8-B-A (0.5)

d. automatic transfer of SM-4 to the Division 3 EDG (0.75)

Ref: WNP-2 System and Procedures; AC Distr.; pg. 15 and 16
for a, b, d; AOP 4.7.1.9, Loss of Power to SM-8,
pg. 3 of 4 for c.

a- this question is unfair, plant operators typically do not memorize relay numbers. They would be expected to know that an 86 relay is a lockout. I wrote all of the AC training procedures (originals) and I'm not sure I could answer this question. Also 86XU also trips 86XIU for an additional six operations which should be acceptable answers

b- Same as above

c. operators are not required to memorize 480V distribution

d. this is the only section of 2.5 that the operator should be expected to answer

2.5 b answer list should include 1. closes all S breakers 2 trip & lock out 500KV bbs
3 locks out exciter field breaker 4 locks out 500KV bbs sync circuit
5 locks out main generator 6 removes gen ground protection 7 blocks N bbs trip ann.

2.6 Concerning the Standby Liquid Control System (SLC):

- a. Give three (3) of the four automatic actions which occur when the SLC System Control Switch is placed in the "Sys A" position? (1.0)
- b. What is the purpose behind the SLC storage tank heater? (1.0)
- c. There is a SLC pump trip on low flow (TRUE or FALSE)? (0.5)

2.6 (any 3, 0.33 each) (1.0)

- a.
 1. Both SLC squib-valves fire.
 2. RWCU-V-4 isolates.
 3. Both SLC storage tank outlet valves open.
 4. SLC-P-1A starts (if at least one suction valve is open).
- b. Maintains solution temperature high enough to prevent precipitation of the sodium pentaborate. (1.0)
- c. False (0.5)

Ref: WNP-2 System and Procedures, SLC; pg. 9, 13, 10, respectively.

no comment

2.7 Answer the following questions concerning the Low Pressure Core Spray System (LPCS):

- a. What is the rated flow of the main LPCS pump? (0.75)
- b. In shutting down LPCS to standby readiness, is the injection shut off valve (LPCS-V-5) closed before stopping the LPCS pump (Yes or No)? Why? (1.25)
- c. The check valve located inside the drywell is motor-operated (Yes or No)? (0.5)
- d. What are the interlocks associated with the auto-opening of the LPCS-V-5 injection valve? (0.75)

this part was not asked for in the question

- 2.7 a. 6350 gpm at 128 psid reactor to suppression chamber. (0.75)
- b. No (0.5); to ensure no voids are left in the discharge line which would cause water hammer upon subsequent pump restart (0.75) (1.25)
- c. No (0.5)
- d. No undervoltage on SM-7 and reactor pressure <470 psig. (0.75)

Ref: WNP-2 System and Procedures, LPCS; pg. 8, 11-12, 5, and 7, respectively.

- b. operators are not required to memorize steps in operating procedures, also, per PPM 2.4.3 (LPCS operating procedure) Page 9 of 15, rev 2 step E 2 and E 4 call for closing discharge valve LPCS-V-5 and then stopping the pump.
- d. The only opening condition taught as an interlock is that the pressure in the reactor must be less than 470 psig. Power supply to a motor is not considered to be an interlock.

2.8 With regard to the Plant Service Water System (TSW):

- a. What is the purpose of the Chlorine System? (1.0)
- b. What causes starting of the ^{TSW} TBCCW pumps and opening of discharge valves? (1.5)
- c. Since TSW provides cooling only to non-essential equipment, why must the plant be shutdown when neither TSW pump can be started? (0.75)

- 2.8 a. To inject chlorine to retard the growth of algae within TSW systems. (1.0)
- b. Pumps start on low pressure (<80 psig) on alternate pump or undervoltage on SM 85 (75) for 15 sec (0.75); discharge valve opens when pump starts if control switch in auto (0.75). (1.5)
- c. Because components cooled by TSW are essential for continued operation of the secondary and primary plant. (0.75)

Ref: WNP-2 System and Procedures, TSW; pg. 3, (4) 7, respectively.

- a. to retard growth of organisms should also be acceptable
- b. time delay should not be required as a specific in the answer. also, TSW pump selected for auto start will come on after the diesel has powered its bus on loss of offsite power.

2.9 Concerning the CRD Hydraulic System, give the appropriate values for the following:

a. Insert drive water pressure at 400 psig reactor pressure. (0.5)

b. Cooling water pressure at 400 psig reactor pressure. (0.5)

2.9 1. $Rx + 260 = 660$ psig (0.5)

2. $Rx + 20 = 420$ psig (0.5)

Ref: WNP-2 System and Procedures; CRDH; pg. 2.

2 - this is taught as $Rx + 20$ to 30

2.10 Concerning the Condensate Storage and Transfer System (CSTS):

- a. What is the minimum level that must be maintained in CST tanks at all times? (0.5)
- b. Why is this minimum level required? (1.0)

- 2.10 a. 135,000 gal (6'8") (0.5)
- b. To provide suction for RCIC and HPCS systems to ensure immediate availability of sufficient condensate for ECCS and shutdown (1.0)

Ref: WNP-2 System and Procedures; CSTS; pg. 1 and 13.

no comment

TABLE 1
REACTOR SCRAMS

*Manual bypassed TTVs
Ref just # 2.4*

Scram	First Alarm	Second Alarm	Scram Setpoint	Bypass	Scram Logic
Manual	-	-	-	-	1/2 twice
Shutdown Mode	-	-	-	-	1/1 once
SRM Hi-HI	1×10^5 cps	-	2×10^5 cps	*Shorting links installed	1/4 once
IRM Inop	-	-	-	In Run Mode	1/4 twice
IRM Hi-Hi	108/125	-	120/125	In Run Mode	1/4 twice
APRM Hi-Hi	12%	-	15%	In Run Mode	1/3 twice
APRM Hi-Hi	$.66W_R + 42$	-	$.66W_R + 51\%$	-	1/3 twice
APRM Hi-Hi	-	-	118%	Not in "RUN"	1/3 twice
APRM Inop	-	-	-	-	1/3 twice
Scram Discharge Volume	A-525'2-3/8" B-524'5-13/16"	527'1-1/2"	529'6"	In S/D or refuel, and keylock switch bypassed	1/2 twice
Reactor Water Level	-	-	13"	-	-
Reactor High Pressure	1021 psig	-	1037 psig	-	1/2 twice
Drywell High Pressure	0.1 psig (low)	1.5 psig	1.68 psig	-	1/2 twice
MSL High Radiation/Inop	1.5 x Normal	-	3 x Normal	-	1/2 twice
MSIV Closure	-	-	6% Closed	Not in Run	Any 3/4 MSLs isolated
Turbine Throttle Valve Closure	-	-	5% Closed	30% Turbine First Stage Pressure	Any 3/4 TTVs
Turbine Governor Valve Fast Closure	-	-	1250# supply oil pressure	30% Turbine First Stage Pressure	1/2 twice

*NOTE: With Shorting Links removed, any one SRM, IRM, or APRM scram signal from any channel will result in a full scram (non-coincidence logic).

J. Westgren

3.0 Instruments and Controls (25.0)

3.1 With regard to the Rod Worth Minimizer System (RWM):

- a. Under what two (2) conditions will the Select Error alarm light be lit? (1.0)
- b. Above LPAP, what alarms remain operative? (1.5)
- c. TRUE or FALSE: A rod block is applied upon the second insertion error. (0.5)

- 3.1 a. Whenever a selected control rod is not in the currently latched group (0.5) or one currently positioned so as to cause a withdraw or insert error (0.5) (1.0)
- b. Inop/Reset; Withdraw block; Insert block (1.5)
- c. False (0.5)

Ref: WNP-2, System Description, RWM, pp. 18, 20, 18, respectively.

b. your answer is true only for a "hardware failure" (Ref pp 20 above) ~~this question answer does not~~ otherwise - during its normal function - no alarms are operable, which should be acceptable answer since hardware failure was not addressed.

- 3.2 For the events listed, match the action(s) that will occur in the Recirculation System. Assume that the pumps are running in high speed. (An action may be used more than once)

(2.5)

Events:

1. Suction or discharge valves <90% open
2. Vessel hi pressure (ATWS)
3. Feedflow <30% with FCV ~~<18% open~~
4. Reactor vessel low level (Level 3)
5. RPT

Actions:

- a. Fast Speed Trip
- b. Slow Speed Trip
- c. LFMG start.

- 3.2 (0.5) each

(2.5)

1. A, B
2. A, B
3. A, C
4. A, C
5. A, C

Ref: WNP-2, RRC, pg. 39.

✓

3.3 According to Monthly Operational Bulletins:

- a. How did the failure, on two shifts, to check chart movement on wetwell level recorder CMS-LR/RR-4 contribute to loss of wetwell level? (1.0)
- b. Why is it important to ensure that local temperature indicators at the nitrogen supply shed and in the reactor building are monitored? (1.0)

- 3.3 a. The recorder had been, in fact, inadvertently de-energized so that annunciator alarm switches activated by the recorder pen were also O.O.S. (1.0)
- b. No control room monitors exist. If nitrogen temperature gets too low, nitrogen flow onto a 30 in. dia. containment purge header and onto wetwell and drywell purge liner inside containment could cause failure through nitrogen embrittlement. (1.0)

Ref: WNP-2 Monthly OP Bull: April-May, pg. 1; Feb-Mar, pg. 6, respectively.

a+b Unfair question :

MOBs are covered as part of the license

Regular Training NOT as part of HOT license

Training and therefore the operator may not have read this.

See response for question # B.9

3.4 In reference to the Source Range Monitors (SRM):

a. What two (2) types of radiation are separated by the pulse height discriminator (PHD)? Which one causes an output signal from the PHD? (1.0)

b. Indicate (by Yes or No) whether the following trip circuits in the SRM electrical circuitry will generate a signal for use in the RMCS rod block circuitry: (3.0)

1. Downscale
2. Retract Permit
3. Upscale High
4. Upscale High High (shorting links installed)
5. Inop.
6. Reactor period.

3.4 a. Neutron and gamma radiation (0.5); neutrons cause output (0.5) (1.0)

b. (0.5) each (3.0)

1. Yes
2. Yes
3. Yes
4. No
5. Yes
6. No

Ref: WNP-2, System Description, IRM, pg. 14 and 28, respectively.

ok

- 3.5 With regard to the Reactor Manual Control System (RMCS):
- a. The accumulation light starts flashing: (1.5)
 1. What is the cause(s) of this?
 2. What causes the light changing to "steady on"?
 - b. In mode 5 under what conditions, and in what manner will Select block be indicated? (1.5)
 - c. Can an overtravel alarm be received if the control rod is connected to its drive unit (Yes or No)? (0.5)

- 3.5 a. 1. High-level (5 cc) or low N₂ pressure (970 psig) (0.75)
 2. Operator acknowledges alarm with "Accumulation Trouble Acknowledge" pushbutton (0.75) (1.5)
- b. Any rod is not fully inserted (0.75); SELECT BLOCK amber light (0.75) (1.5)
- c. No. (0.5)

Ref: WNP-2, System Description, RMCS, pp. 7, 11, and 18, respectively.

- a. setpoints were not asked for in question
- b. also accept " if the operator withdraws one control rod.

3.6 With regard to the RCIC system:

- a. RCIC pump flow indicator RCIC-FIS-2 has two contacts. What is the purpose of each contact? (2.0)
- b. For monitoring steam flow to the RCIC turbine:
 - 1. How many differential pressure switches (DPS) are used to monitor steamflow? (0.5)
 - 2. Where are they located? (0.75)
- c. Should the RCIC-V-8 and RCIC-V-63 keylocked control switches be left in OPEN or CLOSED position when resetting any isolation signal? (0.5)

- 3.6 a 1 a. ~~2~~ - "the equipment name of steam pump flow indicator" (1) this is inpt. design rel. to flow control.
2. b. 2 - 0 " ~~partial credit~~ need specific location as seen in answer for full credit
2. c. ~~open~~ should be "close" position.
- b. 1. → a delete a "ok!" (2.0)
2. (0.5)
- ... to the RHR (0.375) (0.75)
- c. Open ← wrong! (0.5)

Ref: WNP-2, System Description, RCIC, pp. 27, 27; and SOP 2.4.6, pg. 2 of 29, respectively.

- a. This question is not realistic - operators do not memorize FIS switch numbers!! and not the number of contacts and each one's specific function - but the function as viewed from his indications.
- b. 2 should also accept, ~~all~~ "all located in flow elbows inside drywell"
- c. they should be in "Close" position (Ref PPM 2.4.6 Rev2 Limitation D)

- 3.7 Concerning vessel instrumentation, state whether the following are TRUE or FALSE:
- a. The Fuel Zone Range level indicators are calibrated cold. (0.5)
 - b. Level 1 (-128") will initiate NSSSS isolation groups: 1, 2, 3, 4, and 7. (0.5)
 - c. The reference leg design of the Level Indicators have been designed to compensate for extreme temperature transients. (0.5)
 - d. Jet pumps 5, 10, 15, and 20 were individually flow calibrated prior to installation. (0.5)
 - e. Pressure measured at the core inlet plenum is also used as input to the CRDH system. (0.5)

- 3.7 (0.5) each (2.5)
- a. False
 - b. False
 - c. False
 - d. True
 - e. False

Ref: WNP-2, System Description, NBI, pp. 57, 56, 5, 7, and 33, respectively.

- 3.8 With regard to the Power Range Neutron Monitoring System (PRMS):
- What four (4) subsystems make up the PRMS? (0.75)
 - Which three (3) trips are input to RPS from the PRMS? (0.75)
 - For what two (2) conditions and for which components do the white indicators next to the heat flux meters below the full core display become lit? (0.75)

- 3.8 a. RBM, Flow Unit System, APRM, LPRM (0.75)
- b. APRM upscale thermal, APRM upscale neutron, inop. (0.75)
- c. Failed (0.25) bypassed conditions from LPRMs associated with rod selected (0.25). *NO* (0.75)

Ref: WNP-2, System Description: LPRM, pg. 1; APRM, pg. 36; LPRM, pg. 4; respectively.

b. also accept 15% APRM u/s trip in
with Mode Sw. in Startup

c. Manual Bypass or Peripheral rod selected
or APRM Ref signal < 30%

For the RBM system (Ref S+P RBM pg 33
Vol II Tab 7)

3.9 Which type of detector (scintillation, ion chamber, fission chamber or Geiger Mueller) is used in the following process radiation measurements?

- a. Main steam line (0.5)
- b. Off-gas post-treatment (0.5)
- c. Reactor building main exhaust (0.5)

- 3.9 a. Ion chamber (0.5)
- b. Geiger-Mueller (0.5)
- c. Geiger-Mueller (0.5)

Ref: WNP-2 Syst. Descript., PRM, pg. 4, 3, 4, respectively.

OK

4.0 Procedures - Normal, Abnormal, Emergency and Radiological Control (25.0)

- 4.1 With regard to General Operating Procedure 3.1.2, Reactor Plant Cold Startup:
- a. What action(s) should the operator take to prevent RWCU pump trip on low flow. (1.0)
 - b. Why should you avoid heat-up rates that demand a high reject temperature. (1.0)
 - c. How will RPV water level stability be indicated? (1.0)

Weighted too heavily for a single paragraph out of one operating procedure.

- 4.1
- a. By adjusting reject valves RWCU-V-31 and V-33 as required. (1.0)
 - b. This will cause high RWCU F/D inlet temperatures and RWCU F/D isolation at 140°F. (1.0)
 - c. By a small output signal on the RFW-FCU-10 controller. (1.0)

Ref: WNP-2 PPM 3.1.2., pg. 8 of 18.

4.2 With regard to performing Rod or Minimizer (RWM) initiation in accordance with the System Operation Procedures for RWM (2.1.4):

- a. How does the operator verify the RWM is not in "rod test"? (1.0)
- b. What happens when the INOP/RESET pushbutton is depressed before the "System Initialize" pushbutton is depressed? (1.0)
- c. What happens when the INOP/RESET pushbutton is depressed after the system is initialized? (1.0)

- 4.2
- a. By depressing the TEST/SELECT button, observing illumination, depressing again and observing the light goes out. (1.0)
 - b. Any previous alarm ("Comp/Program") is reset, the Comp light and the RWM/Program lights are illuminated. (1.0)
 - c. The RWM and program lights extinguish. (1.0)

Ref: WNP-2 SOP 2.1.4, pg. 2-3 of 3.

Comments:

1. This procedure is not used for operation of the RWM. The volume 7 surv. procedure is used to verify RWM operability and to operate that piece of equipment.
2. Does not assess the operators knowledge of system operation. It checks "which lights come on or off"?
3. Wrote memorization of procedures not req'd per Examiners standard 202, B-4.

4.3 Relative to the Emergency Operating Procedure for RPV Pressure Control (RPV/P)(5.1.2):

- a. List any three (3) of the five entry conditions. (1.5)
- b. What are the four (4) systems to be used to augment the main turbine bypass valves for controlling pressure below 1075 psig? Give any limiting condition on the use of these systems. (3.0)

4.3 a. any 3; 0.5 points each (1.5)

- 1) RPV water level below +13.0 in.
- 2) RPV pressure >1037 psig
- 3) Drywell pressure >1.68 psig
- 4) A condition requiring MSIV isolation
- 5) A condition requiring reactor scram and power is above 5% or cannot be determined.

b. (3.0)

- 1) SRV's (0.6), if suppression pool water level >17 ft (0.3)
- 2) RCIC (0.6)
- 3) RWCU (recirculation through heat exchanger and blowdown modes) (0.6); if no boron has been injected into RPV (0.3)
- 4) Main steam line drains (MS-V-16, -19, -21) (0.6)

Ref: WNP-2 EOP 5.1.2, pp. 1 and 6-7 of 8.

Valve #'s
Should not be req'd
for full credit.

- 4.4 a. The plant is in the process of starting up (Condition 2) with all systems and components normal except that the "A" IRM has previously failed high and was subsequently bypassed. The "E" IRM now loses power and is declared inoperative. May the plant continue in this condition for an extended period of time without being in violation of Tech. Specs? Also give the appropriate action statement. (1.5)
- b. Could you place the mode switch in run (Condition 1) to bypass the action statement in part "A"? (0.5)

- 4.4 a. Yes (0.5); place the RPS A channel in the tripped position within one hour (1.0). (1.5)
- b. No (0.5). (0.5)

Ref: WNP-2 i/s, pp. 3/4 3-1; 3/4 0-1, respectively.

answer should be "yes"

PER Tech. Spec. on IRM operability the provision of Tech. Spec 3.0.4 is not applicable for this action statement!

- 4.5 Assuming a loss of feedwater heaters while operating at 100% power, according to Abnormal Operating Procedure 4.2.7.2:
- Give three (3) of the four events that could have caused this? (2.25)
 - What change would you expect to see in the Main Generator MW (increase, decrease)? (0.5)
 - What is the first immediate operator action you should take? (0.75)

4.5 a. (Any 3 @ 0.75 each) (2.25)

- Heater isolation on high water level.
- Turbine trip.
- (System malfunction resulting in the) isolation of or more feedwater heaters.
- (System malfunction resulting in the) closure of extraction steam line valves for one or more feedwater heaters.

b. Increase ← *This is okay if #1 only is accepted. (on a turb. trip, you don't worry about this because a loss of F.W. heating occurs with every turbine trip.* (0.5)

c. Reduce reactor power via recirculation flow control *every* (0.75)

Ref: WNP-2 AOP 4.2.7.2, pg. 1 of 2.

only answer #1 is found in the abnormal procedure listed as reference. #2 is not even applicable, #3 & 4 are the same thing and they result from #1.

4.6 With regard to the operating procedure for 250V DC Distribution System (SOP 2.7.7), give any three (3) of the four indications that the operator will have if the tie to distribution bus S2-1 has been completed.

(2.25)

4.6 (Any 3 @ 0.75 each)

(2.25)

1. "250 VDC LOSS, BATT B2-1 FAIL" alarm on board "C" in Control Room clears.
2. "250V VDC BATT B2-1 GND" alarm on board "C" remains cleared.
3. Bus S2-1 voltage reads 220 to 250V on board "C".
4. Bus S2-1 ground lamps on board "C" are on.

Ref: WNP-2 LSOP 2.7.7, pg. 2 of 5.

This question requires total recall of procedure. This is not req'd per Examiner's Standard 202 .B-4 (pg 2 of 6)

4.7 Reactor coolant leakage into the primary containment from unidentified sources shall not exceed (1) gpm and the total coolant leakage shall not exceed (2) gpm.

(1.0)

4.7 1. 5 gpm

(0.5)

2. 25 gpm

(0.5)

Ref: WNP-2 Tech Specs., pg. 3/4 4-9.

4.8 The Reactor Operator reports that "GEN BUS DUCT TEMP HIGH" and "GEN BUS DUCT CLR FLOW LOW" have activated and that bus duct temperatures are increasing. The failure of which component(s) is the most probable cause?

(1.0)

4.8 The TSW solenoid supply valve

(1.0)

Ref: WNP-2 AOP 4.5.6.1, pg. 2 of 2.

4.9 According to AOP 4.8.3.2 "Loss of all RCCW," if no RCCW pumps can be started during power operation, a rapid increase will occur in (Fill in).

(0.5)

4.9 Drywell pressure

(0.5)

Ref: WNP-2 AOP 4.8.3.2, pg. 3 of 3.

4.10 According to the Abnormal Operation Procedures for Fires (4.12.4.1), one indication, other than fire alarm, will be fire header pressure fluctuation (TRUE or FALSE)?

(0.5)

4.10 True

(0.5)

Ref: WNP-2 AOP 4.12.4.1, pg. 1 of 2.

4.11 According to the Limitations stated in the Operating Procedures for the Reactor Core Isolation System (SOP 2.4.6), what must you do if manual isolation is required at any time that system initiation is not sealed in? (0.75)

4.11 Close the isolation valves using their respective control switches. (0.75)

Ref: WNP-2 SOP 2.4.6, pg. 2 of 28.

4.12 With regard to Administrative Procedures:

- a. There must be two (2) licensed operators in the Control Room at all times (TRUE or FALSE)? (0.5)
- b. During new fuel handling operations, a licensed operator must be on the refueling floor (TRUE or FALSE)? (0.5)

4.12 a. False (0.5)

b. False (0.5)

Ref: WNP-2 Admin. Proc: 1.3.2, pg. 3 and 6.2.3, pg. 2, respectively.

This could easily be confused with the licensed operator requirement of T.S. 6.2.2.d. which require an SKO for core alt's. Should give credit for a true answer as well.

4.13 According to Standing Order/Night Orders (Admin. Proced. 1.3.1), under what conditions can the reactor operator shut the reactor down without being instructed by the Shift Manager or required by the Emergency Procedures? (1.0)

4.13 When safety of reactor is in jeopardy or when operating parameters exceed any RPS setpoint and autoshutdown does not occur. (1.0)

Ref: WNP-2 Admin. Proc. 1.3.1, pg. 2.

4.14 With regard to the Health Physics Program, what are the whole body exposure limits for the following:

(1.0)

- a. Administrative exposure limits (day, quarter, year)
- b. Lifesaving actions

- 4.14 1. 300 mrem/day (0.17); ~~1000 mrem/quarter (0.17);~~
5000 mrem/year (0.16)
2. 75 rem (0.5)

(1.0)

Ref: WNP-2 Health Physics Program, 3.1.5, pg. 4 of 5.

quarterly limit is 1250 mrem/qtr
see PPM 1.11.3, pg 3 of 25