ENCLOSURE 3

SCE&G

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John L. Skolds
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December 3, 1990
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Mr. M. E. Ernstes
U. S. Nuclear Regulatory Commission
Region II, Suite 2900
101 Marietta Street, N. W.
Atlanta, GA 30323

Dear Mr. Ernstes:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
SENIOR REACTOR OPERATOR WRITTEN EXAM QUESTION COMMENTS

Enclosed are the Virgil C. Summer Nuclear Station comments on the Senior Reactor Operator written examination administered on November 26, 1990. These comments are those discussed with you, Ron Aiello, and Sandy Lawyer in the exit meeting conducted on November 29, 1990.

Also enclosed is the signed security agreement from Mr. Henry Manos, who reviewed the written examination November 7-9.

If additional information is required, please contact us at your convenience.

Very truly yours,

John L. Skolds

WRH:JLS:lcd
Enclosures

c: D. R. Moore
K. W. Woodward
NPCF

V. C. SUMMER NUCLEAR STATION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

EXAMINATION COMMENTS

QUESTION 017

QUESTION:

017 Which one of the following is the pressure at which the RHR will start injecting following a LOCA?

a. 104 psig

b. 152 psig

c. 250 psig

d. 650 psig

ANSWER KEY RESPONSE:

b. 152 psig

REFERENCE:

VCS: AB-10, ECCS, p. 28, Enabling Objective 4

SUGGESTED CORRECT RESPONSE:

b. 152, psig, OR

c. 250 psig

REASON:

Undoubtedly, the actual shutoff head discharge pressure of the RHR pump is 152 psig per the above reference. However, in an actual LOCA, which is the condition specified in the question, the EOP's, would be implemented. Throughout the EOP's, RHR flow is always verified when RCS pressure is at or below 250 psig. Per the attached plant specific setpoint document, OAG 103.2, this pressure represents shutoff head plus allowances for normal channel accuracy. Either answer should be accepted since one represents component performance data and the other constitutes an operator verified parameter associated with the component.

QUESTION 17

U) Highest steam line safety valve setpoint [1235 psig].

V) Lowest steam line safety valve setpoint [1176 psig].

W) Steam Generator PORV setpoint [1130 psig].

X) Steam Generator PORV controller setpoint 25 psi below lowest safety valve setpoint [1150 psig] [setpoint 8.8].

Y) Steam Generator PORV controller setpoint 25 psi below lowest safety valve setpoint, minus 25 psi [1120 psig]. CHG A

Z) Steam Generator saturation pressure corresponding to 350°F plus 20°F, plus allowances for normal channel accuracy and post accident transmitter errors [370 psig].

AA) Steam Generator pressure which prevents accumulator nitrogen injection [140psig].

AB) Steam Generator pressure which prevents accumulator nitrogen injection plus a control margin [240 psig].

AC) Normal accumulator pressure plus 25 psi [660 psig]. CHG A

AD) Shutoff head pressure of the high head Safety Injection pumps plus allowances for normal channel accuracy [N/A].

AE) Shutoff head pressure of the high head Safety Injection pumps plus allowances for normal channel accuracy and post accident transmitter errors [N/A]. CHG B

AF) * Shutoff head pressure of the low head Safety Injection pumps plus allowances for normal channel accuracy [250 psig].

AG) Shutoff head pressure of the low head Safety Injection pumps plus allowances for normal channel accuracy and post accident transmitter errors [250 psig].

AH) Shutoff head pressure for the Condensate Pumps for providing the necessary Feedwater Booster Pump suction pressure [350 psig].

AI) RCS pressure which prevents accumulator injection [800 psig]. CHG B

*AF) Per IMS-94B-302, 152 psig plus channel accuracy listed in Calculation #194 yields 242 psig (Use 250 psig for readability). CHG A

*AG) Same as (AF) per CGSS-23302-DE.

AH) The Feedwater Booster Pumps develop a maximum differential head of 288 psid per 1MS-17-114. Normal Feedwater Booster Pump suction pressure due to elevation head and deaerator pressure is 100 psig. A value of 350 psig Condensate Pump discharge pressure will provide enough margin to ensure the Feedwater Booster Pumps can overcome Steam Generator pressure and discharge flow losses.

AI) Background for this setpoint in REE 21679, Attachment 32. CHG B

*Channel uncertainty is added (+) to the RHR shutoff head to ensure an RHR pump is not prematurely stopped or left idle.

QUESTION 017

	ERG	FOOTNOTE	EOP
S)	FR-I.3	(3)	EOP-18.2
T)	FR-C.1	(11)	EOP-14.0
	FR-C.2	(17)	EOP-14.1
	E-1	(19)	EOP-2.0
	ES-1.2	(28)	EOP-2.1
	ECA-1.1	(28)	EOP-2.4
	ECA-2.1	(31)	EOP-3.1
	ECA-3.1	(39)	EOP-4.2
	ECA-3.2	(35)	EOP-4.3
U)	F-0.3	(3)	EOP-12.0
	FR-H.2	(1)	EOP-15.1
V)	F-0.3	(5)	EOP-12.0
	FR-H.4	(1)	EOP-15.3
W)	ECA-0.0	(3)	EOP-6.0
X)	E-3	(3)	EOP-4.0
Y)	ES-3.1	(11)	EOP-4.1A
	ES-3.2	(17)	EOP-4.1B
	ES-3.3	(17)	EOP-4.1C
	ECA-3.1	(28)	EOP-4.2
	ECA-3.2	(22)	EOP-4.3
	ECA-3.3	(22)	EOP-4.4
Z)	E-3	(11)	EOP-4.0
AA)	ECA-0.0	(7)	EOP-6.0
	ECA-1.1	(12)	EOP-2.4
	FR-C.1	(8)	EOP-14.0
	FR-C.2	(13)	EOP-14.1
AB)	ECA-0.0	(8)	EOP-6.0
AC)	ECA-1.1	(11)	EOP-2.4
AD)	E-0	(2)	EOP-1.0
	ES-1.1	(2)	EOP-1.2
	ECA-2.1	(18)	EOP-3.1
AE)	E-0	(3)	EOP-1.0
	ES-1.1	(3)	EOP-1.2
	ECA-2.1	(19)	EOP-3.1
AF)	E-0	(4)	EOP-1.0
	E-1	(13)	EOP-2.0
	ES-1.2	(2)	EOP-2.1
	E-3	(9)	EOP-4.0
	ECA-1.1	(8)	EOP-2.4
	ECA-2.1	(9)	EOP-3.1
	ECA-3.1	(3)	EOP-4.2
	ECA-3.2	(2)	EOP-4.3
	FR-C.2	(2)	EOP-14.1
	FR-C.3	(2)	EOP-14.2
	FR-H.1	(18)	EOP-15.0
AG)	E-0	(5)	EOP-1.0
	E-1	(14)	EOP-2.0
	ES-1.2	(3)	EOP-2.1
	E-3	(10)	EOP-4.0
	ECA-1.1	(9)	EOP-2.4
	ECA-2.1	(10)	EOP-3.1
	ECA-3.1	(4)	EOP-4.2
	ECA-3.2	(3)	EOP-4.3
	FR-C.2	(3)	EOP-14.1
	FR-C.3	(3)	EOP-14.2
	FR-H.1	(19)	EOP-15.0
AH)	FR-H.1	(9)	EOP-15.0
AI)	ECA-3.2	(36)	EOP-4.3

QUESTION 017

CAUTION: RCS pressure should be monitored. If RCS pressure decreases to less than (13) psig [(14) psig FOR ADVERSE CONTAINMENT], the low-head SI pumps must be manually restarted to supply water to the RCS.

PURPOSE: To alert the operator to possible manual action requirements as a result of his actions in the following step

BASIS:

Low-head SI pumps should be stopped if RCS pressure is above their shutoff head, since they will not be delivering flow. However, if RCS pressure decreases to less than (13) psig [(14) psig for adverse containment], then the pumps will have to be restarted manually since no automatic signal is available.

ACTIONS:

Determine if RCS pressure decreases to less than (13) psig [(14) psig for adverse containment]

INSTRUMENTATION:

RCS pressure indication

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

- o (13) Enter plant specific value for the shutoff head pressure of the low-head SI pumps, plus allowances for normal channel accuracy.
- o (14) Enter plant specific value for the shutoff head pressure of the low-head SI pumps, plus allowances for normal channel accuracy and post accident transmitter errors.

STEP: Check If Low-Head SI Pumps Should Be Stopped

PURPOSE: To stop the low-head SI pumps if RCS pressure is above their shutoff head to prevent damage to the pumps

BASIS

Upon safety injection initiation all safeguard pumps are started regardless of the possibility of high RCS pressure with respect to the low-head safety injection pump shutoff head. On low-head systems where the pump recirculates on a small volume circuit there is concern for pump and motor overheating. Shutdown of the pump and placement in the standby mode, when the RCS pressure meets the criteria outlined in this step, allows for future pump operability.

If SI has not been previously reset and the low-head SI pumps should be stopped, SI should be reset prior to stopping the pumps. SI can be reset regardless of containment pressure.

ACTIONS:

- o Determine if RCS pressure is greater than (13) psig [(14) psig for adverse containment]
- o Determine if RCS pressure is stable or increasing
- o Reset SI signal if necessary
- o Stop low-head SI pumps and place in standby

INSTRUMENTATION:

- o RCS pressure indication
- o Low-head SI pumps status indication
- o SI signal reset status indication

CONTROL/EQUIPMENT:

- o Low-head SI pump switches
- o SI reset switches

KNOWLEDGE:

This step is a continuous action step as indicated by the CAUTION preceding it.

QUESTION 017

PLANT-SPECIFIC INFORMATION:

o (13) Enter plant specific value for the shutoff head pressure of the low-head SI pumps, plus allowances for normal channel accuracy.
o (14) Enter plant specific value for the shutoff head pressure of the low-head SI pum , plus allowances for normal channel accuracy and post accident transmitter errors.

QUESTION 017

QUESTION 023

QUESTION:

023 Which one of the following describes the effect that actuation of the containment spray system has upon containment hydrogen concentration during a LOCA?

a. DECREASES due to the decrease in containment temperature.

b. DECREASES due to the "scrubbing" action of the water droplets.

c. INCREASES due to the increase in containment humidity.

d. INCREASES due to a chemical reaction between NaOH and aluminum.

ANSWER KEY RESPONSE:

d. INCREASES due to a chemical reaction between NaOH and aluminum.

REFERENCE:

VCS: AB-9, ESF, p. 38, Enabling Objective #13

SUGGESTED ACTION:

Delete question entirely.

REASON:

Reference listed for question clearly shows that the reaction involving hydrogen production from aluminum is:

$$2Al + 3H_2O \rightarrow Al_2O_3 + 3H_2$$

The general industry position supports that it is indeed water that chemically reacts with aluminum to form aluminum oxide and hydrogen. This corrosion reaction is accelerated by the high containment temperature conditions and is enhanced by the pH found in containment at the time (8.0 - 10.5). There is, however, no chemical reaction between NaOH and aluminum.

Since the Reactor Building has some finite leak rate, some of the radioactive fission products escape to the environment. If the fuel pellets do not melt, the remainder of the I, Xe, and Kr produced by the fuel remain in the pellet rather than escaping when the clad fails.

Throughout this event, generation of hydrogen from various sources cause a build-up of the hydrogen concentration in the Reactor Building. Hydrogen in concentrations between 4.1 percent and 18.2 percent are flammable. From 18.2 percent to 59 percent an explosive mixture exists. From 59 percent to 74 percent the mixture is again flammable. Above 74 percent there is not enough oxygen to support combustion. Action should be taken to preclude the generation of this gas, and to reduce its concentration in the Reactor Building. The major sources of hydrogen are the following:

- The zirconium-water reaction, as described previously.

- Corrosion of plant materials now in a water environment inside the Reactor Building. The corrosion of aluminum ($2AL + 3H_2O \rightarrow AL_2O_3 + 3H_2$) is of special interest. For this reason, the quantity of aluminum in the Reactor Building should be limited as much as practicable in accordance with Regulatory Guide 1.7. Oxidation reactions involving other metals in solution will also occur, each having its own H_2 production rate.

- Radiolysis of core and sump water. Although water radiolysis is a complex process involving many reactions, the overall process can be expressed by

  $$2H_2O \rightarrow 2H_2 + O_2$$

  Gamma decay from fission products produces the energy necessary for this decomposition of water.

Figures AB9.24 and AB9.25 show the accumulation of hydrogen as a function of time, and curves for reduction in hydrogen concentration. Limiting the zirconium-water reaction by keeping clad temperatures below 2200°F is responsible

Lesson Outline | Instructor Notes

(2) Following a LOCA, any hydrogen generation from this reaction was predicted to be complete the first day.

(3) Westinghouse assumes a 2 percent reaction.

(4) NRC assumes 5 percent.

(5) If ECCS works, only 0.1 percent.

(6) At TMI-2, estimates are between 20 and 70 percent.

b. Corrosion of Plant Materials

(1) In an alkaline borate solution, most plant materials do not show significant rates of corrosion. Aluminum and zinc are two notable exceptions.

SEQUOYAH NUCLEAR PLANT TRAINING MATERIAL

(2) Aluminum in containment is very reactive in an alkaline borate solution (pH 9.3 - 10.0).

(a) $2Al + 3H_2O \rightarrow Al_2O_3 + 3H_2$

(b) Aluminum is present in the intermediate and power range detectors, in the process instruments and controls equipment, and the control rod drive mechanisms.

(3) Zinc is also very reactivity in alkaline solutions. It is used in floor gratings and electrical trays and conduits.

c. Core Radiolysis — Objective B.12

(1) Water in the core is subjected to radiation and heat from the fuel fission products.

QUESTION 023

QUESTION 057

QUESTION:

057 Which one of the following is the minimum level at which steam generators must be maintained above to ensure adequate heat transfer area and continuation of the decay heat removal process following a reactor trip?

a. 38% NARROW range level

b. 4% NARROW range level

c. 50% WIDE range level

d. 20% WIDE range level

ANSWER KEY RESPONSE:

d. 20% WIDE range level

REFERENCE:

VCS: IB-3, EFW p. 2, Enabling Objective #8

SUGGESTED CORRECTED RESPONSE:

b. 4% NARROW range level

REASON:

The reference which was used to write the question was the emergency feedwater system handout (IB-3). This material had not been updated to reflect setpoints which were changed during the recent EOP rewrite. Formerly, 38% narrow range level was used to verify adequate heat sink along with EFW flow. Likewise, the old setpoint for initiating bleed and feed of the RCS as an alternate means of heat removal was 20% WIDE range level. These setpoints, as documented in the attached references, were recalculated and now stand at 4% NARROW RANGE and 10% WIDE range respectively. Because these setpoints are now used, the only correct choice was (b.). Incidentally, the training material referenced has been changed to reflect these new values, but was not issued prior to our sending the NRC the exam reference material. The EOP's sent did have the correct data.

QUESTION 57

BACKGROUND

A) Steam Generator level indication in the narrow range plus instrument uncertainty ensures an adequate inventory for heat removal. Any indicated level above the lower tap ensures coverage of the U-tubes. Per Calculation #194, Rev 1, normal narrow range accuracy is 3.76%. A value of 4.0% should be used for readability. A value of 12% is used for the minimum control band to control level above the Steam Generator low-low level alarm. CHG A

B) Same as (A) with an additional margin for post accident errors. A 50% maximum is imposed to provide margin to Steam Generator overfill for Feedwater control. This value is 33% per Calculation # 194, Rev 1. CHG A

C) Steam Generator level is maintained at 38% (no load value) during natural circulation to ensure a stable heat sink.

D) The Hi Hi Steam Generator level setpoint (82.4%) is primarily used to provide margin for filling a ruptured Steam Generator with cold feed flow to cool metal in the upper region while ensuring the Steam Generator does not overfill. This is a protection setpoint and normal channel accuracy has been included. A value of 82% is used for readability.

E) Same as (D). The -19% post accident error results in indicated level being less than actual level. By using 65%, margin is provided to prevent overfill.

F) Pressurizer level just in range plus a normal channel uncertainty of 3.77% is used to verify adequate RCS inventory (Use 4%). During situations where a steam vent path is established from the Pressurizer vapor space and where RCS subcooling is not indicated, Pressurizer level may not be a true indication of RCS inventory. This can result from steam generated in the reactor vessel, passing through the Pressurizer surge line and preventing the water inventory of the Pressurizer from draining into the RCS loops. This holdup of water can result in a stable or even increasing indicated Pressurizer level while RCS water inventory is actually decreasing. Pressurizer level should be relied on only with hot leg or core exit subcooling present. In SI termination steps in the ERGs, Pressurizer level is only checked after adequate RCS subcooling is confirmed.

G) Same as (F) with an additional 39% margin for post accident errors. An upper limit of 50% is imposed to ensure margin to filling the Pressurizer for RCS inventory control.

H) A Pressurizer level just covering the heaters plus uncertainty is verified prior to restoring normal letdown. A minimum value of 18% is required to reset the letdown isolation signal. The actual level which covers the heaters is 13.0%. Considering the 3.77% normal channel accuracy, 18% should be used for this value (Calculation #194).

I) Same as (H) with an additional 39% margin for post accident error. An upper limit of 50% is imposed to ensure margin to filling the Pressurizer for RCS inventory control.

J) A 17% Pressurizer level isolates letdown and de-energizes all heaters. This level is used to verify charging operation after a reactor trip.

QUESTION 057

USE

	ERG	FOOTNOTE	EOP
A)	E-0	(8)	EOP-1.0
	ES-0.1	(3)	EOP-1.1
	ES-0.4	(20)	N/A
	E-1	(4)	EOP-2.0
	ES-1.1	(11)	EOP-1.2
	ES-1.2	(5)	EOP-2.1
	ES-1.4	(13)	N/A
	E-3	(4)	EOP-4.0
	ES-3.1	(6)	EOP-4.1A
	ES-3.2	(6)	EOP-4.1B
	ES-3.3	(6)	EOP-4.1C
	ECA-0.0	(4)	EOP-6.0
	ECA-0.1	(9)	EOP-6.1
	ECA-0.2	(3)	EOP-6.2
	ECA-2.1	(3)	EOP-3.1
	ECA-3.1	(6)	EOP-4.2
	ECA-3.2	(5)	EOP-4.3
	ECA-3.3	(5)	EOP-4.4
	F-0.3	(6)	EOP-12.0
	FR-S.1	(5)	EOP-13.0
	FR-C.1	(5)	EOP-14.0
	FR-C.2	(11)	EOP-14.1
	FR-H.1	(6)	EOP-15.0
	FR-H.3	(4)	N/A
	FR-H.5	(1)	EOP-15.4
	FR-P.1	(2)	EOP-16.0
	FR-P.2	(2)	EOP-16.1
B)	E-0	(9)	EOP-1.0
	E-1	(5)	EOP-2.0
	ES-0.4	(14)	N/A
	ES-1.1	(12)	EOP-1.2
	ES-1.2	(6)	EOP-2.1
	ES-1.4	(6)	N/A
	E-3	(5)	EOP-4.0
	ES-3.1	(7)	EOP-4.1A
	ES-3.2	(7)	EOP-4.1B
	ES-3.3	(7)	EOP-4.1C
	ECA-0.0	(5)	EOP-6.0
	ECA-0.1	(10)	EOP-6.1
	ECA-0.2	(4)	EOP-6.2
	ECA-2.1	(4)	EOP-3.1
	ECA-3.1	(7)	EOP-4.2
	ECA-3.2	(6)	EOP-4.3
	ECA-3.3	(6)	EOP-4.4
	F-0.3	(1)	EOP-12.0
	FR-S.1	(6)	EOP-13.0
	FR-C.1	(6)	EOP-14.0
	FR-C.2	(12)	EOP-14.1
	FR-H.1	(7)	EOP-15.0
	FR-H.3	(5)	EOP-15.2

	ERG	FOOTNOTE	EOP
B)	FR-H.5	(2)	EOP-15.4
	FR-P.1	(3)	EOP-16.0
	FR-P.2	(3)	EOP-16.1
C)	ES-0.2	(3)	EOP-1.3
D)	ES-3.1	(9)	N/A
	ES-3.2	(15)	N/A
	ES-3.3	(15)	N/A
	ECA-3.1	(26)	N/A
	ECA-3.2	(20)	N/A
	ECA-3.3	(1)	EOP-4.4
	F-0.3	(4)	EOP-12.0
	FR-H.3	(3)	EOP-15.2
E)	ES-3.1	(10)	N/A
	ES-3.2	(10)	N/A
	ES-3.3	(16)	N/A
	ECA-3.1	(27)	N/A
	ECA-3.2	(21)	N/A
	ECA-3.3	(2)	EOP-4.4
F)	E-0	(15)	EOP-1.0
	E-1	(10)	EOP-2.0
	ES-0.4	(11)	EOP-1.0
	ES-0.4	(11)	EOP-1.1
	ES-0.4	(11)	EOP-1.3
	ES-0.4	(11)	EOP-1.4
	ES-0.4	(11)	EOP-1.5
	ES-1.1	(6)	EOP-1.2
	ES-1.2	(17)	EOP-2.1
	ES-1.4	(3)	EOP-1.2
	ES-1.4	(3)	EOP-2.0
	ES-1.4	(3)	EOP-2.1
	ES-1.4	(3)	EOP-2.2
	ES-1.4	(3)	EOP-2.3
	E-3	(17)	EOP-4.0
	ES-3.1	(3)	EOP-4.1A
	ES-3.2	(3)	EOP-4.1B
	ES-3.3	(3)	EOP-4.0
	ES-3.3	(3)	EOP-4.1A
	ES-3.3	(3)	EOP-4.1B
	ES-3.3	(3)	EOP-4.1C
	ECA-0.0	(17)	EOP-6.0
	ECA-0.1	(4)	EOP-6.1
	ECA-2.1	(15)	EOP-3.1
	ECA-3.1	(21)	EOP-4.2
	ECA-3.2	(37)	EOP-4.3
	ECA-3.3	(8)	EOP-4.4
	FR-H.1	(12)	EOP-15.0

QUESTION 057

STEP: Verify Tota: AFW Flow - GREATER THAN (6) GPM

PURPOSE: To ensure AFW flow to the steam generators

BASIS:

AFW flow is necessary for secondary heat sink. If SG level is in the narrow range in at least one SG, a heat sink is available. Therefore, AFW flow is needed only to maintain level. If adequate AFW flow for decay heat removal cannot be established, the transition to the FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, is necessary to establish an alternate source of feed flow or an alternate heat sink.

ACTIONS:

- o Determine if total AFW flow greater than (6) gpm
- o Determine if SG narrow range level greater than (8)% ((9)% for adverse containment)
- o Determine if total AFW flow greater than (6) gpm cannot be established
- o Manually start pumps and align valves as necessary
- o Transfer to FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, step 1

INSTRUMENTATION:

Total AFW flow indication for each steam generator
Narrow range level indication for each steam generator

CONTROL/EQUIPMENT:

Switches for:
- o AFW pumps
- o AFW valves

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

(6) Enter the minimum safeguards AFW flow requirement for heat removal, plus allowances for normal channel accuracy (typically one MD AFW pump capacity at SG design pressure).

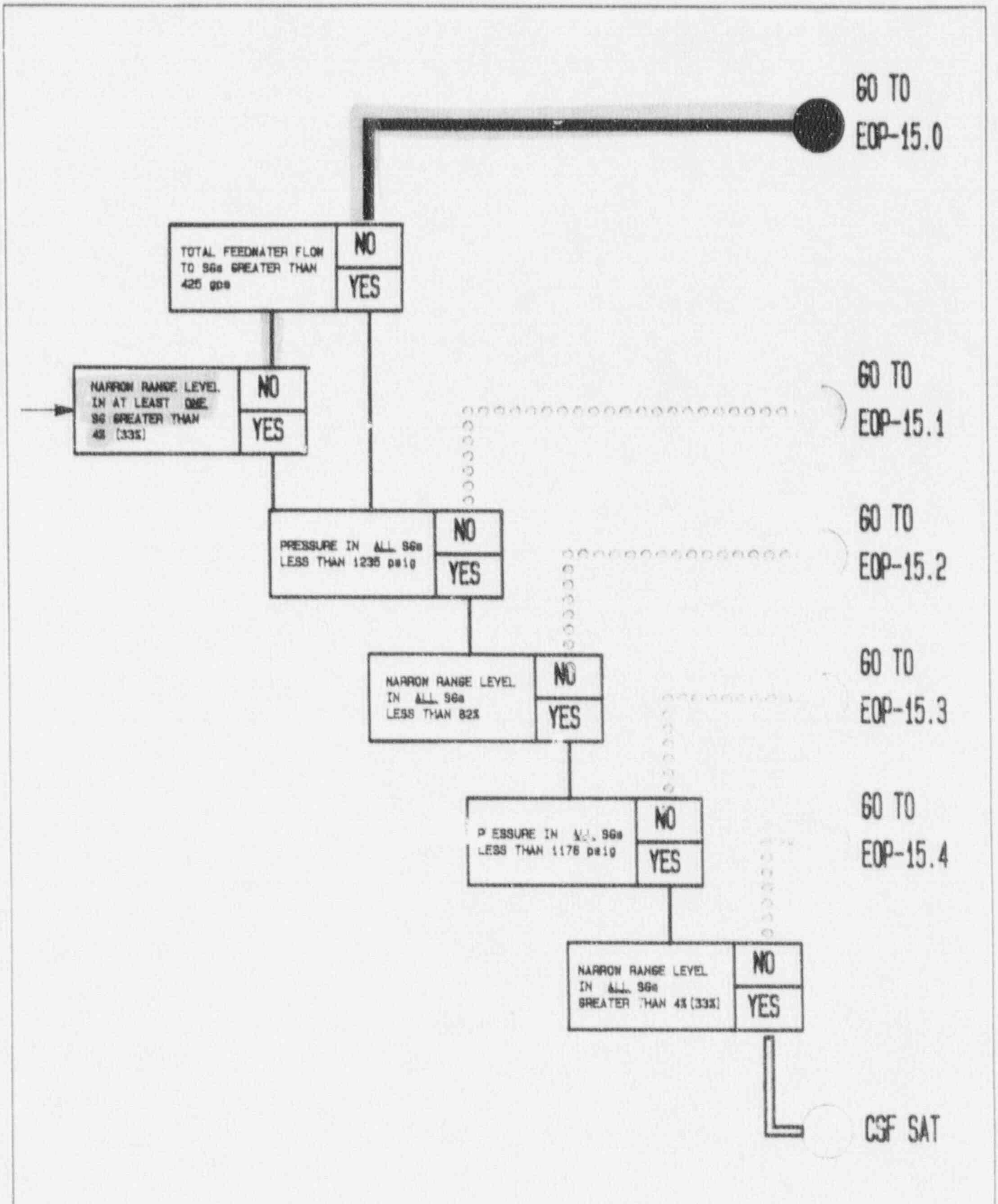
This flow is equivalent to the minimum AFW flow design requirement that must be delivered to the intact steam generators as assumed in the main feedline break safety analysis.

(8) Enter plant specific value showing SG level just in the narrow range, including allowances for normal channel accuracy.

(9) Enter plant specific value showing SG level just in the narrow range, including allowances for normal channel accuracy, post accident transmitter errors, and reference leg process errors, not to exceed 50%.

QUESTION 057

HEAT SINK



QUESTION 057

RESPONSE TO LOSS OF SECONDARY HEAT SINK

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION

OPERATOR ACTIONS

CAUTION

- If total EFW flow is LESS THAN 425 gpm due to operator action, this procedure should NOT be performed.
- If Wide Range level in any two SGs is LESS THAN 10% [35%] OR PZR pressure is GREATER THAN 2335 psig due to loss of secondary heat sink, RCPs should be tripped. Steps 11 through 18 should be immediately initiated for bleed and feed cooling.
- If a NON-FAULTED SG is available, feed flow should NOT be reestablished to any FAULTED SG.

NOTE

Conditions for implementing Emergency Plan Procedures should be evaluated using EPP-001, ACTIVATION AND IMPLEMENTATION OF EMERGENCY PLAN.

ACTION/EXPECTED RESPONSE	ALTERNATIVE ACTION
1 Check if a secondary heat sink is required:	
a. Verify RCS pressure is GREATER THAN any NON-FAULTED SG pressure. ☐	a. GO TO EOP-2.0, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1. ☐
b. Verify RCS Hot Leg temperature is GREATER THAN 350°F. ☐	b. Try to place the RHR System in service while continuing with this procedure: 1) Ensure RCS pressure is LESS THAN 425 psig. ☐ 2) Place the RHR System in service. REFER TO SOP-115, RESIDUAL HEAT REMOVAL. ☐ IF adequate cooling with the RHR System is established, THEN RETURN TO the Procedure and Step in effect. ☐

QUESTION 057

QUESTION 079

QUESTION:

079 Which one of the following describes how an independent verification of position for a manually operated "Locked Throttled" safety injection valve must be done? (Assume process parameters indicate that the valve is indeed throttled and not closed.)

a. Move the valve slightly in the closed direction and then return it to its original position.

b. Inspect the last valve lineup sheet for verification signature and compare recorded valve position with the required position.

c. Observe initial placement of the valve in the position the valve is to be throttled to.

d. Verify the tamper seal is properly installed and has not been broken.

ANSWER KEY RESPONSE:

d. Verify the tamper seal is properly installed and has not been broken.

REFERENCE:

VCS, SAP-153, p. 9

SUGGESTED ACTION:

Accept "c" in addition to "d."

REASON:

The question did not specify the initial conditions as to whether the independent verification involved the initial setup of the throttled valve in its required position or a check of a valve which had been throttled previously. Either case requires independent verification but the methodology differs somewhat. The reference, page 9 of SAP-153, supports choice "d" since it assumes the case of a special locked throttled valve where the position was set much earlier. Page 6 of SAP-153, Section 6.1.15, specifies the procedure used when verifying the initial positioning of the locked throttled valve. SI-90-04, Notes 3.3 and 3.4, supports "c" and "d" as correct choices.

B. Valves that are to be verified closed will be manipulated in the closed direction only as necessary to verify the valve is fully closed, and not binding or difficult to operate. Care must be exercised, however, to avoid overtorquing the valve operator and damaging the valve seat. If any doubt exists, the Shift Supervisor should be contacted for resolution.

C. Valves that are required to be verified in a throttled position will normally be manipulated in the closed direction, with the operator counting the number of turns required to fully close the valve. The operator will then re-open the valve to its properly throttled position.

6.4.2 Locked Valves

A. Verification techniques for locked valves will be performed as in Section 6.4.1 (Unlocked Valves). In addition, the following requirements apply:

1. The person performing the independent verification will ensure the locking device is be properly installed and documented in accordance with the Locked Valve Program (Reference 3.3).

2. The time interval between the initial verification or positioning of a locked valve and the independent verification will normally be less than one shift to reduce the possibility of component mispositioning errors. The independent verification should be completed with the locking device verified on the same shift that initially verified or positioned the valve in order to facilitate the smooth transfer of plant status during shift turnover.

#79
Choice "d" described

3. Independent verification of "special locked throttled" valves will consist of a check of the locking device to ensure that it is properly installed. In addition, a determination that the valve is indeed throttled and not closed will be performed by means other than actual operation of the valve. Other means of position verification are summarized in Section 6.4.2.B below.

QUESTION 079

SPECIAL INSTRUCTION (continued)

1.8 All locked valves using a chain/seal tab combination, chain/padlock combination, or chain operated handwheel/seal tab combination will have their chains color coded in the following manner:

a. Green color coded chains for valves locked in the CLOSED position.

b. Red color coded chains for valves locked in the OPEN position.

c. Yellow color coded chains for valves locked in a THROTTLED position.

d. Silver color coded chains for valve handwheels locked in the NEUTRAL position.

2.0 CHANGING THE POSITION OF A LOCKED VALVE

2.1 The Shift Supervisor shall review the reason for the removal of the locking device and the desired position of the valve. If he concurs, he will then sign under the Removal Section of ATTACHMENT II.

2.2 An operator will then remove the locking device and place the valve in the desired position and sign under the Removal Section of ATTACHMENT II.

2.3 The Administrative Operator on shift will erase the serial number of the removed locking device from ATTACHMENT I and initial the Removal Section of ATTACHMENT II.

2.4 ATTACHMENT II will accompany the Danger Tag Sheet or STP requiring removal of the locking device, or will be placed in the Active Tracking Section of the Locked Valve Book if no Danger Tag Sheet or STP exists.

3.0 SECURING A LOCKED VALVE IN ITS REQUIRED POSITION

3.1 The Control Room Supervisor ensures the Description Section of ATTACHMENT II is filled out (using information from ATTACHMENT I).

3.2 The Control Room Supervisor then issues the locking device filling in the serial number and denoting padlock or seal under the Installation Section of ATTACHMENT II.

Choices "c" + "d"
#79

NOTE 3.3 and 3.4

"c"

(If the valve is being locked in a throttled position, the Independent Verifier should observe the first operator position the valve;) however, he must still independently (verify the locking device properly installed.) ← "d"

3.3 An operator will position the valve to its required position and install the locking device, then sign under the Installation Section of ATTACHMENT II.

3.4 A second operator will perform a verification of the required position and locking device installation, then sign under the Installation Section of ATTACHMENT II. The valve position must be positively verified.

QUESTION 079

6.1.13 The "Locked Valve Program" provides the throttled valve setting as either the number of turns open or the number of turns closed from the full open position for each throttled component. An exception to this are valves throttled to establish a specified flow or by indication, valves with tack welded locking devices that have specified flows established by system characteristics. In the event a system operating condition prohibits closing a throttled component to verify its position and the act of fully opening the component would not unduly upset the system, the number of turns throttled closed from full open may be used in lieu of the normal method of counting the turns open from fully closed. CHG A

6.1.14 The Technical Specifications relating to the required open or closed positions of certain components must be considered on all component manipulations. If the act of verifying the position of a component violates the designated position of the component required by Technical Specifications for the plant operating condition, positive control of the operability of the valve must be maintained at all times during the component manipulation. Technical Specification requirements should be reviewed for applicability.

#79
Choice "c" described.

6.1.15 When the operation of a throttled valve is necessary to determine its position, having a verifier observe the initial valve operator's action is preferable to having both persons independently operate the valve. This second valve operation would effectively nullify the first and would therefore serve no purpose.

6.1.16 Independent verification requirements may be waived by the Shift Supervisor if excessive radiation exposures would result. As a guideline, an exposure > 10 mrem to conduct the independent verification would be considered excessive. Individual situations should be determined on a case-by-case basis by the Shift Supervisor for those components not previously exempted by the "Local IV Exempt" designation on the system line-up checklists. In these situations, an alternate means of independent verification that does not involve radiation exposure (such as observing process parameters) should be considered.

QUESTION 079

QUESTION 084

QUESTION:

084 In accordance with EPP-012, Onsite Personnel Accountability and Evacuation, which one of the following correctly lists the reasons that personnel should be evacuated to an offsite holding area?

a. Potential personnel and/or vehicle contamination, radiation exposure considerations ONLY.

b. Need for additional personnel on short notice, radiation exposure considerations ONLY.

c. Potential personnel and/or vehicle contamination, need for additional personnel on short notice, radiation exposure considerations ONLY.

d. Potential personnel and/or vehicle contamination, need for additional personnel on short notice, radiation exposure considerations, a bomb threat in an unidentified area inside the protected area.

ANSWER KEY RESPONSE:

c. Potential personnel and/or vehicle contamination, need for additional personnel on short notice, radiation exposure considerations ONLY.

REFERENCE:

VCS, EPP-012, p. 6

SUGGESTED ACTION:

Delete question entirely.

REASON:

Site evacuation to an offiste holding area is discussed per page 6 of EPP-012 as referenced. However, the ED/TED has discretionary power, per attached page 2 of EPP-012, to evacuate all or part of the plant staff whether it be to their personal residence or an offsite holding area. Use of the term "ONLY" in choice "c" implies that this discretionary component does not exist and that the three reasons listed are all-inclusive.

4.0 CONDITIONS AND PREREQUISITES

4.1 Protected Area Evacuation may be ordered by the ED/IED when conditions warrant such action.

4.2 When personnel are ordered to evacuate the PA, they should have in their possession the personal effects necessary to evacuate to an Off-Site Holding Area if so directed.

4.3 A specific area within the plant will be evacuated under the following conditions:

a. Confirmed report of a significant radioactive spill in a work area.

b. Confirmed report of an unexpected increase in the level of radiation or airborne activity in a work area.

c. One or more area radiation monitors in a single building reach their "Warning" setpoint.

d. A fire in a work area.

e. As designated by the ED/IED.

4.4 All non-essential personnel will be evacuated from the plant site under the following conditions:

a. Site Area Emergency

b. General Emergency

c. As directed by the ED/IED

4.5 Sounding of the Reactor Building Evacuation Alarm will be cause to evacuate the Reactor Building. The Reactor Building Evacuation Alarm will sound:

a. Automatically - by Source Range (N-31 and N-32) Nuclear Instrumentation.

b. Manually - by the Reactor Building Evacuation Pushbutton located in the control room.

4.6 Evacuation Signs are throughout the plant. Red Signs are in the Radiation Control Area and direct personnel to the 412' elevation for personnel decontamination if applicable. Green Signs are throughout the remainder of the plant and direct personnel out of the PA by the way of the PAP or AAP.

4.7 If decontamination cannot be done expeditiously in the plant decontamination facility, personnel who are found to

QUESTION 084

QUESTION 092

QUESTION:

092 Which one of the following is added to the steam generator bulk water to limit corrosion due to hideout?

a. hydrazine

b. ammonia

c. boron

d. silicon

ANSWER KEY RESPONSE:

c. boron

REFERENCE:

VCS, IB-4, Secondary Chemistry and Chemistry Control, p. 16

SUGGESTED ACTION:

Delete question entirely

REASON:

Per Chemistry Procedure CP-806, Boron is used to inhibit steam generator tube denting as well as stress cracking corrosion. As an agent to prevent tube denting it is fairly exclusive, but as a corrosion inhibits it is one of many chemicals added. The attached reference describes how hydrazine and ammonia are used to establish proper pH and limit oxygen, which ultimately limits corrosion. Had the question asks about hideout with respect to tube denting then "c" would have been a clear choice.

1.0 PURPOSE

1.1 This procedure provides a method of conditioning the Steam Generators with Boric Acid prior to operation above 30% reactor power.

2.0 REFERENCES

2.1 The Morpholine/Boric Acid Application Document for South Carolina Electric and Gas Company. V.C. Summer Nuclear Power Plant. (Westinghouse)

2.2 CP-613, "Steam Generator Chemistry Control".

3.0 DISCUSSION

3.1 Boric Acid has been found to inhibit Steam Generator tube denting and stress cracking corrosion. The inhibiting mechanisms that have been attributed to Boric Acid addition is can be any combination of the following processes:

- Buffering the localized pH of an alkaline environment by forming a borate ion.
- Dilution of an alkaline environment
- Formation of a passive borate layer to blanket the metal and prevent access by corrosive species.

3.2 This procedure shall be performed following shutdowns of 2 weeks or longer.

4.0 PRECAUTIONS AND PREREQUISITES

4.1 The plant is operating at no greater than 30% reactor power.

4.2 Steam Generator chemistry is stable and in specification per CP-613 for exceeding 30% reactor power except for the boron concentration.

4.3 A minimum volume of 500 gallons containing 7000 to 7500 ppm Boron is ready for injection to the suction of the operating Condensate Pump.

4.4 The batch addition tank is connected to the operating Condensate Pump suction drain valve.

The control of corrosion in the secondary systems is accomplished by the use of all volatile treatment chemistry control, the condensate polishing demineralizers, and continuous steam generator blowdown. The all volatile chemistry control utilizes volatile chemicals which are injected into the Condensate or Emergency Feedwater Systems. Oxygen is scavenged by the use of hydrazine, the pH is elevated by the use of ammonia, and the polishing demineralizers remove ionic and solid impurities. The dissolved and suspended impurities are also controlled by minimizing the corrosion of the secondary systems and the in-leakage of circulating water. The continuous steam generator blowdown prevents the build-up (concentration) of impurities in the steam generator.

Condensate and Feedwater Chemistry Control

The chemistry control of the secondary plant begins with the Condensate System. During startup, condensate flow is passed through the polishing filter/demineralizers, which remove ionic impurities and suspended solids. Up to 50 percent flow, all condensate passes through the condensate polishing filter/demineralizers.

Any oxygen dissolved in the condensate is removed by the addition of hydrazine (N_2H_4) to the Condensate System. The oxygen is scavenged by the following reaction:

$$N_2H_4 + O_2 \rightarrow 2H_2O + N_2$$

The hydrazine is continuously injected into the Condensate System downstream of the condensate polishing demineralizers, but upstream of the steam packing condenser. Injection at this point ensures oxygen control of 100 percent of the condensate flow and avoids unnecessary depletion of the demineralizer resin beds. The condensate temperature at the injection point is about 105°F. It increases to about 295°F by the exit of the low pressure feedwater heaters. This oxygen-scavenging reaction occurs relatively slowly at temperatures below 100°F and reaches an optimum reaction rate at temperatures between 150°F and 180°F.

QUESTION 092

The use of hydrazine for control of oxygen provides an added benefit because hydrazine elevates the system pH (Fig. IB4.1) by the following reaction:

$$N_2H_4 + H_2O \rightarrow N_2H_5^+ + OH^-$$

Because hydrazine is a weak base, it does not raise the pH sufficiently by itself at low temperatures (< 200°F). However, at temperatures in excess of 400°F, the hydrazine rapidly decomposes into ammonia (NH_3) by the following reaction:

$$2N_2H_4 \rightarrow 2NH_3 + N_2 + H_2$$

The ammonia in turn combines with water to form hydroxyl ions and elevates the pH even more. Ammonia is a stronger base than hydrazine. However, because of the very low concentration of hydrazine, the formation of hydroxyl ions by both of the above reactions is still not sufficient to maintain the condensate and feedwater at the desired pH.

The pH of the Condensate and Feedwater Systems is elevated by the addition of ammonia (NH_3) to the Condensate System. The following reaction describes the formation of the hydroxyl ions, which cause a basic pH (Fig. IB4.2):

$$NH_3 + H_2O \rightleftarrows NH_4^+ + OH^-$$

The ammonia is continuously injected into the Condensate System downstream of the condensate polishing demineralizers and upstream of the steam packing condenser. Injection at this point ensures pH control of 100 percent of the system flow and also prevents excessive depletion of the demineralizer resin beds. Because of its volatile nature, some of the ammonia is transported into the steam system and eventually returns to the Condensate System. Ammonia injection rate is controlled manually by the water treatment operator as demand indicates.

Condensate polishing demineralizers are used to filter the condensate and to remove icnic impurities. Up to 50% of full condensate flow is passed through the powdex beds by either the condensate (or possibly by the kidney loop pumps in the future). Particulates are mechanically filtered out: the beds are changed when D/P becomes excessive. The resin is a mixed bed type that removes both cations and anions. Resin depletion is indicated by excessive conductivity in the discharge. Spent resin is discharged to the environment or (if radioactive) to solid waste.

<u>Condensate and Feedwater Chemistry Sampling</u> (Figures IB4.3 and IB4.4)

All Turbine Building chemistry sample lines converge at a sample rack and analyzer panel XSR-1 in the Water Treatment Building. The equipment used for sample collection and analysis consists of a sample rack, a recorder analyzer panel, and a refrigeration unit. There are condensate sample points located locally at the discharge side of the condensate pumps.

The samples from the Condensate and Feedwater Systems that are hotter than 150°F are cooled by primary cooling coils to reduce the sample temperature. These coolers, which are cooled by turbine room closed-cycle cooling water, are located at the sample rack.

The sample rack has interchangeable tubing that allows flow from any given point to pass through the analyzers. The rack has analyzers for pH, conductivity, dissolved oxygen, hydrazine, and sodium.

The sample sink at the rack has two drain systems, one for recovery and one for waste. Recovery flow goes to the miscellaneous drain tank in the Condensate System. This flow consists of conductivity, grab, and pH samples. Waste from the hydrazine, dissolved oxygen, and sodium analyzers goes to the Floor Drain System.

ENCLOSURE 4

SRO Examination

Question 17

NRC Resolution: Comment not accepted. The question asked for the pressure at which the RHR pumps will first start injecting which would be the shutoff head pressure. The facility references the ERG which refers to this value plus channel inaccuracies. The value of 250 psig (which was rounded up from 242 psig for readability) ensures that the low-head pumps are not injecting and should be secured. An operator would be in error if he expected the RHR pumps to be injecting into the RCS with pressure at 250 psig.

Question 23

NRC Resolution: Comment not accepted. Although there is an important chemical reaction between aluminum and water, this question was testing the candidates knowledge of the reaction between NaOH and aluminum.

$$2Al + 2NaOH + 2\ H_2O \dashrightarrow 2NaAlO_2 + 3H_2$$

NaOH which is added to containment spray reacts with aluminum to produce hydrogen in containment.

Question 57

NRC Resolution: Comment accepted. The facility supplied reference (IB-3) was in error due to not having been updated since the last EOP rewrite. The correct answer was 10% wide range. Since the correct answer was not one of the choices the question was deleted.

Question 79

NRC Resolution: Comment accepted. The answer key was changed to include both (c) and (d) as correct responses.

Question 84

NRC Resolution: Comment not accepted. This question specifically asked for a reason for evacuating to an <u>offsite holding area</u>. The facility reference discusses any protected area evacuation.

Question 92

NRC Resolution: Comment not accepted. Facility reference IB-4, "Secondary Chemistry and Chemistry Control", states that "Boron is added to the Steam Generator bulk water to limit corrosion due to hideout". Steam generator tube denting is one of the effects of hideout as described in the facility reference. This is limited by adding boron.

ENCLOSURE 5

SIMULATOR FIDELITY REPORT

Facility Licensee: South Carolina Electric and Gas Company

Facility Docket No.: 50-395

Operating Tests Administered on: November 27, 1990

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 non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed :

ITEM	DESCRIPTION
1DB voltmeter	Would always read zero. There was no documentation on the meter to tell the operator that it was not functioning properly. Operators ignored this discrepancy on subsequent scenarios.
Rod group 2D	Did not function. This created a distraction for one crew DRPI while they dealt with the malfunction.

On two occasions, the model degenerated and the scenario had to be stopped prior to the scheduled ending point. The scenario was designed to test the operators on the transition to Cold Leg Recirculation fo.lowin, a large Loss of Coolant Accident.

One scenario was designed to test the operators on EOP-15.0, "Response to Loss of Secondary Heat Sink". Due to a decay heat model which was adding very little energy to the RCS, it was not possible to simulate the conditions which would require them to enter this procedure.