August 27, 1999

Carolina Power & Light Company
ATTN: Mr. D. E. Young
Vice President
H. B. Robinson Steam Electric Plant Unit 2
3581 West Entrance Road
Hartsville, SC 29550

SUBJECT: NRC EXAMINATION REPORT NO. 50-261/99-301

Dear Mr.Young:

On July 26 - 29, 1999, the Nuclear Regulatory Commission (NRC) administered examinations to employees of your company who had applied for licenses to operate the H.B.Robinson Steam Electric Plant Unit 2. At the conclusion of the examination, the examiners discussed the examination questions and preliminary findings with those members of your staff identified in the enclosed report.

A Simulation Facility Report is included in this report as Enclosure 2. Enclosure 3 is the Facility Post-Examination Comments. Enclosure 4 is the NRC Resolution of post-examination comments. A copy of the written examination questions and answer key, as noted in Enclosure 5, was provided to the members of your training staff at the conclusion of the examination.

All four of the senior reactor operators (SRO) passed the written and operating examinations. All three of the reactor operator (RO) applicants failed the written examination. One of the RO applicants failed the operating examination as well. We recommend that your staff review the individual examination reports to determine if adjustments to the training program, as well as individual remediation, are needed.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

Should you have any questions concerning this letter, please contact me at (404) 562-4638.

Sincerely,

Original signed by
Harold o. Christensen

Harold O. Christensen, Chief
Operator Licensing and
Human Performance Branch
Division of Reactor Safety

Docket No. 50-261
License No. DPR-23

Enclosures:
1. Report Details
2. Simulation Facility Report
3. Facility Recommendations
4. NRC Resolution of Recommendations
5. Written Examinations and Answer Keys (SRO / RO) (Document Control Desk Only)

cc w/encls:
J. W. Moyer
Director, Site Operations
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Electronic Mail Distribution

Plant General Manager
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Electronic Mail Distribution

Terry C. Morton, Manager
Performance Evaluation and
Regulatory Affairs CPB 9
Electronic Mail Distribution

H. K. Chernoff, Supervisor
Licensing/Regulatory Programs
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Electronic Mail Distribution

R. L. Warden, Manager
Regulatory Affairs
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Electronic Mail Distribution

Virgil R. Autry, Director
Div. of Radioactive Waste Mgmt.
Dept. of Health and Environmental Control
Electronic Mail Distribution

R. Mike Gandy
Division of Radioactive Waste Mgmt.
S. C. Department of Health and Environmental Control
Electronic Mail Distribution

Mel Fry, Director
Division of Radiation Protection
N. C. Department of Environment, Health and Natural Resources
Electronic Mail Distribution

William D. Johnson
Vice President & Corporate Secretary
Carolina Power & Light Company
Electronic Mail Distribution

John H. O'Neill, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N. Street, NW
Washington, DC 20037-1128

Peggy Force
Assistant Attorney General
State of North Carolina
Electronic Mail Distribution

Robert P. Gruber
Executive Director
Public Staff - NCUC
P. O. Box 29520
Raleigh, NC 27626-0520

Public Service Commission
State of South Carolina
P. O. Box 11649
Columbia, SC 29211

A. Williams
Manager - Training
H. B. Robinson Steam Electric Plant
Unit No. 2
3581 West Entrance Road
Hartsville, SC 29550

Distribution w/encl: (See page 4)

Distribution w/encl:
R. Subbaratnam, NRR
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# U. S. NUCLEAR REGULATORY COMMISSION

## REGION II

| | |
|---|---|
| Docket No.: | 50-261 |
| License No.: | DPR-23 |
| Report No.: | 50-261/99-301 |
| Licensee: | Carolina Power and Light Company |
| Facility: | H. B. Robinson Steam Electric Plant |
| Location: | 3581 West Entrance Road<br>Hartsville, SC 29550 |
| Dates: | July 23 - 29, 1999 |
| Examiners: | Michael E. Ernstes, Chief License Examiner<br>Larry Mellen, License Examiner<br>Paul Steiner, License Examiner |
| Approved by: | Harold O. Christensen, Chief<br>Operator Licensing and Human<br>Performance Branch<br>Division of Reactor Safety |

Enclosure 1

# EXECUTIVE SUMMARY

H. B. Robinson Steam Electric Plant
NRC Examination Report Nos. 50-261/99-301

During the period July 26-29, 1999, NRC examiners conducted an announced operator licensing initial examination in accordance with the guidance of Examination Standards, NUREG-1021, Interim Revision 8. This examination implemented the operator licensing requirements of 10 CFR §55.41, §55.43, and §55.45.

Four senior reactor operator applicants and three reactor operator applicants received written examinations and operating tests. The NRC administered the operating tests during the week of July 26, 1999. Your staff administered the written examination on July 23, 1999.

Operations

- The as-submitted written examination and operating tests met the guidelines of NUREG-1021. (Section O5.1)

- Four of seven (57%) applicants passed the examination. All of the SRO applicants passed. All of the RO applicants failed the written examination. One of the RO applicants failed the operating test as well. (Section O5.1)

## Report Details

Summary of Plant Status

Unit 2 was at 100% power during the exams. One of the applicants inadvertently shut off the 'D' Station Air Compressor in the plant while taking his walkthrough examination. Shift operators immediately responded and restarted the compressor with little effect on the plant.

## I. Operations

### O5 Operator Training and Qualifications

O5.1 Initial Licensing Examinations

a. Scope

NRC examiners conducted regular, announced operator licensing initial examinations during the week of July 26, 1999. The examiners administered examinations developed by members of the H.B.Robinson Steam Electric Plant Unit 2 training staff under the requirements of an NRC security agreement, in accordance with the guidelines of the Examination Standards (ES), NUREG-1021, Revision 8. Four Senior Reactor Operators (SRO) and three Reactor Operators (RO) license applicants received written examinations and operating tests.

b. Observations and Findings

(1) Written Examination

Your staff submitted 127 multiple choice written questions for NRC review. The RO and SRO written examinations shared 73 questions with 27 additional questions for each test that were license level specific.

Your exam authors were in frequent contact with the NRC chief examiner for guidance on the examination development process. Participation of the Manager of Operations in the examination review also greatly aided in developing a valid examination. In general, the submitted examinations were well written in that they were technically accurate and offered plausible distractors. The chief examiner made changes to a relatively small number of questions to meet the guidelines of NUREG-1021.

All three RO applicants failed the written examination. All four SRO applicants passed.

After the exam, your staff recommended a change to the answer key for four of the written examination questions. We accepted two of these recommendations. Resolution of these comments is described in Attachment 4.

(2) Operating Tests

Your staff submitted three simulator scenarios for NRC review. The simulator tests followed the guidelines of NUREG-1021. The walkthrough examination sets submitted by your staff contained job performance measures (JPMs) that generally met the NUREG-1021 guidelines. A substitution was made for one JPM that was similar to one of the scenario events. Two JPMs were modified by the NRC examiner to add more meaningful evaluation. These modifications were due to a change in NUREG-1021, Revision 8 which required the JPMs to have sufficient performance requirements to evaluate the applicants knowledge without the use of prescripted questions. This change is described in section D.3.b of ES-301.

All applicants passed the simulator portion of the operating test. One RO failed the walkthrough portion while the other six passed. All of the RO applicants demonstrated a lack of familiarity with the normal operating temperature of the Pressurizer Relief Tank (PRT) by failing to take steps to cool the PRT during a JPM.

c. <u>Conclusions</u>

The quality and level of difficulty of the licensee's examination submittal met the guidelines of NUREG 1021. Four of seven applicants passed the examination. All three RO applicants failed the written examination and one RO applicant failed the operating test as well. Detailed applicant performance comments were transmitted under separate cover for management review and to allow appropriate applicant remediation.

## V. Management Meetings

### X1 Exit Meeting Summary

An exit interview was conducted on July 29, 1999, to reiterate the purpose of the site visit and to discuss the findings. The licensee had no comments and the examiner received no dissenting comments. No proprietary information was received.

## PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee</u>
*T. Cleary, Manager, Operations
*T. Natale, Manager, Operations Training
*A. Williams, Manager, Training
*D. Young, Vice President, Robinson Nuclear Plant

<u>NRC</u>
*A. Hutto, Resident Inspector

*Attended Exit Interview

## ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>
None

<u>Closed</u>
None

<u>Discussed</u>
None

## PARTIAL LIST OF PERSONS CONTACTED

Licensee
*T. Cleary, Manager, Operations
*T. Natale, Manager, Operations Training
*A. Williams, Manager, Training
*D. Young, Vice President, Robinson Nuclear Plant

NRC
*A. Hutto, Resident Inspector

*Attended Exit Interview

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened
None

Closed
None

Discussed
None

# SIMULATION FACILITY REPORT

Facility Licensee: H.B. Robinson Steam Electric Plant

Facility Docket No.: 50-261

Operating Tests Administered on: July 26 - 29, 1999

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information that may be used in future evaluations. No licensee action is required in response to these observations.

While conducting the simulator portion of the operating tests, the following items were observed (if none, so state):

| <u>ITEM</u> | <u>DESCRIPTION</u> |
|---|---|
| None | |

# ENCLOSURE 3

# FACILITY RECOMMENDATIONS

**CP&L**

**Carolina Power & Light Company**
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

Serial: RNP-RA/99-0150

AUG 5 1999

Mr. Luis A. Reyes
Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, S. W., Suite 23T85
Atlanta, Georgia 30303-8931

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

POST EXAMINATION COMMENTS FOR INITIAL OPERATOR
LICENSE EXAMINATIONS ADMINISTERED DURING JULY 1999

Dear Mr. Reyes:

The Attachment to this letter provides comments on the initial NRC license examination administered from July 23 - 29, 1999 at H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The enclosed comments are submitted in accordance with NUREG-1021, Revision 8, "Operator Licensing Examiner Standards," ES, Section 402, "Administering Written Examinations at Power Reactors," which requires a restatement of the question, answer, reference, comment and a recommendation for resolution of the comment, and supporting references for that resolution.

If you have any questions concerning this matter please contact Mr. H. K. Chernoff.

Sincerely,

James Harrison for

R. L. Warden
Manager - Regulatory Affairs

PMY/pmy
Attachment

c: Document Control Desk
NRC Resident Inspector
R. Subbaratnam, NRC, NRR
M. Ernstes, NRR, HOHB

United States Nuclear Regulatory Commission
Attachment to Serial: RNP-RA/99-0150
29 Pages

# H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 POST EXAMINATION COMMENTS FOR INITIAL OPERATOR LICENSE EXAMINATIONS ADMINISTERED DURING JULY 1999

JPM - 009a ; Response to Power Range N-44 Failure and Remove from Service

SRO question # 3, RO Question # 1

SRO question # 75, RO Question # 73

SRO Question # 88, RO Question # 92

SRO Question #94, RO Question # 96

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
POST EXAMINATION COMMENTS FOR INITIAL OPERATOR
<u>LICENSE EXAMINATIONS ADMINISTERED DURING JULY 1999</u>

A. Copy of pages from JPM CR-009.a including Step 24.
B. Recommendation for Step 24
C. References used in recommendation

# REGION II
# LICENSE EXAMINATION
# JOB PERFORMANCE MEASURE

## JPM CR-009.a

Respond to Power Range N-44 Failure
and Remove From Service

**CANDIDATE** ______________________________

**EXAMINER** ______________________________

Approved By: ______________________ Date: ____________

| | |
|---|---|
| STEP 22: ROD STOP BYPASS Switch: BYPASS PR 44 (9th Step)<br><br>STANDARD: On the Miscellaneous Control & Indication Panel , the operator places the ROD STOP BYPASS Switch to the BYPASS PR 44 position.<br><br>COMMENTS: | **CRITICAL STEP**<br><br>____ SAT<br><br>____ UNSAT |
| STEP 23: COMPARATOR CHANNEL DEFEAT Switch: SELECT PR 44 (10th Step)<br><br>STANDARD: On the Miscellaneous Control & Indication Panel, the operator places the COMPARATOR CHANNEL DEFEAT Switch to the SELECT PR 44 position<br><br>COMMENTS: | **CRITICAL STEP**<br><br>____ SAT<br><br>____ UNSAT |
| STEP 24: DETECTOR CURRENT COMPARATOR DRAWER: UPPER and LOWER SECTION Switch: SELECT PR 44*** (11th Step)<br><br>STANDARD: On the DETECTOR CURRENT COMPARATOR DRAWER, the operator selects PR 44 with the Upper and Lower Section switches.<br><br>COMMENTS: | **CRITICAL STEP**<br><br>____ SAT<br><br>____ UNSAT |
| STEP 25: NI-44 INSTRUMENT POWER FUSES**: REMOVED (12th Step)<br><br>STANDARD: Operator determines this step is not required.<br><br>EXAMINER'S NOTE: **This action is N/A if power is > P-10 or the reactor is in MODES 3 through 6 (ITS Table 3.3.1-1)**<br><br>COMMENTS: | ____ SAT<br><br>____ UNSAT |

Recommendation for Step 24

JPM - 009.a ; Respond to Power Range N-44 Failure and Remove From Service

Administered to the RO and SRO Instant Candidates

Step 24 of the JPM is a critical step that requires the operator to place the Upper and Lower Section switches on the Detector Current Comparator Drawer to the N44 position.

Based on the note annotated by the three asterisks (***) for this step, and feedback from Reactor Engineering it is acceptable to not reposition these switches as directed by OWP-11 if the operator first conducts a verification that the QPTR input from the affected Power Range detector is operable. It is requested, that if the student performed a verification that the input to QPTR was still operable that it be acceptable to not place the Upper and Lower Section switches to the N44 position.

Based on Operations Management expectations it is also acceptable based on the conditions of this failure to position the switches to the N44 position as directed by OWP-11.

CP&L

C
Continuous
Use

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3
PART 10

OPERATIONS WORK PROCEDURE

# OWP-011

# *NUCLEAR INSTRUMENTATION (NI)*

REVISION 12

VALVE, BREAKER, SWITCH LINEUP

| COMPONENT DESCRIPTION | POSITION FOR MAINTENANCE | INIT | VERI | RESTORED POSITION | INIT |
|---|---|---|---|---|---|
| POWER RANGE CHANNEL NI-44 | | | | | |
| ROD STOP BYPASS Switch | BYPASS PR 44 | ____ | | OPERATE | ____ |
| COMPARATOR CHANNEL DEFEAT Switch | SELECT PR 44 | ____ | | NORMAL | ____ |
| DETECTOR CURRENT COMPARATOR Drawer: | | | | | |
| UPPER SECTION Switch | SELECT PR 44 | ____ | *** | NORMAL | ____ |
| LOWER SECTION Switch | SELECT PR 44 | ____ | *** | NORMAL | ____ |
| NI-44 INSTRUMENT POWER FUSES ** | REMOVED | ____ | *____ | INSTALLED | ____ |
| Bistable Light LOW POW RANGE HI FLUX NC44P ** | ILLUM | ____ | | EXTNG | ____ |

* IF Bistable Light is illuminated PRIOR to tripping Bistable Switch, THEN perform an independent verification of Bistable Switch position.

** This action is N/A if Power is greater than P-10 OR the reactor is in MODES 3 through 6 (ITS Table 3.3.1-1).

*** Only required to be performed if the Power Range Neutron Flux input to QPTR is inoperable.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
POST EXAMINATION COMMENTS FOR INITIAL OPERATOR
LICENSE EXAMINATIONS ADMINISTERED DURING JULY 1999

SRO question #3, RO question #1

A. Copy of RO question #1, answer and references.
B. Recommendation for questions
C. References for recommendations

*

1. Given the following plant conditions:

- Control Rod H-8 from Control Bank "D" (CBD) has dropped into the core
- A runback has occurred and the operators have stabilized the plant at 67% RTP
- CBD @188 steps
- The operators are preparing to recover rod H-8

Which ONE (1) of the following describes the operability of Control Rod H-8 at this time?

The rod is considered:

A. operable because it can be moved by it's mechanism.

B. operable because it is providing the assumed reactivity that would be available upon a reactor trip.

C. inoperable because it is not trippable.

D. inoperable because it is more than 7.5 inches out of alignment with it's bank.

Recommendation for questions.
Question #1 - RO
Question #3 - SRO

**Choice "B" should be the correct answer.**

Choice "A" - the control rod is operable because it is trippable, but based on the question stem, the rod has not been moved. The operability of the control rod is not determined by the ability of it's mechanism to move the rod. Therefore, Choice "A" is incorrect.

Choice "B" - the control rod is operable because it is trippable. The control rod is providing the negative reactivity that would be available following a reactor trip. In this situation, the dropped control rod inserted negative reactivity into the core that was balanced by the positive reactivity insertion from the power reduction. The reduction in power has created a state point where the Power Defect is less. So, during a trip of the reactor, the rods need to overcome less of the Power Defect to provide for an adequate shutdown margin.

Choice "C" - For a control rod to be inoperable, per Technical Specifications this means that the control rod is not trippable. The control rod in this case is obviously trippable since it is fully inserted into the core. Therefore, Choice "C" is incorrect.

Choice "D" - a control rod is not inoperable if out of alignment with the rest of the control bank. The rod is misaligned, and as long as the rod is trippable then it is operable. Therefore, Choice "D" is incorrect

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be as follows:

a. For bank demand positions ≥ 200 steps, each rod shall be within 15 inches of its bank demand position, and

b. For bank demand positions < 200 steps, each rod shall be within 7.5 inches of the average of the individual rod positions in the bank.

APPLICABILITY: MODES 1 and 2.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more rod(s) inoperable. | A.1.1 Verify SDM is within the limits provided in the COLR.<br><br>OR<br><br>A.1.2 Initiate boration to restore SDM to within limit.<br><br>AND<br><br>A.2 Be in MODE 3. | 1 hour<br><br>1 hour<br><br>6 hours |

(continued)

## B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are described in the UFSAR (Ref. 1) and 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups

(continued)

BASES

BACKGROUND (continued)

that are moved in a staggered fashion, but always within one step of each other. HBRSEP has four control banks and two shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues for the remaining control banks. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Analog Rod Position Indication (ARPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The ARPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six steps.

(continued)

BASES

BACKGROUND (continued)

The maximum uncertainty of the ARPI System is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and ARPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches (Ref. 4 and 6).

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

There be no violations of:

a. specified acceptable fuel design limits, or
b. Reactor Coolant System (RCS) pressure boundary integrity.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted in excess of its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat generation rates (LHGRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

LCO

CHOICE "B"

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability to meet SDM) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required time on a valid signal. CRDM malfunctions that result in inability to move a rod (e.g., rod urgent failures), which do not impact trippability, do not necessarily result in rod inoperability.

CHOICE "C" CAN NOT BE CORRECT BECAUSE ROD FULLY INSERTED

The requirement to maintain the rod alignment to within the specified limits is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in

(continued)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
POST EXAMINATION COMMENTS FOR INITIAL OPERATOR
LICENSE EXAMINATIONS ADMINISTERED DURING JULY 1999

SRO question #75, RO question #73

A. Copy of RO question # 73, answer and references.
B. Recommendation for questions
C. References for recommendations

*

73. Given the following plant conditions:

- Mode 1 at 100% RTP
- Vacuum pump "A" is running
- Vacuum pump "B" is selected to AUTO

Which ONE (1) of the following describes the correct operation of the vacuum pumps?

A. At 25.5 inches Hg decreasing, "B" automatically starts and at 27.0 inches Hg increasing, "B" automatically stops.

B. At 25.5 inches Hg decreasing, "B" automatically starts and "B" must be manually stopped and returned to AUTO.

C. All running pumps will shift to "hogging" mode at 25 inches Hg decreasing.

D. All running pumps will shift to "jetting" mode at 27 inches Hg increasing.

Recommendation for questions.
Question #73 - RO
Question #75 - SRO

**Choice "B" should be the correct answer.**

Choice "A" - the vacuum pumps will automatically start at 25.5" but will not automatically stop at 27". This is the reset point which allows the pumps to be manually secured. Therefore, Choice "A" is incorrect.

Choice "B" - the vacuum pumps will automatically start at 25.5" . Once the reset point of 27" is reached, this will allow the pumps to be manually secured. Details from the system engineer support that the vacuum pumps must be manually secured after the reset point of 27" is reached.

Choice "C" - All pumps will shift to the hogging mode at 23.5" vice 25". Therefore, Choice "C" is incorrect.

Choice "D" - All pumps will shift to the jetting mode at 25" vice 27". Therefore, Choice "D" is incorrect.

4.7 Seal Water Booster Pumps

The seal water booster pumps are normally under automatic control of differential pressure switch (DPS-1447), there is a local panel located at the pumps. This panel provides for selecting the pump (A or B) to the RUN position and places the other pump in a standby or backup mode. Also located on this panel are individual start switches for the pumps. If a pump is selected to run, but fails to start automatically, the local start button can be pushed to start the pump. If the start button is pushed and differential pressure switch (DPS-1477) does not call for them to be running, the pump will stop after the button is released. The pump will continue to run as long as the start button is depressed.

DPS-1447 measures DP between booster pump discharge and MFW pump suction. At 30 psid decreasing the lead pump will auto start after a 30 second delay, the lag pump will auto start after 40 second delay. At 100 psid increasing the pump will stop.

Each pump has a local indication of discharge pressure and also a low suction pressure switch (PSL-1475-1 for pump A and PSL-1475-2 for pump B) which prevents the pump from starting if there is not sufficient suction pressure for them.

4.8 Polisher Secondary Bypass AOV, QCV-10426

- Key lock switch on Polisher Panel with three positions:

  CLOSE - will close valve if no open signals present and RTGB switch is in LOCAL

  AUTO - auto signals for opening valve will function

  - interlocked with HCV-1459, LP Heater Bypass valve, to open if HCV-1459 opens

  OPENS - opens valve

- RTGB switch has LOCAL position to provide local control at the Polisher Panel and OPEN position to open valve

4.9 Vacuum Pumps

During initial phase of drawing a vacuum both pumps are running in the "hogging" mode. At approximately 25" Hg Vac both pumps, via automatic valve operation, will swap over to the "jetting" mode. At normal operating vacuum one pump may be stopped and placed in standby. If vacuum decreases the standby pump would auto start at ~ 25.5" Hg Vac and reset at 27" Hg Vac increasing vacuum. At 23.5" decreasing the running pumps would automatically swap to "hogging" mode.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
POST EXAMINATION COMMENTS FOR INITIAL OPERATOR
LICENSE EXAMINATIONS ADMINISTERED DURING JULY 1999

SRO question #88, RO question #92

A. A. Copy of RO question #92, answer and references.
B. Recommendation for questions
C. References for recommendations

92. Given the following plant conditions:

- Mode 3, after a trip that occurred 5 hours ago
- Pre-startup preparations are in progress, criticality scheduled for 8 hours from now
- Reactor trip breakers are open
- An Estimated Critical Condition has been prepared for the startup
- ECC RCS boron is 670 ppm
- Mode 3 SDM is 720 ppm
- Present RCS boron concentration is 680 ppm

Which ONE (1) of the following describes the required action to take to adjust RCS boron concentration?

A. Borate to 720 ppm, then withdraw SD Bank "A"

B. Withdraw SD Bank "A", then borate to 720 ppm

C. Dilute to 670 ppm, then withdraw SD Bank "A"

D. Withdraw SD Bank "A" then dilute to 670 ppm

Recommendation for questions
Question #92 - RO
Question #88 - SRO

**Choices "A" should be the correct answer.**

Choice "A" - based on the plant conditions given in the stem, the operators would transition from EPP-4, Reactor Trip Response, to GP-004, Post Trip Stabilization. At this time, all control rods are fully inserted. Step 8.36 of GP-004 is the first procedure guidance provided with respect to boration or rod motion. This step requires boration to Mode 3 if the reactor startup will be greater than 12 hours after trip (question stem provides information indicating greater than 12 hours). Therefore, the correct sequence of events would be to borate then withdraw Shutdown Bank "A".

Choice "B" - this is the reverse order of Choice "A" and is not allowed by the procedures based on plant conditions provided in the question. Therefore, Choice "B" is incorrect.

Choice "C" - this is the reverse order of Choice "D" and is not allowed by the procedure to dilute prior to withdrawal of Shutdown Bank "A". Therefore, Choice C" is incorrect.

Choice "D" - would be correct if you were in GP-003, Normal Plant Startup from Hot Shutdown to Critical, and actually preparing to take the reactor critical. In this situation you would then ensure Shutdown Bank "A" withdrawn and then dilute to ECC boron concentration. Therefore, Choice "D" is incorrect.

INIT

8.34 Perform Attachment 10.1 to reset the SR HI FLUX AT SHUTDOWN Alarm. ____

8.35 Monitor the neutron flux level **AND** count rate as the Xenon transient progresses to verify that core flux multiplication does not result in criticality. ____

**NOTE:** If the count rate on either Source Range channel increases by a factor of two or more during any step involving a Boron concentration change, the operation shall be stopped immediately and suspended until a satisfactory evaluation of the situation has been made.

The COLR identifies the required Shutdown Margin (SDM) based on plant conditions. The required Boron Concentration can be determined using Powertrax or the Plant Curve Book using the SDM identified in the COLR.

The following step is performed to ensure the plant remains in MODE 3 after Xenon decay.

8.36 **IF** the time to return to critical is in excess of 12 hours **AND** a plant cooldown is **NOT** required, **THEN** borate the RCS to achieve a Boron Concentration which is greater than or equal to the Boron Concentration needed to provide the required SDM for MODE 3 with Shutdown Bank "A" fully withdrawn. ____ CHOICE "A"

**NOTE:** The slide switches for enabling/disabling the print function of points on NR-53, EXCORE NIS RECORDER, are located on the rear of the recorder chassis.

8.37 Enable the print function of the following points on NR-53, EXCORE NIS RECORDER, by removing the related slide switches from the BYPASS position. (ACR 93-329)

- PT. 1 CH I SR CPS (slide switch #1) ____
- PT. 3 CH II SR CPS (slide switch #3) ____

8.38 Initiate makeup to the CST, as required, to maintain level greater than 50%. ____

8.39 Place the desired number of Condensate Polisher Beds in service using OP-501. ____

8.40 Verify the MSC DRAIN COLLECTING TANK level is normal. ____

8.41 Verify the Bowser Filter Unit is operating correctly. ____

## 3.0 RESPONSIBILITIES

N/A

## 4.0 PREREQUISITES

4.1 The EOP Network has been exited

## 5.0 PRECAUTIONS AND LIMITATIONS

5.1 The shutdown margin existing at the time the Reactor is shutdown shall not be reduced during the subsequent Xenon transient. This margin is determined by the Boron concentration and withdrawn rods which can be tripped. Positive reactivity addition due to Xenon decay occurs approximately 12 hours after shutdown.

CHOICE "B"

5.2 Shutdown Bank "A" shall be at the fully withdrawn position whenever reactivity is being changed by Boron or Xenon changes, RCS temperature changes, or Control Rods, other than Shutdown Bank "A". The following exceptions to this rule may be applied:

| |
|---|
| **NOTE:** The COLR identifies the required Shutdown Margin (SDM) based on plant conditions. The required Boron Concentration can be determined using Powertrax or the Plant Curve Book using the SDM identified in the COLR. |

5.2.1 The RCS has been borated and confirmed by sampling, to be at least at the Boron Concentration needed to provide the required SDM for MODE 3 and is being maintained at MODE 3. Approval of the Manager - Operations, or his designated alternate, shall be given for Shutdown Bank "A" to be inserted.

5.2.2 The RCS has been borated and confirmed by sampling, to be at least at the Boron Concentration needed to provide the required SDM for MODE 5. Approval of the Manager - Operations, or his designated alternate, shall be given for Shutdown Bank "A" to be inserted.

5.2.3 If Shutdown Bank "A" cannot be withdrawn, the RCS shall be borated as required IAW 5.2.1 or 5.2.2.

5.3 Following a significant (10 ppm or more) change in RCS Boron concentration, additional PZR heaters should be energized. This will permit opening of the PZR spray valves and allow the Boron concentration between the PZR and the RCS loops to equalize.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
POST EXAMINATION COMMENTS FOR INITIAL OPERATOR
LICENSE EXAMINATIONS ADMINISTERED DURING JULY 1999

SRO question # 94, RO question # 96

A. Copy of RO question #96, answer and references.
B. Recommendation for questions
C. References for recommendations

96. Given the following plant conditions:

- Mode 6
- A CV purge is being established per OP-921, CONTAINMENT AIR HANDLING
- The Containment Personnel Airlock Doors will not remain open throughout the purge

Which ONE (1) of the following describes the effect this will have on the Auxiliary Building?

The Auxiliary building will:

A. pressurize unless HVS-1, Auxiliary Building Supply Fan, is running.

B. pressurize unless HVS-1, Auxiliary Building Supply Fan, is secured.

C. depressurize unless HVS-1, Auxiliary Building Supply Fan, is running.

D. depressurize unless HVS-1, Auxiliary Building Supply Fan, is secured.

Recommendation for questions.
Question #96 - RO
Question #94 - SRO

**Choices "B" and "C" are correct answers**

Choice "A" - If HVS-1 is running, this would increase the pressure in the Auxiliary Building. This is the reverse order of Choice "C". Therefore, Choice "A" is incorrect.

Choice "B" - this choice is supported by the caution prior to step 8.2.2.1 in OP-921, Containment Air Handling. This is a correct choice.

Choice "C" - this choice is essentially the same answer and choice "B". In general, if HVS-1, Auxiliary Building Supply fan is not running the Auxiliary Building pressure will be more negative. This supports choice "C" in that the Auxiliary Building will depressurize (pressure will become more negative) unless the fan is running.

Choice "D" - If HVS-1 is secured, this would decrease the pressure in the Auxiliary Building. This is the reverse order of Choice "B". Therefore, Choice "D" is incorrect.

CHOICE "B"

8.2.2 Instructions

INIT

**CAUTION**

- If the Containment Personnel Airlock doors will not remain open during the duration of the Purge, the Auxiliary Building will pressurize unless HVS-1 not operating.
- During Refueling operations, at least one of the Containment Personnel Airlock doors is required to be closed. (ITS LCO 3.9.3)

1. IF the Containment Personnel Airlock doors will not remain open during the duration of the Purge, THEN perform the following:
   a. Stop HVS-1, AUXILIARY BUILDING SUPPLY FAN, IAW OP-906, Section 8.1. ______
   b. Commence monitoring E1 and E2 Room temperature as directed by APP-010-A5, HVS-1 TROUBLE Alarm. ______
2. Place the Purge or Refuel Valves Control Switch to the REFUEL Position. ______
3. IF BOTH HVE-15 AND HVE-15A are secured, THEN verify D-5, HVE-15 and 15A discharge damper, is CLOSED. ______

INIT

## 8.0 INFREQUENT OPERATION

### 8.1 Stopping HVS-1 (ACR 93-130)

**NOTE:** This Section has been screened in accordance with PLP-037 criteria and determined Not Applicable to PLP-037.

This section may be used anytime it is desirable to stop HVS-1 to establish a negative pressure in the Auxiliary Building OR when stopping HVS-1 is deemed necessary to ensure proper ventilation conditions exist.

8.1.1 This revision has been verified to be the latest revision available.

| Name (Print) | Initial | Signature | Date |
|---|---|---|---|
| | | | |

8.1.2 Verify AUX BLDG EXH FAN, HVE-2A OR HVE-2B is OPERATING. ____

**CAUTION**

The maximum allowable E1/E2 Emergency Bus Room temperature is 122 °F.

**NOTE:** Stopping HVS-1 may cause E1/E2 Emergency Bus Room temperature to increase. APP-010-A5, HVS-1 TROUBLE, provides guidance for monitoring and maintaining E1/E2 Emergency Bus Room temperature less than 122 °F.

8.1.3 Open AUX BUILDING SUPPLY FAN, HVS-1, breaker on MCC-5, CMPT 7J. ____

8.1.4 Install a CAUTION tag on HVS-1 breaker identifying the reason for opening the breaker. CAUTION tag # ____________ ____

## NRC RESOLUTION OF FACILITY RECOMMENDATIONS

**OPERATING TEST:**

JPM - 009a:
Recommendation accepted. The intent of the JPM was to evaluate the applicants ability to operate and understand the Nuclear Instrumentation. If the applicant verified that the QPTR inputs from the failed Nuclear Instrument were valid, it was acceptable to not place the Upper and Lower Section switches on the Detector Current Comparator Drawer to the N44 position.

If the applicant bypassed the QPTR inputs without evaluating them, it was regarded as a lack of understanding of the system operation but was not considered a critical step in completion of the overall task.

**WRITTEN EXAMINATION:**

R0 #1 / SRO #3:
Recommendation not accepted. No change to the answer key. The answer remains C.

A control rod inserted in the core, it is not available to provide the negative reactivity needed to meet the reactivity insertion rate assumed in accident analysis.

Section 4.6 of the H. B. Robinson Updated Final Safety Analysis Report (UFSAR) states that,

> "The control rod drive assemblies for the full length rods provide a fast insertion rate during a 'trip' of the RCC assemblies which results in a rapid shutdown of the Reactor Coolant System. This rate is based on the results of various reactor emergency analyses, including instrument and control delay times and the amount or reactivity that must be inserted before deceleration of the RCC assembly occurs."

With a control rod dropped into the core, it would not be able to contribute to the insertion rate as described in UFSAR Section 15.0.6, "RCCA Insertion Characteristics Used in the Analysis".

H.B.Robinson Technical Specifications define OPERABLE / OPERABILITY, as follows:

> "A system, subsystem, train component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s)..."

A control rod that is dropped fully into the core is not capable of fulfilling its safety function of adding negative reactivity during a trip. Therefore, a control rod dropped into the core would not be operable.

RO #73 / SRO #75:
Recommendation accepted. The answer key was changed from A to B.

RO #92 / SRO #88:
Recommendation accepted. The answer key was changed from B to A.

RO #96 / SRO #94:
Recommendation not accepted. Answer B remains as the only correct answer.

The question tested generic K/A 2.3.9, "Knowledge of the process for performing a containment purge." To correctly answer this comprehension level question the applicant needed to understand the desired status of the Auxiliary Building pressure and relate it to the process for performing a containment purge.

The question asked for the effect that establishing a containment purge with the personnel airlock doors closed will have on the Auxiliary Building. The Auxiliary Building pressure is maintained less than atmospheric pressure so that air will leak into the Auxiliary Building instead of creating a potential release path by air leaking out to the environment. The negative pressure is maintained by the Auxiliary Building Exhaust fans HVE-2A and HVE-2B which draw air from the Auxiliary Building ventilation system and discharge it to the plant stack.

When a containment purge is established in accordance with OP-921, Containment Air Handling, one of the CV purge fans is started. These fans, HVE-1A and HVE-1B, draw air from containment and discharge it to the plant stack. With the personnel airlock doors closed, air is not drawn into containment from the Auxiliary Building but from the outside via the Outdoor Air Makeup System. The increased flow through the stack creates a backpressure from the stack on the Auxiliary Building exhaust fans which causes the Auxiliary Building to pressurize.

Under no conditions will establishing a CV purge with the containment personnel airlock doors closed cause the Auxiliary Building depressurize. Therefore, distractors "C" and "D", which state that the Auxiliary Building will depressurize are not correct.

In the licensee's examination submittal, they justified distractor "C" to be incorrect. They explained that this distractor represented a plausible misconception that the Auxiliary Building were normally pressurized vice under a vacuum.

Choosing answer "C" demonstrated a misunderstanding of the both the expected status of the Auxiliary Building pressure and the required ventilation line up for performing a containment purge.