PSEG Nuclear LLC
Hope Creek Initial License Operator Training NRC Written Examination
Administered 3/18/2002

Post Exam Changes:

Exam Record #	RO/SSRO Exam Question #	Facility Recommendations
4	4 RO Only	**Delete question. No correct answer**. Answer choice "C" does not limit flow to only "system" flow. HC.OP-SO.BC-0002 step 5.2.31 and 5.2.33 allow throttling flow with F003A(B) which are NOT affected by the stated bus loss. Also, as stated in Caution 5.2.31 and 5.2.33, the F003A(B) may be used to throttle flow through the heat exchanger. System flow is normally established with the F015A(B). In the question, both F015 A and B valves lose power from the 10A404 bus loss.
13	11/12	**Recommend accept 2 correct answers B and D.** The Technical Specification Bases 3/4 3.9 supports answer B, which states the reason for the high RPV Level Main Turbine trip is to prevent Main Turbine damage. The bases is brief and does not contain detailed discussion of the high level trip. Lesson Plan 0301-000.00H-000002-15 page 13 and 14 supports answer choice D which states HPCI, RCIC, and Reactor Feed Pumps are tripped to prevent RPV overfill and flooding the main steam lines, then states the significant safety concerns if overfill occurred including "Stressing of the reactor main steam line nozzles, steam line snubbers, pipe supports and hangers as a result of: - The weight of water in the main steam lines; and the dynamic transient loads caused by water flow in the main steam lines." This is further supported by NRC INFO NOTICE 88-77, which is also referenced in the lesson plan and addresses RPV overfill and flooding the main steam lines. This is further supported by the bases in Improved Standard Technical Specifications.
31	21/28	**Recommend Answer Key change to D.** Upon trip of the CRD Pump, HCU Accumulator charging water check valves (V115) begin leaking through the valve seats, causing the accumulator pistons to move. The N2 gas pressure lowers when the accumulator piston moves. (See P&ID M-47 Sheet 1) The surveillance requirements of HC.OP-IS.BF-0103 demonstrate the leakage rate is low enough to prevent accumulator trouble alarms for greater than 2 minutes. The accumulator alarms when nitrogen gas side pressure lowers to 940 psig. The original assumption was that there was no leakage past the check valves, however, as stated previously there is some leakage past the check valves that would affect gas pressure and cause it to lower.

Exam Record #	RO/SRO Exam Question #	Facility Recommendations
46	34 RO Only	**Delete question.** RO candidates were not provided the HC Event Classification Guide to correlate the ALERT level EOP 103/4 entry condition to a radiation level. ECG Section 6.1 provides that correlation. Without the ECG, the question becomes a **Level of Difficulty 5 memory** question relying on memory of wording contained in Lesson Plan 0302-000.00H-000127-12. The text for Learning Objective 2 states: "The entry condition for Radioactivity Release Control corresponds to an action level defined in the site Emergency Plan." ECG Bases document states that the ECG Initiating Condition is entered when radioactive release rates reach levels corresponding to 200 times 10CFR20, Appendix B Limits. These levels are high enough that they will not occur during normal operation, but still low enough that the immediate health and safety of the general public is not threatened by the release."
48	36/43	**Recommend Answer Key change to D.** The Hydrogen alarms are set to alarm at 2.0% Hydrogen concentration on the H2/O2 Analyzers. Conditions provided in the stem indicate the reactor core would be degraded and producing hydrogen. EOP 102 Step PC/H3 directs the Hydrogen Recombiners to be placed in service if H2 concentration reaches 0.5%. Placing the H2 Recombiners in service IAW step PC/H3 would be the required action. (Answer D) Hydrogen alarms are clear indicating Hydrogen Concentration is less than 2.0%, therefore, EXIT EOP 102 and enter SAG is not required. (Answer B)
68	55 RO Only	**Delete question. Level of Difficulty 5.** The question requires memorization of the prerequisite 2.6.2 of HC.OP-SO.SB-0001. The procedure should have been referenced in the question stem.
73	59/60	**Delete question. No correct answer.** LPRM 32-33-C is assigned to LPRM Group A IAW HC.RE-ST.SE-0003 Attachment 1. All answer choices affect an APRM. The answer would have been correct if the LPRM chosen belonged to APRM C or D. Additionally, the candidates were not given a reference to determine which APRM the LPRM was assigned. The readings of the LPRM and APRM before the failure occurred would also be necessary to determine if the average went up or down after the failure.

Exam Record #	RO/SRO Exam Question #	Facility Recommendations
106	87/83	**Delete question. No correct answer**. Immediate Operator Action of HC.OP-AB.ZZ-0129 3.1 states "If smoke OR toxic gases are detected in the control room air supply, isolate the Control Room ventilation and place CREF in the RECIRC MODE" . Keyed answer D contains part of that answer. Pressing the Control room EMER FILTER UNIT A and B RECIRC MODE pushbuttons alone will not start CREF or place CREF in the RECIRC MODE. CREF must be running for the Recirc Dampers to open.
116	92 RO Only	**Delete question. No Correct answer.** IAW NC.NA-AP.ZZ-0049, Definitions 7.2. "Formal declaration of Suspension of Core Alterations or Fuel Handling is performed by the Refueling SRO or a condition required by Technical Specifications." None of the answer choices contained conditions required by Technical Specifications that would require Suspension of Core Alterations. Additionally, Lesson Plan 302-000-00H-000113-10 Obj 66 states… "Determine the conditions under which handling of fuel must be suspended, IAW NC.NA-AP.ZZ-0049. (SRO ONLY)". Therefore the question is not appropriate for the RO candidates.
120	94/93	**Delete question for candidates: Breslin, Hernandez, Klass, and Panagotopulos** due to question contained a typographical error that allowed **no correct answer choice**. These 4 students had completed and turned in their exams prior to the discovery of the error. The error was in the stem, Entry 2, Neutron dose should have read 24 instead of 54 mrem. Calculating the answer based on the error resulted in a remaining dose of 1491 mrem. (Reference NC.NA-AP.ZZ-0024) Previous history TEDE = DDE + CEDE; TEDE = 210 + 45; TEDE = 255 mrem Todays dose: Gamma dose + Neutron dose Entry 1 + Gamma dose + Neutron dose Entry 2 = DDE Entry 1 = (52 + 24) + Entry 2 = (124 + 54) Todays dose = 76 + 178 = 254 mrem Remaining dose = Admin limit (2000 mrem TEDE) – Previous history (255 mrem) – todays dose (254 mrem) Remaining dose = 1491 mrem The closest answers were A: 1488 mrem and B: 1521 mrem, both of which are incorrect for the question asked. Once the error was identified, it was made known to the 4 remaining candidates; Hanna, Kopsick, McKeown, and Baker. The correction was then written on the board. The question will remain valid for these 4 candidates.

Exam Record #	RO/SRO Exam Question #	Facility Recommendations
126	98/97	**Recommend accept 2 correct answers A and C.** SH.OP-AP.ZZ-0102 step 5.5.2 supports answer C. However, there are numerous 100 series Abnormal procedures that are operational transients. Abnormal procedures HC.OP-AB.ZZ-0138 Main Turbine Trip and HC.OP-AB.ZZ-0110 Loss of an RPS Channel are examples of operational transients. Answer A stated "100 series are operational transient procedures" which is technically correct. There is currently a major effort underway to correct identified deficiencies to enhance these procedures and governing documents.
129	100/100	**Delete question. No correct answer.** Procedure HC.OP-IO.ZZ-0008 was revised to remove reference to using HCU Accumulator pressures as verification means. The procedure now uses SPDS, CRIDS or RMCS Activity Control Cards to verify rods full in if forced to leave the Control Room.

Hope Creek ILOT 2002-01 NRC Exam

JPM RO-A.4

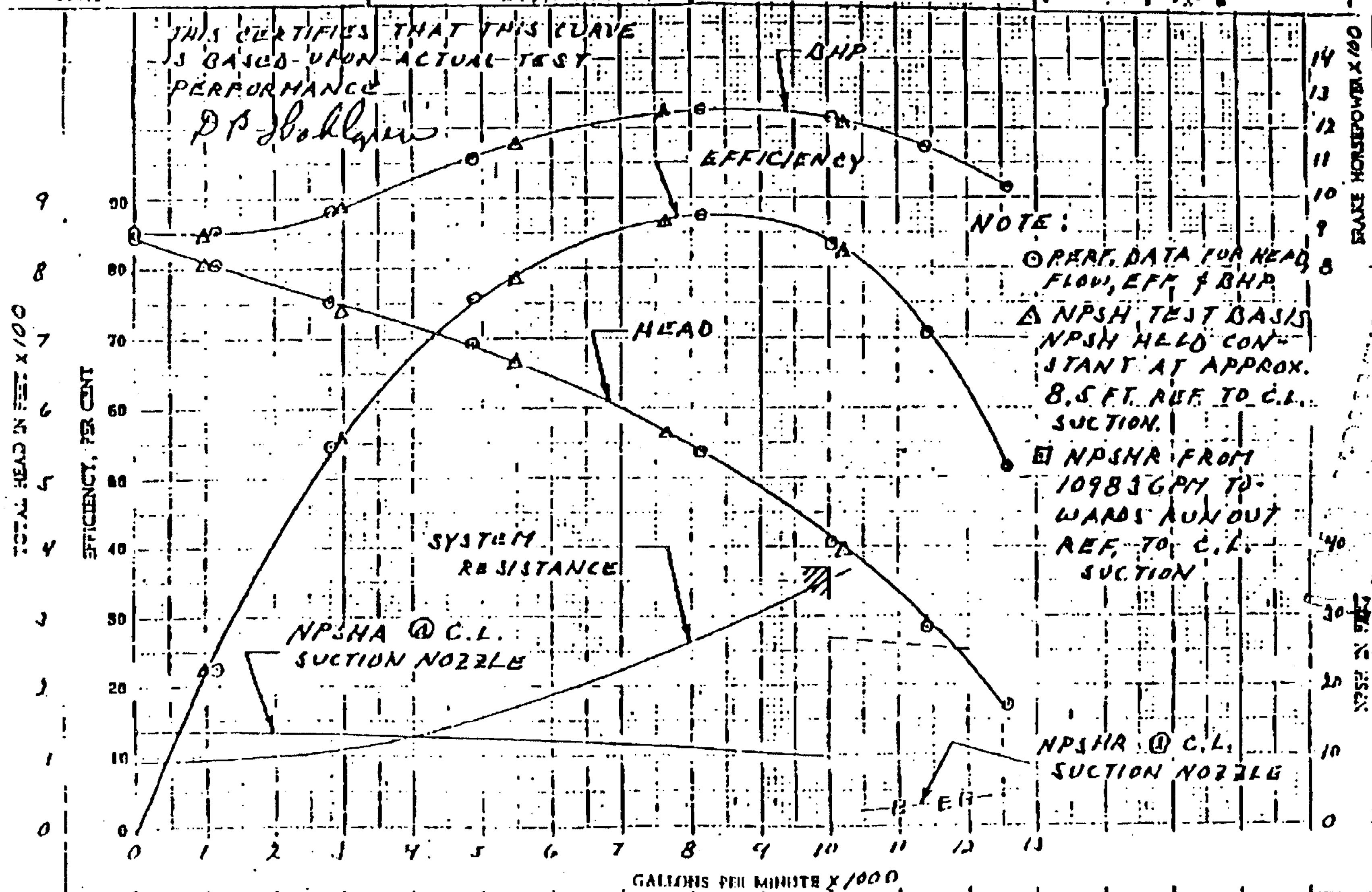
The NRC requested validation of the safety significance and critical nature of the step to circle LPCI flow rate in administrative JPM RO-A.4. The JPM required the RO candidates to perform the licensed operator review of an operational status board for a given set of plant parameters following a small break LOCA event. The control room integrated display (CRIDs) computer was not available requiring the candidates to use alternate (backup) hardwire indication to verify and validate the plant parameters provided on the status board. The JPM, as written, combined verification and validation of the plant status information and circling recorded LPCI flow rates into a single "critical step."

Upon post examination review, it was determined that for this JPM's initial conditions, circling the LPCI flow rates had little safety significance. From a technical perspective, the reasoning is as follows:

- Given the initial JPM conditions and plant status parameters recorded, it is physically impossible for LPCI to be injecting into the RPV at the flow rates indicated. Reactor pressure was at 325#, which is nearly shutoff head for the RHR pumps, and based on the pump curves (attached) the maximum injection flow would be about 500 gpm, not the 10000 gpm indicated for both LPCI pumps. In addition, several other parameters such as the reactor still being at 325# 20 minutes after a small break LOCA indicate that the leak is very small and it is clear that RPV level in being maintained in the normal band with feedwater flow (feedpumps or secondary condensate pumps). Thus, circling LPCI flow to indicate LPCI is operating in a mode other than injection, although required by procedure, is of low safety significance and its omission is of low consequence (i.e. it does not provide clarifying information).
- Additionally, if the candidate had circled the LPCI flow rates indicating that LPCI was operating in a mode other than RPV injection, the procedure does not require that the alternate LPCI mode be recorded on the status board so the receiver of the completed status board would still need to ask for clarifying information (the alternate LPCI modes are: suppression pool cooling, suppression chamber sprays, or drywell sprays). A procedure change request has been generated (attached) to enhance this procedure to require not only circling LPCI flows but to annotate the mode of operation as well.

CRIDS
KNOW THAT WITH HARDWIRED INDICATION
LOCATION

CUSTOMER G.E.A.P.E.D PROPOSAL NO. 086-360/61 ITEM — SPECIAL NOTES S/N 1172136 TESTED IN CUSTOMER SHELL	DESIGN CONDITIONS GPM 10,000 EFF — T.H.(FT) 370 BHP — SG 1.0 RPM 1170 DRIVER HP 1250 HP MOTOR WITH 1.15 S.F.	Ingersoll-Rand DRAWN BY B.P.W. 24 FEB. 76 DATE	CURVE N-747 REV. 0 PUMP 34APHD-4 AP202

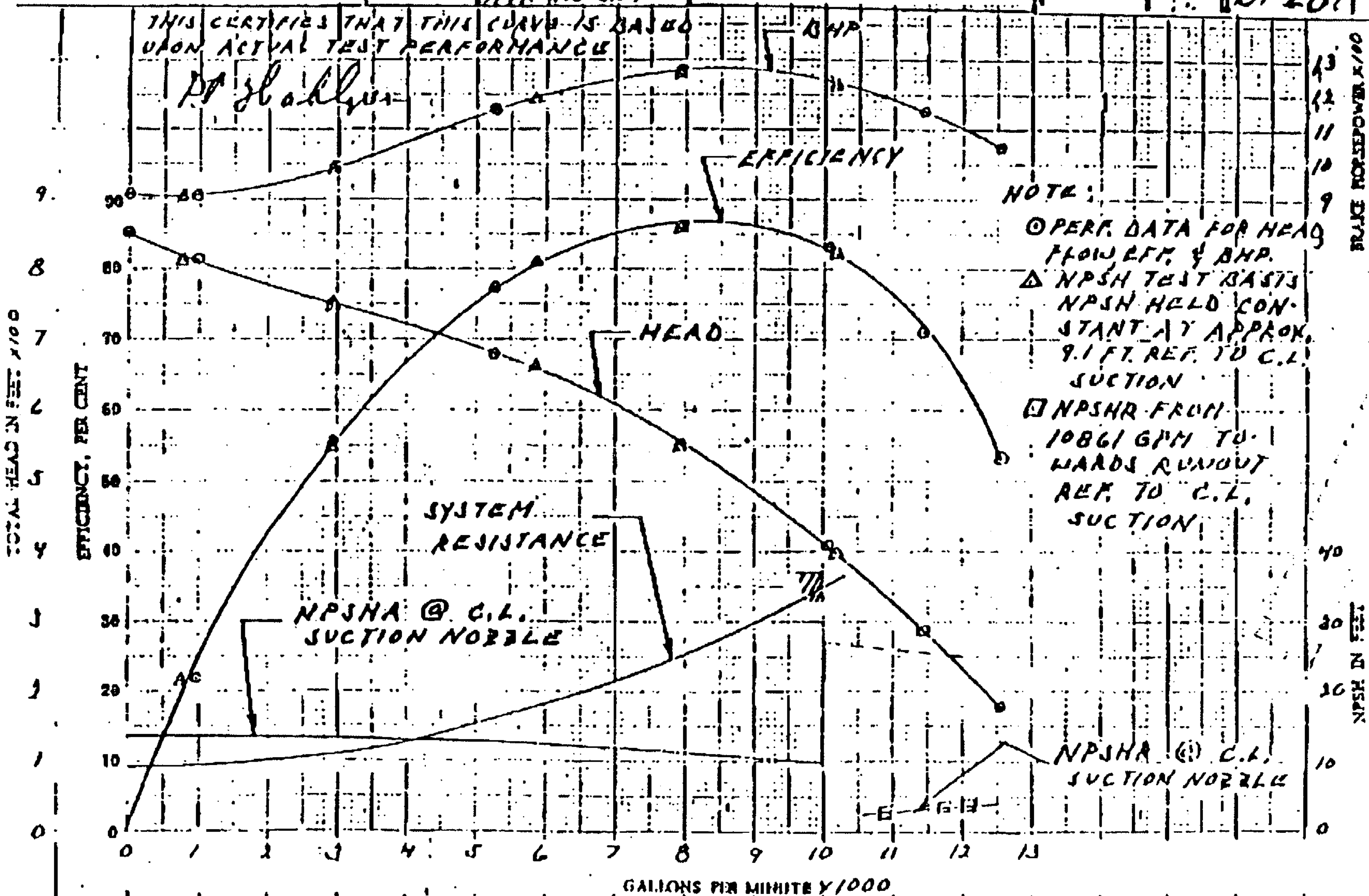


Appendix G

1441 B-7

Sheet 1 of 6

CUSTOMER C.F.A.P.E.D. PROPOSAL NO. 006-36076 ITEM — SPECIAL NOTES SN-1172137 TESTED IN CUSTOMER SHELL	DESIGN CONDITIONS GPM 10,000 EFF — T.H.(FT) 370 BHP — SG. 1.0 RPM 1170 DRIVER HP 1250HP MOTOR WITH 1.15 S.F.	Ingersoll-Rand DRAWN BY D.P.W. DATE 31 DEC. 76	CURVE N-751, Rev. 0 PUMP 34APHO-4 BP201



sheet 2 of 6

PSEG

Notification Overview
Run Date: 04/01/2002
Run Time: 12:11:32
Page: 1 of 1
Notification 20095417

Notification	20095417				
Notification type	N1				
Description	Enhancement to HC ECG Att 8				
Nuc. Maint. Request					
Reporter	NUNFC X-1267				11:52:18
Notification date	04/01/2002				
Start date	04/01/2002	*End date*			
Start time	11:52:18	*End time*	00:00:00		
Priority	*Sig. Level* X	*Main WorkCtr.*	O-EP00		
Funct. location	ECG-ATT.08				
SECONDARY COMMUNICATOR LOG					
Equipment					
Assembly					
Order					
PM planner grp	099	*Nuclear Default*			

04/01/2002 12:02:10 NICOLA CONICELLA (NUNFC)
Enhance Hope Creek ECG Attachment 8, secondary communicator log, page 6 of 8, Operational Status Board - Hope Creek, to require annotating the alternate mode that A or B RHR pumps are operating in if not injecting into the RPV in LPCI mode (i.e. suppression pool cooling, suppression chamber sprays, drywell sprays, or shutdown cooling). The status board currently requires that LPCI flow is circled if not injecting but does not require that the specific mode be annotated. This enhancement was a result of post-NRC license exam review for Hope Creek ILOT class 2002-01. Assign to C. Banner of the emergency preparedness group.

End of report

Given the following:

- The reactor is in Operational Condition 4
- "A" RHR Pump is in Shutdown Cooling at rated flow
- 10A404 4.16KV 1E Bus trips on bus differential overcurrent

Which one of the following describes the effect the bus loss will have on Shutdown Cooling?

a. The Shutdown Cooling common suction line isolates and CANNOT be reset

b. The AP228 Jockey pump trips causing Shutdown Cooling Loop "A" to lose keepfill

c. Both "A" and "B" Shutdown Cooling Loops lose ability to adjust flow

d. "B" Reactor Recirc Pump discharge valve automatically opens bypassing core flow

Answer	c	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002

Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	1	295003G409

295003	Partial or Complete Loss of A.C. Power	**Record Number**	4

2.4	Emergency Procedures and Plan		
2.4.9	Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.	3.3	3.9

Explanation of Answer: "A" RHR SDC Return valve F015A is powered from "D" Channel 1E 480VAC. Loss of D Bus fails this valve as is. Adjusting flow via RHR HX outlet valve and /or bypass valve is not proceduralized. AP228 provides keepfill to HPCI only. B RRP disch valve is controlled by NON 1E power.

Reference Title

HC.OP-SO.BC-0002

HC.OP-SO.SM-0001

Learning Objectives

000028E008	(R) Given a system which physically connects to or is required to support the operation of the RHR System or components therein, explain the function of the supporting system, IAW the RHR System Lesson Plan.

Material Required for Examination

Question Source:	New	**Question Modification Method:**	

Question Source Comments:

CAUTION 5.2.31 — both 18" lines so if bypass full open you are not controlling gpm, just temp.

A. **If BC-HV-F003A(B) RHR HX A(B) OUTLET VLV is used to throttle RHR flow through the RHR Heat Exchanger, then BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV must be fully open. [CD-503B]**

B. **If BC-HV-F003A(B) RHR HX A(B) OUTLET VLV is CLOSED fully, then RHR INLET TO HX A(B) TE N004A(B) (CRIDS point A2380(A2382) or TR-R605 point 1(2)), will not be accurate.**
When BC-HV-F003A(B) is fully closed then utilize RHR DISCH FROM HX A(B) TE N027A(B) (CRIDS point A2381(A2383) or TR-R605 point 3(4)), to MONITOR system temperature.

C. **BC-HV-F003A(B) RHR HX A(B) OUTLET VLV should be opened slowly to avoid a "cold slug" discharge into the recirculation piping.**

5.2.31 **PERFORM** the following as necessary to maintain the Shutdown Cooling return to RPV temperature relatively constant, as monitored on TR-R605 point 4
OR CRIDS A2383, while maintaining the required RHR Shutdown Cooling flow, simultaneously: ______

A. IF temperature is increasing,
THEN, **PERFORM** the following: ______

1. Slowly **THROTTLE** OPEN on the BC-HV-F003A(B) RHR HX A(B) OUTLET VLV. ______

2. IF the BC-HV-F003A(B) is fully opened,
THEN, **THROTTLE** CLOSED BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV ______

Continued Next Page

5.2.31 (Continued)

B. IF temperature is decreasing,
THEN, **PERFORM** the following: ______

1. **THROTTLE** OPEN on the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV. ______

2. IF the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV is fully open,
THEN, **THROTTLE** CLOSED on the BC-HV-F003A(B) RHR HX A(B) OUTLET VLV. ______

C. **PERFORM** the following to maintain shutdown cooling flow: ______

THROTTLE BC-HV-F015B RHR LOOP B RETURN TO RECIRC as necessary to maintain required flow on FI-R603B or FR-R608(B) - CRIDS A3139. ______

D. **PERFORM** the following to maintain head spray flow: ______

THROTTLE BC-HV-F023B RHR LOOP B HEAD SPRAY OUTBD ISLN MOV as necessary to maintain 300 gpm on FI-R607. **[CD-935E]** ______

5.2.32 **MAINTAIN** the RPV Metal Temperatures and RHR SDC System water temperature difference < 240°F.
REFERENCE Crids point A2383
OR Recorder TR-R605 - point 4 for RHR SDC system,
AND CRIDS points A3569 & A3578 for Vessel flange. ______

NOTE 5.2.33

Cooldown until the final desired reactor coolant temperature is reached (90°F -110°F is recommended, although other temperature(s) within TS limits may be used to support integrated plant operations).

CAUTION 5.2.33

A. If BC-HV-F003A(B) RHR HX A(B) OUTLET VLV is used to throttle RHR flow through the RHR Heat Exchanger, then BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV must be fully open.

B. If BC-HV-F003A(B) RHR HX A(B) OUTLET VLV is CLOSED fully, then RHR INLET TO HX A(B) TE N004A(B) (CRIDS point A2380(A2382) OR TR-R605 point 1(2)), will not be accurate when BC-HV-F003A(B) is fully closed, then utilize RHR DISCH FROM HX A(B) TE N027A(B) (CRIDS point A2381(A2383) OR TR-R605 point 3(4)), to MONITOR system temperature. [CD-503B]

C. BC-HV-F003A(B) RHR HX A(B) OUTLET VLV should be opened slowly to avoid a "cold slug" discharge into the recirculation piping.

5.2.33 **PERFORM** the following as necessary to initiate the cooldown AND Control the cooldown rate, while maintaining the required RHR Shutdown Cooling flow, simultaneously: **[CD-133B]** ______

A. **PERFORM** the following to increase the cooldown rate: ______

1. Slowly **THROTTLE** OPEN on the BC-HV-F003A(B) RHR HX A(B) OUTLET VLV. ______

2. IF the BC-HV-F003A(B) is fully open, THEN, **THROTTLE** CLOSED on the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV. ______

Continued Next Page

5.2.33 (Continued)

B. **PERFORM** the following to decrease the cooldown rate: ______

1. **THROTTLE** OPEN on the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV. ______

2. IF the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV is fully open,
THEN, **THROTTLE** CLOSED on the BC-HV-F003A(B) RHR HX A(B) OUTLET VLV. ______

C. **PERFORM** the following to maintain Shutdown Cooling flow: ______

THROTTLE the BC-HV-F015A(B) RHR LOOP A(B) RET TO RECIRC, as necessary, to maintain the required RHR Shutdown Cooling flow. ______

5.2.34 **RECORD** Reactor Vessel temperatures and pressures IAW Integrated Operating Procedure HC.OP-IO.ZZ-0004(Q); Shutdown from Rated Power to Cold Shutdown
AND HC.OP-DL.ZZ-0026(Q), Attachment 3s; Surveillance Log. ______

Given the following:

- The reactor is in Operational Condition 4
- "A" RHR Pump is in Shutdown Cooling at rated flow
- 10A404 4.16KV 1E Bus trips on bus differential overcurrent

Which one of the following describes the effect the bus loss will have on Shutdown Cooling?

a. The Shutdown Cooling common suction line isolates and CANNOT be reset

b. The AP228 Jockey pump trips causing Shutdown Cooling Loop "A" to lose keepfill

c. Both "A" and "B" Shutdown Cooling Loops lose ability to adjust flow

d. "B" Reactor Recirc Pump discharge valve automatically opens bypassing core flow

Answer c **Exam Level** R **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295003G409

295003 Partial or Complete Loss of A.C. Power **Record Number** 4

2.4 Emergency Procedures and Plan

2.4.9 Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies. 3.3 3.9

Explanation of Answer "A" RHR SDC Return valve F015A is powered from "D" Channel 1E 480VAC. Loss of D Bus fails this valve as is. Adjusting flow via RHR HX outlet valve and /or bypass valve is not proceduralized. AP228 provides keepfill to HPCI only. B RRP disch valve is controlled by NON 1E power.

Reference Title

HC.OP-SO.BC-0002

HC.OP-SO.SM-0001

Learning Objectives

000028E008 (R) Given a system which physically connects to or is required to support the operation of the RHR System or components therein, explain the function of the supporting system, IAW the RHR System Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

CAUTION 5.2.31 – both 18" lines so if bypass full open you are not controlling spn., just temp.

A. **If BC-HV-F003A(B) RHR HX A(B) OUTLET VLV is used to throttle RHR flow through the RHR Heat Exchanger, then BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV must be fully open. [CD-503B]**

B. **If BC-HV-F003A(B) RHR HX A(B) OUTLET VLV is CLOSED fully, then RHR INLET TO HX A(B) TE N004A(B) (CRIDS point A2380(A2382) or TR-R605 point 1(2)), will not be accurate.**
When BC-HV-F003A(B) is fully closed then utilize RHR DISCH FROM HX A(B) TE N027A(B) (CRIDS point A2381(A2383) or TR-R605 point 3(4)), to MONITOR system temperature.

C. **BC-HV-F003A(B) RHR HX A(B) OUTLET VLV should be opened slowly to avoid a "cold slug" discharge into the recirculation piping.**

5.2.31 **PERFORM** the following as necessary to maintain the Shutdown Cooling return to RPV temperature relatively constant, as monitored on TR-R605 point 4
OR CRIDS A2383, while maintaining the required RHR Shutdown Cooling flow, simultaneously: _____

A. IF temperature is increasing,
THEN, **PERFORM** the following: _____

1. Slowly **THROTTLE** OPEN on the BC-HV-F003A(B) RHR HX A(B) OUTLET VLV. _____

2. IF the BC-HV-F003A(B) is fully opened,
THEN, **THROTTLE** CLOSED BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV _____

Continued Next Page

5.2.31 (Continued)

B. IF temperature is decreasing,
THEN, **PERFORM** the following: ______

1. **THROTTLE** OPEN on the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV. ______

2. IF the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV is fully open,
THEN, **THROTTLE** CLOSED on the BC-HV-F003A(B) RHR HX A(B) OUTLET VLV. ______

C. **PERFORM** the following to maintain shutdown cooling flow: ______

THROTTLE BC-HV-F015B RHR LOOP B RETURN TO RECIRC as necessary to maintain required flow on FI-R603B or FR-R608(B) - CRIDS A3139. ______

D. **PERFORM** the following to maintain head spray flow: ______

THROTTLE BC-HV-F023B RHR LOOP B HEAD SPRAY OUTBD ISLN MOV as necessary to maintain 300 gpm on FI-R607. **[CD-935E]** ______

5.2.32 **MAINTAIN** the RPV Metal Temperatures and RHR SDC System water temperature difference < 240°F.
REFERENCE Crids point A2383
OR Recorder TR-R605 - point 4 for RHR SDC system,
AND CRIDS points A3569 & A3578 for Vessel flange. ______

NOTE 5.2.33

Cooldown until the final desired reactor coolant temperature is reached (90°F -110°F is recommended, although other temperature(s) within TS limits may be used to support integrated plant operations).

CAUTION 5.2.33

A. If BC-HV-F003A(B) RHR HX A(B) OUTLET VLV is used to throttle RHR flow through the RHR Heat Exchanger, then BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV must be fully open.

B. If BC-HV-F003A(B) RHR HX A(B) OUTLET VLV is CLOSED fully, then RHR INLET TO HX A(B) TE N004A(B) (CRIDS point A2380(A2382) OR TR-R605 point 1(2)), will not be accurate when BC-HV-F003A(B) is fully closed, then utilize RHR DISCH FROM HX A(B) TE N027A(B) (CRIDS point A2381(A2383) OR TR-R605 point 3(4)), to MONITOR system temperature. [CD-503B]

C. BC-HV-F003A(B) RHR HX A(B) OUTLET VLV should be opened slowly to avoid a "cold slug" discharge into the recirculation piping.

5.2.33 **PERFORM** the following as necessary to initiate the cooldown AND Control the cooldown rate, while maintaining the required RHR Shutdown Cooling flow, simultaneously: **[CD-133B]** ______

A. **PERFORM** the following to increase the cooldown rate: ______

1. Slowly **THROTTLE** OPEN on the BC-HV-F003A(B) RHR HX A(B) OUTLET VLV. ______

2. IF the BC-HV-F003A(B) is fully open, THEN, **THROTTLE** CLOSED on the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV. ______

Continued Next Page

5.2.33 (Continued)

B. **PERFORM** the following to decrease the cooldown rate: ______

1. **THROTTLE** OPEN on the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV. ______

2. IF the BC-HV-F048A(B) A(B) RHR HX SHELL SIDE BYP MOV is fully open,
THEN, **THROTTLE** CLOSED on the BC-HV-F003A(B) RHR HX A(B) OUTLET VLV. ______

C. **PERFORM** the following to maintain Shutdown Cooling flow: ______

THROTTLE the BC-HV-F015A(B) RHR LOOP A(B) RET TO RECIRC, as necessary, to maintain the required RHR Shutdown Cooling flow. ______

5.2.34 **RECORD** Reactor Vessel temperatures and pressures IAW Integrated Operating Procedure HC.OP-IO.ZZ-0004(Q); Shutdown from Rated Power to Cold Shutdown AND HC.OP-DL.ZZ-0026(Q), Attachment 3s; Surveillance Log. ______

A malfunction of the Digital Feedwater Level Controller has resulted in an INCREASING reactor water level. The Reactor Feedwater Pumps are automatically tripped on a high reactor water level signal to prevent:

a. feed pump damage due to increasing pump discharge flow rate and head.

b. main turbine damage due to water impingement on turbine blades.

c. reactor vessel damage due to completely filling and overpressurizing the vessel.

d. main steam line piping and hanger damage due to filling the main steam lines.

Answer	b	Exam Level	B	Cognitive Level	Memory	Facility	Hope Creek	Exam Date:	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				295008K304
295008	High Reactor Water Level							Record Number	13

AK3. Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL:

AK3.04 Reactor feed pump trip: Plant-Specific 3.3 3.5

Explanation of Answer: Feedpumps are tripped to prevent reactor overfill and damage to the main turbine.

Reference Title

TC Bases 3/4.3.9

Learning Objectives

000002E008	(R) Given a list of reactor vessel pressure and/or level setpoints determine the automatic action that occurs IAW the Lesson Plan.

Material Required for Examination

Question Source:	INPO Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	QID #6574 Dresden 03/11/1996		

Figure 6

Obj. 6d

3) Steam carryunder - mixing small amounts of steam from the reactor vessel steam dome area and steam separators with water in the downcomer area.

a) Caused by steam entering the bulk water region below the steam dryer skirt due to low water level

b) This results in lower downcomer water density which:

- Decreases indicated water level
- Decreases recirculation pump NPSH
- Decreases jet pump NPSH
- Reduces core inlet subcooling

c) Steam carryunder increases with as reactor water level decreases

Obj. 6c

4) Moisture carryover - mixing of small amounts of moisture with the steam exiting the reactor vessel steam dome to the main steam lines

a) Caused by reduced moisture separator/steam dryer efficiency due to high water level

b) Excessive carryover will damage the main turbine and feedwater pump turbine blading.

c) Moisture carryover increases with increasing water level.

c. LEVEL 8 (+54")

1) Above LEVEL 8, gross moisture carryover can occur, therefore protection of downstream components is necessary.

2) A LEVEL 8 trip initiates the following actions:

a) Main turbine trip

b) Reactor Feed Pump turbine (RFPT) trip.

c) RCIC and HPCI turbine trips.

3) The main turbine is tripped to protect it from blade damage caused by water impingement. The HPCI, RCIC and RFP turbines are tripped to prevent overfilling the vessel and flooding the main steam lines.

NRC IN 88-77

Obj. 11a, b, c

4) The NRC regards a reactor vessel overfill event as a significant safety concern, identifying the following four safety issues:

a) Hydrodynamic effects of water or two-phase fluid being discharged through the SRVs. This process could damage the SRVs.

b) Stressing of the reactor vessel main steam line nozzles, steam line snubbers, pipe supports and hangers as a result of:

- The thermal transient caused by colder water flowing into the hot main steam line and reactor vessel;
- The weight of water in the main steam lines; and
- The dynamic transient loads caused by water flow in the main steam lines.

c) Potential for MSIVs not to close if the main steam lines are filled with water

d) Placing the plant in a condition that has not been analyzed in the Final Safety Analysis Report (FSAR)

d. LEVEL 7(+39")

1) At LEVEL 7, moisture carryover in the main steam lines is expected to increase significantly.

3/4.3.8 DELETED

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of a high reactor vessel water level (Level 8) to mitigate potential damage to the main turbine.

Given the following:

- The plant is at 37% power
- Both CRD pumps are tripped on low suction pressure
- The Reactor Building Operator is swapping CRD suction filters
- CRD ACCUM TROUBLE Overhead Annuciator C6-D4 is clear
(Assume NO other operator actions)

Which one of the following describes the effect on gas pressure in the HCU Accumulators 2 minutes following the pump trip?

a. Stays the same because reactor pressure holds the charging water check valve closed

b. Stays the same because accumulator pressure holds the charging water check valve closed

c. Lowers because the reactor scrams

d. Lowers because the accumulator piston moves when charging water header pressure is lost

Answer	b	Exam Level	B	Cognitive Level	Comprehension	Facility	Hope Creek	Exam Date:	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				295022K203
295022	Loss of CRD Pumps							Record Number	31

AK2.	Knowledge of the interrelations between LOSS OF CRD PUMPS and the following:		
AK2.03	Accumulator pressures.	3.4	3.4

Explanation of Answer: Charging water check valve 115 maintains water volume on a loss of charging pressure from the CRD pumps. N2 gas pressure will remain the same as long as the check valve holds. If the check valve does not hold, the piston will stroke and N2 pressure will drop causing low accumulator pressure alarm.

Reference Title

HC.OP-IS.BF-0103 Purpose

Lesson Plan 00006

Learning Objectives

000006E017	(R) Given the appropriate procedure or access to the procedure, summarize the accumulator trouble alarms and their setpoints associated with each CRD HCU and how these problems may impact CRDH System Operation, IAW the Lesson Plan.

Material Required for Examination			
Question Source:	New	Question Modification Method:	
Question Source Comments:			

CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.0 PURPOSE

The purpose of this procedure is to demonstrate during plant refuel, the operability of the Control Rod Drive Accumulator Charging Water Check Valve, 1-BF-V115, as required by Technical Specification 4.0.5. This is performed by verifying each individual accumulator check valve maintains the associated accumulator pressure above the alarm setpoint for greater than or equal to 2 minutes, starting at normal operating pressure, with no Control Rod Drive Pump operating. **[CD-270A]**

2.0 PREREQUISITES

2.1 Charging Water Check Valve Exercise Test

2.1.1 Permission to perform this test has been obtained from the OS/CRS as indicated by the completion of Attachment 1, Section 1.0. ______

2.1.2 All personnel involved in the performance of this procedure, should complete Attachment 1, Section 3.0, prior to performing any part of this procedure. ______

2.1.3 No other testing <u>OR</u> maintenance is in progress that would adversely effect the performance of this test. ______

> **NOTE 2.1.4**
>
> All Control Rod Drive Scram Accumulators need not be OPERABLE to perform this test provided INOPERABLE accumulators are tracked IAW HC.OP-AP.ZZ-0108, Removal and Return of Equipment to Service, and this surveillance is listed in Part B of Attachment 1, Action Statement Log Sheet as required to restore the equipment to operability.

2.1.4 The Control Rod Drive system is in service <u>AND</u> all OPERABLE Hydraulic Control Units are charged to normal operating pressure IAW HC.OP-SO.BF-0001(Q), CRD Hydraulic System Operation. ______

2.1.5 All insertable control rods are inserted except for rods removed IAW Technical Specifications 3.9.10.1 and/or 3.9.10.2. ______

2.1.5 The plant is in Condition 4 or 5. ______

ACCEPT

Which one of the following gaseous radioactive release limits corresponds to the EOP-104 entry condition?

a. 500 mRem to the Thyroid CEDE

b. 5000 mRem to the Thyroid CEDE

c. 2 times 10CFR 20 Appendix B limits

d. 200 times 10CFR 20 Appendix B limits

Answer	d	Exam Level	R	Cognitive Level	Memory	Facility	Hope Creek	Exam Date:	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions			RO Group	2	SRO Group	1		295038A203
295038	High Off-Site Release Rate							Record Number	46

EA2.	Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE:		
EA2.03	Radiation levels	3.5	4.3

Explanation of Answer

JUSTIFICATION:
CORRECT - IAW ECG Section 6 and Lesson plan 0302-000.00H-000127, the alert value is 200 times the 10CFR20 Appendix B value

Reference Title

ECG Section 6.0

LP 0302-000.00H-000127

Learning Objectives

000127E002	Given a set of plant conditions, analyze and determine if entry conditions into HC.OP-EO.ZZ-0103/4 exists.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

GET RO E-PLAN L-PLAN

L.P. NO.: 0302-000.00H-000127-12
0303
0304
0305
PREVIOUS L.P. NO.: 0302-000.00H-000127-11
0302-000.00H-000128-09

NUCLEAR TRAINING CENTER
LESSON PLAN

PROGRAM TITLE:	HOPE CREEK LICENSED OPERATOR TRAINING
COURSE SECTION/MODULE:	OPERATING PROCEDURES
TOPIC/SUB MODULE:	EMERGENCY OPERATING PROCEDURES
LESSON:	HC.OP-EO.ZZ-0103/4 REACTOR BUILDING & RAD RELEASE CONTROL
DURATION:	2 HOUR (INITIAL) 1 HOUR (REQUAL)
PREREQUISITES:	HOPE CREEK SYSTEMS LP NO: 0302-000.00H-000121
JTA NO. OR QUALIFICATION STATEMENT NO.:	
AUTHOR:	F. W. Berg DATE: 06/09/99

REVIEW/APPROVAL SIGNATURES

SUBMITTED BY: Pete Doran DATE: 06/22/99

BARGAINING UNIT REPRESENTATIVE: ____________ DATE: ________
Badge #

PRINCIPAL TRAINING SUPERVISOR: ____________ DATE: ________
Badge #

LINE SUPERVISOR: ____________ DATE: ________
Badge #

COPY RECEIVED
WORD PROCESSING INITIALS: ____________

INSTRUCTOR REFERENCES:

1. BWROG Emergency Procedure Guidelines/Severe Accident Guidelines
2. BWROG Emergency Procedure Guidelines/Severe Accident Guidelines, Appendix A, B, C and D
3. HCGS Plant Specific Technical Guidelines
4. HCGS Technical Specifications
5. HCGS Event Classification Guide, NUMARC Revision
6. HCGS HC.OP-EO.ZZ-0103/4 Reactor Building & Rad Release Control
7. HCGS HC.OP-EO.ZZ-0103 Reactor Building & Rad Release Control Flowchart

TRAINING MATERIAL REQUIRED:

Lesson Plan
HC.OP-EO.ZZ-0103/4 Flowcharts (Current Revision)
HC.OP-EO.ZZ-0103/4 Bases document
Transparencies (If Applicable)
Procedures referenced will be available in the classroom for student reference

STUDENT HANDOUTS:

Lesson Plan 0302-000.00H-000127
HC.OP-EO.ZZ-0103/4 Flowcharts (Current Revision)
HC.OP-EO.ZZ-0103/4 Bases document
Procedures referenced will be available in the classroom for student reference

SPECIAL CLASSROOM REQUIREMENTS:

No special consideration for the classroom size or arrangement are required.

LEARNING OBJECTIVES:

TERMINAL LEARNING OBJECTIVES:

Provided a scenario or plant/system status or previous plant condition, the trainee will be able to:

1.0 Perform actions required to implement HC.OP-EO.ZZ-0103/4, for Secondary Containment Control. (1-6)

2.0 Recognize the entry condition(s) of HC.OP-EO.ZZ-0103/4, Reactor Building & Rad Release Control and utilize the flowchart to control reactor building parameters. (1-6)

3.0 Derive an expected actuation or continuation of the event based on basic knowledge presented during previous training. The trainee's response shall be consistent with and contain the essential elements identified on an answer key. (1-6)

ENABLING LEARNING OBJECTIVES:

Note: All ELOs are required to be covered during initial training. Those with an "R" prefix are the minimum required to be covered during requal training.

1. State the three purposes of the Reactor Building & Rad Release Control procedure, HC.OP-EO.ZZ-0103/4.

2. Given a set of plant conditions, analyze and determine if entry conditions into HC.OP-EO.ZZ-0103/4 exists.

R3. Define the term "Maximum Safe Operating Temperature".

R4. Define the term "Maximum Safe Operating Radiation Level".

R5. Define the term "Maximum Safe Floor Level".

R6. Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step.

b. The purpose of this step is to direct the operator to restore Reactor Building parameters to normal levels, which will assure the Reactor building integrity and to limit any potential releases to the environment.

c. The operator remains at this step until the exit criteria of the RB-1 override statement are met.

IV. RAD RELEASE CONTROL

A. The procedure is entered when radioactive release rates reach levels corresponding to 200 times 10CFR20, Appendix B Limits. These levels are high enough that they will not occur during normal operation, but still low enough that the immediate health and safety of the general public is not threatened by the release.

B. HC.OP- EO.ZZ-0103/4 isolates primary system discharges and controls RPV pressure through sequentially executed steps as required to minimize the offsite release of radioactivity. These steps provide the interface between individual events specifically addressed by the site Emergency Plan and the symptomatic control of RPV, primary containment, and reactor building parameters.

V. RAD RELEASE CONTROL PROCEDURE

Obj. 2

A. Conditions for Entry

The entry conditions to this procedure is:

GASEOUS RADIOACTIVE RELEASE ABOVE **ALERT**.

B. Procedural Steps

1. RR-1 Retention Override Steps

IF while executing the following steps:	THEN:
SAG entry is required	**EXIT** this procedure AND **ENTER** SAG

a. If primary containment flooding is required or hydrogen above 2% is generated, all EOPs are exited and the SAGs are entered. The SAGs then remain in effect until an emergency no longer exists.

b. Review HC.OP-EO.ZZ-0103/4 bases for these steps.

Obj. 6

2. RR-2 **RESTORE** Rad Release Rate below ALERT level

a. Entry from RR-1, or any of the following:

- Yes response on step RR-6
- No response on RR-10
- Completion of actions from steps RR-9 and RR-11

b. The entry condition for Radioactivity Release Control corresponds to an action level defined in the site Emergency Plan.

Table 3

c. Refer to section 6.1 of HCGS ECG

d. Review HC.OP-EO.ZZ-0103/4 bases for this step.

Obj. 6

3. RR-3 Can Release Rate be Maintained below ALERT Level

a. Entry from step RR-2

b. A **yes** response directs operator to exit procedure.

c. A **no** response directs operator to additional actions in step RR-4

d. Review HC.OP-EO.ZZ-0103/4 bases for this step.

Obj. 6

4. RR-4 Retention Override Steps

IF while executing the following steps:	THEN:
Turbine Bldg Ventilation System is shutdown	**RESTART** Turbine Bldg Ventilation System

a. Entered on a **no** response from RR-3.

b. Review HC.OP-EO.ZZ-0103/4 bases for this step.

Obj. 6

5. RR-5 Except systems required to:

- Assure adequate core cooling
- Shutdown the reactor

ISOLATE all primary systems discharging into areas outside Primary Containment and Rx Bldg

a. Entry from RR-4

b. Review HC.OP-EO.ZZ-0103/4 bases for this step.

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

Initiating Condition	Any **Unplanned** Release of Gaseous Radioactivity to Environment that Exceeds 200 Times the Radiological Technical Specifications for 15 minutes or longer		Any **Unplanned** Release of Gaseous Radioactivity to the Environment that Exceeds 200 times the 10CFR20, Appendix B limits for 30 minutes or longer	
OPCON	All	All	All	All
EAL #	**6.1.2.a**	**6.1.2.b**	**6.1.2.c**	**6.1.2.d**
EMERGENCY ACTION LEVELS	Dose Assessment **IF** Dose Assessment indicates <u>EITHER</u> one of the following at the MEA or beyond as calculated on the SSCL: • TEDE 4-Day Dose $\geq$ **2.0E+01 mRem** • Thyroid-CDE Dose $\geq$ **6.8E+01 mRem** based on Plant Vent effluent sample analysis and <u>NOT</u>on a default Noble Gas to Iodine Ratio	Field Measured Dose Rate **IF** Dose Rate measured at the Protected Area Boundary or beyond <u>EXCEEDS</u> **5 mRem/hr**	Sample Analysis **IF** Gaseous effluent release sample analysis for <u>ANY</u> one of the following indicates a concentration of: • **FRVS:** **$\geq$ 1.13E-01 µCi/cc Total Noble Gas** **$\geq$ 2.71E-05 µCi/cc I-131** • **NPV:** **$\geq$ 2.43E-02 µCi/cc Total Noble Gas** **$\geq$ 5.81E-06 µCi/cc I-131** • **SPV:** **$\geq$ 2.27E-03 µCi/cc Total Noble Gas** **$\geq$ 5.44E-07 µCi/cc I-131**	Alarm Indications **IF** Valid High Alarm received from <u>ANY</u> one of the following Plant Effluent RMS Channels: • **FRVS Noble Gas** (Grid 1/3; 9RX680) • **NPV Noble Gas** (Grid 1/3; 9RX590) • **NPV Iodine** (Grid 3; 9RX601) • **SPV Noble Gas** (Grid 1/3; 9RX580) • **SPV Iodine** (Grid 3; 9RX605) • **HTV Noble Gas** (Grid 3; 9RX516) **AND** Total Plant Vent release rate <u>EXCEEDS</u> <u>EITHER</u>one of the following limits: • **4.80E+05 µCi/sec Total Noble Gas** • **1.15E+02 µCi/sec I-131** (NPV & SPV <u>ONLY</u>)
	AND		**AND**	
			Dose Assessment results <u>NOT</u> available	
			AND	
	Release is ongoing for $\geq$ **15 minutes**		Release is ongoing for $\geq$ **30 minutes**	
	THEN		**THEN**	
Action Required	Refer to Attachment 2 **ALERT**			

Given the following:

- A large break LOCA has occurred inside the Drywell
- Multiple equipment failures occurred
- Drywell pressure is 15 psig
- Steam cooling was required until water level was restored above TAF with Fire Water
- The Containment H2/O2 Analyzers were placed in service
- The High Hydrogen alarms are clear

Which one of the following actions is required IAW EOP-102?

a. Vent the Drywell because Hydrogen concentration is above 2%

b. Exit EOP-102 and enter SAG because Hydrogen concentration is above 2%

c. Vent the Suppression Chamber because Hydrogen concentration is below 2%

d. Place the Hydrogen Recombiners in service because Hydrogen concentration is below 2%

Answer	b	Exam Level	B	Cognitive Level	Application	Facility	Hope Creek	Exam Date:	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				500000K303
500000	High Containment Hydrogen Concentration							Record Number	48

EK3. Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS:

EK3.03 Operation of hydrogen and oxygen recombiners 3.0 3.5

Explanation of Answer: High H2 alarms come in at 2% Hydrogen. Since the H2 concentration is above 2%, EOP-102 step PC/H1 directs exit from EOP-102 and enter SAG

Reference Title

EOP-102 step PC/H1

Learning Objectives

00126AE004	Recall the reasons why the following are used for determining the entry condition and / or subsequent actions IAW the Primary Containment Control - Drywell Lesson Plan. a. Drywell Pressure b. Average Drywell Temperature c. H2 and O2 concentrations in the drywell

Material Required for Examination: EOP Flowcharts without entry conditions

Question Source: New **Question Modification Method:**

Question Source Comments:

IF while executing the following step:	THEN:
Primary containment isolation occurs	**PLACE** H_2/O_2 analyzers back in service, if necessary.
H_2/O_2 analyzer system is OR becomes unavailable after warmup	**SAMPLE** drwl AND supp chamber for H_2 AND O_2
H_2 concentration exceeds **2 %**	**EXIT** this procedure and **ENTER** SAG

PC/H-1

MONITOR H_2 AND O_2 concentrations in the Supp Chamber AND the Drwl

PC/H-2

IF H_2 concentration in the Drwl reaches .5%, **PLACE** the H_2 recombiners in service.

PC/H-3

Which one of the following describes when the Reactor Mode Switch Shutdown position scram may be bypassed?

a. When moving the mode switch from REFUEL to SHUTDOWN

b. When moving the mode switch from SHUTDOWN to REFUEL

c. When testing the "One Rod Out Interlock"

d. When a control rod blade is being uncoupled

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				212000G123
212000	Reactor Protection System							**Record Number**	68
2.1	Conduct of Operations								
2.1.23	Ability to perform specific system and integrated plant procedures during different modes of plant operation.							3.9	4.0

Explanation of Answer: The Reactor Mode Switch Shutdown position scram may be bypassed to move the MS from refuel to Shutdown when all control rods are fully inserted or the reactor is defueled.

Reference Title
HC.OP-SO.SB-0001 Prereq 2.6.2

Learning Objectives	
000022E004	(R) From memory, identify the parameters which initiate a Reactor Scram, list the scram initiation setpoints for each identified parameter, and determine when the parameter is bypassed, IAW the Lesson Plan.

Material Required for Examination			
Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

SRO
specific
problem
- look at
PR

PSEG NUCLEAR L.L.C.

HOPE CREEK GENERATING STATION

HC.OP-SO.SB-0001(Q) - Rev. 16

REACTOR PROTECTION SYSTEM OPERATION

USE CATEGORY: II

FIELD COPY EXISTS

REVISION SUMMARY

1. Order 80013429 (Pro-Trac #2306) This procedure has been revised to include the words "Field Copy Exists" at the top. This change has been incorporated to provide a second verification for the clerks during implementation of procedures.
2. IAW Generic procedure change identified under CR 990219121, (Closed To File), Limitations 3.2.2 was removed from this procedure. The direction contained in the Limitations regarding partial procedure performance and sequential performance of steps is adequately contained within higher tier administrative procedures including NC.NA-AP.ZZ-0001(Q) and NC.NA-WG.ZZ-0001(Q).
3. All procedure references to WCCS have been deleted. Editorial change, conforms to existing procedures.
4. **Order 80007129** DCP 4EC-3192 (CD-547) Editorial Changes
   - Step 5.3.6.K.1 has been revised to read "OR the RWM Shutdown Confirmation Screen." versus Process Computer may need OD-3 EDIT run for a rescan.
   - Step 5.2.2 [3rd bullet] has been revised to change "NSSS CRT" to CMS. Procedure reviewer comment.
5. **Order 80018874**
   Revised step 5.1.7 wording to restore to as written in revision 10 when it was distinct from similar direction in step 5.1.2.D. Revision 11 attempted to make the steps similarly written for consistency but inadvertently introduced a typographical error in describing steps 5.1.7 check as an "unloaded' check, when in actuality it had been correctly identified as a "loaded' check in revision 10 and prior.
6. Based on writer review all occurrences of the action term "CHECK" have been replaced with "VERIFY".

This entire revision can be considered editorial in nature.

IMPLEMENTATION REQUIREMENTS

None

Effective date 8/13/01

APPROVED: ______________________ 081001
Manager - Hope Creek Operations   Date

REACTOR PROTECTION SYSTEM OPERATION

TABLE OF CONTENTS

REACTOR PROTECTION SYSTEM OPERATION

1.0 PURPOSE

This procedure outlines the steps necessary for the operation of the Reactor Protection System (RPS).

2.0 PREREQUISITES

2.1 System Operability Observation

2.1.1 All personnel performing any steps in this procedure should complete Attachment 1, Section 2.0, prior to performing any part of this procedure. ______

2.1.2 Reactor Protection System Electrical Lineup is aligned for the applicable plant condition. ______

2.2 RPS Scram

Automatic
OR Manual Scram initiated. ______

2.3 Resetting RPS Trips

2.3.1 All personnel performing any steps in this procedure should complete Attachment 1, Section 2.0, prior to performing any part of this procedure. ______

2.3.2 Automatic
OR Manual Half
OR Full Scram initiated and the
initiating trip signal(s) have cleared. ______

2.3.3 Reset ARI/RRCS prior to resetting the Scram,
IF it had initiated.

2.4 Transfer of RPS Bus A(B) Power - from RPS MG Set A(B) to RPS Alternate Transformer A(B)

2.4.1 All personnel performing any steps in this procedure should complete Attachment 1, Section 2.0, prior to performing any part of this procedure. ______

2.4.2 Reactor Protection System Electrical Lineup for the Alternate Feed is aligned for the applicable plant condition. ______

2.4.3 The opposite RPS Bus B(A) is NOT being supplied by its RPS Alternate Feed. ______

2.4.4 IF the Main Turbine is shutdown (Turbine Stop and/or Control Valves closed), THEN the EOC-RPT bypass switches are in the BYPASS position. ______

2.5 **Transfer of RPS Bus A(B) Power - from RPS Alternate Transformer A(B) to RPS MG Set A(B)**

2.5.1 All personnel performing any steps in this procedure should complete Attachment 1, Section 2.0, prior to performing any part of this procedure. ______

2.5.2 Reactor Protection System Electrical Lineup for the RPS MG Set A(B), to be placed in service, is aligned for the applicable plant condition. ______

2.5.3 IF the Main Turbine is shutdown (Turbine Stop and/or Control Valves closed), THEN the EOC-RPT bypass switches are in the BYPASS position. ______

2.6 **Bypassing Reactor Mode Switch Scram**

2.6.1 Permission to perform this test has been obtained from the OS/CRS. ______

2.6.2 All Control Rods are inserted OR Reactor is defueled. ______

2.6.3 No Core Alterations are in progress. ______

2.6.4 Banana Jacks have been installed on the terminals to be jumpered in order to facilitate the required jumpering. ______

NOTE 5.6

This section temporarily bypasses the Reactor Mode Switch in SHUTDOWN scram signal, until the signal is automatically bypassed after the 10 second time delay. This is not a normal evolution and should only be used to bypass this scram signal when placing the Reactor Mode Switch from REFUEL to SHUTDOWN. **[PR 981028191]**

The Mode Switch shall be considered inoperable while the jumpers are installed. The Mode Switch should be considered operable when removal of the jumpers has been independently verified and documented on Attachment 5.

5.6 Bypassing Reactor Mode Switch Scram

5.6.1 **ENSURE** all prerequisites of Section 2.6 are satisfied
AND **INITIAL** Attachment 5 to document completion and verification. ______

5.6.2 **DIRECT** I&C to perform the following: ______

A. **CONNECT** a jumper between terminal Z 9
AND terminal ZZ 23
in BAY A of 10-C609
AND I&C and Verifier
INITIAL Attachment 5. ______

B. **CONNECT** a jumper between terminal A 9
AND terminal BB 23
in BAY F of 10-C609
AND I&C and Verifier
INITIAL Attachment 5. ______

C. **CONNECT** a jumper between terminal Z 9
AND terminal ZZ 23
in BAY A of 10-C611
AND I&C and verifier
INITIAL Attachment 5. ______

D. **CONNECT** a jumper between terminal A 9
AND terminal BB 23
in BAY F of 10-C611
AND I&C and verifier
INITIAL Attachment 5. ______

TRAINING MATERIAL REQUIRED:

Lesson Plan

Transparencies

Technical Specifications

STUDENT HANDOUTS:

Reactor Protection System Lesson Plan 0301-000.00H-000022

Instructor References (as required)

SPECIAL CLASSROOM REQUIREMENTS:

No special considerations for classroom size or arrangements are necessary

LEARNING OBJECTIVES:

Provided a scenario of plant/system status or previous plant conditions associated with the operation of the Reactor Protection system, in accordance with the applicable trainee handout and system operating procedures, the trainee will:

1.0 State the purpose/design criteria of the system. (1,2,3,4,8,13,14)

2.0 Identify system configuration including major system components, flowpaths, and protective devices. (2,3,6,8,14)

3.0 Identify electrical power sources (4.16 KV, 480 VAC, 250 VDC, 1E, etc.), actuation signals, interlocks and alarms associated with system operation. (2,4,6,7)

4.0 Describe Technical Specifications requirements associated with the system. (13,18)

5.0 Critique plant problems and industry events associated with the system. (12)

6.0 Use plant procedures associated with system operation. (6,10,11,16)

7.0 Evaluate system interrelationships. (2,4,6,11)

8.0 Describe system configuration following emergency system actuation. (4,5,9,11,14)

ENABLING LEARNING OBJECTIVES:

NOTE: All ELOs are required to be covered during initial training. Those with an "R" prefix are the minimum required to be covered during continuing/requal training.

1. From memory, state the purpose of the Reactor Protection System (RPS), IAW the Lesson Plan.

2. Given the Hope Creek approved Electrical Load List and applicable electrical drawings:

   a. Identify the normal and alternate sources of power to RPS Bus A and RPS Bus B, IAW the Lesson Plan.

   b. Identify the power supplies to RPS MG Set A and RPS MG Set B, IAW the Lesson Plan.

3. From memory, state the purpose of the RPS MG Set Flywheel, IAW the Lesson Plan.

R4. From memory, identify the parameters which initiate a Reactor Scram, list the scram initiation setpoints for each identified parameter, and determine when the parameter is bypassed, IAW the Lesson Plan.

R5. From memory, define the following terms:

a. Scram, IAW the Lesson Plan.

b. Half-scram, IAW Lesson Plan.

R6. Given the appropriate system operating procedure explain the effects on the reactor protection system when the power source is transferred from normal to alternate, and vice versa, IAW the RPS System Operating Procedure.

7. From memory, state the purpose of the time delay after a scram, IAW the Lesson Plan.

8. Given labeled diagrams/drawings of the RPS trip logics, explain the purpose of the RPS shorting links, IAW the Lesson Plan.

R9. Given plant conditions, evaluate the response of RPS to an electrical failure, IAW the Lesson Plan.

10. From memory, explain how a scram can be manually initiated, IAW the RPS System Operating Procedure.

11. From memory, state the immediate operator action(s) for a Loss of RPS channel IAW HC.OP-AB.ZZ-0110.

R12. Given plant problems/industry events associated with the Reactor Protection System.

a. Discuss the root cause of the plant problem/industry events, IAW the Lesson Plan.

b. Discuss the HCGS design and/or procedural guidelines that mitigate/reduce the likelihood of the problem/industry event, IAW the Lesson Plan.

c. Discuss the "Lessons Learned" from the plant problem/industry event, IAW the Lesson Plan.

Table 1

REACTOR SCRAMS

	Parameter	Scram Setpoints	Bypassed
1.	IRM Neutron Flux Upscale	120/125 of full scale	Rx. Mode Sw in RUN
2.	APRM Neutron Flux Upscale	15% of Rated Thermal	Rx. Mode Sw in RUN
3.	Flow Biased Simulated Th. Pwr. Upscale	.66(W-ΔW)+51% Clamped-@ 113.5%	
4.	Fixed Neutron Flux Upscale	118%	Rx Mode Sw Out of RUN
5.	Reactor Vessel Steam Dome Pressure - High	1037 psig	
6.	Reactor Vessel Water Lvl - Low Level 3	≤ 12.5" (Level 3)	
7.	Main Steam Line Isolation Valve	< 92% open	Rx Mode Sw Out of RUN
8.	Primary Containment Pressure High	1.68 psig	
9.	Scram Discharge Volume water level - High	Elev. 110' 10.5"	Rx Mode Sw in S/D or REFUEL & Disch. Vol. Scram Bypassed
10.	Turbine Stop Valve - Closure	≤ 95% open	<30% Rx Power (135.7 psig Turb 1st Stage Pressure)
11.	Turbine Control Valve Fast Closure Trip Oil Pressure Low	530 psig (ETS Oil Pressure)	<30% Rx Power (135.7 psig Turb 1st Stage Pressure)
12.	Rx. Mode Switch in Shutdown	N/A	6 (+/-4) seconds after the Mode Switch is in SHUTDOWN
13.	Manual Scram	N/A	
14.	SRM High Count Rate	$2x10^5$ cps	Shorting Links Installed
15.	IRM Inoperative	• High Voltage P.S. Low, • Function Switch Not In Operate, • Module Unplugged	Rx Mode Switch in RUN
16.	APRM Inoperative	• Function Switch Not in Operate • <14 LPRM Inputs • Module unplugged	

Given the following:

- Local Power Range Monitor (LPRM) detector 32-33-C has just failed downscale
- Subsequently, Control Rod 30-31 is selected

Which one of the following describes the effect of the failure on the associated APRM and RBM channels?

The LPRM input:

a. will be automatically bypassed and removed from the APRM only. The APRM and RBM readings will be lower than actual power.

b. will be automatically bypassed and removed from both the APRM and RBM. The APRM and RBM readings will remain the same.

c. will be automatically bypassed and removed from the APRM only. The APRM reading will remain the same and the RBM reading will be lower than actual power.

d. will be automatically bypassed and removed from the RBM only. The APRM and the RBM readings will be lower than actual power.

Answer	d	Exam Level	B	Cognitive Level	Comprehension	Facility	Hope Creek	Exam Date:	03/12/2002
Tier:	Plant Systems			RO Group	1	SRO Group	1		215005K305
215005	Average Power Range Monitor/Local Power Range Monitor System							Record Number	73
K3.	Knowledge of the effect that a loss or malfunction of the APRM/LPRM will have on following:								
K3.05	Reactor power indication							3.8	3.8

Explanation of Answer

The LPRM must be manually bypassed to remove from the APRM averaging circuit. The LPRM is automatically bypassed in the RBM Count Circuit if the detector is reading <4%. Since the LPRM is still feeding the APRM avg, the indicated avg will be lower. Since the control rod is selected after the LPRM fails downscale so the gain change circuit will null to the now lower APRM reference signal.

Reference Title

HC.OP-SO.SF-0002

Learning Objectives

000017E008 Given the applicable drawing, determine how the Rod Block Monitor (RBM) System interrelates with the following systems:

a. Local Power Range Monitoring (LPRM) System
b. Average Power Range Monitoring (APRM) System
c. Recirculation Flow Units
d. 120 VAC Instrument Power System
e. 120 VAC Un-interruptible Power Supply System
f. Reactor Manual Control System (RMCS)

IAW the Rod Block Monitor (RBM) System Lesson Plan

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: QID# 12556 Limerick 11/10/1995

ATTACHMENT 1
Calculation/Data Sheets
Page 1 of 8

APRM CHANNEL A

LPRM Location	GAF*	EV (V)	I_o (uA)	Calibration Verified By	I_c (uA)
32-49B					
16-33B					
48-33B					
32-17B					
16-49D					
48-49D					
32-33D					
16-17D					
48-17D					
24-57A					
08-41A					
40-41A					
24-25A					
56-25A					
40-09A					
40-57C					
24-41C					
56-41C					
08-25C					
40-25C					
24-09C					

Pre-calibration APRM power level: ___________ V

Post-calibration APRM power level: ___________ V

* **Calibrated LPRM Reading instead of the GAF**

ATTACHMENT 1
Calculation/Data Sheets
Page 2 of 8

APRM CHANNEL B

LPRM Location	GAF*	EV (V)	I_o (uA)	Calibration Verified By	I_c (uA)
08-49B					
40-49B					
24-33B					
56-33B					
08-17B					
40-17B					
24-49D					
08-33D					
40-33D					
24-17D					
56-17D					
32-57A					
16-41A					
48-41A					
32-25A					
16-09A					
48-09A					
16-57C					
32-41C					
16-25C					
48-25C					
32-09C					

Pre-calibration APRM power level: ___________ V

Post-calibration APRM power level: ___________ V

* **Calibrated LPRM Reading instead of the GAF**

ATTACHMENT 1
Calculation/Data Sheets
Page 3 of 8

APRM CHANNEL C

LPRM Location	GAF*	EV (V)	I_o (uA)	Calibration Verified By	I_c (uA)
24-57B					
08-41B					
40-41B					
24-25B					
56-25B					
40-09B					
40-57D					
24-41D					
56-41D					
08-25D					
40-25D					
24-09D					
16-49A					
48-49A					
32-33A					
16-17A					
48-17A					
32-49C					
16-33C					
48-33C					
32-17C					

Pre-calibration APRM power level: ____________ V

Post-calibration APRM power level: ____________ V

* **Calibrated LPRM Reading instead of the GAF**

APRM CHANNEL D

LPRM Location	GAF*	EV (V)	I_o (uA)	Calibration Verified By	I_c (uA)
32-57B					
16-41B					
48-41B					
32-25B					
16-09B					
48-09B					
16-57D					
32-41D					
16-25D					
48-25D					
32-09D					
24-49A					
08-33A					
40-33A					
24-17A					
56-17A					
08-49C					
40-49C					
24-33C					
56-33C					
08-17C					
40-17C					

Pre-calibration APRM power level: __________ V

Post-calibration APRM power level: __________ V

* **Calibrated LPRM Reading instead of the GAF**

ATTACHMENT 1
Calculation/Data Sheets
Page 5 of 8

APRM CHANNEL E

LPRM Location	GAF*	EV (V)	I_o (uA)	Calibration Verified By	I_c (uA)
16-49B					
48-49B					
32-33B					
16-17B					
48-17B					
32-49D					
16-33D					
48-33D					
32-17D					
40-57A					
24-41A					
56-41A					
08-25A					
40-25A					
24-09A					
24-57C					
08-41C					
40-41C					
24-25C					
56-25C					
40-09C					

Pre-calibration APRM power level: __________ V

Post-calibration APRM power level: __________ V

* **Calibrated LPRM Reading instead of the GAF**

ATTACHMENT 1
Calculation/Data Sheets
Page 6 of 8

APRM CHANNEL F

LPRM Location	GAF*	EV (V)	I_o (uA)	Calibration Verified By	I_c (uA)
24-49B					
08-33B					
40-33B					
24-17B					
56-17B					
08-49D					
40-49D					
24-33D					
56-33D					
08-17D					
40-17D					
16-57A					
32-41A					
16-25A					
48-25A					
32-09A					
32-57C					
16-41C					
48-41C					
32-25C					
16-09C					
48-09C					

Pre-calibration APRM power level: ___________ V

Post-calibration APRM power level: ___________ V

*** Calibrated LPRM Reading instead of the GAF**

ATTACHMENT 1
Calculation/Data Sheets
Page 7 of 8

LPRM GROUP A

LPRM Location	GAF*	EV (V)	I_o (uA)	Calibration Verified By	I_c (uA)
40-57B					
24-41B					
56-41B					
08-25B					
40-25B					
24-09B					
24-57D					
08-41D					
40-41D					
24-25D					
56-25D					
40-09D					
32-49A					
16-33A					
48-33A					
32-17A					
16-49C					
48-49C					
32-33C					
16-17C					
48-17C					

* **Calibrated LPRM Reading instead of the GAF**

ATTACHMENT 1
Calculation/Data Sheets
Page 8 of 8

LPRM GROUP B

LPRM Location	GAF*	EV (V)	I_o (uA)	Calibration Verified By	I_c (uA)
16-57B					
32-41B					
16-25B					
48-25B					
32-09B					
32-57D					
16-41D					
48-41D					
32-25D					
16-09D					
48-09D					
08-49A					
40-49A					
24-33A					
56-33A					
08-17A					
40-17A					
24-49C					
08-33C					
40-33C					
24-17C					
56-17C					

* **Calibrated LPRM Reading instead of the GAF**

Given the following:

- The plant is operating at 100% power
- "A" Control Room HVAC train and Chilled Water system is running
- A light haze with an acrid odor is noticed in the Main Control Room
- No alarms are received that could explain the origin of the haze and odor
- HC.OP-AB.ZZ-0129, High Radiation, Smoke or Toxic Gases in the Control Room Air Supply is entered

Based on plant conditions, which one of the following is an immediate action IAW HC.OP-AB.ZZ-0129?

a. Verify that the Control Room Supply Ventilation has automatically isolated

b. Verify that the "A" Control Room Emergency Filter Unit automatically started

c. Press the CONTROL ROOM EMER FILTER UNIT A and B OA pushbuttons

d. Press the CONTROL ROOM EMER FILTER UNIT A and B RECIRC MODE pushbuttons

Answer	d	Exam Level	B	Cognitive Level	Memory	Facility	Hope Creek	Exam Date:	03/12/2002
Tier:	Plant Systems	RO Group	2	SRO Group	2				290003K501
290003	Control Room HVAC							Record Number	106

K5. Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC

K5.01 Airborne contamination (e.g., radiological, toxic gas, smoke) control 3.2 3.5

Explanation of Answer

Press the CONTROL ROOM EMER FILTER UNIT A and B RECIRC MODE pushbuttons. For a toxic gas in the Control Room Supply, isolate Control Room Ventilation and place CREF in the Recirc Mode.
INCORRECT - Press the CONTROL ROOM EMER FILTER UNIT A and B OA pushbuttons. CREF must be in the Recirc Mode for a toxic gas event.
INCORRECT - Verify that the Control Room Supply Ventilation has automatically isolated. Toxic gas will not automatically isolate Control Room Ventilation. Only high rad.
INCORRECT -Verify that the "A" Control Room Emergency Filter Unit automatically started. Does not automatically start on toxic gas, only high rads.

Reference Title

HC.OP-AB.ZZ-0129

Learning Objectives

0AB129E002	(R) From memory, recall the Immediate Operator Actions for High Radiation, Smoke or Toxic Gases in the Contorl Room Air Supply, Abnormal Operating Procedure.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: VISION BANK QID# Q61261

Approved: [signature] 1/29/98
Operations Manager - HCO Date

CATEGORY II

HIGH RADIATION, SMOKE OR TOXIC GASES IN THE CONTROL ROOM AIR SUPPLY

PSE&G CONTROL COPY #

1.0 **SYMPTOMS**

1.1 Alarms

a. RADIATION MONITORING ALARM/TRBL

b. CONTROL RM AIRBORNE ACTIVITY HI

c. FIRE PROT PANEL 10C671

1.2 Increased activity reading for the control room air supply radiation monitors.

1.3 Smoke in the Control Room air supply

1.4 Pungent odor in the Control Room air supply

2.0 **AUTOMATIC ACTIONS**

2.1 Control room ventilation isolates and Control Room Emergency Filter Unit (CREF) starts (2×10^{-5} micro curies/cc) in 0A Mode.

3.0 **IMMEDIATE OPERATOR ACTIONS**

3.1 If smoke OR toxic gases are detected in the control room air supply, isolate the Control Room ventilation and place the CREF in the RECIRC MODE.

3.2 Ensure that all appropriate automatic actions are complete.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

4.1 Ensure that all appropriate immediate operator actions are complete.

4.2 IF the control room atmosphere becomes smoke filled OR is suspected of being contaminated by toxic gases OR airborne radioactivity, don protective clothing AND respiratory equipment as necessary.

4.3 IF high radiation is detected in the air supply intake request the Radiation Protection Department to survey the control room AND limit access as necessary.

4.4 Notify Site Protection IF smoke OR toxic gases are detected in the control room air supply.

4.5 Notify plant personnel of the high radiation, smoke, OR toxic gas in the control room air supply.

CAUTION 4.6

Protective clothing and respiratory equipment maybe required for the personnel involved.

4.6 Determine the source of the high radiation, toxic gas, OR smoke AND initiate corrective action.

CAUTION 4.7

Control Room personnel should not remove protective clothing OR respiratory equipment until Radiation Protection Department has lifted the requirements.

4.7 Implement the HCGS Emergency Plan IF the appropriate entry criteria is satisfied.

4.8 After the source of the contaminated atmosphere has been located AND isolated, ventilate the control room.

4.9 IF it is determined that it is necessary to evacuate the control room, implement procedure HC.OP-AB.ZZ-O130(Q).

5.0 DISCUSSION

5.1 The Control Room Ventilation System will automatically isolate on high radiation in the outside air supply or a LOCA signal.

5.2 The ISOLATE pushbutton for the control room supply (CRS) unit will close one inlet isolation damper to each CRS unit and the Control Area Exhaust (CAE) fan, trip the CAE fan, and start the associated CREF unit.

Continued Next Page

5.3 (Continued)

5.3 The CREF has two modes of operation:

a. OA MODE (pressurizing) is an automatic mode following detection of high airborne radioactivity in the control room normal air intake, in which 1000 cfm of outside air is mixed with 3000 cfm of control room return air, thus pressurizing the control room.

b. RECIRC MODE (recirculation or isolation) is when the outside air intake isolation damper for the CREF unit is closed. In this mode 4000 cfm of return air is circulated through the CREF unit without the introduction of outside air. However, this mode is not used following a radiological accident.

5.4 The loss of airflow in the CREF system automatically trips and isolates the operating train and alarms in the main control room. Manual operation is required to start the standby train. Loss of airflow <u>AND/OR</u> high pressure differential across the filter train are alarmed in the main control room.

5.5 The protective clothing described in Step 4.2 is non-radiological protective clothing. Use of Radiological protective clothing is restricted to the RCA only.

5.6 The existence of this procedure fulfills the requirements of the following Closing Documents:

CD-181X	NHO HSAR F06-0126-00
CD-176X	NHO HSAR F06-0119-00
CD-175X	NHO HSAR F06-0118-00

5.3 **Manual Isolation**

NOTE 5.3

A. The system can be manually isolated 50 sec after a Process Inhibit Signal is generated by a LOP or LOCA.

B. Isolation of Control Area Supply Unit should be for the running unit.

B8r6n

5.3.1 **ENSURE** that all prerequisites have been satisfied IAW Section 2.3. ______

5.3.2 **PRESS** HD-9598A(B) CONTROL AREA ISOLATION DAMPERS A(B) ISOLATE PB. ______

5.3.3 **OBSERVE** that CONTROL RM EMER FILTER UNIT A(B)V400 START is ON. ______

NOTE 5.3.4

The Control Room Emergency Filter Unit is only intended to be operated in the recirculation mode IF smoke or toxic gases are detected in the control room air supply. This mode is not automatically initiated following a radiological accident

5.3.4 IF required to operate Control Room Emergency Filter Unit in recirculation mode,
PERFORM the following: ______

A. **PRESS** CONTROL RM EMER FILTER UNIT A(B) RECIRC MODE. ______

B. **OBSERVE** that OA DAMPER HD-9593A(B) CLOSED is ON. ______

END

Given the following:

- Tech Spec compliance has been verified IAW "Refueling Operations". [HC.OP-IO.ZZ-0009]
- Multiple Control Rod Drive Mechanisms are being removed IAW Technical Specification 3.9.10.2
- Spiral Fuel offload is in progress per directions of Reactor Engineers and Fuel Handling Control Core Alteration forms. [HC.RE-FR.ZZ-0001]
- 14 Fuel Assemblies are remaining in the Vessel

Which one of the following conditions would require a formal declaration of Suspension of Core Alterations as described in plant procedures?

a. Spent Fuel Storage Area Radiation Monitor in alarm while transporting LPRMS through the Cattle Shute

b. All SRMs indicate between 2.1 & 2.6 cps

c. Mode Switch position change from Shutdown to Refuel for Rod Speed adjustments per system operating procedure

d. Refueling Bridge Platform surveillance identifies Frame Mounted hoist up travel stops are out of Technical Specification tolerance

Answer	a	Exam Level	R	Cognitive Level	Application	Facility	Hope Creek	Exam Date:	03/12/2002
Tier:	Generic Knowledge and Abilities			RO Group	1	SRO Group	1		294001G227
GENERIC								Record Number	116

2.2	Equipment Control		
2.2.27	Knowledge of the refueling process.	2.6	3.5

Explanation of Answer

Justification
HC.OP-IO.ZZ-0009, directs use of NC.NA-AP.ZZ-0049, for direction on formal suspension of fuel handling activities, adverse radiological conditions are one of the criteria.
Additionally, Refuel Radiation Area Alarms is an entry condition for HC.OP-AB.ZZ-0101 "Irradiated Fuel Damage" which directs suspension of all refueling operations.
Other choices are all within the Allowable Technical Specification boundaries for Core Alterations.

Reference Title

NC.NA-AP.ZZ-0049

Learning Objectives

00112IE004	(R) Apply Precautions, Limitations and Notes while executing the REFUELING OPERATIONS Integrated Operating Procedure

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Vision Bank QID# Q58930		

the following terms, IAW NC.NA-AP.ZZ-0037:

a. SPCC [Spill Prevention, Control, and Countermeasure Plan]

b. DPCC [Discharge Prevention, Containment, and Countermeasure Plan]

55. Provided access to control room references, initiate chemical item classification permit, IAW NC.NA-AP.ZZ-0038.

56. Given access to control room references, evaluate a CICP and determine limitations on use/disposal prior to start of job, IAW NC.NA-AP.ZZ-0038.

57. State the purpose of Station Aids, IAW NC.NA-AP.ZZ-0044.

58. Determine if a proposed Station Aid may be posted, IAW NC.NA-AP.ZZ-0044.

59. Define the following terms, IAW NC.NA-AP.ZZ-0049:

    a. Core Alterations

    b. Suspension of Fuel Handing (or Core Alterations

60. State the responsibilities of the Refueling SRO, IAW NC.NA-AP.ZZ-0049. **(SRO ONLY)**

61. State the responsibilities of the Refueling Bridge Operator, IAW NC.NA-AP.ZZ-0049.

62. State the responsibilities of the Control Room Refuel Monitor, IAW NC.NA-AP.ZZ-0049.

63. Provided access to control room references, determine the required actions if a fuel assembly is found in an incorrect core location, IAW NC.NA-AP.ZZ-0049.

64. State the minimum fuel handling crew requirement for fuel handling involving core alterations, IAW NC.NA-AP.ZZ-0049.

65. State the minimum fuel handling crew requirement for non-core alteration fuel handling, IAW NC.NA-AP.ZZ-0049.

66. Determine the conditions under which handling of fuel must be suspended, IAW NC.NA-AP.ZZ-0049. **(SRO ONLY)**

67. Determine the types of maintenance activities that will typically require a Post Maintenance Test and/or Operability Retest, IAW NC.NA-AP.ZZ-0050 and NC.NA-TS.ZZ-0050.

68. Provided access to control room references, determine the conditions that would necessitate changing Post Maintenance Testing or

5.2.4 (Continued)

G. All material nonconformances identified during the performance of Fuel Handling shall be documented in accordance with NAP-0.

H. (Hope Creek) All LPRM and Control Blade replacements shall be performed in accordance with procedures in the sequence prescribed by Reactor Engineering.

I. (Hope Creek) During irradiated fuel movement, the time spent with fuel in transit between the spent fuel pool and reactor pressure vessel shall be minimized to avoid the potential for high radiation doses in the upper regions of the drywell. **[CD-612X]**

J. (Salem) During irradiated fuel movement, the time spent with fuel in the transfer tube shall be minimized to avoid the potential for high radiation doses in the transfer tube area.

K. When sufficient manpower exists, Fuel Handling should be conducted using more than one Fuel Handling crew per operating shift. This allows the periodic rotation of Fuel Handling crew personnel and decreases the individual radiation dose received.

5.2.5 The suspension of Fuel Handling may be directed by the OS/CRS or Refueling SRO as appropriate. Prior to the resumption of Fuel Handling, applicable Technical Specification Surveillances shall be verified current. The following conditions require the suspension of Fuel Handling activities:

- Refueling floor radiological conditions that require the termination of refuel work activities.
- Any non-compliance with the applicable unit Technical Specifications governing the performance of Fuel Handling involving Core Alterations.
- Any neutron monitoring channel indicates unexpected increasing count rate.
- Any fuel damage occurs including the dropping, bumping, scraping or general mishandling of fuel or other suspended loads during handling.
- (Hope Creek) For any reactor core fuel loading, if any control rod or associated control rod drive mechanism has been removed from the reactor core or pressure vessel.

Du16

L.P. NO.: 0302-000.00H-000113-10
0303
0304
0305
PREVIOUS L.P. NO.: 0302-000.00H-000113-09

NUCLEAR TRAINING CENTER
LESSON PLAN

PROGRAM TITLE:	HOPE CREEK OPERATOR TRAINING
COURSE SECTION/MODULE:	OPERATING PROCEDURES
TOPIC/SUB MODULE:	ADMINISTRATIVE PROCEDURES
LESSON:	NUCLEAR COMMON, HCGS OPERATIONS, AND HCGS STATION ADMINISTRATIVE PROCEDURES
DURATION:	VARIES DEPENDING ON PROCEDURE(S)
PREREQUISITES:	
JTA NO. OR QUALIFICATION STATEMENT NO.:	

AUTHOR: Peter Doran DATE: 01/06/00

REVIEW/APPROVAL SIGNATURES

SUBMITTED BY: Peter Doran DATE: 01/06/00

BARGAINING UNIT REPRESENTATIVE: DATE:

PRINCIPAL TRAINING SUPERVISOR: DATE:

LINE SUPERVISOR: DATE:

COPY RECEIVED
WORD PROCESSING INITIALS:

INSTRUCTOR REFERENCES:

Number	Title
A. PROCEDURES	
NC.NA-AP.ZZ-0001	Nuclear Procedure System
NC.NA-AP.ZZ-0002	Nuclear Department Organization
NC.NA-AP.ZZ-0005	Station Operating Practices
NC.NA-AP.ZZ-0009	Work Control Process
NC.NA-AP.ZZ-0013	Control of Temporary Modifications
NC.NA-AP.ZZ-0015	Safety Tagging Program
NC.NA-AP.ZZ-0020	Control of Nonconforming Components and Structures
NC.NA-AP.ZZ-0023	Scaffolding Program
NC.NA-AP.ZZ-0024	Radiation Protection Program
NC.NA-AP.ZZ-0025	Operational Fire Protection
NC.NA-AP.ZZ-0037	Environmental Control
NC.NA-AP.ZZ-0038	Chemical Control Program
NC.NA-AP.ZZ-0044	Station Aids and Labels
NC.NA-AP.ZZ-0049	Conduct of Fuel Handling
NC.NA-AP.ZZ-0050	Station Testing Program
NC.NA-AP.ZZ-0059	10CFR50.59 Reviews and Safety Evaluations
NC.NA-AP.ZZ-0069	Work Control Coordination
NC.NA-AP.ZZ-0070	Inservice Testing and MOV Testing Programs
NC.NA-AP.ZZ-0071	Fuel Integrity Program
NC.NA-AP.ZZ-0083	Transient Loads
NC.NA-AP.ZZ-0084	Conduct of Infrequently Performed Tests or Evolutions
HC.SA-AP.ZZ-0002	Station Organization
HC.SA-AP.ZZ-0021	Station Cleanliness Program
HC.RA-AP.ZZ-0051	Leakage Reduction Program
SH.OP-AP.ZZ-0107	Shift and Relief Turnover
HC.OP-AP.ZZ-0005	Department Operating Practices
HC.OP-AP.ZZ-0012	Technical Specification Surveillances
HC.OP-AP.ZZ-0101	Post Reactor Scram/ECCS Actuation Review and Approval Requirements
HC.OP-AP.ZZ-0102	Use of Operations Department Procedures
HC.OP-AP.ZZ-0108	Operability Assessment and Equipment Control Program
HC.OP-AP.ZZ-0109	Equipment Operational Control
HC.OP-AP.ZZ-0110	Use and Development of Operating Logs
PSEG Nuclear Safety Manual (section 9.0)	Confined Space Entry
HC.OP-EO.ZZ-103	Reactor Building Control
SH.OP-DD.ZZ-0004	Operations Standards
HC.OP-DD.ZZ-0067	Personnel Qualification and Training

Number	Title
NC.WM-AP.ZZ-0000	Notification Process
NC.WM-AP.ZZ-0001	Work Management Process
NC.WM-AP.ZZ-0003	Regular Maintenance Process
NC.NA-TS.ZZ-0050	Maintenance Testing Matrix

B. DOCUMENTS

DOCUMENT	DESCRIPTION
SER 29-82 (PTS-306) (CD-515A)	INPO Significant Event Report 29-82; Subject: Trip of Both Reactors at Site following Condensate Pump Trip to be discussed with NC.NA-AP.ZZ-0013
AR-M20-89-0109	Potential/Impact of Unauthorized Mods; to be discussed with NC.NA-AP.ZZ-0013
DCP 4HC-0214	Deletes requirements to use post trip log for review and disables automatic log printout
SOER 85-5 (PTS-1437) (CD-265E)	INPO Significant Operating Experience Report: Internal Flooding of Power Plant Buildings
INPO IS 912	SOER 85-5 Recommendation Clarification
IE Notice 85-27	Notification to the NRC Operations Center and Report Events in Licensee Event Reports
NLR-I91426	ESF System Actuation Reportability
NLR-I91148	Engineered Safety Features System Actuations
ASME	Section XI, Division 1, Articles IWP and IWV
CD-358F	Inservice testing of DG fuel oil transfer pumps
10CFR50.54(x) and (y)	Code of Federal Regulations
10CFR50.59	Code of Federal Regulations
IN 87-21	Shutdown order issued because Licensed Operators asleep while on duty
IN 88-20	Unauthorized individual manipulating controls and performing control room activities
IN 92-30	Falsification of Records
GEK-90300	Reactor Assembly and Servicing Equipment, Vol. II Part 2
FSAR	Auxiliary Systems, Vol. 13, Chapter 9.1
T/S	HCGS Technical Specifications

TRAINING MATERIAL REQUIRED:

1. Lesson Plan
2. NC.NA-AP.ZZ-0001, Nuclear Procedure System
3. NC.NA-AP.ZZ-0002, Nuclear Department Organization
4. NC.NA-AP.ZZ-0005, Station Operating Practices
5. NC.NA-AP.ZZ-0005, Station Operating Practices
6. NC.NA-AP.ZZ-0013, Control of Temporary Modifications
7. NC.NA-AP.ZZ-0015, Safety Tagging Program
8. NC.NA-AP.ZZ-0020, Control of Nonconforming Components and Structures
9. NC.NA-AP.ZZ-0023, Scaffolding Program
10. NC.NA-AP.ZZ-0024, Radiation Protection Program
11. NC.NA-AP.ZZ-0025, Operational Fire Protection Program
12. NC.NA-AP.ZZ-0037, Environmental Control
13. NC.NA-AP.ZZ-0038, Chemical Control Program
14. NC.NA-AP.ZZ-0044, Station Aids and Labels
15. NC.NA-AP.ZZ-0049, Conduct of Fuel Handling
16. NC.NA-AP.ZZ-0050, Station Testing Program
17. NC.NA-AP.ZZ-0059, 10CFR50.59 Reviews and Safety Evaluation
18. NC.NA-AP.ZZ-0069, Work Control Coordination
19. NC.NA-AP.ZZ-0070, Inservice Testing and MOV Testing Programs
20. NC.NA-AP.ZZ-0071, Fuel Integrity Program
21. NC.NA-AP.ZZ-0083, Transient Loads
22. HC.SA-AP.ZZ-0002, Station Organization
23. HC.SA-AP.ZZ-0021, Station Cleanliness Program
24. SH.OP-AP.ZZ-0107, Shift and Relief Turnover
25. HC.OP-AP.ZZ-0005, Department Operating Practices
26. HC.OP-AP.ZZ-0012 Technical Specification Surveillances
27. HC.OP-AP.ZZ-0101, Post Reactor Scram/ECCS Actuation Review and Approval Requirements
28. HC.OP-AP.ZZ-0102, Use of Operations Department Procedures
29. HC.OP-AP.ZZ-0108, Operability Assessment and Equipment Control Program
30. HC.OP-AP.ZZ-0109, Equipment Operational Control
31. HC.RA-AP.ZZ-0051, Leakage Reduction Program
32. SH.OP-DD.ZZ-0004, Operations Standards
33. HC.OP-DD.ZZ-0067, Personnel Qualification and Training
34. PSEG Nuclear Safety Manual

STUDENT HANDOUTS:

1. Lesson Plan / Terminal and Enabling Learning Objectives
   LATEST REVISION OF THE FOLLOWING PROCEDURES
2. NC.NA-AP.ZZ-0001, Nuclear Procedure System
3. NC.NA-AP.ZZ-0002, Nuclear Department Organization
4. NC.NA-AP.ZZ-0005, Station Operating Practices
5. NC.NA-AP.ZZ-0005, Station Operating Practices
6. NC.NA-AP.ZZ-0013, Control of Temporary Modifications
7. NC.NA-AP.ZZ-0015, Safety Tagging Program
8. NC.NA-AP.ZZ-0020, Control of Nonconforming Components and Structures
9. NC.NA-AP.ZZ-0023, Scaffolding Program
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16. NC.NA-AP.ZZ-0050, Station Testing Program
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19. NC.NA-AP.ZZ-0070, Inservice Testing and MOV Testing Programs
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22. HC.SA-AP.ZZ-0002, Station Organization
23. HC.SA-AP.ZZ-0021, Station Cleanliness Program
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26. HC.OP-AP.ZZ-0012 Technical Specification Surveillances
27. HC.OP-AP.ZZ-0101, Post Reactor Scram/ECCS Actuation Review and Approval Requirements
28. HC.OP-AP.ZZ-0102, Use of Operations Department Procedures
29. HC.OP-AP.ZZ-0108, Operability Assessment and Equipment Control Program
30. HC.OP-AP.ZZ-0109, Equipment Operational Control
31. HC.RA-AP.ZZ-0051, Leakage Reduction Program
32. SH.OP-DD.ZZ-0004, Operations Standards
33. HC.OP-DD.ZZ-0067, Personnel Qualification and Training
34. PSEG Nuclear Safety Manual

CLASSROOM REQUIREMENTS:

No special considerations for the classroom size or arrangements are required.

INSTRUCTIONAL OBJECTIVES:

1.0 Provided access to control room references and/or a scenario of plant/system status or previous plant conditions associated with the operation of Hope Creek Generating Station, the trainee will interpret the applicable Nuclear Common Administrative Procedure to ensure plant operations are being conducted, and are in compliance with, the requirements of the following procedures:

A. NC NA-AP ZZ-0001, Nuclear Procedure System
B. NC NA-APZZ-0002, Nuclear Department Organization
C. NC.NA-AP-ZZ-0005, Station Operating Practices
D. NC. NA-AP-ZZ-0009, Work Control Process
E. NC. NA-AP-ZZ-0013, Control of Temporary Modifications
F. NC.NA-AP-ZZ-0015, Safety Tagging Program
G. NC.NA-AP-ZZ-0020, Control of Nonconforming Components and Structures
H. NC.NA-AP-ZZ-0023, Scaffolding Program
I. NC.NA-AP-ZZ-0024, Radiation Protection Program
J. NC.NA-AP-ZZ-0025, Operational Fire Protection Program
K. NC.NA-AP-ZZ-0037, Environmental Control
L. NC.NA-AP-ZZ-0038, Chemical Control Program
M. NC.NA-AP-ZZ-0044, Station Aids and Labels
N. NC.NA-AP-ZZ-0049, Conduct of Fuel Handling
O. NC.NA-AP-ZZ-0050, Station Testing Program
P. NC.NA-AP-ZZ-0059, 10CFR50.59 Reviews and Safety Evaluation
Q. NC.NA-AP-ZZ-0069, Work Control Coordination
R. NC.NA-AP-ZZ-0070, Inservice Testing and MOV Testing Programs
S. NC.NA-AP-ZZ-0071, Fuel Integrity Program
T. NC.NA-AP-ZZ-0083, Transient Loads
U. NC.NA-AP.ZZ-0084, Conduct of Infrequently Performed Tests
V. NC.WM-AP.ZZ-0000, Notification Process
W. NC.WM-AP.ZZ-0001, Work Management Process
X. NC.WM-AP.ZZ-0003, Regular Maintenance Process
Y. NC.NA-TS.ZZ-0050, Maintenance Testing Matrix

2.0 Provided access to control room references and/or a scenario of plant/system status or previous plant conditions associated with the operation of Hope Creek Generating Station, the trainee will interpret the applicable HCGS Operations Administrative Procedure to ensure plant operations are being conducted, and are in compliance with, the requirements of the following procedures:

A. SH.OP-DD.ZZ-0004, Operations Standards

B. HC.OP-AP.ZZ-0012 Technical Specification Surveillances

C. HC.OP-DD.ZZ-0067, Personnel Qualification and Training

D. HC.OP-AP.ZZ-0005, Department Operating Practices

E. SH.OP-AP.ZZ-0107, Shift Turnover Responsibilities

3.0 Provided access to control room references and/or a scenario of plant/system status or previous plant conditions associated with the operation of Hope Creek Generating Station the trainee will interpret the applicable HCGS Station Administrative Procedure to ensure plant operations are being conducted and are in compliance with the requirements of the following procedures:

A. HC.RA-AP.ZZ-0051,Leakage Reduction Program

4.0 Provided access to control room references and/or a scenario of plant/system status or previous plant conditions associated with the operation of Hope Creek Generating Station, the trainee will interpret **PSEG Nuclear Safety Manual , section 9.0** for Confined Space Entry, to ensure plant operations are being conducted, and are in compliance with, the requirements of the procedure.

ENABLING LEARNING OBJECTIVES:

1. State who may delegate authority to an individual, when that function is not delineated in the individual's position description, IAW NC.NA-AP.ZZ-0002.
2. Describe the process of telephone approval and documents requiring signature approval IAW NC.NA-AP.ZZ-0002.
3. Explain, from memory, whom and when the OS may call-out and the approvals required to be obtained prior to making such call-outs, IAW NC.NA-AP.ZZ-0002
4. Determine the requirements for a licensed operator to be present "at the controls" at all times during operation of the facility, IAW NC.NA-AP.ZZ-0005, and HC.OP-AP.ZZ-0005.
5. Given a copy of the control room layout, identify the area denoted "at the controls," IAW NC.NA-AP.ZZ-0005.
6. Determine who is permitted (including conditions) to manipulate controls which directly or indirectly affect reactivity or power level, IAW NC.NA-AP.ZZ-0005, and HC.OP-AP.ZZ-0005.
7. Given plant conditions and/or access to control room references, determine the following IAW NC.NA-AP.ZZ-0005, and HC.OP-AP.ZZ-0005:
    a. The level of licensing required for the OS, CRS, and RO/PO.
    b. Minimum shift manning requirements for all plant conditions.
    c. Normal shift staffing levels.
    d. When a person can serve a dual role as CRS/STA or OS/STA
8. From memory, choose the correct operator response to instrument indications, IAW NC.NA-AP.ZZ-0005, and HC.OP-AP.ZZ-0005.
9. Provided access to control room references, apply the overtime guidelines IAW NC.NA-AP.ZZ-0005.
10. Provided access to control room references, determine the requirements for maintaining an operator license active, IAW HC.OP-DD.ZZ-0067.
11. Explain, from memory, the circumstances and approval required for Licensed Operators to deviate from Technical Specifications or license conditions, IAW NC.NA-AP.ZZ-0005 and 10CF50.54(x).

12. Describe the control room restrictions pertaining to the following IAW NC.NA-AP.ZZ-0005:
    a. Control room access during normal and transient operation.
    b. The conduct of plant related technical or administrative business or personal business.
    c. Access for non-shift personnel
13. State the conditions which require Operations Manager notification, IAW NC.NA-AP.ZZ-0005 and SH.OP-AP.ZZ-0004
14. Determine the requirements for Independent Verification IAW NC.NA-AP.ZZ-0005.
15. Describe how to perform an Independent Verification IAW NC.NA-AP.ZZ-0005
16. Provided access to control room references, determine the Reactor Shutdown considerations IAW SH.OP-DD.ZZ-0004.
17. Given a set of conditions, determine when a Motor Operated Valve must be declared Inoperable due to manual operation, IAW NC.NA-AP.ZZ-0005.
18. Given a set of conditions, determine when a manually operated valve shall be locked, and the correct method to perform this function, IAW NC.NA-AP.ZZ-0005
19. State the purpose of the Notification Process, IAW NC.WM-AP.ZZ-0000,
20. Describe the responsibilities of All NBU Personnel, IAW NC.WM-AP.ZZ-0000.
21. Given access to control room references, describe how to validate a corrective maintenance action request, IAW NC.WM-AP.ZZ-0003.
22. IAW NC.WM-AP.ZZ-0000 describe the actions for OS/CRS review of a Notification to determine the following: **(SRO ONLY)**
    a. If operability is required
    b. Determine Operability
    c. Reportability requirements
23. State the responsibility of the OS/CRS for the work control process IAW NC.NA-AP.ZZ-0009.
24. State the two purposes of an EMIS tag, IAW NC.WM-AP.ZZ-0000

25. Determine the proper method of completing and hanging an EMIS tag, IAW NC.WM-AP.ZZ-0000.

26. Provided access to control room references, determine when it is necessary to obtain OS/CRS permission prior to starting work, IAW NC.WM-AP.ZZ-0001.

27. Determine the proper method of initiating a Work Package for Unscheduled LCO entry due to TS Equipment Declared INOPERABLE IAW NC.WM-AP.ZZ-0001.

28. Determine the responsibilities of the Job Supervisor, IAW NC.NA-AP.ZZ-0009.

29. Given plant conditions, evaluate plant status to determine if work may be conducted as scheduled, IAW NC.WM-AP.ZZ-0001 for the following:. (SRO ONLY)
    a. Minor Maintenance
    b. Work Package

30. Determine the proper method of closing and completing a Work Package, IAW NC.WM-AP.ZZ-0001.

31. Given plant problems/industry events associated with station maintenance:
    a. Discuss the root cause of the plant problem/industry event, IAW SOER 85-5.
    b. Discuss HCGS procedural guidelines that mitigate/reduce the likelihood of the problem/industry event at HCGS, IAW:
        - OP-EO.ZZ-103, Reactor Building Control EOP
        - NC.NA-AP.ZZ-0015, Safety Tagging Program.
        - NC.WM-AP.ZZ-0001, Work Management Process
        - NC.NA-AP.ZZ-0050, Station Testing Program
    c. Discuss the "lessons learned" from the problem/event, IAW INPO SOER 85-5.

32. Explain the appropriate actions to be taken prior to authorizing performance of a Surveillance and/or Inservice Test, IAW NC.WM-AP.ZZ-0003 and HC.OP-AP.ZZ-0012. **(SRO ONLY)**

33. Explain the appropriate action to be taken when Surveillance and/or Inservice test results are determined to be:
    a. acceptable
    b. unacceptable

    IAW NC.WM-AP.ZZ-0003 and HC.OP-AP.ZZ-0012

34. State the appropriate actions to be taken when Temporary Modification Tags are missing, IAW NC.NA-AP.ZZ-0013.

35. State the meaning of the following terms, IAW NC.NA-AP.ZZ-0013:
    a. Temporary Modification
    b. Lifted Lead
    c. Electrical Jumper
    d. Independent Verification

36. Given the procedure for Control of Temporary Modifications and a specific group of proposed actions, determine whether or not a Temporary Modification is required, IAW NC.NA-AP.ZZ-0013. **(SRO ONLY)**

37. Provided with a Troubleshooting Plan for which a Temporary Modification Package (TMP) is not required and access to control room references, determine whether the Troubleshooting Plan should be approved, IAW NC.NA-AP.ZZ-0013 and SH.OP-AP.ZZ-0008. **(SRO ONLY)**

38. Provided with a Temporary Modification Package (TMP) which affects floor drain plug status and access to control room references, determine whether the Temporary Modification Package should be approved, IAW NC.NA-AP.ZZ-0013. **(SRO ONLY)**

39. Identify the responsibilities of the OS/CRS for Control of Temporary Modifications for the following, IAW NC.NA-AP.ZZ-0013: **(SRO ONLY)**
    a. Installation of a T-Mod
    b. Reviewing installed T-mods
    c. Removing T-Mods
    d. Preparing a T-Mod for an alarm bypass
    e. Expedited Temp Mod packages

40. Explain the responsibilities of the "User" of Measuring and Test Equipment, IAW NC.NA-AP.ZZ-0022.

41. Describe what the worker is acknowledging when signing a RWP prior to use, IAW NC.NA-AP.ZZ-0024, Radiation Protection Program.

42. State the definition of the following terms, IAW NC.NA-AP.ZZ-0024:
    a. Contaminated Area
    b. High Radiation Area
    c. Locked High Radiation Area
    d. Radiation Area
    e. Restricted Area

f. Very High Radiation Area
g. Airborne Radioactivity Area
h. Declared Pregnant Woman (DPW)
i. Total Effective Dose Equivalent (TEDE)

43. State the responsibilities of the following personnel for issuance of keys to locked High Radiation Areas, IAW NC.NA-AP.ZZ-0024:
    a. Key Holder
    b. OS
    c. Radiation Protection Supervisor

44. Given a set of exposure conditions, identify the personnel responsible for approval of the following dose extension, IAW NC.NA-AP.ZZ-0024:
    a. Yearly Dose Extension
    b. Declared Pregnant Women Dose Extension
    c. Lifetime Dose Extension

45. State the actions of an individual discovering a fire, IAW NC.NA-AP.ZZ-0025.

46. Explain the controls utilized over the following, IAW NC.NA-AP.ZZ-0025:
    a. Combustible Material
    b. Flammable Liquids and Gases
    c. Ignition Sources

47. Explain the controls utilized over impairments to fire protection systems, IAW NC.NA-AP.ZZ-0025.

48. Determine where ignitable metals are used at HCGS, IAW NC.NA-AP.ZZ-0025.

49. Explain the use of the Transient Combustible Load Allowances Table, IAW NC.NA-AP.ZZ-0025.

50. Explain the responsibilities of the Job Supervisor for entry into cleanliness Zone II, IAW NC.NA-AP.ZZ-0031. **(SRO ONLY)**

51. State the requirements of Zone II cleanliness controls based on work scope as delineated in NC.NA-AP.ZZ-0031.

52. State when a Pre-startup Walkdown is required as delineated in NC.NA-AP.ZZ-0031.

53. Determine the Group that the OS will initially notify to respond to a major spill, IAW NC.NA-AP.ZZ-0037. **(SRO ONLY)**

54. Provided access to control room references, determine the meaning of

PSEG NUCLEAR L.L.C.

HOPE CREEK GENERATING STATION

HC.OP-IO.ZZ-0009(Q) - Rev. 24

REFUELING OPERATIONS

USE CATEGORY: I

A. Biennial Review performed Yes ___ No ___ N/A ✓

B. Change Package(s) and Affected Document Number(s) incorporated into this revision.

- CP No. ________ CP Rev. No. ___ AD No. ________ AD Rev. No. ___ or None ✓

C. OTSC(s) incorporated into this revision:

- OTSC No(s) ________________ or None ✓

REVISION SUMMARY

1. Based on request made under **Order 80033269 (T/S Amendment 137)** all references to Refueling Operations Technical Specifications 3/4.9.4 Decay Time, 3/4.9.5, Communications, and 3/4.9.6, Refueling Platform, were replaced with new Updated Final Safety Analysis Report references. As part of T/S Amendment 137, the above mentioned T/Ss and their associated bases are being moved to section 9.1.4.2.12 of the UFSAR. These changes are reflected in procedure steps 1.0, 2.1.1, 3.3, 3.4, Note 5.1, steps 5.1.10, 5.1.12, 5.1.13, Caution 5.3.1.E, steps 5.3.1.E.8, 5.3.1.E.10, and associated signoffs and references. These changes can be considered editorial based on allowances in NC.DM-AP.ZZ-0001(Q) for incorporating a change that has already been reviewed and approved in accordance with another approved process.
2. Based on Writers Review, the following editorial changes were made:
   - Converted Note 5.1.4, item A, into step 5.1.4 due to the action contained therein to record time and date of shorting link removal.
   - Updated reference to SH.OP-AP.ZZ-0108(Q), in step 5.1.14.B.
   - Added RECORDS section 6.0.
   - Removed reference to HC.OP-MD.KE-0001(Q), Refueling Platform 7-Day Operational Checks, at step 5.1.12.A.1. These checks are now captured within HC.OP-ST.KE-0001, Refuel Interlock Operability Functional Test, which is verified current under 5.1.12.B.
   - Reformatted signoff under step 5.1.12.A.2.

IMPLEMENTATION REQUIREMENTS Effective date 3/18/02

Implementation of T/S Amendment 137.

APPROVED: ____________________ 031502

Manager - Hope Creek Operations **Date**

REFUELING OPERATIONS

TABLE OF CONTENTS

REFUELING OPERATIONS

START TIME ________________ DATE ________________ BY ________________

TERMINATION TIME ________________ DATE ________________ BY ________________

COMPLETION TIME ________________ DATE ________________ BY ________________

1.0 PURPOSE

This procedure provides a mechanism for ensuring that the TECH SPEC / UFSAR requirements are satisfied prior to conducting CORE ALTERATIONS. In addition this procedure provides a means of identifying and coordinating in-vessel activities with other plant activities while the plant is in OPERATIONAL CONDITION 5 - Refueling. **[CD-443X]**

2.0 PREREQUISITES

2.1 Plant Initial Conditions

2.1.1 The plant is in OPERATIONAL CONDITION - 5.
All TECH SPEC / UFSAR requirements for this operating condition have been satisfied AND are being maintained. ______

2.1.2 The Reactor Vessel preparations for refueling have been completed IAW HC.OP-IO.ZZ-0005(Q), Cold Shutdown to Refueling, AND indicated on Attachment 1. ______

2.1.3 The Reactor Cavity (AND Dryer/Separator Pool if applicable) have been flooded to a level equal to the normal level of the Spent Fuel Pool.
The Fuel Pool Gates have been removed. ______

2.1.4 One of the following is in-service, if required, to remove decay heat AND provide Reactor Core circulation: ______

- Residual Heat Removal System in Shutdown Cooling IAW HC.OP-SO.BC-0002(Q), Decay Heat Removal Operation. ______
  OR
- Alternate Decay Heat Removal operation IAW Attachment 5 ______
  OR
- Alternate Fuel Pool Cooling Assist Mode of RHR operation IAW HC.OP-SO.BC-0002(Q). (For Full Core Offload and the transition to and from a Full Core Offload. This mode of operation should be available prior to Offloading ≈ 1/3 of the core). ______

2.2 Zone II housekeeping cleanliness requirements have been established for the Spent Fuel Pool, Reactor Cavity, AND Dryer/Separator Pools IAW NC.NA-AP.ZZ-0031(Q), Artificial Island Inspection/Housekeeping Program. ______

2.3 A list of CORE ALTERATIONS AND in-vessel tests, inspections, and maintenance activities has been obtained that are unique to the current refueling outage from the Outage Planning Department. ______

3.0 PRECAUTIONS AND LIMITATIONS

3.1 In the event that plant conditions require a delay during some part of this procedure, the Operations Superintendent/Control Room Supervisor (OS/CRS) shall retain this procedure UNTIL it is continued OR terminated. ______

3.2 IF this procedure is terminated PRIOR to completion, THEN the OS/CRS shall note the reason, time, AND date of termination on this procedure.. ______

3.3 The TECH SPEC / UFSAR requirements described in Section 5.1 shall be satisfied PRIOR to the start of any activity resulting in a CORE ALTERATION. ______ |

3.4 IF the CORE ALTERATION requirements of TECH SPEC / UFSAR CAN NOT be maintained, THEN the following are required: ______ |

3.4.1 The OS/CRS shall direct the Refuel Floor Supervisor (RFS) to suspend those activities resulting in a CORE ALTERATION. ______

3.4.2 The RFS shall direct personnel performing CORE ALTERATIONS to place hoisted fuel OR core components in a stable configuration AND suspend subsequent CORE ALTERATIONS. ______

3.5 Section 5.2 of this procedure describes tests, inspections, AND CORE ALTERATIONS. The procedure steps described in this section are NOT required to be performed in order. The exact sequence of test, inspections, AND CORE ALTERATIONS is determined by the schedule of events prepared by the Outage Planning Department. ______

3.6 To prevent affecting the Reactor Shutdown Margin DO NOT allow Reactor temperatures to go below 70°F(when fuel is in the vessel). Cooldown below 68°F could result in an invalidation of Shutdown Margin calculations which are based in part on the Reactor being in the shutdown condition; cold i.e. 68°F AND Xenon free. ______

3.7 CORE ALTERATIONS AND in-vessel activities have the potential for affecting the Reactor shutdown margin, exposing personnel to high levels of radiation, contamination, AND other safety hazards. The OS/CRS AND Control Room personnel shall: ______

3.7.1 Be aware of all in-progress tests, inspections, AND CORE ALTERATIONS. ______

3.7.2 Direct, control,
AND coordinate the alignment
AND operation of plant systems with the activities in-progress on the refuel floor. ______

3.8 A designated "spotter" is required for all bridge activities which require any grapple to be loaded. ______

3.9 NO individual should perform bridge activities for greater than six consecutive hours. ______

3.10 Testing of IST Valves need NOT begin in 48 hours after Cold Shutdown when entering a Refueling Outage, provided all valves required to be tested during Cold Shutdown/Refueling will be tested before Plant Startup. ______

- All Cold Shutdown, Refueling Outage,
  AND Containment Deinerted valves shall be tested before startup from Refueling Outages, unless testing has been completed within the previous 92 days. ______

- IF an outage lasts beyond 92 days,
  THEN all Cold Shutdown testing shall be completed
  AND all Cold Shutdown testing shall continue such that all applicable components have been tested within the last 92 days of the shutdown. ______

4.0 EQUIPMENT REQUIRED

None

5.3 **Administrative Controls for Suspension and Resumption of CORE ALTERATIONS**

NOTE 5.3.1

Suspension of CORE ALTERATIONS shall be a formal declaration by the Refuel Floor SRO, OR a suspension of CORE ALTERATIONS as required by Technical Specifications.
Section 5.2.5 of NC.NA-AP.ZZ-0049(Q), Conduct of Fuel Handling, discusses the suspension of fuel handling.

5.3.1 IF CORE ALTERATIONS were suspended,
THEN **PERFORM** the following
PRIOR to resumption of CORE ALTERATIONS: ______

A. **RECORD** the time, date,
AND reason for suspending CORE ALTERATIONS in Remarks section of Attachment 1. ______

NOTE 5.3.1.B

Surveillance requirements shall be re-done as required prior to resuming CORE ALTERATIONS.

B. **VERIFY** the surveillance requirement completion on Refueling Daily Log of HC.OP-DL.ZZ-0026(Q), Surveillance Log, are current. ______

C. IF CORE ALTERATIONS are suspended due to required maintenance on Refuel Position Interlocks,
THEN **PERFORM** the applicable Section(s) of HC.OP-ST.KE-0001(Q), Refuel Interlock Operability Test, as a retest, to ensure interlock operability
PRIOR to resuming CORE ALTERATIONS
AND **RECORD** the time, date,
AND **ENTER** initials on Attachment 4 to indicate when this surveillance requirement is satisfied. **[T/S 4.9.1.3]** ______

D. WHEN ready for resumption of CORE ALTERATIONS,
THEN **REQUEST** OS/CRS to complete items 1 & 2 of Attachment 4. ______

Continued Next Page

5.3.1 (Continued)

CAUTION 5.3.1.E

A. If the original TECH SPEC / UFSAR surveillance requirements for starting CORE ALTERATIONS are maintained current during the time when actual CORE ALTERATIONS are NOT taking place, then the requirement to perform the surveillance PRIOR to CORE ALTERATIONS is redundant and is NOT necessary for resumption of CORE ALTERATIONS.

B. If the TECH SPEC / UFSAR surveillance requirements have NOT been maintained during the period of time when CORE ALTERATIONS are NOT actually occurring, then the "Prior to CORE ALTERATIONS" surveillance requirement is necessary to comply with the associated TECH SPEC / UFSAR requirement.

E. **PERFORM** the following as applicable
AND **RECORD** time, date,
AND **ENTER** initials on Attachment 4 as applicable to indicate when surveillance requirement is satisfied: **[T/S 4.9.1.1.a.2]** ______

1. IF required **(if applicable within last 7 days before resumption of CORE ALTERATIONS)**, THEN **DEMONSTRATE** the Source Range Monitors are OPERABLE by requesting I&C to complete the following as required: **[T/S 4.9.2.b.2]** ______

- HC.IC-FT.SE-0001(Q), NUC Instrument System Source Range Monitor ______
- HC.IC-FT.SE-0002(Q), NUC Instrument System Source Range Monitor ______
- HC.IC-FT.SE-0003(Q), NUC Instrument System Source Range Monitor ______
- HC.IC-FT.SE-0004(Q), NUC Instrument System Source Range Monitor ______
- HC.IC-FT.SE-0025(Q), NIS SRM Fuel Loading Non Coincident Trips ______

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5.3.1.E (Continued)

2. **ENSURE** that at least 22 feet 2 inches of water is being maintained over the Reactor Vessel flange **(within the last 24 hours) [T/S 4.9.8]** ______

3. **VERIFY** that the Reactor Mode Switch is locked (key removed) in the SHUTDOWN OR REFUEL position (within the last 12 hours, unless Mode Switch was unlocked, then within the last 2 hours). **[T/S 4.9.1.1.a.2, 4.9.1.1.b]** ______

4. **VERIFY** Reactor Protection System shorting links are removed OR adequate Shutdown Margin has been demonstrated IAW T/S 3.1.1 **(within the last 8 hours) [T/S 4.9.2.d]** ______

5. **VERIFY** the following PRIOR to the start of removal of a single Control Rod, OR the associated Control Rod Drive Mechanism from the Core OR Reactor Pressure Vessel **(within the last 4 hours)**: **[T/S 4.9.10.1]** ______

   a. The Reactor Mode Switch is OPERABLE IAW Attachment 2 of HC.OP-DL.ZZ-0026(Q), Surveillance Log. **[CD-404E]** ______

   b. The SRM channels are OPERABLE IAW Attachment 2 of HC.OP-DL.ZZ-0026(Q), Surveillance Log. **[CD-404E]** ______

   c. The SHUTDOWN MARGIN requirements IAW Attachment 2 of HC.OP-DL.ZZ-0026(Q), Surveillance Log. **[CD-404E]** ______

Continued Next Page

5.3.1.E.5 (Continued)

d. All other control rods in a five-by-five array centered on the Control Rod being removed are inserted
AND electrically
OR hydraulically disarmed
OR the four Fuel Assemblies surrounding the Control Rod
OR Control Rod Drive Mechanism to be removed from the Core
AND/OR Reactor Vessel
are removed from the core cell. ______

e. All other control rods are fully inserted. ______

f. All fuel loading operations are suspended. ______

g Restricted Core Alteration Forms provided by procedure HC.RE-FR.ZZ-0001(Q) shall be available to enhance administrative control during Control Rod withdrawal during refueling. ______

h. The Reactor Mode Switch is Locked (key removed) in the SHUTDOWN
OR REFUEL position. ______

6. **VERIFY** the following prior to the start of removal of any number of Control Rods,
OR the associated Control Rod Drive Mechanisms from the Core
OR Reactor Pressure Vessel
(within the last 4 hours): [T/S 4.9.10.2.1] ______

a. The Reactor Mode Switch is OPERABLE IAW Attachment 2 of HC.OP-DL.ZZ-0026(Q), Surveillance Log. **[CD-404E]** ______

b. The SRM channels are OPERABLE IAW Attachment 2 of HC.OP-DL.ZZ-0026(Q), Surveillance Log. **[CD-404E]** ______

c. The SHUTDOWN MARGIN requirements IAW Attachment 2 of HC.OP-DL.ZZ-0026(Q), Surveillance Log. **[CD-404E]** ______

Continued Next Page

5.3.1.E.6 (Continued)

d. All other control rods are either inserted <u>OR</u> have the surrounding four Fuel Assemblies removed from the core cell. **[T/S 4.9.10.2.1.d]** ______

e. The four Fuel Assemblies surrounding each Control Rod and/or Control Rod Drive Mechanism to be removed from the core and/or Reactor Vessel are removed from the core cell. **[T/S 4.9.10.2.1.e]** ______

f. All fuel loading operations are suspended. ______

g Restricted Core Alteration Forms provided by procedure HC.RE-FR.ZZ-0001(Q) shall be available to enhance administrative control during Control Rod withdrawal during refueling. ______

h. The Reactor Mode Switch is Locked (key removed) in the SHUTDOWN <u>OR</u> REFUEL position. ______

7. **VERIFY** SRM Channel Count Rate is at least 3 cps as demonstrated by HC.OP-ST.SE-0005(Q). **(within the last 4 hours) [T/S 4.9.2.c]** ______

8. **VERIFY** the Reactor has been subcritical for at least 24 hours as indicated by the date <u>AND</u> time when all Control Rods were fully inserted as recorded in Attachment 1. **(within the last 4 hours) [UFSAR 9.1.4.2.12.1]** ______ |

9. **VERIFY** all Control Rods are fully inserted, <u>OR</u> the withdrawal of one Control Rod under the control of the Reactor Mode Switch REFUEL position one-rod-out interlock may be withdrawn. **(within last 2 hours for starting, within last 12 hours for continuation) [T/S 4.9.3.a & b]** ______

Continued Next Page

APPROVED: [signature]
Operations Manager

3/16/98
Date

CATEGORY II

IRRADIATED FUEL DAMAGE

113

1.0 SYMPTOMS

1.1 Alarms

A. REFUELING FL AIRBORNE ACTIVITY HI

B. R B AIRBORNE ACTIVITY HI

C. RADIATION MONITORING ALARM/TRBL

D. NEW FUEL CRITICALITY RAD HI

2.0 AUTOMATIC ACTIONS

2.1 Reactor Building Ventilation System Isolation on the following:

Reactor Building Exh Hi Rad (1.0 X 10-3 uci/cc)
Refuel Floor Exh Hi Rad (2.0 X 10-3 uci/cc)

2.2 Filtration, Recirculation and Ventilation System (FRVS) automatic start.

3.0 IMMEDIATE OPERATOR ACTIONS

3.1 **SUSPEND** all refueling operations.

3.2 **ENSURE** all appropriate automatic actions are complete.

4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 **ENSURE** all appropriate immediate operator actions are complete. ______

4.2 **EVACUATE** all unnecessary personnel from the Reactor Building. ______

4.3 **ENSURE** that secondary containment is in effect. ______

4.4 **DIRECT** the Radiation Protection Department to take air samples
AND control access to the reactor building and refuel floor, if necessary. ______

4.5 **DETERMINE** the FRVS release rate
AND **ACTIVATE** the appropriate emergency plan. ______

5.0 DISCUSSION

5.1 A damaged fuel assembly attached to the fuel handling grapple should be set down in the fuel pool storage area
OR isolated in the defective fuel storage container if a high area radiation condition does not exist.

5.2 A failure of fuel cladding during refueling operations will release gaseous fission products to the reactor building. The severity of this accident will depend upon the exposure history of the fuel bundles.

5.3 Operation of the spent fuel pool cooling system, with irradiated fuel damage, can result in increased radiation levels in the spent fuel pool cooling piping.

#16

PSE&G NUCLEAR BUSINESS UNIT

NC.NA-AP.ZZ-0049(Q) - REV. 1

CONDUCT OF FUEL HANDLING

SPONSOR ORGANIZATION: Hope Creek Reactor Engineering

REVISION SUMMARY: Biennial Review performed Yes X No ____.
This is an editorial revision. Minor editorial, reference, cross-reference and organizational changes that reflect present processes are not shown with revision bars. Editorial changes shown with revision bars are:

1. Sections 3.13, 5.1.1, 5.1.4.B, 5.2.2.B.4 and 5.2.4.D, added the following "or electronic equivalent". This was added to clarify that an electronic equivalent, e.g., software program, may be used in lieu of the fuel tag board. (CR 951216070 CRCA 2)
2. Section 3.3, deleted former third bullet, unnecessary reference to Radiation Protection Technician responsibilities.
3. Deleted former Section 3.6.4, these duplicated those in Section 3.5.
4. Section 3.6, deleted responsibility specific to Technical Specifications adherence. This is not required to be stated, this is a requirement of the license.
5. Section 3.18, deleted responsibility for revising procedures. This does not have to be stated.
6. Section 8.0, added cross-references: NC.NA-AP.ZZ-0000(Q), Action Request Process; NC.NA-AP.ZZ-0003(Q), Document Management Program.
7. Section 8.0, deleted cross-references not applicable to the procedure: NC.NA-AP.ZZ-0006(Q), Incident Report/Reportable Event Program and Quality/Safety Concerns Reporting System; NC.NA-AP.ZZ-0011(Q), Records Management Program; NC.NA-AP.ZZ-0020(Q), Nonconformance Program and NC.NA-AP.ZZ-0032(Q), Preparation, Review and Approval of Procedures.

IMPLEMENTATION REQUIREMENTS: Effective Date: 9/18/97.

APPROVED: ____________________ 9/10/97
Manager - Hope Creek System Engineering — Date

APPROVED: ____________________ 9/14/97
General Manager - Hope Creek Operations — Date

APPROVED: ____________________ 9/17/97
General Manager - Salem Operations — Date

CONDUCT OF FUEL HANDLING

TABLE OF CONTENTS

1.0 PURPOSE

Establish the responsibilities and requirements necessary to ensure Fuel Handling is conducted in a safe and efficient manner. **[CD-644A]**

2.0 SCOPE

2.1 This procedure is applicable to Fuel Handling activities performed on the refuel floor and the Spent Fuel Handling Building (Salem) except as noted in Section 2.2.

2.2 New fuel receipt is outside the scope of this procedure. New fuel receipt is defined as the time fuel arrives at the plant security gate until its initial placement into the spent fuel pool.

3.0 RESPONSIBILITIES

3.1 **Operations Manager** is responsible for:

- Ensuring that Refueling SROs are qualified in accordance with NC.NA-AP.ZZ-0014(Q), Training, Qualification, and Certification (NAP-14). **[CD-168A]**
- Hope Creek, ensuring that Fuel Handling Operators are qualified in accordance with NAP-14. **[CD-168A]**
- Hope Creek, ensuring that approved procedures are available for Refueling Bridge Operation.
- Salem, ensuring that Fuel Handling Operators, not involving core alterations, are qualified in accordance with NAP-14.
- Ensuring approved procedures are available for coordination of refueling operations involving Core Alterations and that these procedures contain prerequisites, documentation of operation's surveillance and log readings for Core Alterations.

3.2 **General Manager - Nuclear Maintenance** is responsible for ensuring maintenance is performed on all equipment related to Fuel Handling activities.

3.3 **Superintendent - Radiation Protection** is responsible for:

- Hope Creek, controlling access to the upper elevations of the drywell during irradiated Fuel Handling in the reactor vessel. **[CD-612X]**
- Salem, controlling access to the transfer tube area during irradiated Fuel Handling in the reactor vessel.

3.4 **Manager - System Engineering** is responsible for:

- Ensuring that Reactor Engineers assigned to the Fuel Handling crews are qualified in accordance with NAP-14.
- At Salem, ensuring that Fuel Handling Operators, involved in core alterations, are qualified in accordance with NAP-14. **[CD-168A]**
- Ensuring procedures are available for:
  ⇒ Fuel Transfer Documents
  ⇒ Fuel Channeling and Dechanneling (Hope Creek)
  ⇒ Local Power Range Monitors (LPRM) removal and installation (Hope Creek)
  ⇒ Control Blade removal and installation (Hope Creek)
  ⇒ Manipulator Crane Operation (Salem)
  ⇒ Insert change outs (Salem)
  ⇒ Fuel Transfer System (Salem)

3.5 **Supervisor - Reactor Engineering** is responsible for:

- Preparing fuel transfer documents.
- Preparing necessary documentation for LPRM and Control Blade removal and installation. (Hope Creek)
- Monitoring all Fuel Handling and associated activities
- Verifying proper location and orientation of all fuel assemblies, as required.
- Maintaining accurate records of the location history for nuclear fuel and core components.
- Providing and maintaining the underwater cameras, video monitors, video recorders, cables and video control equipment used for Fuel Handling activities.
- Prior to the refueling outage, conducting "pre-refueling training and briefing" meetings with personnel involved with the Fuel Handling, Core Alterations, reactor pressure vessel work and in-vessel work. Meeting agenda should include: **[CD-827D]**
  ⇒ General outline of the activities to be performed.
  ⇒ Any unique activities to be performed.
  ⇒ General practices of the refuel floor.
  ⇒ Actions during an emergency situation.
  ⇒ Potential emergency situation and a review of past industry incidents.

3.6 **Refuel Senior Reactor Operator (SRO)** is responsible for:

- Supervising and controlling activities related to Fuel Handling involving Core Alterations. **[CD-217B, CD-897E, CD-168A]**
- Assisting the Fuel Handling Coordinator in resolving problems that halted Fuel Handling.
- Directing Refueling Bridge (Hope Creek) or Manipulator Crane (Salem) operations during Fuel Handling involving Core Alterations in the sequence defined by Reactor Engineering.
- Ensuring compliance with directions from Radiation Protection personnel to minimize the spread of radioactive contamination and reduce personnel radiation exposure. **[CD-897E]**
- Initiating immediate actions during abnormal events, such as irradiated fuel damage, loss of fuel pool inventory or refuel floor high radiation conditions.
- Suspending Fuel Handling, as he deems necessary or as required by procedure. (Refer to Section 5.2.5)
- Controlling access to the Refuel Bridge (Hope Creek) or Manipulator Crane (Salem) to ensure safety and minimize potential personnel exposure. **[CD-528A]**
- Determining whether activities should be suspended if problems or erratic functioning of refueling tools or cranes exist. **[CD-422A]**

3.7 **Refueling Bridge Operator** (Hope Creek), **Manipulator Crane Operator** (Salem), and **Fuel Handling Operator** (Salem)

- Performing nuclear fuel moves in accordance with the sequence for fuel transfer provided by Reactor Engineering.
- Informing the Refuel SRO of any problems or erratic functioning of the refueling tools or cranes. **[CD-422A]**

3.8 **Control Room Refuel Monitor** is responsible for:

- Communicating with the refuel floor personnel during Fuel Handling involving Core Alterations and related activities.
- Monitoring the Nuclear Instrumentation during fuel movement and informing the refuel floor personnel of any unexpected increasing count rate.
- Updating the fuel tag boards or electronic equivalent in the Control Room.

3.9 **Fuel Handling Coordinator** is responsible for:

- Ensuring that work performed on the refuel floor meets the requirements of NC.NA-AP.ZZ-0031(Q), Inspection/Housekeeping Program (NAP-31).
- Reviewing refuel floor schedule to ensure required tools, equipment, materials, etc. are in place to support upcoming activities.
- Attending daily outage meetings and report on refuel floor activities.
- Ensuring that Work Orders associated with Fuel Handling activities are performed in accordance with NC.NA-AP.ZZ-0009(Q), Work Control Process (NAP-9).
- When equipment or tools require maintenance during Fuel Handling activities, ensuring that the repair is being performed expeditiously and documented, as required.

3.10 **Radiation Protection Technician** is responsible for:

- Sampling the refueling floor and the 130" el. of the Fuel Handling building (Salem) air for the detection of airborne radioactive contamination.
- Directing personnel to prevent over-exposure in the event that radiological conditions change on the refuel floor or the Fuel Handling building.
- Advising Fuel Handling personnel of any changes in radiological conditions affecting the task in progress.
- Monitoring Fuel Handling floor activities to ensure that personnel overexposure, contamination or ingestion of radioactive materials does not occur.
- Performing radiation surveys of all tools and equipment to be removed from the reactor cavity, transfer pool (Salem) or spent fuel pool. **[CD-217B, CD-897E]**

3.11 **Spotter** is responsible for aiding the operator in the movement or placement of fuel in the reactor vessel or the spent fuel pool.

3.12 **Upender Operator** in containment is responsible for performing upending operations in containment in accordance with Reactor Engineering procedures. (Salem)

3.13 **Upender Operator** in the Fuel Handling building is responsible for (Salem):

- Performing upending operations in the Fuel Handling building in accordance with Reactor Engineering procedures.
- Aiding the Fuel Handling operator in the movement or placement of fuel in the spent fuel pool.

3.14 **Core Physics Monitor** is responsible for performing Inverse Count Rate Ratio plots during core reload and core shuffle. (Salem only)

3.15 **Gate Valve Operator** is responsible for closing the Transfer Canal Isolation Valve when directed by the Refuel SRO. (Salem only)

3.16 **Tool Control Monitor** is responsible for assuring personnel/material accountability in the reactor cavity and associated areas.

3.17 **Department Managers** are responsible for ensuring that the overtime guidelines specified in NC.NA-AP.ZZ-0005(Q), Station Operating Practices, are utilized for Fuel Handling Crew personnel, under their direction.

3.18 **Manager - Quality Assessment** is responsible for specifying QA hold or notification points in Fuel Handling procedures.

4.0 PROCESS DESCRIPTION

None.

5.0 PROCEDURE

5.1 Fuel Handling Not Involving Core Alterations

5.1.1 Prior to commencing Fuel Handling, not involving core alterations, Reactor Engineering shall:

- Initiate an Action Request in accordance with NC.NA-AP.ZZ-0000(Q), Action Request Process (NAP-0) for the required work activities.
- Ensure that the system operability and appropriate Technical Specification requirements for irradiated Fuel Handling have been satisfied.
- Ensure that appropriate documents are prepared depicting the transfer of fuel. A set of approved documents is required on the refuel floor (Hope Creek) or Fuel Handling building (Salem).
- Verify that the Control Room fuel tag boards or electronic equivalent reflect the location of nuclear fuel at the applicable unit.
- Obtain permission from the Operations Superintendent/Control Room Supervisor (OS/CRS) to commence Fuel Handling Activities.
- Notify the Radiation Protection Department prior to the commencement of Fuel Handling.
- Brief the Fuel Handling crew (non-core alterations).

5.1.2 Fuel Handling Crew for non-core alterations Fuel Handling activities.

A. The minimum crew for non-core alterations Fuel Handling activities is:

- Fuel Crane Operator
- Radiation Protection Technician
- Reactor Engineer
- Spotter

B. The Reactor Engineer may fulfill the duties of the spotter.

C. All members of this crew shall be on the refueling floor (Hope Creek) or in the Fuel Handling building (Salem) during Fuel Handling activities.

5.1.3 The transfer of nuclear fuel shall be performed as described below.

A. Fuel movements shall be performed in accordance with approved procedures.

B. The sequence of fuel movement shall be performed in accordance with the fuel transfer documents.

C. Following the completion of fuel moves, the Fuel Crane Operator and the spotter shall sign the completed fuel move on the fuel transfer documents.

D. Hope Creek, any required fuel dechanneling or channeling operations is performed in the fuel preparation machines by the Operations Department.

E. The removal of tools or equipment stored underwater in the spent fuel pool, cask storage pit (Hope Creek) or transfer pool (Salem) shall not to be allowed without the prior knowledge and consent of the Radiation Protection Technician. The Radiation Protection Technician shall be present for these activities. **[CD-217B, CD-897E]**

F. All material nonconformances identified during the performance of Fuel Handling shall be documented in accordance with NAP-0.

5.1.4 Upon completion of Fuel Handling activities, the Reactor Engineer shall:

A. Notify the OS/CRS and the Radiation Protection Department.

B. Ensure that the Control Room fuel tag boards or electronic equivalent are updated to reflect the location of nuclear fuel at the applicable unit.

5.2 **Fuel Handling Involving Core Alterations**

5.2.1 Refueling operations involving Core Alterations are coordinated through department procedures which provide the following:

A. Prerequisite signatures for the initiation of Core Alterations including system operability requirements. (Operations)

B. Documentation governing the resumption of Core Alterations. (Operations)

C. Refueling logs or other means necessary to document periodic Refueling surveillance items. (Departments which are responsible for Fuel Handling related Surveillances.)

5.2.2 Prior to commencing Fuel Handling, involving core alterations, the following shall be completed:

A. The Refueling SRO shall request permission from the OS/CRS and notify the Radiation Protection Department.

B. Reactor Engineering shall:

1. Prepare the appropriate documents depicting the transfer of fuel.

2. Prepare the necessary documentation to direct the performance of the following tasks:

- LPRM removal and installation (Hope Creek)
- Control blade removal and installation (Hope Creek)
- Fuel Channeling and De-channeling (Hope Creek)
- Insert change outs (Salem)

3. Ensure that copies of the documents for steps 1 and 2 above are available at the following locations:

- Control Room
- Refuel floor
- Fuel Handling Building (Salem)

5.2.2 (Continued)

4. Verify that the Control Room fuel tag boards or electronic equivalent are reflect the location of nuclear fuel on the refuel floor and the Fuel Handling building (Salem).

5. Conduct pre-refueling training and briefing.

C. The Fuel Handling Coordinator shall ensure that Work Orders are prepared for the required work activities.

D. The Refueling SRO shall direct Radiation Protection Technicians to establish restricted access to the upper regions of the drywell (Hope Creek).

5.2.3 Fuel Handling Crew for Fuel Handling activities involving core alterations.

A. The following is the minimum crew for Fuel Handling activities involving core alterations. Those designated with an asterisk (*), shall be on the refueling floor during Fuel Handling activities. **[CD-168A]**

- *Refueling SRO
- *Refueling Bridge Operator (Hope creek)
- *Manipulator Crane Operator (Salem)
- *Invessel Spotter
- *Radiation Protection Technician
- Reactor Engineer
- Upender Operator in containment (Salem only)
- Fuel Handling Building Crane Operator (Salem only)
- Radiation Protection technician in the Fuel Handling Building (Salem only)
- Upender Operator in Fuel Handling Building (Salem only)
- Control Room Refuel Monitor
- Core Physics Monitor (Salem only during core reload or shuffle))
- Fuel Handling Coordinator
- Gate Valve Operator
- Tool Control Monitor

B. The Refueling SRO or the Reactor Engineer may fulfill the duties of the Invessel Spotter. This shall be determined by the Refueling SRO.

C. The Invessel Spotter or the Reactor Engineer may fulfill the duties of the Fuel Handling Coordinator.

5.2.4 All Fuel Handling involving core alterations shall be performed as described below.

A. Communications shall be established and maintained between the Control Room Refuel Monitor located in the Control Room and a member of the Fuel Handling Crew on the refuel floor. **[HC & S T/S 3.9.5]**

- Salem, communications is also required with the Fuel Handling Building.
- Hope Creek, it may be necessary to maintain communications with the undervessel area.

B. All fuel movement shall be performed by approved Fuel Handling procedures.

C. The sequence of fuel movement shall be conducted in accordance with the approved fuel transfer documents.

D. Following the completion of individual fuel moves, the following shall be performed:

- The Refueling Bridge Operator (Hope Creek), Manipulator Crane Operator (Salem) and the Refueling SRO (or designee) complete all required signatures on the associated fuel transfer document. The spotter may sign for the operator.
- The Control Room Refuel Monitor located in the Control Room signs the associated Control Room document (independent verification of Control Room documentation is not applicable).
- (Salem) The spotter, in the Fuel Handling Building, signs the associated fuel transfer document.
- Control Room fuel tag boards or electronic equivalent are updated by the Control Room Refuel Monitor.

E. (Hope Creek) Any required fuel dechanneling or channeling operations are performed in the fuel preparation machines by the Operations department per appropriate Reactor Engineering procedures.

F. The removal of tools or equipment stored underwater in the reactor, spent fuel pool, cask storage pit or transfer canal shall not be performed without the prior knowledge and consent of the Radiation Protection Technician. The Radiation Protection Technician shall be present for these activities. **[CD-217B, CD-897E]**

5.2.6 If a fuel assembly is found in an incorrect core location, the following actions shall be taken:

A. If the error is found during a fuel move, return the latched fuel assembly to its original location.

B. Contact the Reactor Engineer to prepare modified fuel transfer documents to put the core in an analyzed configuration, with respect to shutdown margin. Then move the fuel assembly to the analyzed configuration.

C. Terminate Fuel Handling and Core Alterations.

D. Initiate an Action Request to evaluate the consequences of the Fuel loading error in accordance with NAP-0.

E. If a personnel error in the fuel movement occurred, the event shall be reviewed by the Operations Manager. This should involve a review of the refueling qualifications of the Refuel Platform Operator (Hope Creek), Manipulator Crane Operator (Salem) and Refueling SRO (and any Spotter and/or Designee) on the refuel platform at the time.

F. Recovery Fuel Transfer Documents are prepared and provided by the Reactor Engineer.

G. Fuel Handling shall not recommence without the approval of the Operations Manager.

5.2.7 Upon the completion of Fuel Handling involving Core Alterations, Reactor Engineering shall verify the reactor core fuel locations and orientations. (Hope Creek) This should be performed prior to any control rod motion (in that particular cell) once reactor fuel loading operations commence.

5.3 General Housekeeping Considerations

5.3.1 The Fuel Handling Coordinator is responsible for housekeeping and cleanliness on the refuel floor during and following the completion of Refueling activities until operational housekeeping is restored.

5.3.2 Provisions for personnel/material accountability established in accordance with NAP-31 are maintained on the refuel floor and Fuel Handling Building during the performance of Fuel Handling.

5.3.3 The temporary suspension of tools and equipment from the refueling platform or manipulator crane and temporary placement of such items within the reactor cavity should only be allowed if such items are required for work presently in process.

5.3.4 Additional groups involved in Fuel Handling such as ISI personnel or fuel vendor representatives, shall ensure that refuel floor housekeeping and cleanliness is maintained during the performance of work on the refuel floor.

5.3.5 Special care shall be maintained to avoid the contamination of completed Fuel Transfer Documents and documentation of other activities located on the refuel floor.

6.0 RECORDS

6.1 The master transfer documents shall be retained in accordance with NC.NA-AP.ZZ-0003(Q), Document Management Program.

6.2 Completed Work Orders shall be retained in accordance with NAP-9.

6.3 Qualification records shall be retained in accordance with NAP-14.

6.4 Personnel/Material Accountability Control Logs shall be retained in accordance with NAP-31.

7.0 DEFINITIONS

7.1 **Core Alterations** - This definition is specific to Salem or Hope Creek. See the applicable Technical Specification.

7.2 **Suspension of Fuel Handling** (or Core Alterations) - A formal declaration by the Refuel SRO or a condition required by Technical Specifications.

8.0 REFERENCES

8.1 UFSAR, Hope Creek Generating station, Chapter 9.
8.2 UFSAR, Salem Generating station, Chapter 9.
8.3 Hope Creek and Salem Technical Specifications.

8.4 Cross References

8.4.1 NC.NA-AP.ZZ-0000(Q), Action Request Process
8.4.2 NC.NA-AP.ZZ-0003(Q), Document Management Program
8.4.3 NC.NA-AP.ZZ-0005(Q), Station Operating Practices
8.4.4 NC.NA-AP.ZZ-0009(Q), Work Control Process
8.4.5 NC.NA-AP.ZZ-0014(Q), Training, Qualification and Certification
8.4.6 NC.NA-AP.ZZ-0031(Q), Inspection/Housekeeping Program

8.5 Closing Documents

CD-168A (NRC Circular 80-21)	CD-217B (INPO O&MR 111)
CD-442A (INPO SER 59-81)	CD-827D (INPO SOER 84-01R06)
CD-528A (INPO SER 43-82)	CD-897E (INPO SER 12-87)
CD-644A (INPO O&MR 65)	CD-612X (Hope Creek UFSAR F01-0091-01)

An operator has the following exposure history this year until today:

Deep Dose Equivalent (DDE)	210 mrem
Committed Effective Dose Equivalent (CEDE)	45 mrem
Shallow Dose Equivalent (SDE)	33 mrem

Today, the operator was required to make two entries into the Drywell at 5 percent reactor power:

Entry 1: Gamma dose: 52 mrem; Neutron dose: 24 mrem
Entry 2: Gamma dose: 124 mrem; Neutron dose: 54 mrem

How much radiation exposure is available to the operator without extension if he has to make additional entries?

His available Non-Emergency margin for the year is...

a. 1488 mrem

b. 1521 mrem

c. 1599 mrem

d. 1712 mrem

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Generic Knowledge and Abilities **RO Group** 1 **SRO Group** 1 294001G301

GENERIC **Record Number** 120

2.3 Radiological Controls

2.3.1 Knowledge of 10 CFR 20 and related facility radiation control requirements. 2.6 3.0

Explanation of Answer CORRECT ANSWER. Gamma and neutron dose are summed for DDE. DDE and CEDE are summed together to obtain TEDE. The Dose limit without extension is 2000 mrem/year TEDE

Reference Title

NC.NA-AP.ZZ-0024

Learning Objectives

000113E059 a. Identify the personnel responsible for approval of the following dose extension:
Yearly Dose Extension
Declared Pregnant Women Dose Extension
Lifetime Dose Extension

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO EXAM BANK QUESTION ID #3324. Braidwood 1 09/14/1998

An operator has the following exposure history this year until today:

Deep Dose Equivalent (DDE)	210 mrem
Committed Effective Dose Equivalent (CEDE)	45 mrem
Shallow Dose Equivalent (SDE)	33 mrem

Today, the operator was required to make two entries into the Drywell at 5 percent reactor power:

Entry 1: Gamma dose: 52 mrem; Neutron dose: 24 mrem
Entry 2: Gamma dose: 124 mrem; Neutron dose: 54 mrem

Should be 24 mrem, not 54

How much radiation exposure is available to the operator without extension if he has to make additional entries?

His available Non-Emergency margin for the year is...

a. 1488 mrem

b. 1521 mrem

c. 1599 mrem

d. 1712 mrem

Answer:	b	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	03/12/2002
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G301
GENERIC								Record Number:	120

2.3 Radiological Controls

2.3.1 Knowledge of 10 CFR 20 and related facility radiation control requirements. 2.6 3.0

Explanation of Answer: CORRECT ANSWER. Gamma and neutron dose are summed for DDE. DDE and CEDE are summed together to obtain TEDE. The Dose limit without extension is 2000 mrem/year TEDE

Reference Title

NC.NA-AP.ZZ-0024

Learning Objectives:

000113E059	a. Identify the personnel responsible for approval of the following dose extension: Yearly Dose Extension Declared Pregnant Women Dose Extension Lifetime Dose Extension

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO EXAM BANK QUESTION ID #3324. Braidwood 1 09/14/1998

7.21 **Public Dose** - Dose received by a member of the public from station operations, or from another source under the control of PSEG. It does not include occupational dose or dose received from background radiation, as a patient from medical practices, or as a medical research volunteer.

7.22 **Radiation Area** - Any area accessible to personnel with radiation dose rates that exceed 5 mrem/hour DDE at 30 cm from any source.

7.23 **Radiation Work Permit** - A document used to provide workers with instructions for radiological controls associated with work in the RCA.

7.24 **Radiation Worker** - A worker who has completed Radiation Worker Training.

7.25 **Radioactive Sources** - Radioactive material used or stored for calibrating or testing station installed or portable instrumentation.

7.26 **Radioactive Waste** - Licensed radioactive material that has been determined to be no longer useful and that requires disposal.

7.27 **Radiological Effluent Technical Specifications (RETS/ODCM)** - Specifications contained in the Technical Specifications or the Offsite Dose Calculations Manual.

7.28 **Radiologically Controlled Area (RCA)** - An area within the Restricted Area that has the potential for significant personnel radiation exposure and has positive control over personnel access. The term "Radiological Control Area" is synonymous.

7.29 **Restricted Area** - Any area where access is controlled by the licensee to protect individuals from undue risks from exposure to radiation and radioactive materials. The Restricted Area is normally the area within the PSEG security fence (the Protected Area).

7.30 **Self Monitor** - An individual trained in radiation protection procedures who is not a member of the Radiation Protection staff, but who is qualified to use radiation protection instrumentation for personal radiation protection in High Radiation Areas for observation, sampling, or tours.

7.31 **Station ALARA Committee (SAC)** - The committee responsible for overall coordination of the ALARA program within the station. The committee meets at least semi-yearly to review the status of the ALARA Program and is composed of individuals from the major functional departments of the station.

7.32 **(Radiological) Stop Work Order** - A directive halting all work within a specified area, or associated with a specified activity, or performed by a specified group. The directive prevents unplanned radiation exposure or loss of control of radioactive material. The order can be lifted only by Radiation Protection Manager.

7.33 **Total Effective Dose Equivalent (TEDE)** - The sum of the external whole body dose (DDE) and the internal whole body dose (CEDE).

7.34 **Unrestricted Area** - The area outside of the Restricted Area (beyond the "owner-controlled area").

ATTACHMENT 1
ADMINISTRATIVE DOSE CONTROL LEVELS AND EXTENSION REQUIREMENTS
Page 1 of 2

Whole Body Dose Control Levels - TEDE			
Control Level	**Description**	**Action at Control Level**	**Increase Approval**
2000 mrem/year TEDE	Current year dose control level	Dose control level may be increased to 3000 mrem/year	Radiation Protection Supervisor
3000 mrem/year TEDE	Extended current year dose control level.	Dose control level may be increased to 4000 mrem/year	Radiation Protection Manager
4000 mrem/year TEDE	Final current year dose control level (may not be exceeded in non-emergency situations).	Incremental increase up to 4750 mrem.	Vice President - Operations
Dose Control Level to the Lens of the Eye - LDE			
4000 mrem/year LDE	Current year dose control level to the lens of the eye.	Incremental increase, based upon job requirements, up to 12,000 mrem.	Radiation Protection Manager

Whole Body Dose Control Level for the Declared Pregnant Woman (DPW)			
Dose Control Level	**Description**	**Action at Control Level**	**Increase Approval**
50 mrem/month or less TEDE	Monthly dose control level for DPWs.	Total dose for the gestation period should not exceed 450 mrem TEDE. Control level should not be increased beyond 50 mrem in any month unless absolutely necessary.	Radiation Protection Manager
Internal Dose Monitoring Threshold for the Declared Pregnant Woman (DPW)			
50 mrem/year CEDE	Internal dose monitoring threshold for DPWs. Confirmatory monitoring may be provided.	Monitoring threshold shall not be exceeded unless monitoring program is established by Radiation Protection.	Radiation Protection Manager

Which one of the following describes organizational grouping of Abnormal Operating Procedures (ABs) IAW SH.OP-AP.ZZ-0102 "Use of Procedures".

a. 100 series are operational transient procedures

b. 200 series address component failures

c. 300 series apply at all times

d. 000 series address fire and medical emergencies

Answer	c	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G405
GENERIC								**Record Number**	126

2.4	Emergency Procedures and Plan		
2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.	2.9	3.6

Explanation of Answer

Justification
IAW SH.OP-AP.ZZ-0102, section 5.5.2

Reference Title

SH.OP-AP.ZZ-0102, section 5.5.2

Learning Objectives

000113E005	a. Summarize the guidelines for the use of the following types of procedures: Abnormal Operating Procedures Emergency Operating Procedures Alarm Response procedures

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION BANK QID# Q57004

5.4.3 SO and AOP Interrelationship

- In general, the AOP is the preferred method of dealing with abnormal or transient conditions, and will normally take precedence over the implementation of normal operating procedures during these situations.

- The interrelationship is not always clearly defined between the two procedure types and operator skill and knowledge is required to determine this balance. Since every possible situation cannot be addressed in the abnormal operating procedure, the control room supervisor is allowed to exercise good judgment in the concurrent use of normal operating procedures during the implementation of the AOPs. Concurrent use of the normal operating procedure should not impede the progress through the AOPs, or counteract the alignment of the plant in mitigating the transient within the AOP.

5.5 **Hope Creek Abnormal Operating Procedures (AB)**

5.5.1 Purpose - describes the actions to be taken when important parameters or systems are in jeopardy yet, in most cases, the Reactor Protection System has not yet activated.

5.5.2 Types - the Abnormal Operating Procedures have been divided into two groups: operational transients and abnormal conditions.

- 100 Series Abnormal Condition addresses system or component failures which pose significant problems to the operator. In addition, these procedures deal with non-system related problems which may adversely effect operation (i.e., plant fire, flood, etc.). they are brief and rely on the operator's training and knowledge.

- 200 Series Operational Transient procedures are symptom oriented and deal with plant conditions which input directly into the RPS. The procedures are written to the point of entry into the EOPs. They are not written to cover actual system isolation and scram but rather to prevent these occurrences. They include any immediate steps for mitigating the consequences of the transient and subsequent steps for determining the origin of the event.

  The discussion section provides additional information to aid in assessing the transient. In general, the procedure directs the operator to look at key items and activities and relies on the operator's training and systems knowledge for specific component manipulations.

- 300 Series Reactor Power Oscillations applies at all times. The 100 and 200 series may provide entry conditions for the 300 series.

5.5.3 Use - guidelines for use of both types of ABs are as follows:

- All Immediate Actions shall be committed to memory.
- All expected Automatic Actions shall be verified to have occurred.
- If a scram condition or other EOP entry condition is met the appropriate EOP shall be entered.
- The Subsequent Actions shall be performed with the procedure in-hand or at the direction of a person with the procedure in-hand.
- The AB, when in use, should be utilized as part of the Control Room Log(s) and should be marked in a manner so as to allow for re-creation of the event. This should be accomplished by writing on the procedure information pertaining to major steps.

5.6 Alarm Response Procedures (AR)

5.6.1 Purpose - directs operator response to an alarm on an overhead panel, control room console, Plant Computer, or local panel in the plant

5.6.2 Types

- Overhead Alarm Response Procedures - the overhead AR directs the operator in responding to control room overhead alarms. Since many of the overhead ARs represent multiple inputs with a variety of digital or console alarms feeding the alarm, the overhead ARs are prepared as packages containing the overhead summary sheet along with the associated digital and console alarm. However, due to the volume of the packages, the procedures are split in the control room by panel. They are placed within easy reach and provide the operator with quick information on what may have caused the alarm as well as the major concerns associated with the alarm. The digital, console, and non-indicating alarm response procedures provide the operator with more specific information on causes and actions with the alarm.
- Local Alarm Response Procedures - the local ARs direct operator response to an alarm at a local panel.

Which one of the following describes how a scram is verified in accordance with HC.OP-IO.ZZ-0008 Shutdown from Outside the Control Room?

a. HCU nitrogen pressure verified to be less than 800 psig at each HCU

b. Reactor vessel pressure verified less than 920 psig

c. RPS power distribution circuit breakers verified to be open

d. Scram air header pressure verified to be less than 100 psig

Answer	a	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G434
GENERIC								**Record Number**	129

2.4	Emergency Procedures and Plan	
2.4.34	Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications.	3.8 3.6

Explanation of Answer: The scram is verified outside the control room via HCU Accumulator pressures < 800 psig at each HCU

Reference Title

HC.OP-IO.ZZ-0008

Learning Objectives

00112HE004	(R) Apply Precautions, Limitations and Notes while executing the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Exam Bank QID # Q54018

5.0 PROCEDURE

NOTE 5.0

A. Each step should be initialed upon completion of the step.

B. Attachment 2 should be referred to for RSP redundant instrumentation/controls.

C. Attachment 3 should be referred to for placing 'A' Loop RHR in Suppression Pool Cooling.

D. Attachments 6 and 7 should be referred to for plant communications information. When dispatching an operator to a remote shutdown control station, the operator should be provided with a sound-powered phone OR radio to assist with communication.

5.1 Establish Control from Outside the Control Room

5.1.1 **ENSURE** that all prerequisites have been satisfied IAW Section 2.0 of this procedure. ____

NOTE 5.1.2

A. IF the Reactor was NOT scrammed AND the MSIVs are still open, then the Feedwater System AND the Main Turbine Bypass Valves may be regulating Rx level AND Rx pressure at this time.

B. Opening the circuit breakers listed in Step 5.1.2 will deenergize the RPS busses, scramming the plant, AND deenergize the NSSSS busses, closing the MSIVs.

C. 10C410(10C411) RPS PWR Dist. Panels A(B) are located in Control/DG Bldg. El. 54'.

5.1.2 IF the Reactor was NOT scrammed prior to Control Room evacuation, THEN **OPEN** the following circuit breakers: **[CD-987X]** ____

A. CB2A, CB3A, CB5A, CB7A AND CB8A (RPS PWR DIST PNL A 10C410). ____

B. CB2B, CB3B, CB5B, CB7B AND CB8B (RPS PWR DIST PNL B 10C411). ____

5.1.3 IF the Rx scram was NOT verified prior to evacuating the Control Room, THEN **VERIFY** Rods Full In. (SPDS/~~CRIDS~~ (TSC) OR RMCS Activity Control Cards OR other). ______

5.1.4 **NOTIFY** Chemistry to verify that the Hydrogen/Oxygen System has tripped IAW HC.CH-SO.AX-0001(Q). ______

5.1.5 Upon arriving at the RSP, **MONITOR** the RSP System indications AND **CHECK** specifically for the following: ______

A. REACTOR VESSEL PRESSURE PR-7853D (905 - 1045 psig) ______

CAUTION 5.1.5.B

IF the rate of rise of RPV level indicates HPCI is injecting AND the Control Room is unmanned, THEN HPCI will have to be tripped using Attachment 8 when no longer required OR prior to exceeding the high level trip (Level 8). The high level trip may NOT function in the event a fire occurs in the relay room. [CD-012Z]

B. REACTOR VESSEL LEVEL LR-7854 (12.5 - 54 ") ______

C. RCIC System status (standby OR auto-initiated) ______

D. PSV-F013F,H,M SRV status (standby OR cycling open/closed) ______

E. SUPPRESSION CHAMBER WATER TR-3647J (AND M) (average less than 95°F) ______

F. DIESEL GENERATOR 1A(B,C,D)G400 TRIP/CLOSED Status (closed IF a loss of offsite power has occurred). ______

5.1.6 IF a loss of offsite power has occurred, THEN **SEND** an operator to the Diesel Generator Remote Control Panel (Aux. Bldg El. 130') to monitor Diesel Generator operation, AND **IMPLEMENT** HC.OP-AB.ZZ-0135(Q), Loss of Offsite Power, concurrent with this procedure. ______

Nuclear Training Department

Fax Cover Sheet

To:

JOE DiANTONIO
NAME

USNRC
COMPANY

DEPARTMENT

610 337-5320
FAX

610-337-5085
PHONE

TOTAL PAGES (INCLUDING COVER)

From:

ARCHIE FAULKNER
NAME

856-339-3966
PHONE

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☐ ***URGENT!*** ☐ ***FOR YOUR REVIEW*** ☐ ***REPLY ASAP*** ☐ ***PLEASE COMMENT***

Work Order Shop Papers
08/04/2001

PSEG

50020180

Order:	50020180	PERF OP-IS BF-103 CRD ACC CHG VLVS 550D	
Order Type	NUST		
Status	REL PRC SETC		
Notification			
Unit	H1		
Functional Location	H1BF	CONTROL ROD DRIVE HYDRAULIC HOPE CREE	
Equipment			
Assembly			
Location			
Room			
System	BF		
Priority	4	Other	
Main Work Center	O-H	HOPE CREEK OPERATIONS MANAGER	
Status	REL PRC SETC		
Basic Dates:	Start: 10/10/2001	Finish: 10/11/2001	Overdue 03/10/2002
Sfty Rltd/QA Reqd	N		
Sfty Class	N		
Qlty Grp Code			
SEISMIC	N		
EQ	EH2		
Permission to Begin Work		Date:	
		Time: 00:00:00	

Hold

Description of Work

PERF OP-IS BF-103 CRD ACC CHG VLVS 550D
PERFORM CRD ACCUMULATOR CHARGING WATER CHECK VLVS 550D
INSERVICE TEST REFUELING OP-IS BF-0103(O)

*************** LAST THREE COMPLETION DAT***************
19990213 19971120 19960310

*************** TECHNICAL SPECIFICATIONS ***************
4.0.5

*************** MMIS WORK ORDER NUMBER ***************
0089011307l

C.H.O.I.C.E. **SAFETY: The Only**

Commitment Help Oversight Involvement

Operation List Summary
08/04/2001

50020180

OP	Sub Op.	Work Center	Description	Start Date	Work	No	Durtn
0010		O-H	PERF OP-IS.BF-0103 C RD ACC CHG VLVS 550D	10/10/2001	8	2	4

SAFETY: The Only C.H.O.I.C.E.
Commitment, Help, Oversight, Involvement, Communication, Empowerment

Operation Key Info
08/04/2001

PSEG

50020180

Order: 50020180 PERF OP-IS.BF-103 CRD
ACC CHG VLVS 550D **Operation:** 0010 PERF OP-IS.BF-0103 CRD
ACC CHG VLVS 550D **Permit No's** **Work center:** O-H NNUC
Status: REL PRC SETC **Number of People:** 2 **Scheduled**
Dates: **Start:** 10/10/2001 **Finish:** 10/11/2001 **Planned Hours:** 8
Actual Dates: **Start:** **Finish:** **Actual Hours:**
0 **Personnel Number:** ________ **Completion Confirmation Number:**
1006454 **Confirmation Text:**

____
______________________________ **Signature:**

Description of Work:
PERF OP-IS.BF-0103 CRD ACC CHG VLVS 550D
PERFORM CRD ACCUMULATOR CHARGING WATER CHECK VLVS 550D
IN SERVICE TEST-REFUELING OP-IS.BF-103(Q).

*************** MTE ***************
CALIBRATED STOPWATCH

*************** DOCUMENTS ***************
CLASS MANUAL/DRAWING NUMBER SEQUENCE
OP-IS.BF-0103(Q)

*************** CODE JOB PACKAGE ***************

*************** ASSIGNED PLANNER ***************
POWELL 3643

*************** ACTIVITY COMPONENT ID ***************
NONE

HOPE CREEK GENERATING STATION

HC.OP-IS.BF-0103(Q) - Rev. 1

CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

USE CATEGORY: I

REVISION SUMMARY

1. This procedure has been converted from Professional Write to Microsoft Word.
2. The conversion of this procedure required portions of the Attachments to be re-typed.
3. Organizational title changes were made in this revision to bring the procedure in line with guidelines, as contained in NC.NA-AP.ZZ-0002(Q), Nuclear Business Unit Organization, Attachment 1 and are considered editorial based on an allowance in NC.NA-AP.ZZ-0001(Q), Attachment 7 for "changing personnel titles to reflect organizational changes (without changing authority or responsibilities)." Due to the extensive changes, revision bars were omitted.
4. The following changes were made in this revision to bring the procedure in line with the rules governing procedure format, content, and writer/reviewer guidelines, as contained in NC.NA-WG.ZZ-0001(Q), Procedure Writers Guide and can be considered Editorial in nature. Due to the extensive changes, revision bars were omitted.
   - Added "RECORDS" Section 6.0
   - Capitalization and Bolding of action verbs
   - Changed procedure Font from "Arial" to "Times New Roman"
   - Revised Cautions and Note boxes format from margin to margin
   - Moved "Commitment Document" numbers from left margins to end of applicable steps
   - Added Placekeeping/Step completion signoffs throughout procedure
5. This procedure has been revised to add Document Security classification statement "PSEG Internal Use Only" to procedure header.
6. LIST OF EFFECTIVE PAGES has been deleted, this is a generic change.
7. Removed Note 2.0 from Prerequisites Section that stated "Prerequisites within a subsection may be completed in any order". NC.NA-AP.ZZ-0001(Q), Nuclear Department Procedure System, states that Prerequisites need not be completed in order unless specifically stated, so, this Note is no longer required in any Implementing procedure. This change can be considered editorial in nature.
8. CAUTION 2.1.4 has been changed to NOTE 2.1.4. The information contented does not satisfy the criteria for a caution.

(Continued)

IMPLEMENTATION REQUIREMENTS

Effective Date 8/3/98

APPROVED: [signature] for [signature] 7/30/98
Operations Manager **Date**

NOTE 5.1.7

Additional Attachment 3 forms may be used for performing the following steps.

5.1.7 **VALIDATE** the Control Rod Drive Accumulator low pressure alarms by performing the following steps: PJF

A. **LIST** the identification number of each HCU recorded on Attachment 2 from Step 5.1.5.C on Attachment 3.

B. **ELIMINATE** the alarms caused by high water IAW HC.OP-SO.BF-0002(Q), Individual CRD HCU Operation, Subsection 5.6, Draining Accumulator (nitrogen side) of Water. PJF

C. IF the cause for the Accumulator alarm can not be attributed to high water, **ENTER VALID** on Attachment 3, otherwise **ENTER INVALID**.

D. **INITIAL** the space provided on Attachment 3.

E. **RETURN** this procedure to the NCO for the completion of this test.

5.1.8 **RECORD** SAT on Attachment 2 for any Control Rod Drive Accumulator Alarms invalidated in Step 5.1.7, otherwise **RECORD** UNSAT.

5.1.9 **RECORD** SAT on Attachment 2 for all Control Rod Drive Accumulators that did not alarm during the performance of Step 5.1.5.

5.1.10 **INITIAL** the space provided on Attachment 2.

5.1.11 **LOG** test end time in the Control Room log(s). 13:35

5.1.12 **SUBMIT** this procedure to the OS/CRS for review AND completion of Attachment 1.

ATTACHMENT 1
Page 1 of 3

OS/CRS DATA AND SIGNATURE SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.0 **PRETEST INFORMATION**

1.1 **Reason for the Test**

1.1.1 Regular Surveillance ____A____
INITIALS

1.1.2 Retest ________
INITIALS

1.1.3 IF not performing the complete test,
LIST subsections to be performed
AND Accumulators that this procedure is testing.

________ ________ ________ ________
________ ________ ________ ________
____
SUBSECTION(S)

1.2 **Plant Conditions**

1.2.1 Operational Condition ____4____

1.2.2 Reactor Power Level ____ϕ____

1.2.3 GMWe ____ϕ____

1.3 **Permission to Perform the Test**

1.3.1 A review of NC.NA-AP.ZZ-0005(Q); Station Operating Practices for a list of systems requiring an independent verification has been completed. The OS/CRS has placed an N/A in the applicable space(s) on Attachment 2 which DO NOT require an independent verification.

[signature] / 10/9/01 - 1125
OS/CRS DATE-TIME

1.3.2 Permission granted to perform this test.

[signature] / 10/11/01 0016
OS/CRS DATE-TIME

ATTACHMENT 1
Page 2 of 3
OS/CRS DATA AND SIGNATURE SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

2.0 POST TEST INFORMATION

2.1 The data acquired during the performance of this test has been reviewed for completeness and compliance with Technical Specification 4.0.5 and the test is considered:

2.1.1 SATISFACTORY (All acceptance criteria is marked SAT)

/ 10/10/01 1400
OS/CRS DATE-TIME

2.1.2 UNSATISFACTORY (Any test evaluations are marked UNSAT). TAKE action IAW NC.NA-AP.ZZ-0070(Q).

_______________ / _______________
OS/CRS DATE-TIME

2.1.3 Test results which are related to Technical Specification 4.0.5 have been evaluated for acceptability. IF required, an Action Request has been generated to incorporate new baseline data for Inservice Test components contained in this procedure. [CD-463H, PR 951018240]

/ 10.11.01 - 1316
IST IMPLEMENTATION ENGINEER DATE-TIME

2.1.4 Work Order No. _______________

2.1.5 Remarks _______________

ATTACHMENT 1
Page 3 of 3
OS/CRS DATA AND SIGNATURE SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

3.0 PROCEDURE PERFORMER(S) AND VERIFIER(S)

3.1 I have read and understand the steps of this procedure that I am required to perform. (All Departments)

PRINT NAME	SIGNATURE	INITIALS	DATE/TIME
Bill Hickey	[signature]	[initials]	10-9-01/1126
Butler	[signature]	[initials]	10-15-01/1005
[illegible]	[signature]	[initials]	10/10/01 1153
Meyers	[signature]	[initials]	10/11/01 0015
R. Binz III	[signature]	[initials]	10/11/01 1314

ATTACHMENT 2

Page 1 of 10

CONTROL ROOM DATA SHEET

CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 Control Rod Drive Exercise

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
02-19	N/A	SAT		*
02-23	N/A	SAT		*
02-27	N/A	SAT		*
02-31	N/A	SAT		*
02-35	N/A	SAT		*
02-39	N/A	SAT		*
02-43	N/A	SAT		*
06-15	N/A	SAT		*
06-19	N/A	SAT		*
06-23	N/A	SAT		*
06-27	N/A	SAT		*
06-31	N/A	SAT		*
06-35	N/A	SAT		*
06-39	N/A	SAT		*
06-43	N/A	SAT		*
06-47	N/A	SAT		*
10-11	N/A	SAT		*
10-15	N/A	SAT		*
10-19	N/A	SAT		*
10-23	N/A	SAT		*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

CONTROL ROOM DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 Control Rod Drive Exercise

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
10-27	N/A	SAT	[illegible]	*
10-31	N/A	SAT	[illegible]	*
10-35	N/A	SAT	[illegible]	*
10-39	N/A	SAT	[illegible]	*
10-43	N/A	SAT	[illegible]	*
10-47	N/A	SAT	[illegible]	*
10-51	N/A	SAT	[illegible]	*
14-07	N/A	SAT	[illegible]	*
14-11	N/A	SAT	[illegible]	*
14-15	N/A	SAT	[illegible]	*
14-19	N/A	SAT	[illegible]	*
14-23	N/A	SAT	[illegible]	*
14-27	N/A	SAT	[illegible]	*
14-31	N/A	SAT	[illegible]	*
14-35	N/A	SAT	[illegible]	*
14-39	N/A	SAT	[illegible]	*
14-43	N/A	SAT	[illegible]	*
14-47	N/A	SAT	[illegible]	*
14-51	N/A	[illegible]	[illegible]	*
14-55	N/A	[illegible]	[illegible]	*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

ATTACHMENT 2
Page 3 of 10
CONTROL ROOM DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 **Control Rod Drive Exercise**

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
18-03	N/A	SAT	[illegible]	*
18-07	N/A	SAT	[illegible]	*
18-11	N/A	SAT	[illegible]	*
18-15	N/A	SAT	[illegible]	*
18-19	N/A	SAT	[illegible]	*
18-23	N/A	SAT	[illegible]	*
18-27	N/A	SAT	[illegible]	*
18-31	N/A	SAT	[illegible]	*
18-35	N/A	SAT	[illegible]	*
18-39	N/A	[illegible]	[illegible]	*
18-43	N/A	SAT	[illegible]	*
18-47	N/A	SAT	[illegible]	*
18-51	N/A	SAT	[illegible]	*
18-55	N/A	SAT	[illegible]	*
18-59	N/A	SAT	[illegible]	*
22-03	N/A	SAT	[illegible]	*
22-07	N/A	SAT	[illegible]	*
22-11	N/A	SAT	[illegible]	*
22-15	N/A	SAT	[illegible]	*
22-19	N/A	SAT	[illegible]	*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

ATTACHMENT 2
Page 4 of 10
CONTROL ROOM DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 Control Rod Drive Exercise

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
22-23	N/A	SAT	[illegible]	*
22-27	N/A	SAT	[illegible]	*
22-31	N/A	SAT	[illegible]	*
22-35	N/A	SAT	[illegible]	*
22-39	N/A	SAT	[illegible]	*
22-43	N/A	SAT	[illegible]	*
22-47	N/A	SAT	[illegible]	*
22-51	N/A	SAT	[illegible]	*
22-55	N/A	SAT	[illegible]	*
22-59	N/A	SAT	[illegible]	*
26-03	N/A	SAT	[illegible]	*
26-07	N/A	SAT	[illegible]	*
26-11	N/A	SAT	[illegible]	*
26-15	N/A	SAT	[illegible]	*
26-19	N/A	SAT	[illegible]	*
26-23	N/A	SAT	[illegible]	*
26-27	N/A	SAT	[illegible]	*
26-31	N/A	SAT	[illegible]	*
26-35	N/A	SAT	[illegible]	*
26-39	N/A	SAT	[illegible]	*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

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CONTROL ROOM DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 Control Rod Drive Exercise

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
26-43	N/A	SAT	[illegible]	*
26-47	N/A	SAT	[illegible]	*
26-51	N/A	SAT	[illegible]	*
26-55	N/A	SAT	[illegible]	*
26-59	N/A	SAT	[illegible]	*
30-03	N/A	SAT	[illegible]	*
30-07	N/A	SAT	[illegible]	*
30-11	N/A	[illegible]	[illegible]	*
30-15	N/A	SAT	[illegible]	*
30-19	N/A	SAT	[illegible]	*
30-23	N/A	SAT	[illegible]	*
30-27	N/A	SAT	[illegible]	*
30-31	N/A	SAT	[illegible]	*
30-35	N/A	SAT	[illegible]	*
30-39	N/A	SAT	[illegible]	*
30-43	N/A	SAT	[illegible]	*
30-47	N/A	SAT	[illegible]	*
30-51	N/A	SAT	[illegible]	*
30-55	N/A	SAT	[illegible]	*
30-59	N/A	SAT	[illegible]	*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

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CONTROL ROOM DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 Control Rod Drive Exercise

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
34-03	N/A	SAT		*
34-07	N/A	SAT		*
34-11	N/A	[illegible]		*
34-15	N/A	SAT		*
34-19	N/A	SAT		*
34-23	N/A	[illegible]		*
34-27	N/A	SAT		*
34-31	N/A	SAT		*
34-35	N/A	SAT		*
34-39	N/A	SAT		*
34-43	N/A	SAT		*
34-47	N/A	SAT		*
34-51	N/A	SAT		*
34-55	N/A	[illegible]		*
34-59	N/A	SAT		*
38-03	N/A	SAT		*
38-07	N/A	SAT		*
38-11	N/A	SAT		*
38-15	N/A	SAT		*
38-19	N/A	SAT		*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

CONTROL ROOM DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 Control Rod Drive Exercise

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
38-23	N/A	SAT	[illegible]	*
38-27	N/A	SAT	[illegible]	*
38-31	N/A	SAT	[illegible]	*
38-35	N/A	SAT	[illegible]	*
38-39	N/A	SAT	[illegible]	*
38-43	N/A	SAT	[illegible]	*
38-47	N/A	SAT	[illegible]	*
38-51	N/A	[illegible]	[illegible]	*
38-55	N/A	[illegible]	[illegible]	*
38-59	N/A	SAT	[illegible]	*
42-03	N/A	SAT	[illegible]	*
42-07	N/A	SAT	[illegible]	*
42-11	N/A	[illegible]	[illegible]	*
42-15	N/A	SAT	[illegible]	*
42-19	N/A	SAT	[illegible]	*
42-23	N/A	SAT	[illegible]	*
42-27	N/A	[illegible]	[illegible]	*
42-31	N/A	SAT	[illegible]	*
42-35	N/A	[illegible]	[illegible]	*
42-39	N/A	[illegible]	[illegible]	*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

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CONTROL ROOM DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 Control Rod Drive Exercise

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
42-43	N/A	SAT	[illegible]	*
42-47	N/A	SAT	[illegible]	*
42-51	N/A	SAT	[illegible]	*
42-55	N/A	SAT	[illegible]	*
42-59	N/A	SAT	[illegible]	*
46-07	N/A	SAT	[illegible]	*
46-11	N/A	SAT	[illegible]	*
46-15	N/A	SAT	[illegible]	*
46-19	N/A	SAT	[illegible]	*
46-23	N/A	SAT	[illegible]	*
46-27	N/A	SAT	[illegible]	*
46-31	N/A	SAT	[illegible]	*
46-35	N/A	SAT	[illegible]	*
46-39	N/A	SAT	[illegible]	*
46-43	N/A	SAT	[illegible]	*
46-47	N/A	[illegible]	[illegible]	*
46-51	N/A	SAT	[illegible]	*
46-55	N/A	SAT	[illegible]	*
50-11	N/A	SAT	[illegible]	*
50-15	N/A	SAT	[illegible]	*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

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CONTROL ROOM DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 Control Rod Drive Exercise

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
50-19	N/A	SAT	[illegible]	*
50-23	N/A	SAT	[illegible]	*
50-27	N/A	SAT	[illegible]	*
50-31	N/A	SAT	[illegible]	*
50-35	N/A	SAT	[illegible]	*
50-39	N/A	SAT	[illegible]	*
50-43	N/A	SAT	[illegible]	*
50-47	N/A	SAT	[illegible]	*
50-51	N/A	SAT	[illegible]	*
54-15	N/A	SAT	[illegible]	*
54-19	N/A	SAT	[illegible]	*
54-23	N/A	SAT	[illegible]	*
54-27	N/A	SAT	[illegible]	*
54-31	Yes	Sat	[illegible]	*
54-35	N/A	SAT	[illegible]	*
54-39	N/A	SAT	[illegible]	*
54-43	N/A	SAT	[illegible]	*
54-47	N/A	SAT	[illegible]	*
58-19	N/A	SAT	[illegible]	*
58-23	N/A	SAT	[illegible]	*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

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CONTROL ROOM DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

1.1 Control Rod Drive Exercise

HCU NUMBER	STEP 5.1.5 ACCUMULATOR ALARM YES OR N/A	STEP 5.1.8 AND 5.1.9 SAT OR UNSAT (UNSAT FOR VALID, SAT FOR INVALID)	PERF	
58-27	N/A	SAT	[illegible]	*
58-31	N/A	SAT	[illegible]	*
58-35	N/A	SAT	[illegible]	*
58-39	N/A	SAT	[illegible]	*
58-43	N/A	SAT	[illegible]	*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT.

STEP	STOP WATCH M&TE NO.	CAL DUE DATE	NOTES
5.1.4	214052	7/22/02	

STEP	TEST ELAPSED TIME REQUIRED	ACTUAL	PERF	
5.1.5	≥ 2 minutes	2.1 min	[illegible]	*

* Acceptance Criterion - the SAT/UNSAT block must be marked SAT or actual time must be ≥ 2 minutes.

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INPLANT DATA SHEET
CONTROL ROD DRIVE ACCUMULATOR CHARGING WATER CHECK VALVE - REFUEL - INSERVICE TEST

STEP	HCU NUMBER	VALID/INVALID	PERF
5.1.7	54-31	INVALID	

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Applicability:

____	Salem 1	____	Salem 3 (Gas Turbine)	____	NBU Common
____	Salem 2	XXX	Hope Creek		
____	Common to Salem 1 & 2	____	Common to Hope Creek & Salem		

COMPLETION AND APPROVAL

Mark Cirelly	1/21/99	MARK CIRELLY	09/09/00
PREPARER (SIGN)	DATE	NAME (PRINT)	QUAL EXPIRES
John A Thompson	1/21/99	JOHN THOMPSON	04/02/99
PEER REVIEWER (SIGN)	DATE	NAME (PRINT)	QUAL EXPIRES
[signature]	1/21/99	LEN RAJKOWSKI	03/10/99
APPROVAL (SIGN)	DATE	NAME (PRINT)	QUAL EXPIRES

Safety Evaluation No. H99-005

SEIRT REVIEW Date 1/21/99

SORC Chairman: (Hope Creek) [signature] Mtg. No. 99-006 Date 1/22/99
Sta. GM Approval: (Hope Creek) [signature] Date 1/23/99
SORC Chairman: (Salem) N/A Mtg. No. ____ Date ____
Sta. GM Approval: (Salem) N/A Date ____

Safety Evaluation and associated documentation sent to Nuclear Review Board (NRB)
M/C N38: SORC
[UFSAR 17.2.1.1.2.1] Presenter: ____ Date: ____

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1.0 10CFR50.54 PRE-SCREENING

YES	NO	
____	XXX	a. Could the proposed change affect the Quality Assurance Program Description included in the UFSAR? If YES, **STOP**. Contact Quality Assessment for assistance.
____	XXX	b. Could the proposed change affect the Security Plan? If YES, **STOP**. Contact Nuclear Security for assistance.
____	XXX	c. Could the proposed change affect the Emergency Plan? If YES, **STOP**. Contact Emergency Preparedness for assistance.

2.0 10CFR50.59 APPLICABILITY REVIEW - 10CFR50.59 applies because:

2.1 The proposal changes the facility as described in the SAR.

YES XXX NO ____

Explain: Sections 7.2.1.1.11 and 7.2.2.3.7 of the UFSAR describe the RPS scram signal generated when the Reactor Mode Switch is placed the "Shutdown" position. Insofar as the proposed procedure revision provides instructions to bypass this feature, it constitutes a change to the facility as described in the SAR.

2.2 The proposal changes procedures as described in the SAR.

YES ____ NO XXX

Explain: Although the subject procedure is listed in the SAR, it is not described. Therefore, the proposed revision does not change procedures as described in the SAR.

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2.3 The proposal involves a test or experiment not described in the SAR.

YES ______ NO XXX

Explain: The activities associated with the proposed procedure revision do not constitute a test or experiment as defined in NC.NA-AP.ZZ-0008 (Q) and NC.NA-AS.ZZ-0059 (Q). Therefore, it does not involve a test or experiment not described in the SAR.

3.0 LICENSING BASIS DOCUMENTATION

3.1 UFSAR REVISION DETERMINATION - Does the proposal require a UFSAR change?

YES ______ NO XXX

UFSAR Change Notice No. XXX

The proposed procedure revision provides instructions on bypassing the RPS scram signal generated when the Mode Switch is placed in the "Shutdown" position. This will only be done when the reactor is shutdown with all rods inserted. The scram signal will only be bypassed for approximately 30 seconds each time the mode switch is required to be moved to the shutdown position and then restored. Since this will not be a normal mode of operation, and the system is restored to the configuration described in the SAR each time the procedure is performed, a change to the SAR is not required.

3.2 TECHNICAL SPECIFICATION REVISION DETERMINATION - Does the proposal require a Technical Specification change?

YES ______ NO XXX

If a change is required, **STOP**. Contact Nuclear Licensing for assistance in preparation of a License Change Request.

Identify the pertinent Technical Specification sections that were reviewed to make the determination:

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2.2 LIMITING SAFETY SYSTEM SETTINGS – REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

3.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

3.3.6 CONTROL ROD BLOCK INSTRUMENTATION

3.9.1 REFUELING OPERATIONS – REACTOR MODE SWITCH

4.0 DESCRIPTION

4.1 Describe the modification or activity being evaluated and its expected effects.

JUSTIFICATION FOR PROPOSED CHANGE:
When the Reactor Mode switch is placed in the shutdown position, a RPS scram signal is generated and then is automatically bypassed after a six (+/- four) second time delay. This feature is required to be operable in Operational Conditions (Op Con) 1 through 5 in accordance with Specification 3.3.1. With the feature not operable in Op Con 5, the required action is to suspend core alterations and insert all insertable control rods within one hour.

During the course of a refueling outage, it is necessary to periodically move the mode switch between the refueling and shutdown positions for testing and surveillance purposes and to conduct core alterations. However, the resultant scram signals cause CRD scram valves to reposition and scram accumulators to discharge. Empirical experience has shown this to be a major contributor to control rod withdrawal difficulties during startup due to nitrogen intrusion into the CRD HCUs. Additionally, the repeated scram signals place mechanical and hydraulic stresses on internal control rod drive mechanism components that accelerate the degradation of the drives.

DESCRIPTION OF PROPOSED CHANGE:
UFSAR Section 7.2.1.1.10 describes a manual reactor scram as being initiated by use of the four RPS manual pushbuttons. Additionally, the statement is made that "Manual reactor scram is diverse to all automatic reactor trip signals." Section 7.2.1.1.11 then describes the reactor mode switch manual scram signal. Therefore, the scram signal

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generated by the mode switch is redundant to the manual scram pushbuttons, which in turn are redundant to all automatic scram signals.

In order to preclude control rod withdrawal difficulties during startups, the proposed procedure revision is being incorporated to bypass the RPS scram signal generated when the mode switch is shifted from the refuel to the shutdown position. The prerequisites of the procedure require permission to be obtained from the OS/CRS, all control rods to be inserted (or the core to be off-loaded) and no core alterations to be in progress. These prerequisites will ensure that Technical Specification operability is addressed, and that the required actions have been completed prior to bypassing the mode switch scram signal. A note will alert the operator that bypassing the mode switch scram signal will render the switch inoperable in accordance with Technical Specifications. Additionally, the prerequisites require the verification that banana jacks have been installed on the affected terminals in order to facilitate the required jumpering. The installation of the banana jacks will be performed under existing approved procedures. This ensures compliance with the guidance of Regulatory Guide 1.118 and it's referenced documents pertaining to lifted leads and jumpers as discussed in UFSAR Sections 1.8.1.118 and 7.1.2.4, Item #17.

Once the prerequisites are complete, the mode switch position scram signal, and only that signal, is bypassed by installing four jumpers. The mode switch is then placed in the shutdown position. After approximately thirty seconds, the jumpers are removed and the mode switch is returned to operable status. The value of approximately 30 seconds is used to provide sufficient time for the six (+/- four) second timer to time out and then automatically bypass the scram signal. However, a delay in the removal of the jumpers would neither affect nor invalidate this evaluation or its conclusions since there are no time-dependent factors introduced. At this point, the jumpers are no longer required and can be removed. Independent verification of the prerequisites is performed as procedural steps. Installation and removal of the jumpers are documented and second verified in Attachment 5 of the procedure.

DESCRIPTION OF THE RPS LOGIC AFFECTED BY PROPOSED CHANGE:
The part of the RPS logic that is affected by the proposed revision is shown on GE Elementary Drawing PN1-C71-1020-0006, sheet 12. This is the reactor manual scram trip logic for each of the four RPS subchannels (A1, A2, B1 and B2). The normally energized logic feeds the K15A-D relays. If the K15 relays become de-energized, they in turn open the K15 contacts in the automatic scram logic shown on sheet 13, thus de-energizing the K14 contacts and causing a reactor scram. The K15 relays can be de-

energized by the Nuclear Instrumentation (NI) non-coincident trip contacts, the reactor mode switch being placed in the shutdown position or the manual RPS scram pushbuttons. The mode switch part of the logic consists of four contacts in parallel, K32, K34, K35 and K36. Each contact is associated with a mode switch position, (run, startup, refuel and shutdown respectively) and is closed when the switch is in that position. The K36 contact, which is closed when the mode switch is in the shutdown position, also has a time delayed contact in series with it, the K16 contact. These contacts close six (+/- four) seconds after the mode switch is placed in shutdown. Consequently, when the mode switch is placed in shutdown, the K36 contacts close; however, logic power is interrupted for six (+/- four) seconds until the K36 contacts close. This is how the scram signal is generated and then automatically bypassed.

The jumpers installed by the proposed revision are placed across terminals Z-9 and ZZ-23 in the A1 RPS logic, A-9 and BB-23 in the A2 logic, Z-9 and ZZ-23 in the B1 logic and A-9 and BB-23 in the B2 logic. As can be seen on GE Elementary Drawing PN1-C71-1020-0006, sheet 12, jumpers at these locations will provide a logic path around contacts K32, K34, K35 and K36 thus preventing a scram signal when the mode switch is placed in shutdown. The rest of the manual scram logic is unaffected by the jumpers since opening of any of the other contacts will de-energize the respective K15 contacts and result in a scram signal. Additionally, it can be seen on sheets 2 and 12 that there is no other mode switch functions affected by the jumpers. The signals associated with the run, startup and refuel positions are unaffected as seen on sheet 2 and the non-scram signals associated with the shutdown position (to the Nuclear Steam Supply System, Rod Block interlocks, MSIV isolation bypass and Scram Discharge Volume High Level bypass logics) remain totally unaffected. Additionally, the NI and manual RPS push button logics remain unaffected.

4.2 Identify the parameters and systems affected by the change.

The proposed procedure revision affects the RPS scram signal generated when the mode switch is placed in the shutdown position. No other signals or actuations are affected. Refer to preceding discussion in Section 4.1 for technical discussion and references.

4.3 Identify the credible failure modes associated with the change.

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The credible failure modes associated with the proposed procedure revision include incorrect installation or removal of the subject jumpers and failure of the RPS manual scram pushbuttons necessitating use of the mode switch to initiate a manual scram.

4.4 Provide references to location of information used for the Safety Evaluation.

UFSAR Sections 1.8, 7.2, 7.5, 7.7, 15.7, 15.8, 15.9

Technical Specifications 2.2, 3.3.1, 3.9.1, 3.3.6,
GEK-90348B – Reactor Protection System (PN1-A41-8010-0044 (2))

ANSI/ANS-58.9-1981 – Single Failure Criteria for Light Water Reactor Safety-related Fluid Systems

Regulatory Guide 1.47 – BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

Regulatory Guide 1.118 – PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS – REVISION2

NC.NA-AP.ZZ-0008 (Q)

NC.NA-AS.ZZ-0059 (Q)

4.5 Other Discussion, If applicable.

UFSAR Section 1.8.1.47 states that the Hope Creek Generating Station complies with Regulatory Guide (Reg Guide) 1.47, "Bypassed And Inoperable Status Indication For Nuclear Power Plant Safety Systems". The Reg Guide delineates requirements to provide automatic or manually actuated control room indication when the protective action of some part of a protection system has been bypassed or deliberately rendered inoperative. Although the proposed procedure will bypass the reactor mode switch position scram signal, it will only be performed when all control rods have been inserted or the core has been off-loaded. Therefore, the protective action that is affected by the bypass, control rod insertion, will already have occurred. Per reg. Guide 1.47 automatic indication of bypass or inoperable conditions apply when all of the following conditions exist::

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1. The bypass or inoperable condition affects a system that is designed to perform automatically a function that is important to the safety of the public. (Placing the mode switch to shutdown is a manual function, so this condition does not apply)
2. The bypass will be utilized by plant personnel or the inoperable condition can reasonably be expected to occur more frequently than once per year. (Although this procedure will only be performed during an outage, it will occur several times during the outage and therefore applies)
3. The bypass or inoperable condition is expected to occur when the affected system is normally required to be operable. (As controlled by the procedure, the bypass will occur when the mode switch is inoperable and appropriate technical specification action statement will be in affect, so this condition does not apply)

Based on the preceding discussion, It is concluded that although the requirements of the regulatory guide do not apply, the proposed revision is nonetheless in compliance with it.

5.0 USQ DETERMINATION - Is an Unreviewed Safety Question (USQ) involved?

5.1 Which anticipated operational transients or postulated design basis accidents previously evaluated in the SAR are considered applicable to the proposal?

Insofar as the mode switch shutdown position scram is a redundant means of initiating a manual reactor scram, the Anticipated Transient Without Scram event is applicable to the proposed change.

5.2 May the proposal:

a. Increase the probability of an accident previously evaluated in the SAR?

YES ______ NO XXX

DISCUSSION: The proposed procedure revision will facilitate bypassing of the RPS scram signal generated when the mode switch is placed in the shutdown position. This is accomplished by installation of four jumpers. There are no other signals or actuations affected by installation of the specified jumpers. All other scram signals will remain unaffected and the rod block generated when the mode switch is in the shutdown position will remain in effect. In accordance with prerequisites included in the proposed change, the evolution will not be performed unless permission has been obtained from

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the OS/CRS, all control rods have been inserted (or the core is off-loaded) and no core alterations are in progress. Independent verification of these prerequisites is performed as procedural steps. Per the proposed revision, once the mode switch has been placed in shutdown and approximately thirty seconds have elapsed, the jumpers are removed and the system configuration is restored to normal. Since the prerequisites remain in effect until the jumpers are removed, a delay in the removal of the jumpers would have no adverse impact on plant safety. Installation and removal of the jumpers are documented and second verified in accordance with an attachment included in the revision. Based on the preceding, it is concluded that the prerequisites will ensure the proposed procedure will only be performed when the safety function of RPS, control rod insertion, has already been met (or the core is off-loaded) and the probability of a failure of RPS leading to an ATWS event is completely precluded.

Based on the preceding, it is concluded that the proposed change will not affect any accident precursors and therefore will not increase the probability of any accidents previously evaluated in the SAR.

b. Increase the consequences of an accident previously evaluated in the SAR?

YES ______ NO XXX

DISCUSSION: UFSAR Section 7.2.1.1.10 describes a manual reactor scram as being initiated by use of the four RPS manual pushbuttons. Additionally, the statement is made that "Manual reactor scram is diverse to all automatic reactor trip signals." Section 7.2.1.1.11 then describes the reactor mode switch manual scram signal. Therefore, the scram signal generated by the mode switch is redundant to the manual scram pushbuttons, which in turn are redundant to all automatic scram signals.

In accordance with prerequisites included in the proposed change, the evolution will not be performed unless permission has been obtained from the OS/CRS, all control rods have been inserted (or the core is off-loaded) and no core alterations are in progress. Since the reactor is already shutdown, there are no credible accidents in which the manually initiated scram signal could be a potential mitigating factor. Therefore, the consequences of a

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failure to scram (ATWS), including potential radioactive release, are unaffected. It is therefore concluded that the proposed change will not increase the consequences of any accident previously evaluated in the SAR.

5.3 What malfunctions of equipment important to safety that were previously evaluated in the SAR are considered applicable to the proposal?

The scram signal generated by the mode switch is redundant to the manual scram pushbuttons, which in turn are redundant to all automatic scram signals. Therefore, failure of the RPS system to initiate an automatic reactor scram is the most pertinent malfunction.

5.4 May the proposal:

a. Increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

YES ____ NO XXX

DISCUSSION: The RPS logic is de-energized to trip. When the mode switch is placed in the shutdown position, a contact opens which then interrupts power to the logic relay and results in a scram signal. When the scram signal is bypassed in accordance with the proposed revision, jumpers will be installed around the contacts associated with the mode switch position. This will maintain a path for current flow to the rest of the logic train when the switch position contacts open, thus defeating the scram signal. However, the jumpers will have no affect on the rest of the logic since the opening of any of the other contacts associated with automatic scram setpoints and the RPS manual push buttons will still de-energize the logic and result in a reactor scram signal. Therefore, although the mode switch position scram signal is one feature that can be used to mitigate an ATWS event, it will not affect the probability of the system failing to automatically scram the reactor when any of the sensed parameters exceed their specified setpoints during an anticipated transient. The preceding discussion notwithstanding, the proposed procedure will not be performed unless the reactor is already fully shutdown. Requiring either all control rods to be

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inserted or the core to be off-loaded ensures this. Therefore, the conditions leading to an ATWS event are entirely precluded.

UFSAR Section 15.8 provides an assessment of the Anticipated Transient Without Scram (ATWS) event. The discussion focuses on operator response delineated by the Emergency Operating Procedures and the use of "diverse, highly redundant, and very reliable scram systems". The description mentions the following design features: normal scram systems, Alternate Rod Insertion System, ATWS recirculation pump trip, manual rod insertion, Standby Liquid Control System, feedwater runback, and the scram discharge volume. It does not describe or list the reactor mode switch or the associated scram signal.

As a mitigation feature (not described in the SAR), the mode switch is placed in shutdown in accordance with the EOPs in order to insert a redundant scram signal and also to bypass the MSIV isolation interlock if reactor pressure decreases to the specified setpoint. This retains the main condenser as a heat sink after a reactor trip. The proposed change will have no impact on this feature.

Potential errors that could occur during installation and removal of the subject jumpers were assessed as part of this safety evaluation. During installation of the jumpers, the jumpers could be misplaced. The misplaced jumper could:
1. bypass a different function than intended
2. short out a power bus
3. cross-connect circuits such that the logic operates in a different manner.

In any of these scenarios, there is no safety impact. The procedure is only being used when the plant is shut down and after all control rods are verified to be full in or the core is off-loaded. Therefore, the conditions leading to an ATWS event are entirely precluded and the safety function of RPS has already been completed. If a jumper is misplaced, a half scram will occur and be detected when the mode switch is placed in shutdown. RPS is fail-safe and a power bus short would result in a blown fuse and half scram in that channel. A single jumper mistake would be limited to one sub-channel (A1, A2, B1, and B2) due to the physical separation of channels in the RPS panel bays.

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Title: REACTOR PROTECTION SYSTEM OPERATION

Installation and removal of the subject jumpers are documented and second verified in accordance with an attachment included in the revision. Additionally, the prerequisites require the verification that banana jacks have been installed on the affected terminals in order to facilitate the required jumpering. The installation of the banana jacks will be performed under separate station procedures and will provide additional assurance that the jumpers are correctly installed, do not come loose while installed and are properly removed. This ensures compliance with the guidance of Regulatory Guide 1.118 and it's referenced documents pertaining to lifted leads and jumpers as discussed in UFSAR Sections 1.8.1.118 and 7.1.2.4, Item #17. Where temporary alterations are required (e.g., Jumpers and/or Lifted Leads), Hope Creek is committed to follow the guidance in Office of Inspection and Enforcement (IE) Information notice 84-37, "Use of Lifted Leads and Jumpers During Maintenance and Surveillance Testing", which recommends a combination of administrative controls and functional test to verify the restoration of proper system configuration following surveillance test. This allows for additional procedure checks, recordings, and independent verifications as outlined in this referenced procedure. Therefore, it would require multiple human errors for the jumpers to be incorrectly removed. Additionally, it would require multiple equipment failures (i.e. the systems described in UFSAR Section 15.8) in order for this malfunction to impact the consequences of an ATWS. Based on the preceding discussion it is concluded that the proposed change will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

b. Increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

YES ______ NO XXX

DISCUSSION: As discussed in the preceding discussion, the sequence of events which would be required in order to necessitate use of the mode switch to initiate a manual scram are precluded by the proposed procedure revision. Therefore, the consequences of an ATWS event are unaffected.

FORM-3
10CFR50.59 SAFETY EVALUATION

I.D. Numbers/Reference/Revision: HC.OP-SO.SB-0001(Q) REVISION 15
Title: REACTOR PROTECTION SYSTEM OPERATION

The proposed procedure will only be used when all control rods are inserted or the core is off-loaded. Thus, the safety function of RPS, control rod insertion, have already been fulfilled and the conditions leading to an ATWS event are precluded. It is therefore concluded that the proposed revision will not increase the consequences of, or radioactive release associated with, a malfunction of equipment important to safety previously evaluated in the SAR.

5.5 May the proposal:

a. Create the possibility of an accident of a different type from any previously evaluated in the SAR?

YES _____ NO XXX

DISCUSSION: The proposed procedure revision will only be performed when the reactor is shutdown with all rods inserted (or the core off-loaded) and no core alterations in progress. There will be no impact to any mechanical or electrical distribution systems and no potential to cause undesired initiations or failures.

The proposed change will bypass the scram signal associated with the reactor mode switch shutdown position. This will only be done when all rod s are inserted or the core is off-loaded. Once the prerequisites are met, there are no compensatory actions required while the subject scram signal is bypassed. Therefore, there are no manual actions taking the place of automatic actions.

It is therefore concluded that the proposed change will not create the possibility of an accident of a different type from any previously evaluated in the SAR.

b. Create the possibility of a malfunction of a different type from any previously evaluated in the SAR?

YES _____ NO XXX

I.D. Numbers/Reference/Revision: HC.OP-SO.SB-0001(Q) REVISION 15
Title: REACTOR PROTECTION SYSTEM OPERATION

DISCUSSION: The jumpers used to bypass the mode switch position scram signal will have no impact on any other scram signal or mode switch interlock (e.g. Rod block). As discussed previously, incorrect installation of the jumpers or restoration of the mode switch scram feature would require multiple human errors and is therefore not a credible event. Further, even if this were to occur, the RPS manual pushbuttons would still be available to initiate a scram as described in UFSAR Section 7.2.1.1.10. It is therefore concluded that the proposed change will not create the possibility of a malfunction of a different type from any previously evaluated in the SAR.

5.6 Does the proposal reduce the margin of safety as defined in the basis for any Technical Specifications?

YES ______ NO XXX

Discuss the bases for the determinations and identify the pertinent Technical Specification sections that were reviewed to make the determination (use continuation sheets if required).

Specification 2.2, LIMITING SAFETY SYSTEM SETTINGS – REACTOR PROTECTION SYSTEM SETPOINTS, lists the RPS scram signals and their respective setpoints. The reactor mode switch shutdown position is listed as Item #11 on Table 2.2.1-1 with no trip setpoint or allowable value specified. The basis for this specification states that the reactor mode switch shutdown position provides additional manual reactor trip capability. With a RPS setpoint set less conservative than specified (or with the mode switch position scram unavailable), Specification 2.2 requires the affected channel to be declared inoperable and the actions specified in Specification 3.3.1 to be taken. This action will apply when the mode switch scram signal is bypassed.

Specification 3.3.1provides the operability requirements for RPS instrumentation. The basis for this specification states that RPS automatically initiates a reactor scram to: 1) preserve the integrity of the fuel cladding; 2) preserve the integrity of the reactor coolant system; 3) Minimize the energy which must be absorbed following a LOCA; and 4) prevent inadvertent criticality. Table 3.3.1-1 requires the reactor mode switch position scram signal to be operable in Operational Conditions 1 through 5. The actions required to be taken if the mode switch

I.D. Numbers/Reference/Revision: HC.OP-SO.SB-0001(Q) REVISION 15
Title: REACTOR PROTECTION SYSTEM OPERATION

position scram is inoperable are to suspend all core alterations and insert all insertable control rods within one hour. These actions will be required to be taken in accordance with the prerequisites of the proposed revision prior to bypassing the mode switch shutdown position scram. Therefore, the action requirements of Technical Specifications will be satisfied and the function of RPS in shutdown conditions, to prevent inadvertent criticality, will be fulfilled.

Specification 3.9.1 requires the mode switch to be operable and locked in the refuel or shutdown position in operational condition 5. The basis for this specification state that locking the mode switch in refuel or shutdown ensures that the restrictions on control rod withdrawal and refueling platform movement during refueling operations are properly activated. For the shutdown position this applies to the control rod block generated when the mode switch is in the shutdown position. Although the mode switch will be declared inoperable when the shutdown position scram is bypassed, the rod block will be unaffected and will therefore fulfill the intent of this specification. Refer to Section 4.1 for technical discussion and references.

The rod block that is generated when the mode switch is in the shutdown position is required to be operable in Operational Conditions 3 and 4 in accordance with Specification 3.3.6, CONTROL ROD BLOCK INSTRUMENTATION. This function will be unaffected by implementation of the proposed procedure revision. Refer to Section 4.1 for technical discussion and references.

The proposed procedure revision requires the installation of four jumpers; each of which will be installed in a separate bay in two separate panels. The jumpers will be installed into banana jacks, which will be readily accessible. Removal of the jumpers will be documented and independently verified. Based on the simplicity of the task and the ease of accessibility for removal and verification of removal, operability of the mode switch function will be re-established by documentation of jumper removal and independent verification. No other re-tests will be required.

Based on the preceding discussion, it is concluded that the proposed change will not reduce the margin of safety as defined in the basis for any Technical Specifications.

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10CFR50.59 SAFETY EVALUATION
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Revision 0

I.D. Numbers/Reference/Revision: HC.OP-SO.SB-0001(Q) REVISION 15
Title: REACTOR PROTECTION SYSTEM OPERATION

6.0 10CFR50.59(b)(2) REPORT - Provide a brief description of the change and a summary of the Safety Evaluation.

The proposed procedure revision provides instructions on bypassing the RPS scram signal generated when the Mode Switch is placed in the "Shutdown" position. This will only be done when the reactor is shutdown with all rods inserted. The scram signal will only be bypassed for approximately 30 seconds each time the mode switch is required to be moved to the shutdown position and then restored. The safety evaluation concludes that the proposed change does not constitute an unreviewed safety question and no Technical Specification changes are required.

7.0 CONCLUSION

If ALL answers in Section 5 are "NO," the proposal does NOT involve a USQ.

If ANY answer in Section 5 is "YES," the proposal DOES involve a USQ.

Is a USQ involved?

YES _____ NO XXX

If a USQ is involved, refer to NC.NA-AP.ZZ-0035 (Q) and obtain assistance from Licensing for additional processing.

LCR Number: NA

Nuclear Training Department

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HOPE CREEK GENERATING STATION

HC.OP-IO.ZZ-0008(Q) - Rev. 10

SHUTDOWN FROM OUTSIDE CONTROL ROOM

USE CATEGORY: I

REVISION SUMMARY:

REV. 10

1. Attachment 1 - step B.1.5 has been changed from "TCIC" to RCIC.

2. Attachment 5 has been revised to delete that portion containing the curve drawing. This change satisfies the requirements of revision request **OP-95-1139**.

3. The following changes have been incorporates to satisfy the requirements of revision request **OP-95-1154**:

   - Step 5.9.12 has been revised to delete the reference to T.S Fig 3.4.6.1-1 on Attachment 1.
   - Step 5.9.12 has been revised to reference Tech Spec 3.4.6.1

4. Attachment 5 - step 5.4 has been revised to reflect the deletion of the graphics. Reviewer's comment.

5. The following changes have been incorporated as per procedure reviewer's comments:

   - Step 3.1.9.C has been revised to change HV-F007A to HV-F007B. This change conforms with the valve description and the body of this procedure. This procedure only addresses the HV-F007B.
   - Step 3.1.9.D has been revised to delete the reference to HV-F007A. This procedure provides no guidance for lowering Reactor water level utilizing the HV-F007A.

IMPLEMENTATION REQUIREMENTS

This procedure revision is only effective for use after a paper copy is issued to the Control Room. The previous revision remains in effect until then, regardless of the approval date or the DCS update date. A 1/18/96

APPROVED: [signature] 1-19-96
Operations Manager **Date**

USE CATEGORY: I

CONTINUATION SHEET

REVISION SUMMARY:

REV. 10

5. The following changes have been incorporated as per procedure reviewer's comments:

    - Step 5.11.2.G has been restructured to included substeps.
    - Caution 5.1.12.1.D has been revised to indicate "failure to reset NSSSS prior to transferring to the MCR may result in a Shutdown Cooling Isolation" versus "NSSS."
    - Step 5.10.2 has been revised to indicate Reactor coolant temperature is < 120°F prior to route RHR to Radwaste. This change conforms with step 5.10.2.A, 3.1.9.F, 5.11.2.B and Caution 5.11.2.
    - Step 5.9.11 has been revised to add the following:
    -

      "(Maintain an administrative temperature range of 90°F -110°F. Other temperature(s) within TS limits may be used to support specific plant operations, as necessary)." This change conforms with HC.OP-IO.ZZ-0004(Q).

6. Attachment 4 - step 4.1.3 has been revised to correct valve number. The valve referenced to as 1BC-V26 changed to 1BC-V262.

7. The following changes have been incorporated as per reviewer's comments:

    - Step 3.1.9.A has been revised to indicate only the HV-F004B.
    - Step 3.1.9.B has been revised to indicate only the HV-F024B.

SHUTDOWN FROM OUTSIDE CONTROL ROOM

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SHUTDOWN FROM OUTSIDE CONTROL ROOM

TABLE OF CONTENTS

SECTION **TITLE** **PAGE**

SHUTDOWN FROM OUTSIDE CONTROL ROOM

START TIME	____________ DATE ____________	BY ____________	
TERMINATION TIME	____________ DATE ____________	BY ____________	
COMPLETION TIME	____________ DATE ____________	BY ____________	

1.0 PURPOSE

This procedure provides guidelines for the shutdown of the plant from outside the Control Room, and reestablishing control in the Control Room.

2.0 PREREQUISITES

2.1 HC.OP-AB.ZZ-0130(Q), Control Room Evacuation, is complete, if possible.

2.2 Establish communications between each local panel or breaker and the Remote Shutdown Panel as the steps are performed

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Administrative

3.1.1 This procedure is to be used as a guideline for the shutdown of the plant from outside the Control Room. It is not required that each section/step be performed in precise sequence as long as the sections/steps are performed in a timely manner in keeping with the intent of this procedure.

3.1.2 In the event plant conditions require a delay during some part of this procedure, the Senior Nuclear Shift Supervisor/Nuclear Shift Supervisor (SNSS/NSS) shall retain this procedure until it is continued or terminated.

3.1.3 If this procedure is terminated prior to completion, the SNSS/NSS shall note the reason, time, and date of termination on this procedure.

3.1.4 For any unit scram, refer to the Event Classification Guide for the appropriate classification and notifications.

3.1.5 Ensure compliance with the Reactor Coolant System temperature and pressure requirements of Technical Specification 3.4.6.1.

3.1.6 Ensure compliance with the Reactor Vessel and Head Flange temperature limits of Technical Specification 3.4.6.1.d.

3.1.7 Observe the Suppression Chamber temperature requirements of Technical Specification 3.6.2.1.

CD-904B
CD-695A

3.1.8 When the RSP Transfer Switches are placed in EMER, all trips and auto starts associated with the following equipment are bypassed:

A. SACS PUMPS B and D

B. SSWS PUMPS B and D

C. RHR PUMP B

D. RCIC system

(The backup mechanical overspeed trip of 125% rated speed is still provided (RCIC). This turbine trip will close the trip and throttle valve (HV-4282). This overspeed trip must be locally reset to allow relatching of the Turbine Trip Throttle valve. The limitorque must be manually run to the full closed position to relatch the valve. After locally resetting, valve control is restored to the Control Room.

3.1.9 The following precautions and limitations are related to the RHR System:

CD-847E
CD-695A

A. HV-F004B RHR PMP SUPP POOL SUCT MOV will drain the Reactor Vessel to the Suppression Pool if opened in shutdown cooling.

CD-847E
CD-695A

B. HV-F024B RHR LOOP TEST RET MOV will drain the Reactor Vessel to the Suppression Pool if opened in shutdown cooling.

CD-847E
CD-695A

C. HV-F007B B RHR PMP MIN FLOW MOV will drain the Reactor Vessel to the Suppression Pool if opened in shutdown cooling.

CD-847E
CD-695A

D. Opening HV-F007B to lower the reactor water level is only to be done when absolutely necessary. Opening this valve provides the potential for an uncontrolled drainage path from the Reactor to the Suppression Pool.

E. Opening HV-F009 SHUTDOWN COOLING INBD ISLN MOV may cause a decrease in Reactor water Level.

3.1.9. F. Do not allow discharge of water > 120°F to Liquid Radwaste System.(TI-4401, Disch to LRW-DISCH TEMP).

3.1.10 The following precautions and limitations are related to the RCIC System:

A. To prevent RCIC Turbine exhaust piping and check valve vibration problems, RCIC Turbine speed should be rapidly increased to ≥ 2150 RPM.

B. To prevent possible bearing damage, RCIC Turbine speed should be limited to ≥ 2150 RPM.

CD-012Z 3.1.11 If the rate of rise of the Reactor Pressure Vessel level indicates HPCI is injecting and the Control Room is unmanned, HPCI will have to be tripped using Attachment 8 when no longer required or prior to exceeding the high level trip (Level 8). The high level trip (Level 8) may not function in the event a fire occurs in the relay room.

CD-847E 3.1.12 When the RSP transfer switch is placed in EMER, the LOW REACTOR LEVEL/HIGH REACTOR PRESSURE isolation logic associated with shutdown cooling is bypassed.

A. High Reactor pressure will prevent opening of the shutdown cooling valves but will not isolate the valve if pressure rises above setpoint.

3.1.13 Operation of SRVs at low Reactor pressures (below 700 PSIG) may result in failure of an SRV to reclose when required.

3.1.14 During plant Cooldown/Depressurization, monitor similar Rx water level instrumentation for significant deviation, indicating possible reference line De-gasing. Also, terminate all maintenance activities which have the potential of draining the Rx vessel.

CD-473G 3.1.15 The HPCI Exhaust Diaphragms, if ruptured, relieve exhaust steam directly to the Torus Chamber area creating a severe personnel safety hazard. ENSURE all personnel are clear of the Torus Chamber Area prior to starting HPCI and during system operation, except as part of SA-AP.ZZ-0051(Q) walkdowns.

3.1.16 Contact Radiation Protection prior to performing venting and/or draining in this procedure. The individual(s) performing the venting and/or draining shall obtain instructions and approval from the RP Shift Technician or RP Supervisor.

3.2 Other

3.2.1 The precautions and limitations in the appropriate SOP's will be applicable when the SOP's are used in this procedure.

3.2.2 AP211(BP211) A(B) FUEL POOL COOLING PUMP(s) may trip due to channel transfers, Fuel Pool Cooling Demineralizers should be placed in hold as required.

4.0 EQUIPMENT REQUIRED

4.1 Sound powered phones

4.2 Radios

4.3 Keys for Security Doors and MCC Keylock Switches (located in key cabinet, Remote Shutdown Panel Room)

5.0 PROCEDURE

> **NOTE 5.0**
>
> **A.** **Initial each step upon completion of the step.**
>
> **B.** **Refer to Attachment 2 for RSP redundant instrumentation/controls.**
>
> **C.** **Refer to Attachment 3 for placing 'A' Loop RHR in Suppression Pool Cooling.**
>
> **D.** **Refer to Attachments 6 and 7 for plant communications information. When dispatching an operator to a remote shutdown control station, provide the operator with a sound powered phone or radio to assist with communication.**

Initials

5.1 Establish Control from Outside the Control Room

5.1.1 Ensure that all prerequisites have been satisfied IAW Section 2.0 of this procedure. ____

> **NOTE 5.1.2**
>
> **A.** **If the Reactor was not scrammed and the MSIVs are still open, the Feedwater System and the Main Turbine Bypass Valves may be regulating Rx level and Rx pressure at this time.**
>
> **B.** **Opening the circuit breakers listed in step 5.1.2 will deenergize the RPS busses, scramming the plant, and deenergize the NSSSS busses, closing the MSIVs.**
>
> **C.** **10C410(10C411) RPS PWR Dist. Panels A(B) are located in the Control/DG Bldg. El. 54'.**

Initials

CD-987X 5.1.2 If the Reactor could not be scrammed prior to Control Room evacuation OPEN the following circuit breakers

A. CB2A, CB3A, CB5A, CB7A and CB8A (RPS PWR DIST PNL A 10C410).

B. CB2B, CB3B, CB5B, CB7B and CB8B (RPS PWR DIST PNL B 10C411). ____

5.1.3 If the Rx scram was not verified prior to evacuating the Control Room, send an operator to each HCU to check local nitrogen side pressure indicators for a low (< 800 psig) pressure indication verifying the Reactor Scram. ____

5.1.4 Notify Chemistry to verify that the Hydrogen/Oxygen System has tripped IAW HC.CH-SO.AX-0001(Q). ____

5.1.5 Upon arriving at the RSP, monitor the RSP system indications and check specifically for the following: ____

A. REACTOR VESSEL PRESSURE PR-7853D (between 905 psig and 1045 psig) ____

CD-012Z

CAUTION 5.1.5.B

If the rate of rise of the Reactor Pressure Vessel level indicates HPCI is injecting and the Control Room is unmanned, HPCI will have to be tripped using Attachment 8 when no longer required or prior to exceeding the high level trip (Level 8). The high level trip (Level 8) may not function in the event a fire occurs in the Relay Room.

5.1.5. B. REACTOR VESSEL LEVEL LR-7854 (between 12.5" and 54") ____

C. RCIC System status (standby or auto-initiated) ____

D. PSV-F013F,H,M SRV status (standby or cycling open/closed) ____

E. SUPPRESSION CHAMBER WATER TR-3647J (and M) (average less then 95°F) ____

Initials

5.1.5. F. DIESEL GENERATOR 1A(B,C,D)G400 TRIP/CLOSED Status (closed if a loss of offsite power has occurred). ____

5.1.6 If a loss of offsite power has occurred, send an operator to the Diesel Generator Remote Control Panel (Aux. Bldg El. 130') to monitor Diesel Generator operation and implement HC.OP-AB.ZZ-0135(Q), Loss of Offsite Power, concurrent with this procedure. ____

CD-462Y 5.1.7 PLACE the following RSP switches to EMER:

A. CH "A" TRANSFER ____

B. CH "B" TRANSFER ____

C. CH "C" TRANSFER ____

D. CH "D" TRANSFER ____

E. CHANNEL "NON-1E" TRANSFER ____

F. PMP BP202 XFR - B RHR PUMP ____

G. PMP BP502 XFR - B SERV WTR PUMP ____

H. PMP DP502 XFR - D SERV WTR PUMP ____

I. PMP BP210 XFR - B SACS PUMP ____

J. PMP DP210 XFR - D SACS PUMP ____

K. MOTOR BK400 TRANS ____

L. MOTOR BK403 TRANS ____

5.1.8 Ensure appropriate automatic actions have occurred on the transfer IAW Attachment 1. ____

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HOPE CREEK GENERATING STATION

HC.OP-IO.ZZ-0008(Q) - Rev. 11

SHUTDOWN FROM OUTSIDE CONTROL ROOM

USE CATEGORY: I

REVISION SUMMARY:
REV. 11

1. Added initials blocks to Sections 2 and 3 and Attachments 3, 4, 5 and 8.

2. Changed substeps in 4.1.10 of Attachment 4 to bulleted steps.

3. Added page numbers specific to that attachment to each attachment (also to TOC)

4. Changed Step 8.2 in Attachment 8 to Note 8.1.

5. Modified Step 4.1.7.E to notify Radwaste of impending receipt of water at 200°F.

6. Edited Attachment 6 to improve ease of reading; rev bars not used.

5. Deleted BC-HV-F011B, F026B, F052B and HV-4428 from step B.2.1 of Attachment 1 since these valves are permanently out of service, with power removed.

6. Corrected Valve number from HV-F022B to HV-F022 in B.2.2.

7. Incorporated revision request **OP 97-1600 [PR 970630279, PR 970702247]** to correct the valve number from 1EG-HV-2496B TO 1EG-HV-2491A in Attachment 3, Step 3.3.4.

8. Incorporated revision request **OP 96-1601** to add clarity to the operation of Attachment 3, Steps 3.5.1, 3.5.2, 3.5.3, 3.5.6, and 3.5.7. Specific direction is given to operate the key switch at the respective breaker vice the breaker itself to conduct valve manipulations.
Also addresses portion of **OP 95-0352 [PR 960723097]**

(Continued)

IMPLEMENTATION REQUIREMENTS

Effective Date 10/2/97

APPROVED: ____________________ 10/1/97
Operations Manager **Date**

REVISION SUMMARY: **CONTINUATION SHEET**

REV. 11

9. Incorporated revision request **OP 96-1614** and **OP 97-0083 [PR 961122266]** by adding the following:

   - Removed Prerequisites 2.2 which stated Establish communications between each local panel or breaker and the Remote Shutdown Panel as the steps are performed. The content of this prerequisite is captured in Note 5.0.D.

   - Changed the discharge temperature limit for liquids going to liquid radwaste in Precaution 3.1.9 and Steps 5.10.2, 5.10.2.A and 5.11.2.B and Caution 5.11.2 from 120 degrees to 200 degrees. This coincides with HC.OP-SO.BC-0001(Q).
     Also addresses portion of **OP 95-0352, [PR 960723097]**

   - Changed the maximum SACS temperature from 100 degrees to 95 degrees in step 5.9.3. This coincides with HC.OP-SO.EG-0001(Q), Step 3.2.8.

10. Incorporated revision request **OP 96-0057** to add the noun names of the valves operated in Attachment 1, Section D and to change Section D.2.7 from HV-2494D to HV-2494B and from HV-2355D to HV-2355B for proper valve identification.

11. Incorporated reviewer comments **[PR 970514283, PR 970702247]** to include the following changes:

   - Added Note 5.1.8 stating.. If running, chillers BK400 and BK403 will trip when their respective transfer switches are taken to emergency.. to note expected system response.

   - Added Step 5.12.1.D stating.. If RCIC is running from the RSP, ensure the RCIC flow controller in the MCR is matched with actual RCIC flow.. to ensure continuity of monitored readings prior to shifting control from the RSP to the MCR.

   - Changed breaker number 52-40104 to 52-40304 in Attachment 3, Step 3.4.2 to indicate the correct breaker associated with "C" SACS pump.

   - Added a provision to reset PCIS in step 5.12.1.C.

   - Deleted Attachment 8, Step 8.1.2 open circuit breaker 26, HPCI INVERTER VERTICAL BOARD 10C620 because it is a spare.

(Continued)

SHUTDOWN FROM OUTSIDE CONTROL ROOM

USE CATEGORY: I

REVISION SUMMARY: **CONTINUATION SHEET**

REV. 11

12. Incorporated revision request **OP 95-0352 [PR 960723097]** to change the following:

- Deleted Step 5.8.2.D so that the procedure would leave HV-F007B in the open position which is its normal position.
- See revison summary item 8.
- See revision summary item 9, second bullet.
- Revised direction on Attachment 3 at step 3.5.6, and 3.5.7 (previously Note and step 3.5.5) by: converting the note into a step; clarify "hook up' statement in note to mean hooking up of sound powered phone equipment, and added a reference to Attachment 7; clarified instrument to be used in lower relay room; and converted the stated indication to achieve, to one commensurate with the range of the instrument (previously the step directed operator to achieve 83.3% indication, yet the instrument range is from 0 to 30%. The new value listed is ≈ 83.3% of 30 (25%).

13. Based on Department reviewer comments, the following changes were made:

- Removed "Control Room", and added "RSP" to Note 3.1.8.D
- Split precaution 3.1.13 into a note and a step.
- Deleted Precaution 3.1.9.C. This precaution was adequately addressed in 3.1.9.D.
- Added Letters (A, B, C, D) to Note 5.0 to differentiate statements.
- Removed implied interlock from 5.2.1.G for an auto swap of RCIC suctions on a high suppression chamber level.
- Corrected nomenclature of instrument E11-N652A to match field labeling, and as refered to in Attachments 2, 3, and 7.
- Corrected title of Attachment 6 as refered to in the table of contents.
- Added "Local" to step 5.3.1.F.
- Added steps 5.5.1.C and 5.5.1.E to verify repositioned HX Inlet Valves. This action is similar to that found in SOP for placing system in service.
- Changed "CLOSE" to "REPOSITION" and added "as necessary" in step 5.8.2.E.
- Added steps 5.4.1.D and 5.4.1.J to open the HV-2197 Backwash Valves. This action is similar to that found in SOP for placing system in service.
- Added direction to ensure the HV-F003B B RHR HX OUTLET MOV is open in step 5.8.1.G.
- Added new direction at step 5.11.3 for utilizing the A RHR for Torus Level reduction if B RHR is in SDC.

(Continued)

ALL ACTIVE ON-THE-SPOT CHANGES MUST BE ATTACHED FOR FIELD USE
20020402

REVISION SUMMARY: **CONTINUATION SHEET**

REV. 11

13. Based on Department reviewer comments, the following changes were made:

- Changed "HPCI" to "RCIC" in precaution 3.1.16.
- Modified step 4.1.8 of Attachment 4 to match direction for operation of HV-F015 found in RHR SOP.

14. Incorporates revision request **OP-97-0443 (BP 970811169)** by adding Precaution 3.1.12 and step 5.1.7 to warn of the potential effect a fire in the relay room would have on BJ-HV-F008.

15. Incorporates revision request **BP 970908166** by adding precaution 3.1.18 and NOTE 5.8 to warn of a possible water hammer condition if the HV-F024A(B) and/or HV-F027A(B) are open and the associated RHR pump is stopped.

Initials

SHUTDOWN FROM OUTSIDE CONTROL ROOM

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SHUTDOWN FROM OUTSIDE CONTROL ROOM

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Initials

SHUTDOWN FROM OUTSIDE CONTROL ROOM

START TIME ______________ DATE ______________ BY ______________

TERMINATION TIME ______________ DATE ______________ BY ______________

COMPLETION TIME ______________ DATE ______________ BY ______________

1.0 PURPOSE

This procedure provides guidelines for the shutdown of the plant from outside the Control Room, AND for re-establishing control in the Control Room.

2.0 PREREQUISITES

2.1 HC.OP-AB.ZZ-0130(Q), Control Room Evacuation, complete if possible.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Administrative

3.1.1 This procedure is to be used as a guideline for shutdown of the plant from outside the Control Room. It is NOT required that each section/step be performed in precise sequence as long as the sections/steps are performed in a timely manner, in keeping with the intent of this procedure. ______

3.1.2 In the event plant conditions require a delay during performance of this procedure, the Senior Nuclear Shift Supervisor/Nuclear Shift Supervisor (SNSS/NSS) should retain this procedure until it is continued OR terminated. ______

3.1.3 IF it is terminated prior to completion, the SNSS/NSS should note the reason, time, AND date of termination on this procedure. ______

3.1.4 For any unit scram, the Event Classification Guide should be referred to for the appropriate classification AND notifications. ______

3.1.5 Reactor Coolant System temperature AND pressure requirements of T/S 3.4.6.1 shall be complied with. ______

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3.1.6 The Reactor Vessel AND Head Flange temperature limits of T/S 3.4.6.1.d shall be complied with. ____

3.1.7 The Suppression Chamber temperature requirements of T/S 3.6.2.1 shall be complied with. ____

3.1.8 When RSP Transfer Switches are placed in EMER, all trips and auto starts associated with the following equipment are bypassed: **[CD-904B, CD-695A]** ____

A. SACS PUMPS B and D ____

B. SSWS PUMPS B and D ____

C. RHR PUMP B ____

> **NOTE** 3.1.8.D
>
> The RCIC backup mechanical overspeed trip of 125% rated speed is still provided. This turbine trip will close the trip and throttle valve (HV-4282). This overspeed trip must be locally reset to allow relatching of the Turbine Trip Throttle valve. The limitorque must be manually run to the full closed position to relatch the valve. After locally resetting, valve control is restored to the RSP.

D. RCIC system ____

3.1.9 The following are related to the RHR System: ____

A. IF opened during Shutdown Cooling operations, THEN HV-F004B RHR PMP SUPP POOL SUCT MOV will drain the Reactor Vessel to the Suppression Pool **[CD-847E, CD-695A]** ____

B. IF opened during Shutdown Cooling operations, THEN HV-F024B RHR LOOP TEST RET MOV will drain the Reactor Vessel to the Suppression Pool. **[CD-847E, CD-695A]** ____

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3.1.9 (Continued)

C. Opening HV-F007B to lower the reactor water level is to be done ONLY when absolutely necessary. Opening this valve provides the potential for an uncontrolled drainage path from the Reactor to the Suppression Pool. **[CD-847E, CD-695A]** ____

D. Opening HV-F009 SHUTDOWN COOLING INBD ISLN MOV may cause a decrease in Reactor water Level. ____

E. Water >200°F should NOT be discharged to the Liquid Radwaste System (TI-4401, Disch to LRW-DISCH TEMP). ____

3.1.10 The following precautions AND limitations are related to the RCIC System: ____

A. To prevent RCIC Turbine exhaust piping AND check valve vibration problems, RCIC Turbine speed should be rapidly increased to $\geq$ 2150 rpm. ____

B. To prevent possible bearing damage, RCIC Turbine speed should be limited to $\geq$ 2150 rpm. ____

3.1.11 IF the rate of rise of RPV level indicates HPCI is injecting AND the Control Room is unmanned, THEN HPCI will have to be tripped using Attachment 8 when no longer required OR prior to exceeding high level trip (Level 8). High level trip may NOT function in the event a fire occurs in the relay room. **[CD-012Z]** ____

3.1.12 A fire in the relay room can cause BJ-HV-F008, HPCI TEST BYPASS TO CST to spuriously open. If HPCI suction is aligned to the suppression pool, this valve must be closed from it's MCC - 10D251103. ____

NOTE 3.1.13

High Reactor pressure will prevent opening of the shutdown cooling valves but will NOT isolate the valve IF pressure rises above setpoint.

3.1.13 WHEN the RSP transfer switch is placed in EMER, THEN the LOW REACTOR LEVEL/HIGH REACTOR PRESSURE isolation logic associated with shutdown cooling is bypassed. **[CD-847E]** ____

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3.1.14 Operation of SRVs at low Reactor pressures (below 700 PSIG) may result in failure of SRV(s) to reclose when required. ______

3.1.15 During plant Cooldown/Depressurization, similar Rx water level instrumentation should be monitored for significant deviation, indicating possible reference line degassing.
In addition, all maintenance activities having the potential for draining the Rx vessel should be terminated. ______

3.1.16 IF ruptured, the RCIC Exhaust Diaphragms relieve exhaust steam directly to the Torus Chamber area, creating a severe personnel safety hazard.
ENSURE that all personnel are clear of the Torus Chamber Area PRIOR to starting RCIC AND during system operation, except as part of SA-AP.ZZ-0051(Q) walkdowns. **[CD-473G]** ______

3.1.17 Radiation Protection should be contacted prior to performing venting AND/OR draining. The individual(s) performing the venting AND/OR draining should obtain instructions AND approval from the RP Shift Technician or RP Supervisor. ______

3.1.18 IF at any time a situation develops whereby HV-F024A(B) and/or HV-F027A(B) are open with the associated RHR pump not in operation, a potential system drain down will occur. A subsequent start of the RHR pump following this situation will cause water hammer.
Therefore, IF the valves are open
AND the pump either trips or is not running
THEN **ENSURE** both valves are closed,
AND **PERFORM** a system fill & vent
PRIOR to starting the RHR pump. **[970908166]** ______

3.2 Other

3.2.1 The precautions AND limitations in the appropriate SOPs will be applicable when the SOPs are used in this procedure. ______

3.2.2 AP211(BP211) A(B) FUEL POOL COOLING PUMP(s) may trip due to channel transfers. Fuel Pool Cooling Demineralizers should be placed in hold, as required. ______

4.0 EQUIPMENT REQUIRED

- Sound powered phones
- Radios
- Keys for Security Doors AND MCC Keylock Switches (located in key cabinet, Remote Shutdown Panel Room)

Initials

5.0 **PROCEDURE**

NOTE 5.0

A. Each step should be initialed upon completion of the step.

B. Attachment 2 should be referred to for RSP redundant instrumentation/ controls.

C. Attachment 3 should be referred to for placing 'A' Loop RHR in Suppression Pool Cooling.

D. Attachments 6 and 7 should be referred to for plant communications information. When dispatching an operator to a remote shutdown control station, the operator should be provided with a sound-powered phone OR radio to assist with communication.

5.1 **Establish Control from Outside the Control Room**

5.1.1 **ENSURE** that all prerequisites have been satisfied IAW Section 2.0 of this procedure. ______

NOTE 5.1.2

A. If the Reactor was NOT scrammed AND the MSIVs are still open, then the Feedwater System AND the Main Turbine Bypass Valves may be regulating Rx level AND Rx pressure at this time.

B. Opening the circuit breakers listed in step 5.1.2 will deenergize the RPS busses, scramming the plant, AND deenergize the NSSSS busses, closing the MSIVs.

C. 10C410(10C411) RPS PWR Dist. Panels A(B) are located in Control/DG Bldg. El. 54'.

5.1.2 IF the Reactor was NOT scrammed prior to Control Room evacuation, THEN **OPEN** the following circuit breakers: **[CD-987X]** ______

A. CB2A, CB3A, CB5A, CB7A AND CB8A (RPS PWR DIST PNL A 10C410). ______

B. CB2B, CB3B, CB5B, CB7B AND CB8B (RPS PWR DIST PNL B 10C411). ______

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5.1.3 IF the Rx scram was NOT verified prior to evacuating the Control Room, THEN **VERIFY** Rods Full In. (SPDS/CRIDS (TSC) OR RMCS Activity Control Cards OR Other). ____

5.1.4 **NOTIFY** Chemistry to verify that the Hydrogen/Oxygen System has tripped IAW HC.CH-SO.AX-0001(Q). ____

5.1.5 Upon arriving at the RSP, **MONITOR** the RSP system indications AND **CHECK** specifically for the following: ____

A. REACTOR VESSEL PRESSURE PR-7853D (905 - 1045 psig) ____

CAUTION 5.1.5.B

IF the rate of rise of RPV level indicates HPCI is injecting AND the Control Room is unmanned, THEN HPCI will have to be tripped using Attachment 8 when no longer required OR prior to exceeding the high level trip (Level 8). The high level trip may NOT function in the event a fire occurs in the relay room. [CD-012Z]

B. REACTOR VESSEL LEVEL LR-7854 (12.5 - 54 ") ____

C. RCIC System status (standby OR auto-initiated) ____

D. PSV-F013F,H,M SRV status (standby OR cycling open/closed) ____

E. SUPPRESSION CHAMBER WATER TR-3647J (AND M) (average less than 95°F) ____

F. DIESEL GENERATOR 1A(B,C,D)G400 TRIP/CLOSED Status (closed IF a loss of offsite power has occurred). ____

5.1.6 IF a loss of offsite power has occurred, THEN **SEND** an operator to the Diesel Generator Remote Control Panel (Aux. Bldg El. 130') to monitor Diesel Generator operation, AND **IMPLEMENT** HC.OP-AB.ZZ-0135(Q), Loss of Offsite Power, concurrent with this procedure. ____

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PROCEDURE COMMENT FORM

TO: PROCEDURE WRITER Dept: HC OPS CC: ______ Comment Due Date: ______

FROM: ______ Dept: ______

DOCUMENT NO., REVISION, AND TITLE:
HC.OP-IO.ZZ-0008(Q), Rev. 11
SHUTDOWN FROM OUTSIDE CONTROL ROOM

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Page/Para Number	Comments or Recommendations	Accepted Yes	Accepted No	Comment Disposition (A negative disposition requires justification)
Step [illegible] 5.[illegible].3	This step is inadequate for its purpose due to nature of system (RCS) pressure indicated at HCU [illegible] may be as high or higher than [illegible] pressure depending on condition of individual HCU [illegible] new step - VERIFY RODS FULL IN FROM SPDS/ CRIDS FROM TSC OR RMCS ACTIVITY CONTROL CHANS 1,2 THE RODS NOT FULL IN LEDS EXTINGUISHED NOTE FOR ROT. CONTROL ROD INDICATION ALL RODS MUST BE FULL IN. IF ROD [illegible] TO [illegible] LEDS WILL BE LITE INDICATING ALL RODS NOT FULL IN.	✓		Added Sources parenthetically following action. Also allowed "Other" to Sources.

REVIEWER: [signature] DATE: [illegible] 4 97

DISPOSITIONED BY: D. [signature] DATE: 9.10.97