# REQUALIFICATION PROGRAM AUDIT REPORT

Facility Licensee: Nebraska Public Power District (NPPD)
P. O. Box 499
Columbus, NE 68601

Facility: Cooper Nuclear Station (CNS)

Facility Docket No.: 50-298

Facility License No.: DPR-46

Requalification program audit at Cooper Nuclear Station near Brownville, Nebraska

Chief Examiner: John L. Pellet 6-22-84
John L. Pellet, License Examiner Date

Approved by : R. A. Cooley 6-22-84
R. A. Cooley, Section Chief Date


Summary

Requalification Program Audit in May, 1984.

A requalification program audit was conducted in May 1984 via three site visits and two meetings in Region IV offices. On May 1-3, 1984, six written examinations (four senior operators (SRO) and two operators (RO)) and six oral examinations (four SRO and two RO) were administered, to a total of nine licensed personnel. On May 11, 1984, a meeting was held in Region IV offices with corporate-level personnel from NPPD. On May 14, 1984, an unannounced followup site visit was conducted by RIV personnel to review facility grading. On May 21, 1984, a meeting was held in RIV offices with site-level CNS personnel. On May 25, 1984, an unannounced followup visit was conducted by RIV pesonnel to proctor the retake examinations conducted after accelerated retraining for the ROs and SROs. Individual results are not included. The audit found the requalification program to be marginal.

# CNS REQUALIFICATION AUDIT REPORT

## Report Details

1. Persons Examined

   SRO Licensees:
   Six (6) SRO-licensed personnel were examined on either written or oral exams or both.

   RO Licensees:
   Three (3) RO-licensed personnel were examined on either written or oral exams or both.

2. Facility Requalification Program Audit Areas

   Requalification Examination
   Facility examinations were reviewed for compliance with CNS procedures and Appendix A to 10 CFR 55. Facility grading of requalification examinations scoring between 78 and 82 percent overall were reviewed for concurrence with NRC grading and pass/fail decisions.

   Requalification Retake Examinations
   Facility retake examinations were reviewed for content prior to administration. Examination administration was proctored by NRC. Facility grading of retake examinations was reviewed for concurrence with NRC grading and pass/fail decisions.

3. Examiners - May 1-3, 1984

   J. L. Pellet (Chief Examiner), NRC
   D. E. Hill, EG&G Idaho, Inc.

4. Examiner - May 14, 1984

   R. A. Cooley, NRC

5. Examiner - May 25, 1984

   R. A. Cooley, NRC

# CNS REQUALIFICATION AUDIT REPORT

This Audit Report is composed of the sections listed below.

A. Examination Review Meeting Comment Resolution

B. Meeting/Visit Summaries

   1. May 1-3 Site Visit Exit Meeting Summary
   2. May 11 Meeting Summary
   3. May 14 Unannounced Site Visit Summary
   4. May 21 Meeting Summary
   5. May 25 Unannounced Site Visit Summary

C. Program Evaluation

   1. Facility Examination Results
   2. Facility Examination Administration
   3. NRC Examination Results
   4. Facility-NRC Grading Agreement
   5. Overall Program Evaluation

D. Examination Master Copy (SRO/RO Questions and Answers)

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

CNS REQUALIFICATION AUDIT REPORT

## A. EXAMINATION REVIEW MEETING COMMENT RESOLUTION

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

| NRC | UTILITY |
|---|---|
| J. Pellet | R. Beilke |
| D. Hill | R. Jansky |

COMMENTS

1.2.c — Accept text in place of formula.
Resp.: ACCEPT if answer complete.

3.2/6.5 — Accept CSCS/ECCS as one answer.
Resp.: ACCEPT.

3.3 — Overspeed is mechanical.
Resp.: ACCEPT.

4.4/7.4 — Accept FALSE with explanation of area around elevator.
Resp.: ACCEPT

4.5/7.5 — Accept for full credit answer without limits per SWP form since question does not specifically ask for limits.
Resp.: Accept for pass credit (70%). Procedure gives limits as part of determining if SWP req'd.

8.2 — Accept DMNO as full credit per exposure form & question wording.
Resp.: Accept DMNO for 0.75 & C/HP Supervisor for 0.25.

B. <u>MEETING/VISIT SUMMARIES</u>

B.1. May 1-3 Site Visit Exit Meeting Summary

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

| NRC | UTILITY |
|---|---|
| J. Pellet | P. Thomason |
| D. Hill | K. Wire |
| D. DuBois | D. Whitman |
| | R. Beilke |

Mr. Pellet started the discussion by stating that no preliminary results were available but the examinees were superior to those observed in March. The same generic weaknesses were present as observed during the March exams. A general discussion then took place on the areas described below.

1. Specific areas of weakness:

   a. Hand calculation of air release rates was difficult even with the procedure in hand. This is an appropriate oral exam topic since it is a current procedure in place and NRC expects an examinee to be able to work through any procedure with it in hand.

   b. There was some confusion on allowable exposure limits, especially the differences for males versus females.

   c. Mr. Hill expressed concern about key logs but after a discussion of the governing procedure stated that his concern was resolved.

   d. The SRO oral exams again demonstrated that some SRO-licensed individuals were not fully cognizant of the additional responsibilities inherent in the Shift Supervisor (SS) position. This was attributed to the lack of SS time for some examinees.

   e. Procedure knowledge was weak. Examinees had substantial difficulties in identifying appropriate immediate actions for any event - both Emergency and Abnormal event classes. This weakness was much less severe than encountered during the examinations given in March, but it was still an obvious problem. NRC policy is that operators should know all immediate actions from memory. Mr. Pellet explained that this meant an examinee should be able to list all major actions from memory without relying on a panel walkdown to suggest appropriate actions. It is apparent that current CNS training does not accomplish this. All parties agreed that a substantial part of this problem is caused by the poor structure of CNS procedures that often list 10 - 20 immediate actions that are not appropriate for memorization.

f. Examinees could not take plant symptoms and then identify the event in progress and analyze the basic plant response. As above, this problem was less severe than in March. In the discussion that followed it was agreed that this problem was aggravated by the lack of a plant-specific simulator and that the procedure training might not provide sufficient guidance on plant response in addition to the procedures themselves. It was also agreed that implementation of symptom-oriented procedures could aid in resolving the procedure-related weaknesses identified during the last two site visits.

2. The requalification program is receiving greater attention than in the past and the level of NRC involvement during facility requalification examinations can be expected to increase in the future.

3. The schedule for fall exams at CNS needs to be finalized in the near future. In March, the week of October 7 was discussed, but that week contains a Federal holiday which complicates travel. The following 2 weeks are currently being reserved for the next set of CNS exams (October 15 & 22).

4. Current procedures do not help the examinees. Procedures regularly assume corrective actions are successful and do not guide the operator toward alternate success paths. Examinees routinely have difficulty identifying correct actions for multiple failure events during oral exams.

5. After a detailed discussion of NRC authority versus CNS scheduling conflicts, it was agreed that while NRC did, in fact, possess the authority to review requalification retake examinations, we would not require that CNS submit them for our review until after grading of the NRC and facility examinations just administered.

6. Mr. Pellet commented that the facility RO exam appeared to be noticeably less difficult in the areas of theory and systems knowledge than it should. The facility SRO exam appeared to be about the right knowledge level.

7. The utility expressed concern about the amount of time available to complete the NRC-administered examinations (facility exams were without time limit while the NRC exam had a loose time limit of about 5 hours). Mr. Pellet noted that timing had been discussed prior to site arrival, before the start of the exam, with the

examinees at the start of the exam, and during the exam. Also, he noted that no exams were collected at a given time and that time was not called. After a general discussion it was agreed that one individual may have felt undue pressure to finish and this was not the intent of the examiners. The time allowed for the requalification examination should be no more than the six hours allocated for an NRC license examination.

8. Mr. Beilke attempted to discuss one question on the written exam. This topic is not appropriate to the exit meeting and this was so stated by NRC representatives present.

9. Specific sections of the Examiner Standards (NUREG-1021) were discussed during the meeting. Some of the utility personnel present did not understand that these standards are guidance on an acceptable method of implementing the operator licensing function, but, just as with other similar documents, this guidance does not supplant or change the authority and responsibility delegated to the NRC by various chapters of the Code of Federal Regulations, for example, 10 CFR 55.

10. The exit meeting is intended to benefit the utility by giving it advance indication of weak areas encountered and preliminary results that are available at the time of the site exit. Final results of the requalification program audit will be made in Region IV after all results are evaluated.

B.2. May 11 Meeting Summary

On May 11, NRC and NPPD personnel met in Region IV offices to discuss the results of the March replacement and May requalification examinations. The following personnel were present for the meeting.

| NRC | UTILITY |
|---|---|
| J. Collins | L. Kuncl |
| P. Check | J. Pilant |
| R. Denise | |
| E. Johnson | |
| R. Cooley | |
| J. Pellet | |

Mr. Cooley started the discussion by introducing the problems experienced in March and explaining the reason for the requalification program audit in general and this specific audit under discussion. Mr. Pilant expressed the view that the results of these and the March replacement examinations were of concern, especially considering the resources added to the CNS training budget this year. Mr. Collins noted similar NRC concern. A general discussion of utility and NRC perceptions of the problem followed along with corrective actions planned by the licensee. The following corrective actions were presented by NPPD after the group discussion.

1. NRC-administered requalification examinations will be reviewed by the facility training department to identify the reasons for the high failure rate.

2. A consultant with extensive experience in operator training will be retained to perform a thorough review of the CNS training program.

3. The four individuals who did not pass the NRC-administered requalification examination will be removed from licensed duties effective their next shift after today.

4. The results of the facility-administered examination will be reported to the RIV duty officer or Mr. Collins by May 12.

The meeting concluded with all personnel agreeing to work towards an expeditious resolution of this problem.

B.3. May 14 Unannounced Site Visit Summary

On May 14, Mr. R. Cooley made an unannounced site visit to CNS. The purpose of the visit was to regrade the facility-administered requalification examinations where facility grading resulted in marginal overall scores (78% - 82%). This regrading resulted in one additional operator being removed from shift duties and placed in the accelerated retraining program already in progress. The other grades were within NRC guidelines for NRC-facility grade agreement and pass/fail decisions. The facility grading was considered satisfactory. The facility was asked to regrade one other examination and inform the Operator Licensing Staff, RIV of the results. Overall, 5 of the 20 examineees who took the facility-administered requalification examination did not pass it. This is a pass rate of 75 percent which is considered marginal.

B.4. May 21 Meeting Summary

On May 21, NRC and NPPD personnel met in Region IV offices to discuss the results of the March replacement and May requalification examinations.

CNS REQUALIFICATION AUDIT REPORT

The following personnel were present for the meeting.

| NRC | UTILITY |
|---|---|
| P. Check | P. Thomason |
| E. Johnson | D. Whitman |
| J. Jaudon | |
| R. Cooley | |
| D. DuBois | |
| J. Pellet | |

The meeting started with a general discussion on use of nonplant specific references. NRC policy is to use plant-specific material to the extent possible, but since CNS has not sent any material on the theory training and much of the systems material provided was either outdated, incorrect, or not used in the training program, then material from plants other than CNS was used. One purpose of the exam review is to identify questions or answers which are inappropriate due to the use of nonplant specific material. Comments of this nature are also evaluated by the NRC during exam grading.

The corrective actions for general training planned by the facility are as follows:

1. More instructors will be added.

2. The consultant that was retained for the program review will also review and audit the requalification program.

3. Training plan development will be accelerated.

4. CNS will revise their emergency and abnormal procedures during the upcoming extended outage to clarify the distinction between immediate and followup actions.

A general discussion of CNS procedure problems followed with all parties agreeing that the corrective action above was needed. Also, some discussion on long term changes and goals took place.

The next set of replacement exams was scheduled for either the week of October 15 or 22, depending on the final outage schedule. It was agreed that this date would be finalized as soon as possible. Mr. Cooley indicated that three changes could aid future groups of examinees (both replacement and requalification).

1. Show methodology, assumptions, and work to get maximum partial credit.

2. Avoid extremely short answers.

3. Send NRC copies of the actual training material, especially for theory.

Mr. Pellet noted that through an error, the figures referenced in the Master Exam included with the exam report for the March exam were inadvertently ommitted and would be sent via a followup letter. A general discussion of the appeals process followed. Mr. Cooley agreed to review additional technical material relating to the March exam separate from the formal appeals process but noted that any such review, as well as any formal appeal, was a low-priority task.

B.5. May 25 Unannounced Site Visit Summary

On May 25, Mr. R. Cooley made an unannounced site visit to CNS. The purpose of the visit was to proctor the requalification retake examinations being conducted for the eight license-holders that had been in the accelerated retraining program. There were three ROs and five SROs in the group. One RO that failed the NRC-administered requalification examination did not participate in the retraining or retake because he has been removed from shift duties and placed in a position that does not require an NRC license. Therefore, his license will not be renewed. At the completion of the examination, copies of the retake examinations and keys were brought back to RIV for comparison grading with the facility grading. The facility will grade the examinations and telephone RIV with the results the week of May 28.

C. PROGRAM EVALUATION

C.1. Facility Examination Results

The facility requalification results showed four examinees failing the examination and requiring accelerated retraining. NRC review of the facility grading showed one additional candidate as failing. For the facility-administered examination, a total of five examinees out of the twenty who took it did not pass. This is a pass ratio of 75 percent which is considered marginal.

C.2. Facility Examination Administration

Observed portions of the facility's requalification and requalification retake examinations were adequately constructed and administered by facility staff. The requalification exams were reviewed and found to adequately cover the technical subjects required by the facility requalification program and Appendix A to 10 CFR 55. This is considered satisfactory.

C.3. NRC Examinations

NRC requalification examinations were administered to nine licensed personnel at CNS. Three RO's were evaluated. One took the NRC written only, one took the NRC oral only, and one took both the NRC written and the oral. Six SROs were evaluated. Two took the NRC written only, two took the NRC oral only, and two took both the NRC written and oral. Five of the nine licensed personnel evaluated passed the NRC administered examination. Therefore, 55.56 percent of the evaluated operators passed all portions of the examinations administered by NRC examiners. One operator has been removed from license duties permanently. If this is considered then 62.5 percent passed. A pass rate between 60 percent and 80 percent is considered marginal.

C.4. Facility-NRC Grading Agreement

Facility grading of the requalification examination was reviewed for five examinees that facility grading showed as marginal overall (between 78 percent and 82 percent). One examinee failed on the NRC regrade after passing per the facility grading. Facility grading of the eight requalification retake exams was also reviewed by NRC. Pass/fail and overall grading was within satisfactory limits for the retake examinations. The overall pass/fail agreement on facility examination grading reviewed by NRC was at least 80 percent. Grading agreement overall was within 10 percent. These agreement rates are considered satisfactory.

CNS REQUALIFICATION AUDIT REPORT

C.5. Overall Program Evaluation

Facility: COOPER NUCLEAR STATION

Examiner: R. Cooley, J. Pellet, D. Hill

Dates of Evaluation: May 1984

Areas Evaluated: XX Written XX Oral ____ Simulator

Written Examination

1. Evaluation of Facility Examination Results: Marginal
2. Evaluation of Facility Examination Administration: Satisfactory
3. Evaluation of NRC Examination Results (if given): Marginal
4. Evaluation of NRC-facility grading agreement: Satisfactory

Oral Examination

1. Overall Evaluation: Satisfactory
2. Number Observed: None Number Conducted: Six

Simulator Evaluation

1. Overall Evaluation: Not Applicable
2. Number Observed: ____ Number conducted: ____

Overall Program Evaluation

Satisfactory: ____ Marginal: XX Unsatisfactory: ____

*** Pass ratio for NRC written examinations below 80%. ***
*** Pass ratio for facility written examinations below 80%. ***

D. EXAMINATION MASTER COPY (SRO/RO QUESTIONS AND ANSWERS)

The master requalification examination, consisting of RO and SRO questions, answers, and references is an enclosure to this report.

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR ~~LICENSE EXAMINATION~~
REQUALIFICATION EXAMINATION

FACILITY: COOPER

REACTOR TYPE: GE-BWR

DATE ADMINISTERED: MAY 2, 1984

EXAMINER: J. PELLET

APPLICANT: ____________

INSTRUCTIONS TO APPLICANT:

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

| CATEGORY VALUE | % OF TOTAL | APPLICANT'S SCORE | % OF CATEGORY VALUE | CATEGORY |
|---|---|---|---|---|
| 20 | 25 | ____ | ____ | 1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW |
| 17 | 25 | ____ | ____ | 2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS |
| 20.5 | 25 | ____ | ____ | 3. INSTRUMENTS AND CONTROLS |
| 19.5 | 25 | ____ | ____ | 4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL |
| 77 ~~XWXX~~ | | ____ | | TOTALS |

FINAL GRADE ____________%

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

____________
APPLICANT'S SIGNATURE

SECTION 1 (20 pts)

1.1. List three reactivity coefficients of concern at Cooper and state at what point in core life (BOL/EOL) each has their most negative value.
(3.0)

1.2. A. Define subcritical multiplication (1.0)
B. Sketch the neutron flux response to a control rod pull during an approach to critical. (1.0)
C. Explain why the neutron flux takes longer to level off as $k_{eff}$ approaches 1. (1.0)

1.3. A control rod worth 0.002 dk/k is continuously withdrawn from position 0 to 48. Assume reactivity is inserted linearly as the rod is withdrawn and that withdrawal takes 50 seconds. What is the transient period when the control rod is at position 36? (Assume BOL conditions)
(2.0)

1.4 Sketch the production and decay chain for Xe-135. List the time to xenon equilibrium at 100% power and the time to xenon peak after a reactor trip from 100% power.
(2.0)

1.5 The reactor is operating at 100% power when one (1) recirculation pump trips. Indicate how each of the parameters listed below would initially change (increase, decrease, or remain the same) and explain briefly why the change occurs.
A. Reactor power (1.0)
B. Reactor water level (1.0)
C. Feedwater flow (1.0)

1.6 Briefly describe how the SRM's may be used to determine water level in the core following a LOCA and state the level range where they may be used.
(2.0)

1.7 With the reactor at 60% rated power, steam header pressure is 945 psig and reactor pressure is 972 psig. What will be the reactor pressure if one (1) MSIV is closed?
(2.0)

CATEGORY CONTINUED ON NEXT PAGE

1.8 Tube flow through a feedwater heater is 4 Mlb/hr of feedwater flow which enters at $150^o$ F and exits at $200^o$ F. The shell side is supplied with extraction steam at 30 psig which leaves the shell side as drain water at $150^o$ F. What extraction steam flow is required?

(2.0)

1.9 Place the flow types listed below in the proper order of occurrence for a BWR reactor.

1. Bubble flow
2. Single phase forced convection
3. Slug flow
4. Subcooled boiling
5. Annular flow

(1.0)

SECTION 2 (17 pts)

2.1 Explain why the seal purge flow from the CRD system must be shut off prior to isolating a recirc pump.
(1.5)

2.2 Why must the reactor be shut down if a jet pump is determined to be inoperable per tech specs?
(1.0)

2.3 What is the maximum temperature differential allowed between the reactor coolant in the idle and operating loops to permit starting the idle loop? Why is this limit imposed? (Assmue most limiting operating conditions.)
(1.5)

2.4 List three (3) conditions which could threaten the fuel barriers or reactor pressure system boundary that the Reactor Protection system is designed to protect against.
(2.0)

2.5 State the basic purpose for installing the LLS Safety/Relief Valve Control system.
(1.5)

2.6 What is the function of the Standby Liquid Control system and how does it accomplish this function?
(2.0)

2.7 What two variables are controlled to assure no fuel damage if a rod drop accident should occur? What systems or components are used to control these variables?
(2.0)

2.8 Give two (2) functions performed by the Transversing In-core Probe (TIP) system?
(1.0)

2.9 What are the five visual indications which are used to verify that the fuel assembly is properly oriented in the reactor duing fuel handling?
(2.0)

2.10 List the components or structures which make up primary and secondary containment (2 lists required).
(1.0)

2.11 What is meant by <u>recirculation ratio</u> and what is its value at rated power and recirc flow?
(1.5)

SECTION 3 (20.5 pts)

3.1 List the Group IV isolation signals and the actions resulting from this isolation.
(2.5)

3.2 List three (3) plant protection systems which utilize both reactor pressure and water level.
(1.5)

3.3 Give the four diesel generator trip functions available for an auto start signal.
(2.0)

3.4 What are the requirements for arming the Low-Low Set (LLS)?
(2.0)

3.5 Explain how failure of the inservice pressure controller (A) low will affect actual steam pressure if DEH is in MODE 4 at 100% power and initial steam pressure is 955 psig. Give the new steady-state steam pressure.
(1.5)

3.6 Give the relationship between monitor flow and vent flow in the Kaman system for both the normal and accident range monitors.
(2.0)

3.7 List eight (8) rod blocks which are effective only when the mode switch is in either STARTUP or REFUEL. Include setpoints.
(4.0)

3.8 Give four (4) basic reactor vessel parameters monitored by reactor vessel instrumentation. (Do NOT include core parameters.)
(2.0)

3.9 Explain why the main portion of the Feedwater Control system is designed to use single element control when operated at low power levels.
(1.0)

3.10 Explain how and why each of the following would affect indicated reactor level. Assume in all cases a change occurs.
A. Equalizing valve on the level transmitter leaks through.
B. Drywell temperature DECREASES over an extended period of time.
(2.0)

SECTION 4 (19.5 pts)

4.1 When is secondary containment NOT required? (2.0)

4.2 Upon the loss of Instrument Air, at what point does the operator reduce power and scram the reactor? (2.0)

4.3 What neutron monitoring requirements are imposed during core alterations (fuel being moved) during refueling? (1.0)

4.4 True or False? During refueling operations access to the refueling floor will be controlled by a SWP. (0.5)

4.5 List four (4) conditions where it would be necessary to obtain a Special Work Permit (SWP). (2.0)

4.6 Give three (3) conditions which must be met for a jet pump to be considered inoperable (per Technical Specification daily surveillance requirements)? (3.0)

4.7 Give the immediate action(s) required to recover from a reactor scram from power with MSIV's closed. Do NOT include verifications. Assume all other plant conditions are normal (i.e., do not include action(s) prefaced by phrases such as "If (parameter) then···"). Include sequential action(s) such as "After (parameter) recovers then···". (3.0)

4.8 Give the immediate action(s) required to assure that the reactor is in a safe condition for a major loss of feedwater heating (temperature increases more than $60^{\circ}$ F). (1.0)

CATEGORY CONTINUED ON NEXT PAGE

4.9 Give eight (8) systems which are required by technical specifications to be operable (in at least a limited fashion) during fuel handling. Limit answers to <u>support</u> systems only - do <u>NOT</u> include fuel handling systems.

(2.0)

4.10 Give the immediate action(s) required to assure that the reactor is in a safe condition for a relief valve stuck open. Do <u>NOT</u> include verifications. Assume all other plant conditions are normal (i.e., do <u>not</u> include action(s) prefaced by phrases such as "If (parameter) then···"). <u>Include</u> sequential action(s) such as "After (parameter) recovers then···".

(3.0)

**END OF EXAM**

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

FACILITY: COOPER

REACTOR TYPE: GE-BWR

DATE ADMINISTERED: MAY 2, 1984

EXAMINER: J. PELLET

APPLICANT: ______

INSTRUCTIONS TO APPLICANT:

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

| CATEGORY VALUE | % OF TOTAL | APPLICANT'S SCORE | % OF CATEGORY VALUE | CATEGORY |
|---|---|---|---|---|
| 20 | 25 | | | 5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS AND THERMODYNAMICS |
| 20 | 25 | | | 6. PLANT SYSTEMS DESIGN |
| 19.5 | 25 | | | 7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL |
| 19 | 25 | | | 8. ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS |
| 78.5 ~~XXXX~~ | | | | TOTALS |

FINAL GRADE ______%

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

______
APPLICANT'S SIGNATURE

SECTION 5 (20 pts)

5.1. List three reactivity coefficients of concern at Cooper and state at what point in core life (BOL/EOL) each has their most negative value.

(3.0)

5.2. A. Define subcritical multiplication (1.0)

B. Sketch the neutron flux response to a control rod withdrawal during an approach to critical. (1.0)

c. Explain why the neutron flux takes longer to level off as $k_{eff}$ approaches 1. (1.0)

5.3. A control rod worth 0.002 dk/k is continuously withdrawn from position 0 to 48. Assume reactivity is inserted linearly as the rod is withdrawn and that withdrawal takes 50 seconds. What is the transient period when the control rod is at position 36? (Assume BOL conditions)

(2.0)

5.4 Sketch the production and decay chain for Xe-135. List the time to xenon equilibrium at 100% power and the time to xenon peak after a reactor trip from 100% power.

(2.0)

5.5 The reactor is operating at 100% power when one (1) recirculation pump trips. Indicate how each of the parameters listed below would initially change (increase, decrease, or remain the same) and explain briefly why the change occurs.

A. Reactor power (1.0)

B. Reactor water level (1.0)

C. Feedwater flow (1.0)

5.6 Briefly describe how the SRM's may be used to determine water level in the core following a LOCA and state the level range where they may be used.

(2.0)

5.7 With the reactor at 60% rated power, steam header pressure is 945 psig and reactor pressure is 972 psig. What will be the reactor pressure if one (1) MSIV is closed?

(2.0)

CATEGORY CONTINUED ON NEXT PAGE

5.8 Tube flow through a feedwater heater is 4 Mlb/hr of feedwater flow which enters at $150^{o}$ F and exits at $200^{o}$ F. The shell side is supplied with extraction steam at 30 psig which leaves the shell side as drain water at $150^{o}$ F. What extraction steam flow is required?

(2.0)

5.9 Place the flow types listed below in the proper order of occurrence for a BWR reactor.

1. Bubble flow
2. Single phase forced convection
3. Slug flow
4. Subcooled boiling
5. Annular flow

(1.0)

SECTION 6 (20 pts)

6.1 What is the maximum temperature differential allowed between the reactor coolant in the idle and operating loops to permit starting the idle loop? Why is this limit imposed? (Assmue most limiting operating conditions.)

(1.5)

6.2 List three (3) conditions which could threaten the fuel barriers or reactor pressure system boundary that the Reactor Protection system is designed to protect against.

(2.0)

6.3 What are four (4) of the five (5) visual indications which are used to verify that the fuel assembly is properly oriented in the reactor duing fuel handling?

(2.0)

6.4 What is meant by recirculation ratio and what is its value at rated power and recirc flow?

(1.5)

6.5 List three (3) plant protection systems which utilize both reactor pressure and water level.

(1.5)

6.6 Give the four diesel generator trip functions available for an auto start signal.

(2.0)

6.7 What are the requirements for arming the Low-Low Set (LLS)?

(2.0)

6.8 Explain how failure of the inservice pressure controller (A) low will affect actual steam pressure if DEH is in MODE 4 at 100% power and initial steam pressure is 955 psig. Give the new steady-state steam pressure.

(1.5)

6.9 List eight (8) rod blocks which are effective only when the mode switch is in either STARTUP or REFUEL. Include setpoints.

(4.0)

6.10 Explain how and why each of the following would affect indicated reactor level. Assume in all cases a change occurs.

A. Equalizing valve on the level transmitter leaks through.

B. Drywell temperature DECREASES over an extended period of time.

(2.0)

SECTION 7 (19.5 pts)

7.1 When is secondary containment NOT required? (2.0)

7.2 Upon the loss of Instrument Air, at what point does the operator reduce power and scram the reactor? (2.0)

7.3 What neutron monitoring requirements are imposed during core alterations (fuel being moved) during refueling? (1.0)

7.4 True or False? During refueling operations access to the refueling floor will be controlled by a SWP. (0.5)

7.5 List four (4) conditions where it would be necessary to obtain a Special Work Permit (SWP). (2.0)

7.6 Give three (3) conditions which must be met for a jet pump to be considered inoperable (per Technical Specification daily surveillance requirements)? (3.0)

7.7 Give the immediate action(s) required to recover from a reactor scram from power with MSIV's closed. Do NOT include verifications. Assume all other plant conditions are normal (i.e., do not include action(s) prefaced by phrases such as "If (parameter) then···"). Include sequential action(s) such as "After (parameter) recovers then···". (3.0)

7.8 Give the immediate action(s) required to assure that the reactor is in a safe condition for a major loss of feedwater heating (temperature increases more than $60^{\circ}$ F). (1.0)

7.9 Give eight (8) systems which are required by technical specifications to be operable (in at least a limited fashion) during fuel handling. Limit answers to support systems only - do NOT include fuel handling systems. (2.0)

CATEGORY CONTINUED ON NEXT PAGE

7.10 Give the immediate action(s) required to assure that the reactor is in a safe condition for a relief valve stuck open. Do <u>NOT</u> include verifications. Assume all other plant conditions are normal (i.e., do <u>not</u> include action(s) prefaced by phrases such as "If (parameter) then···"). <u>Include</u> sequential action(s) such as "After (parameter) recovers then···".

(3.0)

SECTION 8 (19 pts)

8.1 Fill in the blanks in the paragraph below.
Any event where evacuation of the Control Room is required is basis for implementation of the CNS "__________". Any event requiring evacuation of the Control Room must be classified at least as a(n) "__________".
(Blanks may represent one or more words or numbers.)
(1.0)

8.2 Whose approval is required for an exposure above the whole body yearly administrative limit (for NPPD personnel)? (JOB TITLE)
(1.0)

8.3 Who is required to prepare a Scram Report and under what conditions?
(1.5)

8.4 A. What is the maximum allowable suppression pool temperature during normal operation?
B. What is the maximum allowable suppression pool temperature during testing which adds heat to the suppression pool?
C. What additional restriction(s) apply to the limit in B. above?
(1.5)

8.5 List four (4) scrams which must be operable when the reactor is subcritical, irradiated fuel is in the vessel, and the reactor temperature is less than $212^{O}$ F.
(2.0)

8.6 List five (5) conditions which must be met per Technical Specifications to permit two nonadjacent control rods to be withdrawn from the core in order to perform control rod or control rod drive maintenance.
(3.0)

8.7 A. Who is responsible for filling out the notification of significant events checklist?
B. What time is allowed for notifying the NRC Operations Center if any event occurs as described in 10CFR50.72?
(1.0)

8.8 List eight (8) items to be reviewed per the Shift Supervisors shift turnover log.
(2.0)

CATEGORY CONTINUED ON NEXT PAGE

8.9 A. What requirement is placed on control rods when loading fuel into the core? (non-spiral loading)
B. What requirement is placed on control rods when loading fuel per the spiral reload technique?
(1.5)

8.10 What electrical requirements must be met prior to a startup from cold shutdown?
(3.0)

8.11 A. Who is required to approve a change in the Special Nuclear Material transfer form during a refueling? (JOB TITLE)
B. What is the minimum time after shutdown that irradiated fuel can be handled in or above the reactor?
(1.5)

**END OF EXAM**

CNS REQUAL KEY

SECTION 1 (20 pts)

1.1. List three reactivity coefficients of concern at Cooper and state at what point in core life (BOL/EOL) each has their most negative value.

(3.0)

ANS: 1. Void Coefficient - BOL
2. Doppler Coef. - EOL
3. Moderator Temperature Coef. - BOL
(name - 0.5; B/EOL - 0.5; 3 ans @ 1.0 ea.)

REF: GE BWR Simulator Reactor Physics L. P.

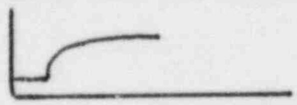
1.2. A. Define subcritical multiplication (1.0)

B. Sketch the neutron flux response to a control rod withdrawal during an approach to critical. (1.0)

c. Explain why the neutron flux takes longer to level off as $k_{eff}$ approaches 1. (1.0)

ANS: A. The process of utilizing source neutrons to maintain a self-sustaining reaction with $k_{eff}$ LT 1.00.

B.



C. $CR= MS = S(1+k+k^2+k^3+\cdots k^n)$

The flux will level off only when $k^n$ approaches 0. This takes more generations as k approaches 1. (formula NOT req'd if text complete - more than above)

REF: GE BWR Simulator Reactor Physics L. P.

1.3. A control rod worth 0.002 dk/k is continuously withdrawn from position 0 to 48. Assume reactivity is inserted linearly as the rod is withdrawn and that withdrawal takes 50 seconds. What is the transient period when the control rod is at position 36? (Assume BOL conditions)

(2.0)

ANS: $\tau = \bar{\beta} - \rho \,/\, \lambda\rho + \frac{\Delta\rho}{\Delta t}$

=(0.0072-0.0015)/(0.1x0.0015+0.002/50)=0.0057/0.00019= 30 sec.
(1.0 for proper formula; 0.5 for constants; 0.5 for math)

REF: GE BWR Simulator Reactor Physics L. P.

1.4 Sketch the production and decay chain for Xe-135. List the time to xenon equilibrium at 100% power and the time to xenon peak after a reactor trip from 100% power.

(2.0)

ANS: $^{135}_{52}Te \xrightarrow{\beta} I \xrightarrow{\beta} Xe \xrightarrow{\beta} Cs \xrightarrow{\beta} Ba$

Xe equil.= about 40 hrs; Xe peak= 8-11 hrs
(chain=1.0; peak=0.5; equil.=0.5)

REF: NRC RTC BWR Systems Manual, p. 1.7-27 to 1.7-30

1.5 The reactor is operating at 100% power when one (1) recirculation pump trips. Indicate how each of the parameters listed below would initially change (increase, decrease, or remain the same) and explain briefly why the change occurs.

A. Reactor power (1.0)
B. Reactor water level (1.0)
C. Feedwater flow (1.0)

ANS: A. Decrease - more voiding adds neg. react.
B. Increase - swell due to more voids & less j.p. flow
C. Decrease - higher level sent to RFPT control sys
(direction 0.33; reason 0.67)

REF: GE BWR Simulator Reactor Physics L. P.

1.6 Briefly describe how the SRM's may be used to determine water level in the core following a LOCA and state the level range where they may be used.
(2.0)

ANS: While inserting SRM's into the core the count rate will rapidly decrease when the detector comes out of the water (due to less n moderation). They may be used from 2.5 ft. below bottom of core to 1.5 ft. above core centerline (# approximate).
(How 1.5; Range 0.5)

REF: GE Degraded Core Recognition Text & CNS SRM L. P.

1.7 With the reactor at 60% rated power, steam header pressure is 945 psig and reactor pressure is 972 psig. What will be the reactor pressure if one (1) MSIV is closed?
(2.0)

ANS: Old dP = 972-945 = 27 psid; New dP = Old x $dAREA^2$ =
27 x 4/3 $^2$= 48
New Pr = 945 + 48 = 993 psig
(formula 1.0; math 1.0)

REF: GE Thermodynamics, Heat Transfer and Fluid Flow Text

1.8 Tube flow through a feedwater heater is 4 Mlb/hr of feedwater flow which enters at 150$^0$ F and exits at 200$^0$ F. The shell side is supplied with extraction steam at 30 psig which leaves the shell side as drain water at 150$^0$ F. What extraction steam flow is required?
(2.0)

ANS: 30 psig = 45 psia; Ms x dh = Mfw x c x dT
Ms = 4 EE06 x 50 / (1172.1-118.0) = 1.9 EE05 lbm/hr
(formula 1.0; steam tables 0.5; math 0.5)

REF: GE Thermodynamics, Heat Transfer and Fluid Flow Text

1.9 Place the flow types listed below in the proper order of occurrence for a BWR reactor.

1. Bubble flow
2. Single phase forced convection
3. Slug flow
4. Subcooled boiling
5. Annular flow

(1.0)

ANS: 2-4-1-3-5
REF: GE Thermodynamics, Heat Transfer and Fluid Flow

SECTION 2 (17 pts)

2.1 Explain why the seal purge flow from the CRD system must be shut off prior to isolating a recirc pump.

(1.5)

ANS: Prevent overpressurized recirc pump/piping on the isolated loop.

REF: CNS Proc 2.1.15, p. 2

2.2 Why must the reactor be shut down if a jet pump is determined to be inoperable per tech specs?

(1.0)

ANS: Increases the blowdown area (during DBA)

REF: CNS TS 3.6.E BASES, p. 150

6.1/
2.3 What is the maximum temperature differential allowed between the reactor coolant in the idle and operating loops to permit starting the idle loop? Why is this limit imposed? (Assume most limiting operating conditions.)

(1.5)

ANS: $50^{o}$ F

To prevent undue stress on the vessel nozzles and bottom head region.

(# 0.5; reason 1.0)

REF: CNS TS 3.6.A, p. 133 & 3.6.A BASES, p. 147

6.2/
2.4 List three (3) conditions which could threaten the fuel barriers or reactor pressure system boundary that the Reactor Protection system is designed to protect against.

(2.0)

ANS:
1. Excessive thermal heat flux.
2. Excessive reactor pressure.
3. Excessive heat energy released to containment (i.e., which must be absorbed by the suppression pool post-accident).

(3 @ 0.666 ea.)

REF: BWR Reactor Operating Fundamentals

2.5 State the basic purpose for installing the LLS Safety/Relief Valve Control system.

(1.5)

ANS: To provide sufficient time between initial and subsequent lifting of any single relief valve for the water level in the discharge piping to drain down to normal level to prevent overpressure conditions during subsequent lifting of a relief valve.

REF: Low-Low Set L. P.

2.6 What is the function of the Standby Liquid Control system and how does it accomplish this function?

(2.0)

ANS: Purpose is to bring the reactor to a full shutdown condition from 100% power independent of the CRD hydraulic system. It accomplishes this by injecting sodium pentaborate ($Na_2B_{10}O_{16} \cdot 10H_2O$) into the reactor vessel.
(1.0 ea. ans)

REF: BWR Operating Fundamentals

2.7 What two variables are controlled to assure no fuel damage if a rod drop accident should occur? What systems or components are used to control these variables?

(2.0)

ANS: Variables - Control rod worth & rod velocity on free fall
Controls - RWM, RSCS; velocity limiter
(var 0.5 ea.; cont. 0.33 ea.)

REF: CRD L. P.

2.8 Give two (2) functions performed by the Transversing In-core Probe (TIP) system?

(1.0)

ANS: Show axial flux @ in-core locations & calibrate the LPRM strings
(2 ans @ 0.5 ea.)

REF: Bwr Reactor Operating Fundamentals

6.3/
2.9 What are four (4) of the five (5) visual indications which are used to verify that the fuel assembly is properly oriented in the reactor duing fuel handling?

(2.0)

ANS:
1. Spring clip at center of four bundle array.
2. Fuel channel spacer buttons located on the inside surface of each four bundle array.
3. Lugs on fuel bundle handles all point to center of array.
4. Bundle serial # can be read from center of array.
5. Core should have cell-to-cell symmetry.

(4/5 ans @ 0.5 ea.)

REF: Core and Fuel Design L. P.

2.10 List the components or structures which make up primary and secondary containment (2 lists required).

(1.0)

ANS: PRI - torus and drywell (with connecting piping, etc.)
SEC - reactor building
(3 ans @ 0.33 ea.)
REF: BWR Reactor Operating Fundamentals

6.4/
2.11 What is meant by <u>recirculation ratio</u> and what is its value at rated power and recirc flow?

(1.5)

ANS: Total core flow / Steam or feedwater flow; about 7.5:1 @ CNS
(ratio 1.0; value 0.5)
REF: CNS Training Manual, Ch. 20, p. 9

SECTION 3 (20.5 pts)

3.1 List the Group IV isolation signals and the actions resulting from this isolation.

(2.5)

ANS: ISOL -
1. HPCI steam line high flow LT/EQ 300% rated
2. HPCI steam line lo press. GT/EQ 100 psig
3. HPCI steam line hi area temp. LT/EQ 200$^{\circ}$ F

ACTS -
1. HPCI steam supply isol vlvs close (MO-15,16)
2. HPCI torus suction vlvs close (MO-58)
3. Exhaust line drain pot vlvs close (AO-70,71)
4. HPCI turbine trip

(isol: 3 ans @ 0.5 ea.; acts: 4 ans @ 0.25 ea.)

REF: CNS TS Table 3.2.A Notes, GOP 2.1.22

6.5/
3.2 List three (3) plant protection systems which utilize both reactor pressure and water level.

(1.5)

ANS: RPS, PCIS, ATWS, & CSCS/ECCS (3 ans @ 0.5 ea.)
REF: BWR Reactor Operating Fundamentals

6.6/
3.3 Give the four diesel generator trip functions available for an auto start signal.

(2.0)

ANS:
1. Incomplete sequence
2. Overspeed (660 rpm - Mechanical)
3. Local manual trip
4. Generator lockout

(4 ans @ 0.5 ea.)

REF: Diesel Generator L. P.

6.7/
3.4 What are the requirements for arming the Low-Low Set (LLS)?

(2.0)

ANS:
1. RV pressure GT scram setpoint (1045 psig)
2. Any one of 8 SRV's open (30 psig tail pipe pressure)

(2 ans @ 1.0 ea.)

REF: LLS L. P.

6.8/
3.5 Explain how failure of the inservice pressure controller (A) low will affect actual steam pressure if DEH is in MODE 4 at 100% power and initial steam pressure is 955 psig. Give the new steady-state steam pressure.

(1.5)

ANS: 'A' failing low causes 'B' to be selected by the HVG & 'B' is normally set 3 psi higher than 'A'. New steam press. is 958 psig.
(# 0.5; expl. 1.0)



REF: DEH System L. P.

3.6 Give the relationship between monitor flow and vent flow in the Kaman system for both the normal and accident range monitors.

(2.0)

ANS: NORMAL - monitor flow is proportional to vent flow
ACC'NT - monitor flow is constant regardless of vent flow
(2 ans @ 1.0 ea.)
REF: Kaman System L. P.

6.8/
3.7 List eight (8) rod blocks which are effective only when the mode switch is in either STARTUP or REFUEL. Include setpoints.

(4.0)

ANS:
1. SRM INOP
2. SRM downscale @ GT/EQ 3 cps
3. SRM high flux @ LT/EQ 10EE05 cps
4. SRM not fully inserted
5. IRM INOP
6. IRM downscale @ GT/EQ 2.5% of scale
7. IRM high flux @ LT/EQ 108/125 scale
8. IRM not fully inserted
9. APRM high flux @ LT/EQ 12%
10. Service platform job crane loaded GT 400 lbs.

(8/10 @ 0.5 ea.)
REF: RBM L. P. Handout

3.8 Give four (4) basic reactor vessel parameters monitored by reactor vessel instrumentation. (Do NOT include core parameters.)

(2.0)

ANS: temperature, level, pressure, & flow (j.p. not recirc/stm/fw)
(4 ans @ 0.5 ea.)
REF: CNS Tr. Man., Ch. 27, p. 2

3.9 Explain why the main portion of the Feedwater Control system is designed to use single element control when operated at low power levels.

(1.0)

ANS: At low feed/steam rates flow instruments are inaccurate and unstable so 3 element mode is bypassed as unstable.
REF: CNS Tr. Man., Ch. 46, p. 4

6.10/
3.10 Explain how and why each of the following would affect indicated reactor level. Assume in all cases a change occurs.
A. Equalizing valve on the level transmitter leaks through.
B. Drywell temperature DECREASES over an extended period of time.

(2.0)

ANS:
A. Indicated level would increase. Reference leg - Variable leg dP would go to zero.
B. Indicated level would decrease. Ref. leg dens. would incr. w/ colder water so ref.-var. dP incr.

(2 ans; 0.25 direction, 0.75 reason)
REF: CNS Tr. Man., Ch. 27, p. 14 - 14a

CNS REQUAL KEY

SECTION 4 (19.5 pts)

4.1 When is secondary containment NOT required? (2.0)

ANS: When all of the follwing are met:
1. Rx subcritical & SDM req'm'nts are met.
2. Rx water temp. LT 212$^{o}$ F & Rx vented.
3. No activity is being performed that could reduce SDM below specified requirements.
4. Irradiated fuel is not being handled in sec. cont.

(4 ans @ 0.5 ea.)

REF: CNS TS 3.7.C.1.

4.2 Upon the loss of Instrument Air, at what point does the operator reduce power and scram the reactor? (2.0)

ANS:
1. More than 1 CR inserts.
2. Air pressure decreases to 50 psig.
3. Unsafe plant conditions caused by loss of air.

(3 ans @ 0.666 ea.)

REF:CNS Proc 5.2.8

4.3 What neutron monitoring requirements are imposed during core alterations (fuel being moved) during refueling? (1.0)

ANS: 2 SRM's OP, 1 in quadrant fuel is being moved & 1 in adjacent

REF: CNS Proc 3.5

4.4 True or False? During refueling operations access to the refueling floor will be controlled by a SWP. (0.5)

ANS: True. Accept FALSE w/ expl. of area around elevator

REF: CNS Proc 3.5, p. 4

4.5 List four (4) conditions where it would be necessary to obtain a Special Work Permit (SWP). (2.0)

ANS:
1. Rad exp. rates GT 100 mr/hr.
2. Airborne activity GT 3EE-9 uci/ml.
3. Gen. contam. levels GT 2200 dpm/100 cm$^2$.
4. Industrial safety cond. hazardous to life.
5. Airborne rad conc. avg'd over 1 wk. GT 25% 10CFR20.

(any 4 ans @ 0.5 ea. - Accept for pass credit w/ no limits)

REF: CNS Proc 9.1.1.4

4.6 Give three (3) conditions which must be met for a jet pump to be considered inoperable (per Technical Specification daily surveillance requirements)?

(3.0)

ANS: 1. Recirc pump flow differs by more than 15% from the established speed - flow characteristics.
2. Indicated core flow rate varies from value derived from loop flow measurements by more than 10%
3. Diffuser to lower plenum DP on an individual jet pump varies from the mean of all jet pump DP's by more than 10%.
(3 ans @ 1.0 ea)

REF: CNS TS 4.6.E.1, p. 137

4.7 Give the immediate action(s) required to recover from a reactor scram from power with MSIV's closed. Do NOT include verifications. Assume all other plant conditions are normal (i.e., do not include action(s) prefaced by phrases such as "If (parameter) then···"). Include sequential action(s) such as "After (parameter) recovers then···".

(3.0)

ANS: 1. Place mode switch in REFUEL.
2. Place mode switch in SHUTDOWN.
3. Start HPCI in the test mode (pressure control/cooldown).
4. Trip one condensate & condensate booster pump.
5. Trip turbine.
6. Acknowledge scram signal.
7. Reset the scram after level recovery.
8. Insert SRM & IRM detectors, change recorders to IRM, & range IRM's on scale.
9. Turn all MSIV switches to close.
10. Close steam inlet valves to SJAE's & trap bypasses on MSIV's.
11. Equalize around MSIV's by opening outer MSIV & drain valves.
12. Open inner MSIV when DP LT 200 psid.
13. Complete scram report.
(1-3,5,8,9 @ 0.4 ea.; 4,6,7,10-13 @ 0.1 ea.)

REF: CNS PROC 2.1.8, p. 2-4

4.8 Give the immediate action(s) required to assure that the reactor is in a safe condition for a major loss of feedwater heating (temperature decreases more than $60^{\circ}$ F).

(1.0)

ANS: Reduce power via recirc flow to at least 20% below the thermal power which existed prior to the reduction in FW heating or minimum flow if reached first (and keep it there by additional flow reductions as necessary).

(1.0)

REF: CNS Proc 2.4.9.4.7, p. 2

4.9 Give eight (8) systems which are required by technical specifications to be operable (in at least a limited fashion) during fuel handling. Limit answers to <u>support</u> systems only - do <u>NOT</u> include fuel handling systems.

(2.0)

ANS:
1. CS 2. RHR
3. SW 4. RHR SW
5. FP 6. ARMS
7. SBGT 8. RB HVAC
9. FPC 10. CR Emer. Fan
11. SLC 12. RBCCW
13. RPS 14. Condensate M/U
15. DG 16. Neutron Monitoring system (SRM)

(any 8/16 @ 0.25 ea.)

REF: CNS Proc 3.5, p. 7

4.10 Give the immediate action(s) required to assure that the reactor is in a safe condition for a relief valve stuck open. Do <u>NOT</u> include verifications. Assume all other plant conditions are normal (i.e., do <u>not</u> include action(s) prefaced by phrases such as "If (parameter) then···"). <u>Include</u> sequential action(s) such as "After (parameter) recovers then···".

(3.0)

ANS:
1. Attempt to close RV by panel switch.
2. Reset ADS logic (via pushbuttons).
3. Pull fuses in Aux Relay Room (& replace when valve closes).
4. Shut down as follows:
   a. Reduce power as fast as possible until @ a low power
   b. Notify load dispatcher of the problem and anticipated reactor scram
   c. Transfer station electrical loads
   d. Manually scram reactor, trip the turbine, and open the field breaker
5. Use HPCI/RCIC for RV level M/U
6. Depressurize reactor to LT 75 psig
7. Start 1 loop of RHR in S/D cooling mode
8. Start 1 loop of RHR in torus cooling mode

(1-3 @ 0.67 ea.; 4a-d @ 0.167 ea.; 5-8 @ 0.08 ea.)

REF: CNS Proc 2.4.2.3.1, p. 2-4

CNS REQUAL KEY

SECTION 5 (20 pts)

Section 5 is a duplicate of section 1. Grade per key on pages 1 - 3.

SECTION 6 (20 pts)

Section 6 is compiled from sections 2 & 3 which appear on pages 4 - 8 of this key. The question cross-reference is given below. Grade per the key on the above pages.

| | | |
|---|---|---|
| 6.1 - 2.3 | 6.5 - 3.2 | 6.9 - 3.7 |
| 6.2 - 2.4 | 6.6 - 3.3 | 6.10 - 3.10 |
| 6.3 - 2.9 | 6.7 - 3.4 | |
| 6.4 - 2.11 | 6.8 - 3.5 | |

SECTION 7 19.5 pts)

Section 7 is a duplicate of section 4. Grade per key on pages 9- 11.

SECTION 8 (19 pts)

8.1 Fill in the blanks in the paragraph below.
Any event where evacuation of the Control Room is required is basis for implementation of the CNS "___________". Any event requiring evacuation of the Control Room must be classified at least as a(n) "___________".
(Blanks may represent one or more words or numbers.)

(1.0)

ANS: Emergency Plan (accept similar)
ALERT
(2 ans @ 0.5 ea.)
REF: CNS Proc 5.2.1, p. 4

8.2 Whose approval is required for an exposure above the whole body yearly administrative limit (for NPPD personnel)? (JOB TITLE)

(1.0)

ANS: Chem/HP Supervisor
Mgr., Div. Nuc. Ops.
(CHPS 0.25; DMNO 0.75)
REF: CNS Proc 9.1.2.1, p. 5

8.3 Who is required to prepare a Scram Report and under what conditions?

(1.5)

ANS: WHO: Shift Supervisor
WHEN: After all scrams when
fuel is in the vessel and
more than one rod is withdrawn.
(who 0.5; when 3 ans @ 0.333 ea.)
REF: CNS Proc 1.4.6.1.2, p. 9

8.4 A. What is the maximum allowable suppression pool temperature during normal operation?
B. What is the maximum allowable suppression pool temperature during testing which adds heat to the suppression pool?
C. What additional restriction(s) apply to the limit in B. above?

(1.5)

ANS: A. $95^o$ F B. $+10^o$ F / $105^o$ F
C. Reduce below normal temperature limit within 24 hours.
(A & B 0.25 ea.; C 1.0)
REF: CNS TS 1.0

8.5 List four (4) scrams which must be operable when the reactor is subcritical, irradiated fuel is in the vessel, and the reactor temperature is less than 212$^{o}$ F.

(2.0)

ANS: 1. Mode switch in S/D.
2. Manual P/B.
3. High flux IRM (120/125)
4. APRM high flux (15%)
(4 ans @ 0.5 ea.)

REF: CNS TS Table 3.1.1 Notes

8.6 List five (5) conditions which must be met per Technical Specifications to permit two nonadjacent control rods to be withdrawn from the core in order to perform control rod or control rod drive maintenance.

(3.0)

ANS: 1. Mode switch in REFUEL.
2. All refuel interlocks operable except 1 rod with. byp.
3. SDM/Reactivity limits met.
4. 2 rods separated by GT 2 cells in any direction.
5. Appropriate # SRM's operable/in service.
(5 ans @ 0.6 ea.)

REF: CNS TS 3.10.A.5

8.7 A. Who is responsible for filling out the notification of significant events checklist?
B. What time is allowed for notifying the NRC Operations Center if any event occurs as described in 10CFR50.72?

(1.0)

ANS: A. Shift Communicator
B. 1 hour
(2 ans @ 0.5 ea.)

8.8 List eight (8) items to be reviewed per the Shift Supervisors shift turnover log.

(2.0)

ANS: Safety System Status Panel, Inop. equip. status board and req'd surv. performed, Containment status board, MWR's in progress, Clearance orders open, Control room operators log, annunciators in alarm status, key control log, jumper log, valve seal log, any abnormal ind. from panel walkdown (e.g., abnormal lineup or out of service)
(any 8 ans @ 0.25 ea.)

REF: CNS Proc 1.4

. . .



8.9 A. What requirement is placed on control rods when loading fuel into the core? (non-spiral loading)

B. What requirement is placed on control rods when loading fuel per the spiral reload technique?

(1.5)

ANS: A. All CR must be fully inserted

B. The control room operator, a licensed operator, & a member of the reactor engineering staff on the refueling floor shall verify that the CR is fully inserted in the cell to be loaded.

(A 0.5; B 1.0)

REF: CNS TS 3.10.A.2 & 4.10.A.5

8.10 What electrical requirements must be met prior to a startup from cold shutdown?

(3.0)

ANS: 1. 345 KV, 69 KV, S/U xformer, & Emergency xformer available to 4160 V emerg. buses 1F & 1G.

2. Both DG oper. w/ 45,000 gal fuel oil in tanks.

3. 4160 V 1F & 1G energized, 480 V 1F & 1G energized.

a. Loss of voltage & aux. relays oper.

b. UV & aux relays oper.

4. All 4 125/250 VDC batteries w/ chg. oper.

5. Power mon. sys. for MG set (or alt.) oper.

(5 ans @ 0.6 ea.)

REF: CS TS 3.9.A

8.11 A. Who is required to approve a change in the Special Nuclear Material transfer form during a refueling? (JOB TITLE)

B. What is the minimum time after shutdown that irradiated fuel can be handled in or above the reactor?

(1.5)

ANS: A. SRO/SS & SRO (both required)

B. 24 hours

(2 ans @ 0.75 ea.)

REF: CNS TS 3.10.D