QUESTION 10

Answer choice C states "Raise level to +80 inches using natural circulation for heat removal."

Natural circulation removes decay heat from the fuel bundles in the core to the bulk coolant. RWCU, C, RHR, D RHR, or Condensate Transfer can be used to remove decay heat from the bulk coolant to the Main Condenser, RACS, or SACS.

Abnormal procedure HC.OP-AB.RPV-0009 Condition E action step E.2 states "Maintain RPV Level ≥ 80 inches BUT ≤ 90 inches." This step is performed if Forced Circulation cannot be established using preferred RHR loops (A or B) or Reactor Recirculation. This step is performed when RWCU, C RHR, D RHR, or Condensate Transfer is required for Alternate Decay Heat removal. The conditions of the stem require Alternated Decay Heat removal methods.

Condition E action E.5 states "Evaluate the following systems for alternate decay heat removal:

- RWCU (Subsequent F)
- C RHR (Attachment 1)
- D RHR (Attachment 2)
- Condensate Transfer (Subsequent G)"

The stem does not provide core exposure history other than shutdown 16 hours ago. The student could assume Beginning of Core Life, End of Life or anywhere in between.

Since the stem does not rule out RWCU or Condensate Transfer operation, RWCU can be assumed in service and can be used in conjunction with natural circulation once level has been raised to 80 - 90 inches. The heat removal means is natural circulation removing heat from the fuel bundles to the bulk reactor coolant then to RACS and the Main Condenser. Under normal operation with RWCU rejecting 69 gpm from CRD injection, with some heat removed through RACS and some removed by replacement water from CRD. Based on stem conditions, RWCU is required for Alternate Decay Heat Removal. RWCU is realigned IAW Subsequent Action step F which opens the cooling water supply valve ED-V035 full open and bypasses the Non Regenerative Heat Exchanger.

Heat removal capability in Alternate Decay Heat Removal Mode is approximately 15 to 16 Million BTU's per hour. (per System Engineering B. Down). Reactor decay heat load at the Beginning of Life (BOL) during the initial startup from a typical 30 day refueling outage is approximately 13 Million BTUs per hour and rises with full power operation history. If the scram was assumed from the startup from refueling outage before the reactor had any significant full power operation, the decay heat load 16 hours after the scram would be well within RWCU Alternate Decay Heat Removal capability. Therefore, answer choice C would also be correct.

The students are not required to know the value of BTU's per hour removal rate, or the BTU generation rated of the core at a particular time of core life. From a procedure user point of view, answer choice C is also correct when applied to Subsequent Action step E.

Recommended action is to accept answer choices A or C as correct answers.

Given the following conditions:

- The plant is in Operational Condition 4 following a forced shutdown 16 hours ago.
- RHR Loop "A" operating in Shutdown Cooling.
- The "B" RHR pump is Cleared & Tagged for motor replacement.
- The "A" RHR pump develops a high vibration and trips on overcurrent.
- HC.OP-AB.RPV-0009, Shutdown Cooling, is entered.

Which of the following will be adequate to maintain Operational Condition 4?

a. Crosstie "C" or "D" RHR pump for heat removal.

b. Maximize RWCU bottom head drain flow.

c. Raise level to +80 inches using natural circulation for heat removal.

d. Inject with Core Spray from the CST to the RPV.

Answer: a | Exam Level: B | Cognitive Level: Application | Facility: Hope Creek | Exam Date: 02/24/2003

Tier: Emergency and Abnormal Plant Evolutions | RO Group: 3 | SRO Group: 2 | 295021A104

295021 Loss of Shutdown Cooling

AA1. Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING:

AA1.04 Alternate heat removal methods 3.7 3.7

Explanation of Answer: Justification
Crosstie "C" or "D" RHR pump for heat removal. correct - RPV-0009 subsequent action E, RHR Pumps C & D may be realigned to provide alternative decay heat removal.
Maximize RWCU bottom head drain flow. -incorrect- Maximizing bottom head drain flow does not provide heat removal adequate to maintain less than 200 degrees.
Raise level to +80 inches using natural circulation for heat removal. -Incorrect- per subsequent action E Natural Circulation does not provide heat removal, only circulation.
Inject with Core Spray from the CST's to the RPV. -incorrect- This is not an approved method of Alternate DHR.

Reference Title
HC.OP-AB.RPV-0009

Learning Objectives	
ABRPV9E007	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Shutdown Cooling.

Material Required for Examination: None

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

Question Source Comments: VISION Bank QID# Q61332 Sig Mod

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
E. Forced Circulation CANNOT be established using preferred RHR loops or Reactor Recirculation. **[CD-693A, CD-178A, CD-973B, CD-100A, CD-076B, CD-065X]** Time: _____	❑ **** NOTE 1**** ❑ E.1 **MONITOR** temperatures IAW DL-0026 Attachment 3s. ❑ ✯ **CAUTION 1** ✯ ❑ E.2 **MAINTAIN** RPV LVL ≥ 80 inches, BUT < 90 inches. ❑ E.3 IF RPV LVL reaches 90 inches, THEN **CLOSE** the MSIV's. ❑ E.4 **ENSURE** T.S. cool down limits are not exceeded. **[T/S 3.4.6.1.b]** E.5 **EVALUATE** the following systems for alternate decay heat removal: ❑ • RWCU (Subsequent F) **[CD-900E]** ❑ • "C" RHR (Attachment 1) ❑ • "D" RHR (Attachment 2) ❑ • CONDENSATE TRANSFER (Subsequent G) ❑ E.6 IF the vessel head is removed, AND the Reactor Cavity is flooded, THEN maximize Fuel Pool Cooling: ❑ • **ENSURE** two Fuel Pool Cooling pumps are in service. (EC) ❑ • **ENSURE** SACS flow aligned through BOTH Fuel Pool Cooling heat exchangers.
F. RWCU is required for Alternate Decay Heat Removal. **[CD-900E]** Time: _____	❑ F.1 **ENSURE** RWCU is in service. (BG) ❑ F.2 **FULLY OPEN** ED-V035. ❑ F.3 IF necessary, THEN Bypass the Regenerative heat exchanger to maximize decay heat removal. (BG)

4. You may bring pens, pencils, and calculators into the examination room. Use black ink to ensure legible copies; dark pencil should be used only if necessary to facilitate machine grading.

5. Print your name in the blank provided on the examination cover sheet and the answer sheet. You may be asked to provide the examiner with some form of positive identification.

6. Mark your answers on the answer sheet provided and do not leave any question blank. Use only the paper provided and do not write on the back side of the pages. If you are using ink and decide to change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

7. If you have any questions concerning the intent or the initial conditions of a question, do *not* hesitate asking them before answering the question. Ask questions of the NRC examiner or the designated facility instructor *only*. When answering a question, do *not* make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question. For example, you should not assume that any alarm has activated unless the question so states or the alarm is expected to activate as a result of the conditions that are stated in the question. Finally, answer all questions based on actual plant operation, procedures, and references. If you believe that the answer would be different based on simulator operation or training references, you should answer the question based on the *actual plant*.

8. Restroom trips are permitted, but only one applicant at a time will be allowed to leave. Avoid all contact with anyone outside the examination room to eliminate even the appearance or possibility of cheating.

9. When you complete the examination, assemble a package including the examination questions, examination aids, answer sheets, and scrap paper and give it to the NRC examiner or proctor. Remember to sign the statement on the examination cover sheet indicating that the work is your own and that you have neither given nor received assistance in completing the examination. The scrap paper will be disposed of immediately after the examination.

10. After you have turned in your examination, leave the examination area as defined by the proctor or NRC examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

11. Do you have any questions?

PART C - GENERIC OPERATING TEST GUIDELINES (CATEGORIES A, B, AND C)

1. If you are asked a question or directed to perform a task that is unclear, you should not hesitate to ask for clarification.

2. The examiner will take notes throughout the test to document your performance, and sometimes the examiner may take a short break for this reason. The amount of note-

ATTACHMENT 2

LICENSEE COMMENTS

Written Question 10

Answer choice C states "Raise level to +80 inches using natural circulation for heat removal."

Natural circulation removes decay heat from the fuel bundles in the core to the bulk coolant. RWCU, C RHR, D RHR, or Condensate Transfer can be used to remove decay heat from the bulk coolant to the Main Condenser, RACS, or SACS.

Abnormal procedure HC.OP-AB.RPV-0009 Condition E, action step E.2 states "Maintain RPV level greater than or equal to 80 inches but less than or equal to 90 inches." This step is performed if forced circulation cannot be established using preferred RHR loops (A or B) or reactor recirculation. This step is performed when RWCU, C RHR, D RHR, or condensate transfer is required for alternate decay heat removal. The conditions of the stem require alternate decay heat removal methods to be used.

Condition E, action E.5 states "Evaluate the following systems for alternate decay heat removal:

- RWCU (subsequent F)
- C RHR (Attachment 1)
- D RHR (Attachment 2)
- Condensate Transfer (Subsequent G)"

The stem does not provide core exposure history other than shutdown 16 hours ago. The student could assume Beginning of Core Life, End of Core Life or anywhere in between.

Since the stem does not rule out RWCU or condensate transfer operation, RWCU can be assumed in service and can be used in conjunction with natural circulation once reactor level has been raised to 80 - 90 inches. The heat removal means is natural circulation removing heat from the fuel bundles to the bulk reactor coolant, then to RACS and the main condenser. Under normal operation with RWCU rejecting 69 gpm from CRD injection, with some heat removed through RACS and some removed by replacement water from CRD. Based on stem conditions, RWCU is required for alternate decay heat removal. RWCU is realigned in accordance with subsequent action step F, which opens the cooling water supply valve ED-V035 full open and bypasses the non regenerative heat exchanger.

Heat removal capability in alternate decay heat removal mode is approximately 15 to 16 Million BTU's per hour. (Per System Engineering B. Down). Reactor decay heat load at the Beginning of Life (BOL) during the initial startup from a typical 30 day refueling outage is approximately 13 Million BTUs per hour and rises with full power operation history. If the reactor automatic shutdown was assumed to occur during a startup from a refueling outage, before the reactor had any significant full power operation, the decay heat load 16 hours after the automatic shutdown would be well within RWCU Alternate Decay Heat Removal capability. Therefore, answer choice C would also be correct.

The students are not required to know the value of BTU's per hour removal rate, or the BTU generation rate of the core at a particular time of core life. From a procedure user point of view, answer choice C is also correct when applied to Subsequent Action step E.

Recommended action is to accept answer choices A or C as correct answers.

ATTACHMENT 3

NRC RESOLUTION OF LICENSEE COMMENTS

Written Question: 10

Comment: The question provides a condition in which the plant has been shutdown for 16 hours when residual heat removal (RHR) shut down cooling is lost. The applicant must determine which condition will be adequate to maintain the plant in operational condition 4, in accordance with HC.OP-AB.RPV-0009, "Shutdown Cooling." The correct answer was (a) to crosstie "C" or "D" RHR pump for core decay heat removal. Answer (c) Raise level to + 80 inches using natural circulation for heat removal was recommended to also be accepted as a correct answer. The basis for accepting (c) is that the reactor water clean up system would be in service with a normal line up and rejecting water to the condenser at 69 gpm following a reactor shutdown. If the plant has just started up following a 30 day refueling outage then the decay heat load would be approximately 13 million BTU per hour. The RWCU heat exchanger, in the alternate decay heat removal lineup will have a capacity of 15 to 16 million BTUs per hour. Therefore, the plant will be able to be maintained in operational condition 4, after increasing level to + 80 inches <u>and</u> reconfiguring the RWCU system to the alternate decay heat removal mode of operation in accordance with the procedure.

NRC Resolution: Based our review the only correct answer is (a) "crosstie "C" or "D" RHR pump for heat removal." This is based on the stem of the question which states that "Which of the following will be <u>adequate</u> to maintain Operational Condition 4." Answer (a) is the only answer that is sufficient to have enough heat removal capacity to maintain the plant in operational condition 4. Answer (c) states raise level to +80 inches using natural circulation for heat removal. This action may initially keep the plant in operational condition 4, but will not be able to maintain the plant in operational condition 4 for long term. Raising level to + 80 inches only enhance heat transfer from the fuel to the coolant, but it will not remove the decay heat from the vessel. Another system(s) will be required together with raising level to remove the decay heat from the vessel. Therefore, (a) is the only correct answer.

ADAMS DOCUMENT COVER SHEET

DOCUMENT TITLE: NRC Letter Transmitting Certificates

ESTIMATED PAGE COUNT: 0
AVAILABILITY: PUBLICALLY AVAILABLE
KEY WORD: NRR-079
DOCUMENT SENSITIVITY: Non-Sensitive
SECURITY: R1-DRS-BS..... Access Level: Owner
NRC Users..... Access Level: Viewer
R1-DRS-OSB.....Access Level: Viewer
DPC..... Access Level: Owner
CKB.....Access Level: Owner

Accession No:_____

The only individual that passed all portions of the exam will receive lic. after he completes his 6 months of onsite experience.

ADAMS DOCUMENT COVER SHEET

DOCUMENT TITLE: Misc. Documents - Contains Personal Information.

ESTIMATED PAGE COUNT: _____

AVAILABILITY: NON-PUBLICALLY AVAILABLE ~~"OR" PUBLICALLY AVAILABLE~~

KEY WORD: NRR-079

DOCUMENT SENSITIVITY: ~~Non-Sensitive "OR"~~ Sensitive

SECURITY:

R1-DRS-BS	Access Level:	Owner
NRC Users	Access Level:	Viewer
R1-DRS-OSB	Access Level:	Viewer
NRR-DIPM-IOLB	Access Level:	Viewer
DPC	Access Level:	Owner
CKB	Access Level:	Owner

Accession No: _____

1. ROI 03-01
2. Exam Item Review
3. Seating Chart
4. Questions during LSRO Written Exam
5. ES-501-1 Post Exam Checklist.

Master File

6/2000	**Operator Licensing Report on Interaction (ROI)**	Interaction No.: 03-01

Subject:	Deferral of Six Months of Site Specific Experience Until After Completion of the LSRO Exam.		
Type of Action:	Waiver: X (deferral) Policy Interpretation:	Request for HQ Action:X	
From:	R. Conte, Chief Operational Safety Branch R1/DRS/OSB	Date:	1/23/03
To:	D. Trimble, Chief, IOHS	Proposed Due Date:	2/24/03
Info.:	ADAMS PKG: ML022680541 ROI Accession # ML030580309		

Background / Issue:

Mr. Terry Beltz is an applicant for the Hope Creek LSRO license exam in March 2003. Prior to taking the exam, Mr. Beltz will not meet the required six months of site specific (Hope Creek) experience. Specifically, PSEG Nuclear, LLC requests that all six months of the site specific experience be deferred until after the LSRO examination.

Recommended Action / Resolution:

Region I recommends that Mr. Beltz be allowed to take the exam in March 2003, and defer the required six months of site specific experience until after the LSRO exam. The license will be issued after successful completion of the LSRO exam, completion of the required six months of site specific (Hope Creek) experience and PSEG has certified, in writing, that he completed the six months of site specific training.

Final Action / Resolution:

The request to defer the six months on-site experience requirement is approved. Complete the licensing action per Section D.3.c of ES-501 of NUREG-1021 after Hope Creek provides documentary evidence that the applicant has completed at least six months of "responsible nuclear power plant experience" on site.

File Subject(s):	10CFR55: NUREG-1021: Specify Other:		
Distribution:	OLBCs, ROI logbook	Post on Web: Yes or No?	
Signatures / Concurrences			
Originator:	R. Conte, Chief, Operational Safety Branch R1/DRS/OSB */RA/*	Date: 1/30/03	
OGC:	N/A (if needed)	Date:	
IOHS CH:	D. C. Trimble, Chief, IOHS/IEHB */RA/*	Date:	02/28/03
IOLB CH:	T. Quay, Chief, IEHB (If needed)	Date:	
Distribution Completed by IOLB Secretary (Initials): CCT		Date:	3/6/03

Course Name: LSRO NRC Written Examination

SAP Event #: NOH06NRCEX SAP Event #: 50763913

DATE EXAM ADMINISTERED: 04/07/03 LESSON PLAN NO.: N/A

NUMBER OF TRAINEES EXAMINED: 3

ADMINISTERED BY: A. Faulkner

LIST EACH EXAM ITEM THAT HAS AN AVERAGE SCORE LESS THAN 70%

ITEM NO.	AVERAGE SCORE %	REVIEW CONCLUSION	CORRECTIVE ACTION
2	33.3	G	Question acceptable
3	33.3	G	Question acceptable
4	33.3	G	Question acceptable
5	0.0	G	Question acceptable
10	66.7	F	Accept A or C See attached explanation.
12	66.7	G	Question acceptable
15	66.7	G	Question acceptable
16	66.7	G	Question acceptable
25	66.7	G	Question acceptable
26	66.7	G	Question acceptable
40	66.7	G	Question acceptable
42	66.7	G	Question acceptable
44	66.7	G	Question acceptable
45	66.7	G	Question acceptable
47	66.7	G	Question acceptable
49	33.3	G	Question acceptable
50	33.3	G	Question acceptable

REVIEW CONCLUSIONS:

A. Insufficient training for the objective tested.
B. Objective not adequately covered in the lesson plan.
C. Poorly worded or invalid enabling objective.
D. Poorly worded or invalid test item or answer.
E. Incorrect answer in the exam key.
F. More than one correct answer.
G. Question acceptable.
H. Other:

REVIEW PERFORMED BY: _____ DATE: 4/8/03

CHANGE INITIATED BY: _____ DATE: 4/8/03

TRAINING SUPERVISOR: _____ DATE: 4/8/03

EXAMINEES NOTIFIED OF RESULTS BY: N/A DATE: N/A

QUESTION 10

Answer choice C states "Raise level to +80 inches using natural circulation for heat removal."

Natural circulation removes decay heat from the fuel bundles in the core to the bulk coolant. RWCU, C, RHR, D RHR, or Condensate Transfer can be used to remove decay heat from the bulk coolant to the Main Condenser, RACS, or SACS.

Abnormal procedure HC.OP-AB.RPV-0009 Condition E action step E.2 states "Maintain RPV Level $\geq$ 80 inches BUT $\leq$ 90 inches." This step is performed if Forced Circulation cannot be established using preferred RHR loops (A or B) or Reactor Recirculation. This step is performed when RWCU, C RHR, D RHR, or Condensate Transfer is required for Alternate Decay Heat removal. The conditions of the stem require Alternated Decay Heat removal methods.

Condition E action E.5 states "Evaluate the following systems for alternate decay heat removal:

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Heat removal capability in Alternate Decay Heat Removal Mode is approximately 15 to 16 Million BTU's per hour. (per System Engineering B. Down)

The students are not required to know the value of BTU's per hour removal rate, or the BTU generation rated of the core at a particular time of core life. From a procedure user point of view, answer choice C is also correct when applied to Subsequent Action step E.

Accept answer choices A or C.

Given the following conditions:

- The plant is in Operational Condition 4 following a forced shutdown 16 hours ago.
- RHR Loop "A" operating in Shutdown Cooling.
- The "B" RHR pump is Cleared & Tagged for motor replacement.
- The "A" RHR pump develops a high vibration and trips on overcurrent.
- HC.OP-AB.RPV-0009, Shutdown Cooling, is entered.

Which of the following will be adequate to maintain Operational Condition 4?

a. Crosstie "C" or "D" RHR pump for heat removal.

b. Maximize RWCU bottom head drain flow.

c. Raise level to +80 inches using natural circulation for heat removal.

d. Inject with Core Spray from the CST to the RPV.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Hope Creek **Exam Date:** 02/24/20(

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 3 **SRO Group** 2 295021A104

295021 Loss of Shutdown Cooling

AA1. Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING:

AA1.04 Alternate heat removal methods 3.7 3.7

Explanation of Answer Justification
Crosstie "C" or "D" RHR pump for heat removal. correct - RPV-0009 subsequent action E, RHR Pumps C & D may be realigned to provide alternative decay heat removal.
Maximize RWCU bottom head drain flow. -incorrect- Maximizing bottom head drain flow does not provide heat removal adequate to maintain less than 200 degrees.
Raise level to +80 inches using natural circulation for heat removal. -Incorrect- per subsequent action E Natural Circulation does not provide heat removal, only circulation.
Inject with Core Spray from the CST's to the RPV. -incorrect- This is not an approved method of Alternate DHR.

Reference Title

HC.OP-AB.RPV-0009

Learning Objectives

ABRPV9E007 (R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Shutdown Cooling.

Material Required for Examination None

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

Question Source Comments: VISION Bank QID# Q61332 Sig Mod

Seating Chart
LSRO Written Examination administration
4/7/2003

NTC Classroom 32/33

Front

Scarpati			
			Beltz

		Fitz-gibbon	

Questions asked during LSRO Examination administration. 4/7/2003

Time	Question #	Candidate	Question	Answer provided
0816	-	Beltz	Is there a formula sheet provided?	No.
0822	37	Scarpati	Figure 3.1.5 missing from references	Figure found in package out of sequence.
0830	13	Fitzgibbon	Not sure what the question is asking.	Answer to the best of your ability.
0842	21	Scarpati	What does "isolated from injection" mean? Does that mean isolated by procedure?	Yes, isolated by procedure.
0905	27	Fitzgibbon	Is IO-5 complete in its entirety?	Answer question based on conditions provided.
1043	27	Fitzgibbon	Are the bullets listed in order of occurrence?	Yes.

	Task Description	Date Complete
1.	Facility written exam comments or graded exams received and verified complete	4/11/03
2.	Facility written exam comments reviewed and incorporated and NRC grading completed, if necessary	4/16/03
3.	Operating tests graded by NRC examiners	4/16/03
4.	NRC Chief examiner review of written exam and operating test grading completed	4/16/03
5.	Responsible supervisor review completed	5/1/03
6.	Management (licensing official) review completed	5/1/03
7.	License and denial letters mailed	5/2/03
8.	Facility notified of results	5/1/03 *
9.	Examination report issued (refer to NRC MC 0610)	5/20/03
10.	Reference material returned after final resolution of any appeals	6/2/03

* A. Faulkner notified, exit meeting on 5/5/03.

HOPE CREEK
EVENT CLASSIFICATION GUIDE
Glossary of Acronyms & Abbreviations
Section ii

AAAG	-	Accident Assessment Advisory Group (Delaware)
AC	-	Alternating Current
ADS	-	Automatic Depressurization System
ALARA	-	As Low As Reasonably Achievable
APRM	-	Average Power Range Monitor
ARI	-	Alternate Rod Insertion
ARM	-	Area Radiation Monitor
ASAP	-	As Soon As Possible
ASM	-	Administrative Support Manager
AS	-	Administrative Supervisor
ATWS	-	Anticipated Transient Without Scram
BKGD	-	Background
BKR	-	Breaker (electrical circuit)
BNE	-	Bureau of Nuclear Engineering (NJDEPE)
CACS	-	Containment Atmosphere Control System
CAS	-	Central Alarm Station
CCPM	-	Corrected Counts per Minute
CEDE	-	Committed Effective Dose Equivalent
CDE	-	Committed Dose Equivalent
CFR	-	Code of Federal Regulations
CIS	-	Containment Isolation System
CNTMT	-	Containment (Barrier)
CP	-	Control Point
CPM	-	Counts Per Minute
CR	-	Control Room
CREF	-	Control Room Emergency Filter System
CRIDS	-	Control Room Integrated Display System
CRD	-	Control Rod Drive
CSS	-	Core Spray System
DC	-	Direct Current
DAPA	-	Drywell Atmosphere Post Accident (Radiation monitor)
DDE	-	Deep Dose Equivalent
DEI	-	Dose Equivalent Iodine
DEMA	-	Delaware Emergency Management Agency
DEP	-	Department of Environmental Protection (NJ)

DID	-	Direct Inward Dial (phone system)
DLD	-	Drywell Leak Detection
DOE	-	Department of Energy
DOT	-	Department of Transportation
DPCC/DCR	-	Discharge Prevention, Containment, & Countermeasures/ Discharge Cleanup & Removal Plan
DPM	-	Disintegrations per Minute
DRCF	-	Dose Rate Conversion Factor
EACS	-	ESF Equipment Area Cooling System
EAL	-	Emergency Action Level
EAS	-	Emergency Alert System (Broadcast)
EC	-	Emergency Coordinator
ECCS	-	Emergency Core Cooling Systems
ECG	-	Emergency Classification Guide
EDG	-	Emergency Diesel Generator
EDO	-	Emergency Duty Officer
EMRAD	-	Emergency Radio (NJ)
ENC	-	Emergency News Center
ENS	-	Emergency Notification System (NRC)
EOC	-	Emergency Operations Center (NJ & DE)
EOF	-	Emergency Operations Facility
EOP	-	Emergency Operating Procedure
EPA	-	Emergency Preparedness Advisor
EPA	-	Environmental Protection Agency
EPC	-	Emergency Preparedness Coordinator
EPIP	-	Emergency Plan Implementing Procedure
EPZ	-	Emergency Planning Zone
EQPT	-	Equipment
ERDS	-	Emergency Response Data System
ERM	-	Emergency Response Manager
ERO	-	Emergency Response Organization
ESF	-	Engineered Safety Feature
ESSX	-	Electronic Switch System Exchange (Centrex)
FC	-	Fuel Clad (Barrier)
FFD	-	Fitness For Duty
FRVS	-	Filtration, Recirculation, and Ventilation System
FTS	-	Federal Telecommunications System (NRC)
GE	-	General Emergency
HCLL	-	Heat Capacity Level Limit
HCGS	-	Hope Creek Generating Station

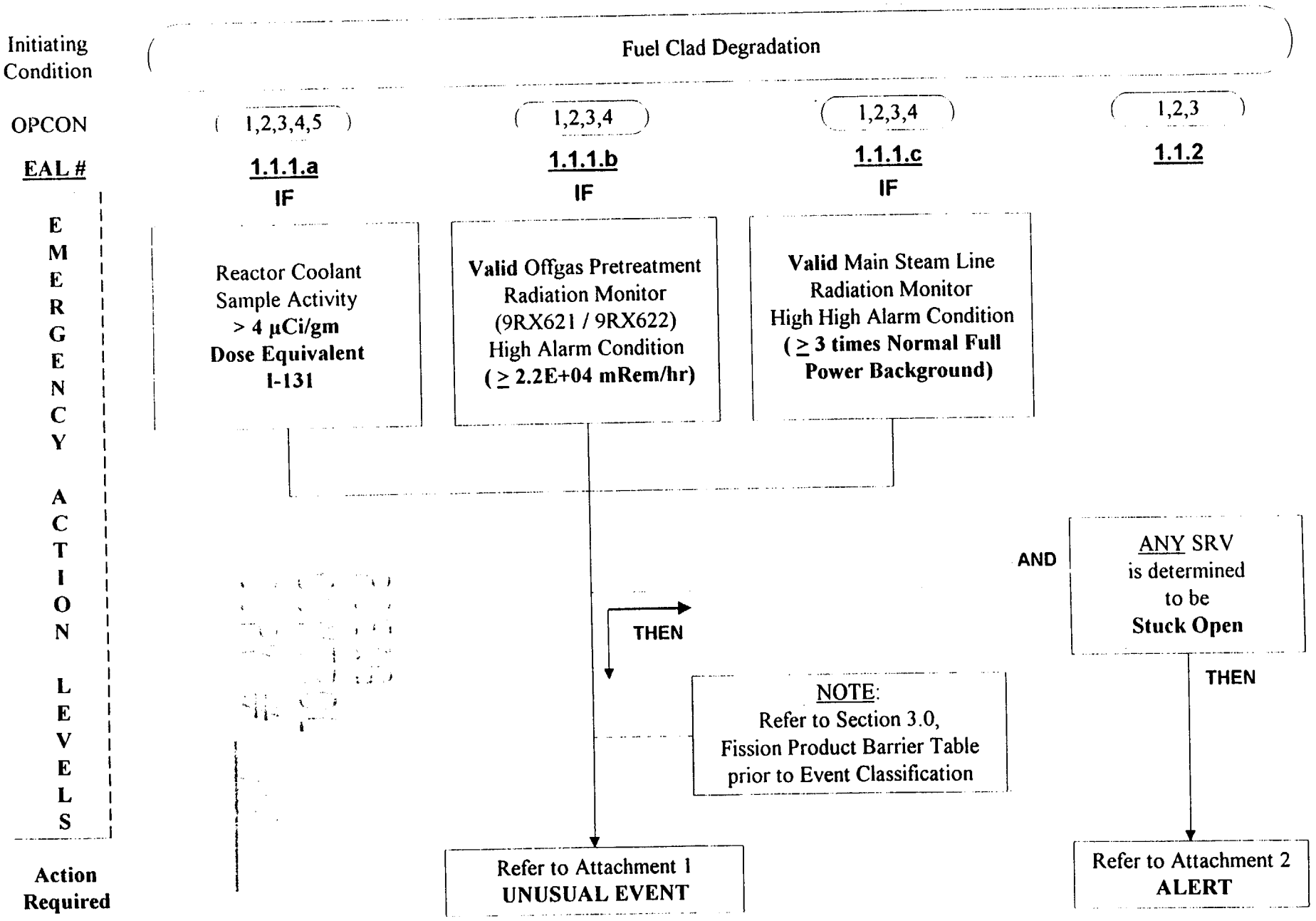
HCTL	-	Heat Capacity Temperature Limit
HEPA	-	High Efficiency Particulate Absorbers
HPCI	-	High Pressure Coolant Injection
HTV	-	Hardened Torus Vent
HVAC	-	Heating, Ventilation & Air Conditioning
HWCI	-	Hydrogen Water Chemical Injection
HX	-	Heat Exchanger
IAW	-	In Accordance With
IC	-	Initiating Condition
ICMF	-	Initial Contact Message Form
IDLH	-	Immediately Dangerous to Life and Health
IRM	-	Intermediate Range Monitor
I/S	-	In Service
KI	-	Potassium Iodide
KV	-	Kilovolt
LAC	-	Lower Alloways Creek
LCO	-	Limiting Condition for Operation
LDE	-	Lens Dose Equivalent
LEL	-	Lower Explosive Limit
LLD	-	Lowest Level Detectable
LOCA	-	Loss of Coolant Accident
LOP	-	Loss of Offsite Power
LPCI	-	Low Pressure Coolant Injection
LPZ	-	Low Population Zone
MCR	-	Main Control Room
MDA	-	Minimum Detectable Amount
MEA	-	Minimum Exclusion Area
MEES	-	Major Equipment & Electrical Status (Form)
MET	-	Meteorological
M.O.U.	-	Memorandum of Understanding
MRO	-	Medical Review Officer
MSIV	-	Main Steam Isolation Valve
MSIVSS	-	Main Steam Isolation Valve Sealing System
MSL	-	Main Steam Line
NAWAS	-	National Attack Warning Alert System
NCO	-	Nuclear Control Operator
NDAB	-	Nuclear Department Administration Building (TB2)
NEO	-	Nuclear Equipment Operator
NETS	-	Nuclear Emergency Telecommunications System

NFE	-	Nuclear Fuels Engineer
NFPB	-	Normal Full Power Background
NJSP	-	New Jersey State Police
NOAA	-	National Oceanographic and Atmospheric Administration
NPV	-	North Plant Vent
NRC	-	Nuclear Regulatory Commission
NSS	-	Nuclear Shift Supervisor
NSSSS	-	Nuclear Steam Supply Shutoff System
NSTA	-	Nuclear Shift Technical Advisor
NUMARC	-	Nuclear Management and Resources Council
NWS	-	National Weather Service
OBE	-	Operating Basis Earthquake
OCA	-	Owner Controlled Area
ODCM	-	Offsite Dose Calculation Manual
OEM	-	Office of Emergency Management (NJ)
OHA	-	Overhead Annunciator
OPCON	-	Operating Condition
OSB	-	Operational Status Board (Form)
OSC	-	Operations Support Center
PAG	-	Protective Action Guideline
PAR	-	Protective Action Recommendation
PASS	-	Post Accident Sample System
PC	-	Primary Containment (Barrier)
PCIG	-	Primary Containment Instrument Gas System
PCIS	-	Primary Containment Isolation System
PSIG	-	Pounds Square Inch Gauge
RAD	-	Radiation
RAL	-	Reportable Action Level
RC	-	Reactor Coolant
RCA	-	Radiologically Controlled Area
RCAM	-	Repair and Corrective Action Mission
RCIC	-	Reactor Core Isolation Cooling
RCS	-	Reactor Coolant System (Barrier)
RHR	-	Residual Heat Removal (Containment Heat Removal)
RM	-	Recovery Manager
RMO	-	Recovery Management Organization
RMS	-	Radiation Monitoring System
RPS	-	Radiation Protection Supervisor
RPS	-	Reactor Protection System
RPV	-	Reactor Pressure Vessel
RRCS	-	Redundant Reactivity Control System

RSM	-	Radiological Support Manager
RWCU	-	Reactor Water Cleanup (System)
SACS	-	Safety Auxiliaries Cooling System
SAE	-	Site Area Emergency
SAM	-	Severe Accident Management
SAS	-	Secondary Alarm Station (Security)
SBO	-	Station Blackout
SCBA	-	Self Contained Breathing Apparatus
SCP	-	Security Contingency Procedure
SDE	-	Shallow Dose Equivalent
SDM	-	Shutdown Margin
SLC	-	Standby Liquid Control
SJAE	-	Steam Jet Air Ejector
SNM	-	Special Nuclear Material
SNSS	-	Senior Nuclear Shift Supervisor
SOS	-	Systems Operations Supervisor (Security)
SPDS	-	Safety Parameter Display System
SPV	-	South Plant Vent
SRM	-	Source Range Monitor
SRPT	-	Shift Radiation Protection Technician
SRV	-	Safety Relief Valve
SSCL	-	Station Status Checklist
SSE	-	Safe Shutdown Earthquake
SSWS	-	Station Service Water System
SSNM	-	Strategic Special Nuclear Material
TAF	-	Top of Active Fuel
TDR	-	Technical Document Room
TEDE	-	Total Effective Dose Equivalent
TIP	-	Traversing Incore Probe
TLV	-	Threshold Limit Value
T/S	-	Technical Specifications
TSC	-	Technical Support Center
TSS	-	Technical Support Supervisor
TSTL	-	Technical Support Team Leader
TSTM	-	Technical Support Team Member
UE	-	Unusual Event
UFSAR	-	Updated Final Safety Analysis Report
UHS	-	Ultimate Heat Sink
USCG	-	United States Coast Guard
VDC	-	Volts Direct Current
WB	-	Whole Body

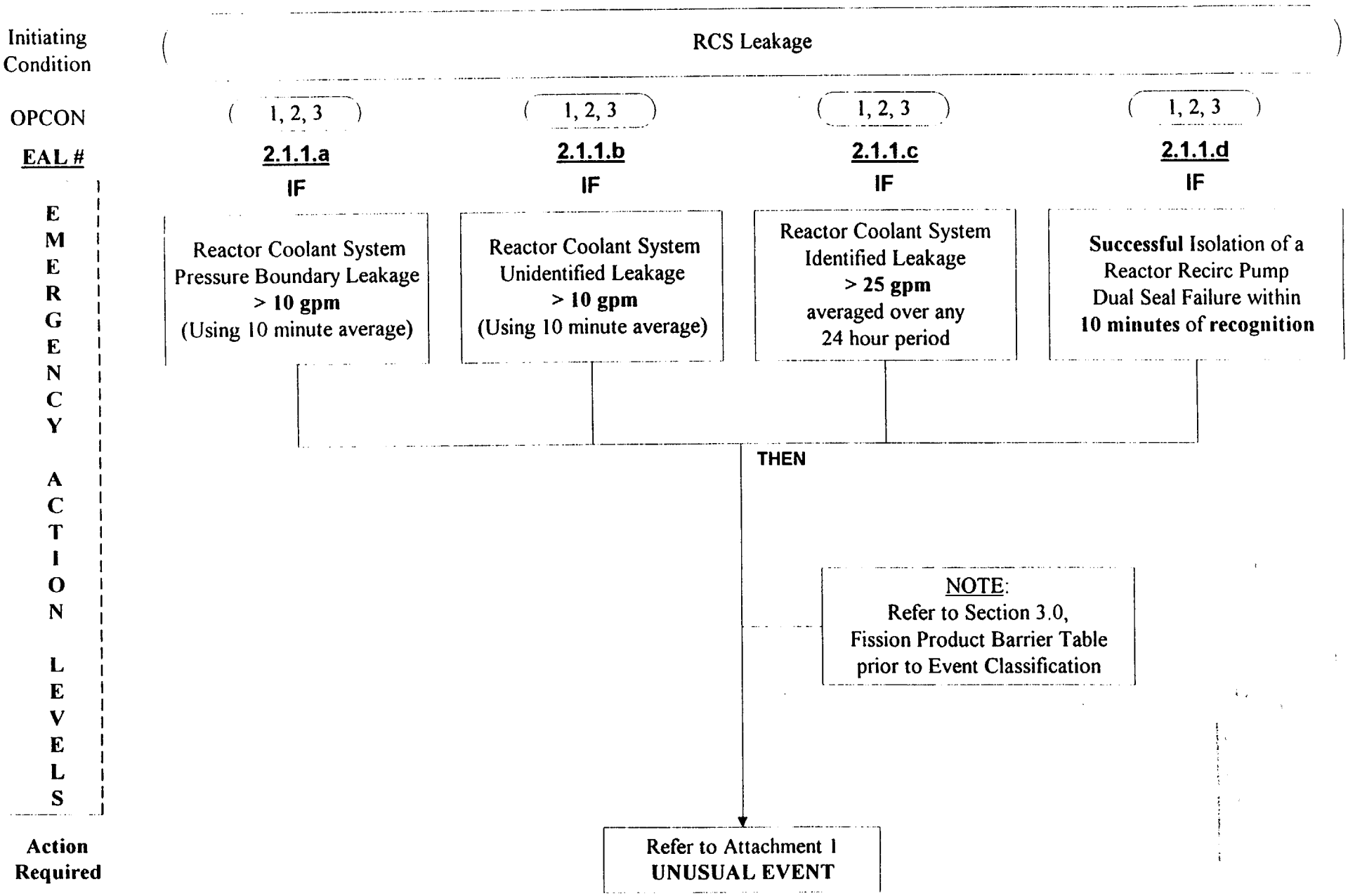
1.0 Fuel Clad Challenge

1.1 RCS Activity



2.0 RCS Challenge

2.1 RCS Leakage



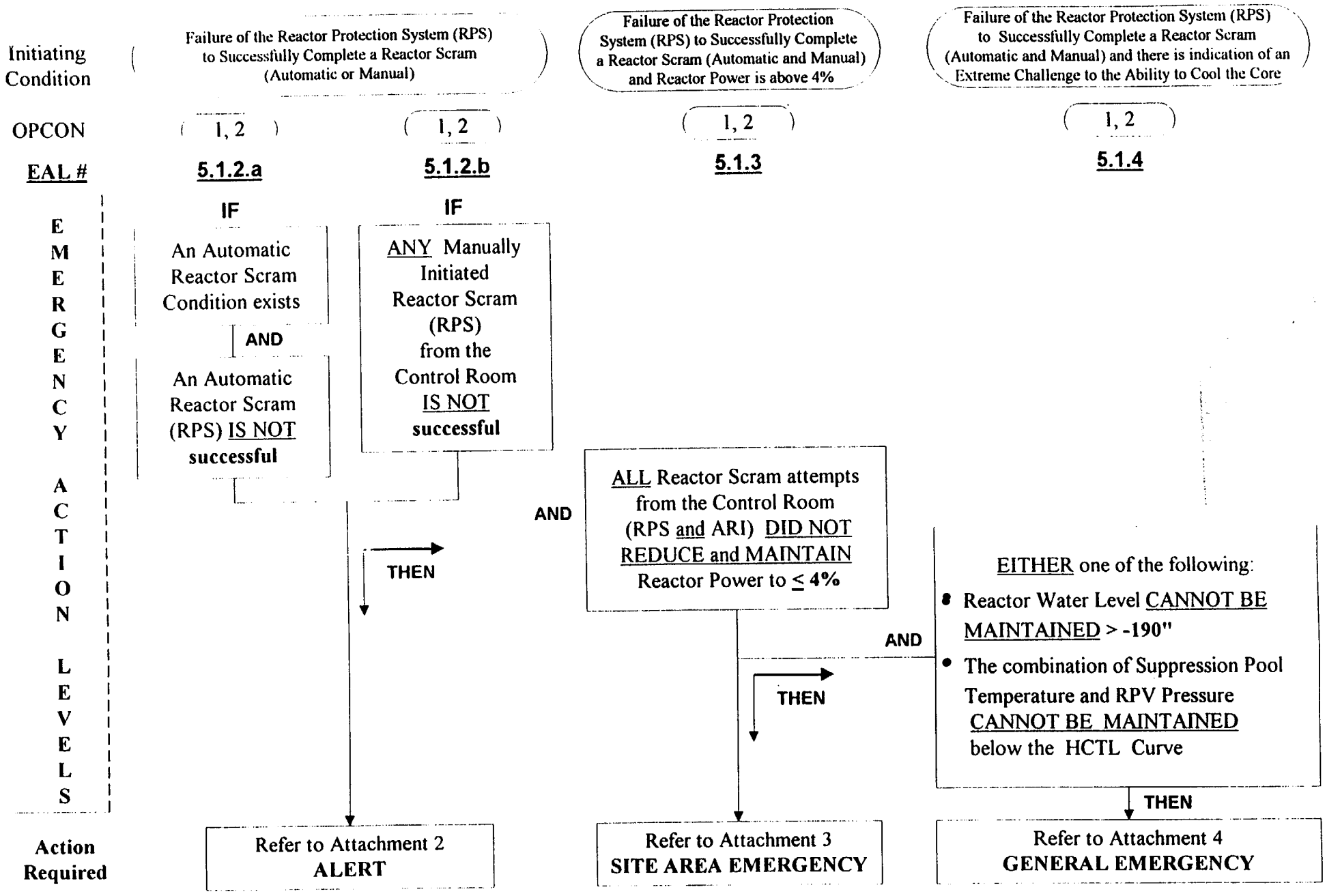
4.0 E C Discretion

4.1 Emergency Coordinator Discretion

Initiating Condition	Other Conditions Exist Which In the Judgment of the Emergency Coordinator Warrant Declaration of an Unusual Event	Other Conditions Exist Which In the Judgment of the Emergency Coordinator Warrant Declaration of an Alert	Other Conditions Exist Which In the Judgment of the Emergency Coordinator Warrant Declaration of a Site Area Emergency	Other Conditions Exist Which In the Judgment of the Emergency Coordinator Warrant Declaration of a General Emergency
OPCON	All	All	All	All
EAL #	4.1.1	4.1.2	4.1.3	4.1.4
EMERGENCY ACTION LEVELS	IF	IF	IF	IF
	Events are in progress or have occurred which, in the judgment of the Emergency Coordinator, indicate a **Potential Degradation of Plant Safety**	Events are in progress or have occurred which, in the judgment of the Emergency Coordinator, indicate EITHER one of the following: • Plant safety systems **(more than one)** are or may be degraded • ANY Plant Vital Structure is degraded or potentially degraded	Events are in progress or have occurred which, in the judgment of the Emergency Coordinator, indicate an Actual or likely major failure of plant functions needed for protection of the public	Events are in progress or have occurred which, in the judgment of the Emergency Coordinator, indicate an Actual or imminent substantial core degradation with the potential for loss of containment
	THEN	AND Increased monitoring of Safety Functions is warranted THEN	THEN	THEN
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**	Refer to Attachment 2 **ALERT**	Refer to Attachment 3 **SITE AREA EMERGENCY**	Refer to Attachment 4 **GENERAL EMERGENCY**

5.0 Failure to Scram

5.1 ATWS



6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

Initiating Condition

- Any **Unplanned** Release of Gaseous Radioactivity to Environment that Exceeds 2 Times the Radiological Technical Specifications for 60 minutes or longer
- Any **Unplanned** Release of Gaseous Radioactivity to the Environment that Exceeds 2 times the 10CFR20, Appendix B limits for 60 minutes or longer

OPCON

- All
- All
- All
- All

EAL #

6.1.1.a — Dose Assessment — IF

Dose Assessment indicates EITHER one of the following at the MEA or beyond as calculated on the SSCL:

- TEDE 4-Day Dose ≥ **2.0E-01 mRem**
- Thyroid-CDE Dose ≥ **6.8E-01 mRem**

based on Plant Vent effluent sample analysis and NOT on a default Noble Gas to Iodine Ratio

6.1.1.b — Field Measured Dose Rate — IF

Dose Rate measured at the Protected Area Boundary or beyond EXCEEDS **.05 mRem/hr** above normal background

6.1.1.c — Sample Analysis — IF

Gaseous effluent release sample analysis for ANY one of the following indicates a concentration of:

- **FRVS:**
  - ≥ **1.13E-03 µCi/cc Total Noble Gas**
  - ≥ **2.71E-07 µCi/cc I-131**
- **NPV:**
  - ≥ **2.43E-04 µCi/cc Total Noble Gas**
  - ≥ **5.81E-08 µCi/cc I-131**
- **SPV:**
  - ≥ **2.27E-05 µCi/cc Total Noble Gas**
  - ≥ **5.44E-09 µCi/cc I-131**

6.1.1.d — Alarm Indications — IF

Valid High Alarm received from ANY 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 EXCEEDS EITHER one of the following limits:

- **4.80E+03 µCi/sec Total Noble Gas**
- **1.15E+00 µCi/sec I-131** (NPV & SPV ONLY)

AND

Dose Assessment results NOT available

AND

Release is ongoing for ≥ **60 minutes**

THEN

Refer to Attachment 1
UNUSUAL EVENT

EMERGENCY ACTION LEVELS

Action Required

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 — Dose Assessment — **IF**

Dose Assessment indicates EITHER one of the following at the MEA or beyond as calculated on the SSCL:

- TEDE 4-Day Dose ≥ **2.0E+01 mRem**
- Thyroid-CDE Dose ≥ **6.8E+01 mRem** based on Plant Vent effluent sample analysis and NOTon a default Noble Gas to Iodine Ratio

6.1.2.b — Field Measured Dose Rate — **IF**

Dose Rate measured at the Protected Area Boundary or beyond EXCEEDS **5 mRem/hr**

AND

Release is ongoing for ≥ **15 minutes**

THEN

6.1.2.c — Sample Analysis — **IF**

Gaseous effluent release sample analysis for ANY one of the following indicates a concentration of:

- **FRVS:**
  ≥ **1.13E-01 µCi/cc Total Noble Gas**
  ≥ **2.71E-05 µCi/cc I-131**
- **NPV:**
  ≥ **2.43E-02 µCi/cc Total Noble Gas**
  ≥ **5.81E-06 µCi/cc I-131**
- **SPV:**
  ≥ **2.27E-03 µCi/cc Total Noble Gas**
  ≥ **5.44E-07 µCi/cc I-131**

AND

Dose Assessment results NOT available

AND

Release is ongoing for ≥ **30 minutes**

THEN

6.1.2.d — Alarm Indications — **IF**

Valid High Alarm received from ANY 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 EXCEEDS EITHERone of the following limits:

- **4.80E+05 µCi/sec Total Noble Gas**
- **1.15E+02 µCi/sec I-131** (NPV & SPV ONLY)

EMERGENCY ACTION LEVELS

Action Required

Refer to Attachment 2
ALERT

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

Initiating Condition: Boundary Dose Resulting from an Actual or **Imminent** Release of Gaseous Radioactivity Exceeds 100 mrem Total Effective Dose Equivalent (TEDE) or 500 mRem Thyroid CDE Dose for the actual or projected duration of the release

OPCON: All

EAL # — **EMERGENCY ACTION LEVELS**

6.1.3.a — Dose Assessment — IF

Dose Assessment indicates EITHER one of the following at the MEA or beyond as calculated on the SSCL:

- TEDE 4-Day Dose ≥ **1.0E+02 mRem**
- Thyroid-CDE Dose ≥ **5.0E+02 mRem**

based on Plant Vent effluent sample analysis and NOT on a default Noble Gas to Iodine Ratio

6.1.3.b — Field Measured Dose Rate — IF

Dose Rate measured at the Protected Area Boundary or beyond EXCEEDS **100 mRem/hr**

AND

Release is expected to continue for ≥ **15 minutes**

6.1.3.c — Field Survey Analysis — IF

Analysis of field survey samples at the Protected Area Boundary indicates EITHER one of the following:

- ≥ **4.36E+02 CCPM**
- ≥ **3.85E-07 µCi/cc I-131**

6.1.3.d — Alarm Indications — IF

Valid High Alarm received from ANY one of the following Plant Effluent RMS Channels:

- **FRVS Noble Gas** (Grid 1/3; 9RX680)
- **NPV Noble Gas** (Grid 1/3; 9RX590)
- **SPV Noble Gas** (Grid 1/3; 9RX580)
- **HTV Noble Gas** (Grid 3; 9RX516)

AND

Total Plant Vent release rate EXCEEDS **7.6E+07 µCi/sec Total Noble Gas**

AND

Dose Assessment results NOT available

AND

Release is ongoing for ≥ **15 minutes**

THEN

Action Required: Refer to Attachment 3 **SITE AREA EMERGENCY**

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

Initiating Condition: Boundary Dose Resulting from an Actual or **Imminent** Release of Gaseous Radioactivity Exceeds 1000 mrem Total Effective Dose Equivalent (TEDE) or 5000 mRem Thyroid CDE Dose for the actual or projected duration of the release

EMERGENCY ACTION LEVELS

OPCON	All	All	All	All
EAL #	**6.1.4.a**	**6.1.4.b**	**6.1.4.c**	**6.1.4.d**
	IF	IF	IF	IF
	Dose Assessment	Field Measured Dose Rate	Field Survey Analysis	Alarm Indications
	Dose Assessment indicates EITHER one of the following at the MEA or beyond as calculated on the SSCL: • TEDE 4-Day Dose ≥ **1.0E+03 mRem** • Thyroid-CDE Dose ≥ **5.0E+03 mRem** based on Plant Vent effluent sample analysis and NOT on a default Noble Gas to Iodine Ratio	Dose Rate measured at the Protected Area Boundary or beyond EXCEEDS **1000 mRem/hr** AND Release is expected to continue for ≥ **15 minutes**	Analysis of field survey samples at the Protected Area Boundary indicates EITHER one of the following: • ≥**4.36E+03 CCPM** • ≥ **3.85E-06 µCi/cc I-131**	**Valid** High Alarm received from ANY one of the following Plant Effluent RMS Channels: • **FRVS Noble Gas** (Grid 1/3; 9RX680) • **NPV Noble Gas** (Grid 1/3; 9RX590) • **SPV Noble Gas** (Grid 1/3; 9RX580) • **HTV Noble Gas** (Grid 3; 9RX516) AND Total Plant Vent release rate EXCEEDS **7.6E+08 µCi/sec Total Noble Gas** AND Dose Assessment results NOT available. AND Release is ongoing for ≥ **15 minutes**

THEN

Action Required: Refer to Attachment 4 **GENERAL EMERGENCY**

6.0 Radiological Releases/Occurrences

6.2 Liquid Effluent Release

Initiating Condition	Any **Unplanned** Release of Liquid Radioactivity to the Environment that Exceeds 2 Times the Radiological Technical Specifications for 60 minutes or longer	Any **Unplanned** Release of Liquid Radioactivity to the Environment that exceeds 200 Times Radiological Technical Specifications for 15 minutes or longer
OPCON	All	All
EAL #	**6.2.1**	**6.2.2**
EMERGENCY ACTION LEVELS	IF **Valid** Cooling Tower Blowdown Effluent Radiation Monitor High Alarm Condition THEN AND Sample analysis of liquid effluent indicates concentration in excess of **2 times Technical Specification limits** AND Release continues for ≥ **60 minutes** after the alarm occurs THEN	AND Sample analysis of liquid effluent indicates concentration in excess of **200 times Technical Specification limits** AND Release continues for ≥ **15 minutes** after the alarm occurs THEN
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**	Refer to Attachment 2 **ALERT**

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

Initiating Condition	Unplanned Increase in Plant Radiation	Release of Radioactive Material or increases in Radiation Levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown	
OPCON	All	All	All
EAL #	6.3.1	6.3.2.a	6.3.2.b
EMERGENCY ACTION LEVELS	IF **Unplanned** rise in radiation levels inside the Protected Area **≥ 1000 times normal** as indicated by EITHER one of the following: • Permanent or portable Area Radiation Monitors • General Area Radiological Survey	IF **Unplanned** Dose Rates **≥ 2000 mRem/hr** in ANY **area of the plant** which requires ACCESS to maintain plant safety functions (EXCLUDING the Main Control Room and CAS) as indicated by EITHER one of the following: • Permanent or portable Area Radiation Monitors • General Area Radiological Survey	IF **Unplanned** Dose Rates **≥ 15 mRem/hr** in EITHER one of the following: • Main Control Room • Security Central Alarm Station (CAS)
	THEN	THEN	
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**	Refer to Attachment 2 **ALERT**	

6.0 Radiological Releases/Occurrences

6.4 Irradiated Fuel Event

Initiating Condition

OPCON

EAL #

EMERGENCY ACTION LEVELS

Unplanned Increase in Plant Radiation

5

6.4.1.a

IF

Uncontrolled water level drop in the Reactor Cavity as indicated by EITHER one of the following:

- Visual Observation
- Reactor Water Level Shutdown Range Level Indicator 1BBLI-R605

All

6.4.1.b

IF

Uncontrolled water level drop in the Spent Fuel Pool as indicated by **Valid** Fuel Pool Low Level Alarm Condition

AND

Visual Observation

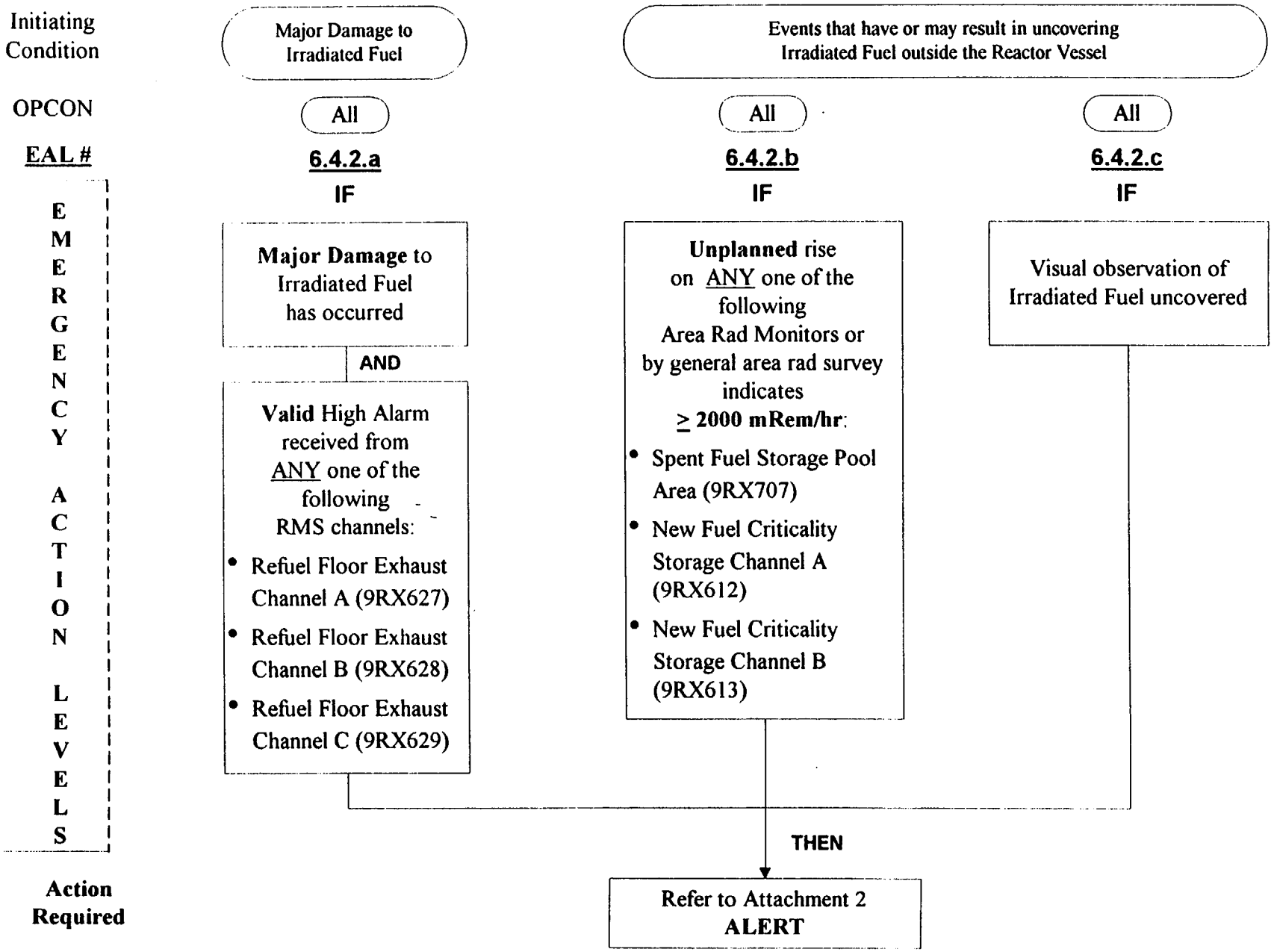
THEN

Refer to Attachment 1
UNUSUAL EVENT

Action Required

6.0 Radiologic.. Releases/Occurrences

6.4 Irradiated Fuel Event



7.0 Electrical Power

7.1 Loss of AC Power Capabilities

Initiating Condition	Loss of All Offsite Power to Vital Buses for greater than 15 minutes	AC power capability to Vital Buses reduced to a Single Power Source for greater than 15 minutes such that any additional single failure would result in a complete loss of all 4.16 KV Vital Buses	Loss of All Offsite Power and All Onsite AC Power to 4.16 KV Vital Buses during either in Cold Shutdown or Refueling for greater than 15 minutes
OPCON	All	1, 2, 3	4, 5, Defueled
EAL #	**7.1.1**	**7.1.2.a**	**7.1.2.b**
EMERGENCY ACTION LEVELS	IF **Unplanned** Loss of Power from Station Service Transformers 1AX501 **AND** 1BX501 to ALL **4.16 KV Vital Buses** AND **> 15 minutes** have elapsed THEN	IF Loss of **4.16 KV Vital Bus Power Sources** (Offsite and Onsite) which results in the **availability** of ONLY **one 4.16 KV Vital Bus Power Source** (Offsite or Onsite)	IF ALL **4.16 KV Vital Buses** are deenergized AND **> 15 minutes** have elapsed THEN
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**		Refer to Attachment 2 **ALERT**

7.0 Electrical Power

7.1 Loss of AC Power Capabilities

Initiating Condition	Loss of All Offsite Power and All Onsite AC Power to All Vital AC Buses during either Power Operation, Startup or Hot Shutdown for greater than 15 minutes	Prolonged Loss of All Offsite and Onsite AC Power to All Vital AC Buses	
OPCON	(1, 2, 3)	(1, 2, 3)	(1, 2, 3)
EAL #	**7.1.3**	**7.1.4.a**	**7.1.4.b**
EMERGENCY ACTION LEVELS	IF ALL **4.16 KV Vital Buses** are deenergized THEN AND > **15 minutes** have elapsed THEN	AND Restoration of Power to **at least one** 4.16 KV Vital Bus within **4 hours** is NOT likely THEN	AND Loss of **any 2** Fission Product Barriers has occurred or is **Imminent** THEN
Action Required	Refer to Attachment 3 **SITE AREA EMERGENCY**	Refer to Attachment 4 **GENERAL EMERGENCY**	

7.0 Electrical Power

7.2 Loss of DC Power Capabilities

Initiating Condition	Unplanned Loss of All Vital 125 VDC Power during either Cold Shutdown or Refueling Mode for greater than 15 minutes	Unplanned Loss of All Vital 125 VDC Power during either Power Operation, Startup or Hot Shutdown for greater than 15 Minutes
OPCON	4, 5, Defueled	1, 2, 3
EAL #	7.2.1	7.2.3
EMERGENCY ACTION LEVELS	IF **Unplanned** degraded voltage condition for ALL **Vital 125 VDC Buses**, such that voltage is < **108 VDC** AND > **15 minutes** have elapsed THEN	IF **Unplanned** degraded voltage condition for ALL **Vital 125 VDC Buses**, such that voltage is < **108 VDC** AND > **15 minutes** have elapsed THEN
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**	Refer to Attachment 3 **SITE AREA EMERGENCY**

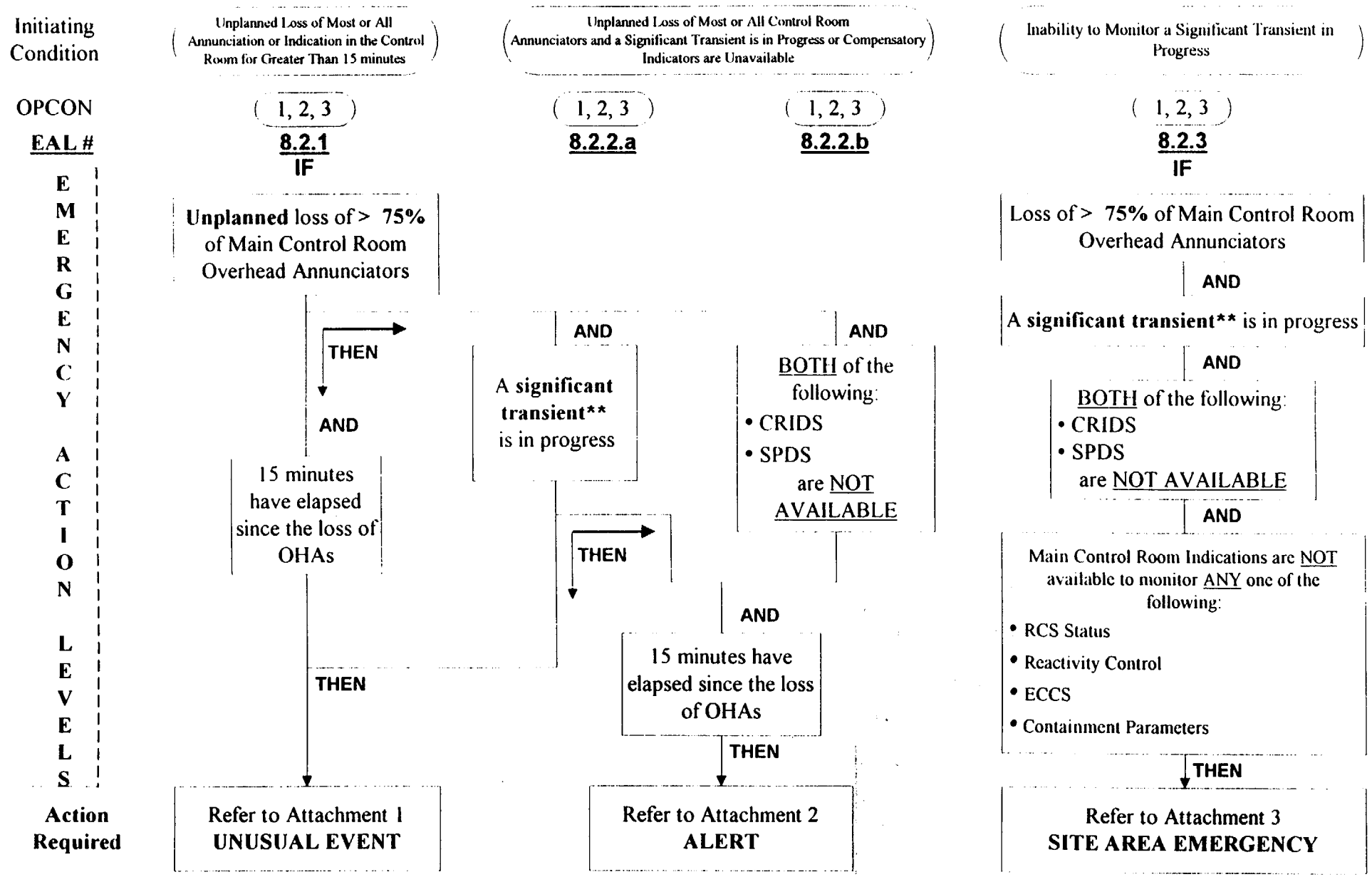
8.0 System Malfunctions

8.1 Loss of Heat Removal Capabililty

Initiating Condition	Inability to Maintain the Plant in Cold Shutdown	Loss of Reactor Water Level that has or will Uncover Fuel in the Reactor Vessel	Complete Loss of Functions Needed to Achieve Cold Shutdown Conditions
OPCON	4, 5	4, 5	1, 2, 3
EAL #	**8.1.2**	**8.1.3.a**	**8.1.3.b**
EMERGENCY ACTION LEVELS	**IF**	**IF**	**IF**
	Unplanned, Complete Loss of ALL Technical Specification required systems **available to provide Decay Heat Removal functions**	Reactor Water Level REACHES -161" **(Top of Active Fuel)**	**Loss** of Main Condenser capabilities, as evidenced by an inability to remove Decay Heat from the Reactor
	AND		**AND**
	RCS Temperature has risen to > 200°F (Excluding a <15 **minutes** rise >200°F with a **heat removal function** restored)		**Loss** of Torus capabilities as evidenced by EITHER one of the following: • Entry into an Unsafe region of ANY one of the following curves: * Heat Capacity Temperature Limit (HCTL) Curve * Heat Capacity Level Limit (HCLL) Curve * Pressure Suppression Pressure (PSP) Curve * SRV Tailpipe Level Limit Curve • Insufficient SRV capacity to reduce RPV pressure
	AND		
	An UNCONTROLLED temperature rise is RAPIDLY approaching **200°F** (with NO **heat removal function** restored)		
	THEN		**THEN**
Action Required	Refer to Attachment 2 **ALERT**		Refer to Attachment 3 **SITE AREA EMERGENCY**

8.0 System Malfunctions

8.2 Loss of Overhead Annunciators



** NOTE: A Significant Transient is based on EC judgment, but includes as a minimum ANY one of the following: RX SCRAM, LOAD REJECTION >25% POWER, ECCS INJECTION, THERMAL POWER OSCILLATION >10% .

8.0 System Malfunctions

8.3 Loss of Communications Capabililty

Initiating Condition

OPCON

EAL #

EMERGENCY ACTION LEVELS

Action Required

Unplanned Loss of All Onsite or Offsite Communications Capabilities

All

8.3.1.a

IF

Unplanned loss of ALL ONSITE communications as evidenced by the loss of ALL of the following systems:

- Direct Inward Dial System (DID)
- Station Page System (Gaitronics)
- Station Radio System

All

8.3.1.b

IF

Unplanned loss of ALL OFFSITE communications as evidenced by the loss of ALL of the following systems:

- Direct Inward Dial System (DID)
- Nuclear Emergency Telephone System (NETS)
- Essx System (Centrex) phone

THEN

Refer to Attachment 1
UNUSUAL EVENT

8.0 System Malfunctions

8.4 Control Room Evacuation

Initiating Condition	Main Control Room Evacuation has been Initiated	Main Control Room Evacuation has been Initiated and Plant Control cannot be established
OPCON	All	All
EAL #	8.4.2	8.4.3
EMERGENCY ACTION LEVELS	IF Main Control Room Evacuation has been initiated THEN	AND **Control of the plant** CANNOT be established from outside the Main Control Room within **15 minutes** THEN
Action Required	Refer to Attachment 2 **ALERT**	Refer to Attachment 3 **SITE AREA EMERGENCY**

8.0 System Malfunctions

8.5 Technical Specifications

Initiating Condition

Inability to Reach Required Operational Condition within Technical Specification Limits

OPCON

1, 2, 3

EAL #

8.5.1

IF

Plant is NOT brought to the REQUIRED Operational Condition within the Technical Specification required time limit

THEN

EMERGENCY ACTION LEVELS

Action Required

Refer to Attachment 1
UNUSUAL EVENT

9.0 Hazards - Internal/External

9.1 Security Threats

Initiating Condition	Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant	Security Event in a Plant Protected Area	Security Event in a Plant Vital Area	Security Event Resulting in Loss of Ability to Reach and Maintain Cold Shutdown
OPCON	All	All	All	All
EAL #	9.1.1	9.1.2	9.1.3	9.1.4
EMERGENCY ACTION LEVELS	IF	IF	IF	IF
	Confirmed security threat directed toward the station as evidenced by ANY one of the following: • Credible threat of malicious acts or destructive device within the Protected Area resulting in **SCP-5** implementation • Credible intrusion or assault threat to the Protected Area resulting in **SCP-5** implementation • Attempted intrusion or assault to the Protected Area resulting in **SCP-7** or **SCP-11** implementation • Malicious acts attempted or discovered within the Protected Area resulting in **SCP-10** implementation • Hostage/Extortion situation that threatens normal plant operations resulting in **SCP-8** implementation • Destructive Device discovered within the Protected Area resulting in **SCP-10** implementation	Confirmed hostile intrusion or malicious acts as evidenced by ANY one of the following: • Discovery of an intruder(s), armed and violent, within the Protected Area resulting in **SCP-6** implementation • Hostage held on-site in a non-vital area resulting in **SCP-8** implementation • Malicious acts attempted or discovered within a Vital Area (which is in the Protected Area) resulting in **SCP-10** implementation that does not meet criteria of 9.1.3	Confirmed hostile intrusion or malicious acts in Plant Vital Areas that involve actual or likely major failures of plant functions needed for the protection of the public as evidenced by any one of the following: • Discovery of an intruder(s), armed and violent, within the Vital Area resulting in **SCP-6** implementation • Destructive device discovered in a Vital Area resulting in **SCP-10** implementation • Malicious act(s) attempted or discovered in a Vital Area resulting in **SCP-10** implementation AND the malicious act(s) is of a magnitude that it SPECIFICALLY results in **Damage** to ANY one of the following: • TWO OR MORE Trains of a **Safety System** • MORE THAN ONE **Safety System** • Any Plant Vital Structure which renders the structure incapable of performing its Design Function AND Damaged Safety System(s) or Plant Vital Structure is required for the present Operational Condition	Security event resulting in the **actual loss of physical control** of EITHER one of the following: • Control Room • Remote Shutdown Panel
	THEN	THEN	THEN	THEN
Action Required	Refer to Attachment 24 **UNUSUAL EVENT (Common Site)**	Refer to Attachment 2 **ALERT**	Refer to Attachment 3 **SITE AREA EMERGENCY**	Refer to Attachment 4 **GENERAL EMERGENCY**

9.0 Hazards - Internal/External

9.2 Fire

EMERGENCY ACTION LEVELS

Initiating Condition	Fire within the Protected Area Boundary Not Extinguished within 15 minutes of Detection		Fire Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown
OPCON	All	All	All
EAL #	9.2.1	9.2.1	9.2.2
	IF	IF	IF
	Valid Fire Alarm is received in the Main Control Room	Report of a fire from personnel at the scene	Fire within ANY one of the following Plant Vital Structures: • Reactor Building • Control/Aux Building • Service Water Intake Structure • Service/Rad Waste Building
		AND	AND
	Fire is within ANY one of the following Plant Structures (EXCLUDING small fires that have NO potential to affect Safety Systems or Protected Area Permanent Plant Structures) • Reactor Building • Turbine Building • Control/Aux Building • Service Water Intake Structure • Service/Rad Waste Building • Low Level Radwaste Interim Storage Facility		The Fire is of a magnitude that it SPECIFICALLY results in Damage to ANY one of the following: • TWO OR MORE subsystems of a Safety System • MORE THAN ONE Safety System • Any Plant Vital Structure which renders the structure incapable of performing its Design Function
	AND		AND
	Fire is NOT extinguished within 15 minutes of EITHER one of the following: • Receipt of a Valid Fire Alarm • Report of a fire from the scene		Damaged Safety System(s) or Plant Vital Structure is required for the present Operational Condition
	THEN		THEN
Action Required	Refer to Attachment 1 UNUSUAL EVENT		Refer to Attachment 2 ALERT

9.0 Hazards - Internal/External

9.3 Explosion

Initiating Condition	Natural and Destructive Phenomena Affecting the Protected Area	Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown
OPCON	All	All
EAL #	9.3.1	9.3.2
EMERGENCY ACTION LEVELS	IF **Confirmed Explosion** within the Protected Area AND Report of visible damage to Plant equipment or Protected Area Permanent Plant Structures THEN	IF **Confirmed Explosion** within ANY one of the following Plant Vital Structures: • Reactor Building • Control/Aux Building • Service Water Intake Structure • Service/Rad Waste Building AND The **Explosion** is of a magnitude that it SPECIFICALLY results in **Damage** to ANY one of the following: • TWO OR MORE subsystems of a **Safety System** • MORE THAN ONE **Safety System** • Any Plant Vital Structure which renders the structure incapable of performing its Design Function AND Damaged Safety System(s) or Plant Vital Structure is required for the present Operational Condition THEN
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**	Refer to Attachment 2 **ALERT**

9.0 Hazards - Internal/External

9.4 Toxic/Flammable Gases

Initiating Condition: Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant

EMERGENCY ACTION LEVELS

OPCON	All	All	All
EAL #	9.4.1.a	9.4.1.b	9.4.1.c
	IF	IF	IF
	Notification by Local, County, or State Officials for the potential need to EVACUATE non-essential personnel due to an Offsite **Toxic Gas** release	**Uncontrolled Toxic Gas** release within the Protected Area in ANY area which does not normally require an atmospheric survey or Respiratory Protection for entry	**Uncontrolled Flammable Gas** release within the Protected Area that RESULTS in Flammable Gas concentrations EXCEEDING **25% of the LEL**
	AND		
	SNSS deems evacuation of non-essential personnel is required		
		AND	
		Routine Plant Operations are IMPEDED based on EITHER one of the following: • **Access restrictions** caused by the **uncontrolled** release • Personnel injuries have occurred as a result of the release	
	THEN	THEN	
Action Required	Refer to Attachment 24 **UNUSUAL EVENT (Common Site)**	Refer to Attachment 1 **UNUSUAL EVENT**	

9.0 Hazards - Internal/External

9.4 Toxic/Flammable Gases

Initiating Condition: Release of Toxic or Flammable Gases Within a Facility Structure Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown Conditions

OPCON: All

EAL #: 9.4.2.a

IF

Uncontrolled Toxic Gas release within ANY one of the following Plant Structures

- Reactor Building
- Turbine Building
- Control/Aux Building
- Service Water Intake Structure
- Service/Rad Waste Building

AND

Toxic Gas concentrations result in ANY one of the following:

- An IDLH atmosphere
- Plant personnel report severe adverse health reactions, including burning eyes, nose, throat, dizziness
- The Threshold Limit Value (TLV) being EXCEEDED

OPCON: All

EAL #: 9.4.2.b

IF

Uncontrolled Flammable Gas release within ANY one of the following Plant Structures

- Reactor Building
- Turbine Building
- Control/Aux Building
- Service Water Intake Structure
- Service/Rad Waste Building

AND

Flammable Gas concentrations EXCEED **50% of the LEL**

AND

Plant personnel are unable to perform actions necessary to complete a Safe Shutdown of the plant without appropriate personnel protection equipment

THEN

Action Required: Refer to Attachment 2 **ALERT**

EMERGENCY ACTION LEVELS

9.0 Hazards - Internal/External

9.5 Seismic Event

Initiating Condition	Natural and Destructive Phenomena Affecting the Protected Area		Natural and Destructive Phenomena Affecting the Plant Vital Area
OPCON	All	All	All
EAL #	**9.5.1.a**	**9.5.1.b**	**9.5.2**
EMERGENCY ACTION LEVELS	IF Seismic Event felt by personnel within the Protected Area	IF **Valid** Actuation of the Seismic Trigger (> **0.01g**) has occurred as verified by the SMA-3 Event Indicator (flag) being **WHITE** on Panel 10-C-673 in the Upper Relay Room	AND **Valid** Actuation of the Seismic Switch (> **0.1g**) has occurred as verified by EITHER one of the following: • **Valid** actuation of Main Control Room Overhead Annunciator C6-C4 • **AMBER** Alarm light on the Seismic Switch Power Supply Drawer is lit on Panel 10-C-673 in the Upper Relay Room
		THEN	THEN
Action Required	Refer to Attachment 24 **UNUSUAL EVENT (Common Site)**		Refer to Attachment 2 **ALERT**

9.0 Hazards - Internal/External

9.6 High Winds

Initiating Condition	Natural and Destructive Phenomena Affecting the Protected Area		Natural and Destructive Phenomena Affecting the Plant Vital Area
OPCON	All	All	All
EAL #	9.6.1.a	9.6.1.b	9.6.2
EMERGENCY ACTION LEVELS	IF Report of a Tornado TOUCHING DOWN within the Protected Area	IF **Sustained** wind speeds **> 75 MPH for 15 minutes**, measured at ANY elevation of the Met Tower THEN	The Wind Speed is of a magnitude that it SPECIFICALLY results in **Damage** to ANY one of the following: • TWO OR MORE subsystems of a **Safety System** • MORE THAN ONE **Safety System** • Rendering ANY of the following structures incapable of performing its Design Function: * Reactor Building * Control/Aux Building * Service Water Intake Structure * Service/Radwaste Building AND Damaged Safety System(s) or Plant Vital Structure is required for the present Operational Condition THEN
Action Required	Refer to Attachment 24 **UNUSUAL EVENT (Common Site)**		Refer to Attachment 2 **ALERT**

9.0 Hazards - Internal/External

9.7 Flooding

Initiating Condition	Internal Flooding in Excess of Sump Handling Capability Affecting Safety Related Areas of the Plant	Internal Flooding Affecting the Operabilityof Plant Safety Systems Required to Establish or Maintain Safe Shutdown
OPCON	All	All
EAL #	9.7.1	9.7.2
EMERGENCY ACTION LEVELS	IF Visual Observation of **Uncontrolled Flooding** that confirms ANY one of the following: • Reactor Building Floor Levels above the Maximum Normal Floor Level (>1") referenced in EOP 103, Secondary Containment Control • Receipt of a SSWS Pump Room Flooded Alarm • Greater than 2" of water in ANY area that contains a **Safety System(s)**, not included above	IF Visual Observation of Flooding within ANY one of the following Plant Vital Structures: • Reactor Building • Control/Aux Building • Service Water Intake Structure • Service/Rad Waste Building AND The Flooding is of a magnitude that it SPECIFICALLY results in **Damage** to ANY one of the following: • TWO OR MORE subsystems of a **Safety System** • MORE THAN ONE **Safety System** • Any of the above listed Plant Vital Structures which renders the structure incapable of performing its Design Function AND Damaged Safety System(s) or Plant Vital Structure is required for the present Operational Condition THEN
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**	Refer to Attachment 2 **ALERT**

9.0 Hazards - Internal/External

9.8 Turbine Failure / Vehicle Crash / Missile Impact

Initiating Condition	Natural and Destructive Phenomena Affecting Certain Structures Within the Protected Area		Natural and Destructive Phenomena Affecting Certain Structures Within the Plant Vital Area
OPCON	All	All	All
EAL #	9.8.1.a	9.8.1.b	9.8.2
EMERGENCY ACTION LEVELS	IF	IF	IF
	Catastrophic damage to the Main Turbine as evidenced by EITHER one of the following: • Main Turbine casing penetration • Main Turbine/Generator Damage potentially releasing Lube Oil or Hydrogen Gas to the Turbine Building	**Vehicle Crash / Missile Impact** with or within ANY one of the following Plant Structures: • Reactor Building • Turbine Building • Control/Aux Building • Service Water Intake Structure • Service/Rad Waste Building	**Vehicle Crash / Missile Impact** with or within ANY one of the following Plant Vital Structures: • Reactor Building • Control/Aux Building • Service Water Intake Structure • Service/Rad Waste Building
			AND
			The **Vehicle Crash / Missile Impact** is of a magnitude that it SPECIFICALLY results in **Damage** to ANY one of the following: • TWO OR MORE subsystems of a **Safety System** • MORE THAN ONE **Safety System** • Any of the above Plant Vital Structures which renders the structure incapable of performing its Design Function
			AND
			Damaged Safety System(s) or Plant Vital Structure is required for the present Operational Condition
	THEN		THEN
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**		Refer to Attachment 2 **ALERT**

9.0 Hazards - Internal/External

9.9 River Level

Initiating Condition

OPCON
EAL #

EMERGENCY ACTION LEVELS

Natural and Destructive Phenomena Affecting the Protected Area

All	All
9.9.1.a	**9.9.1.b**
IF	IF
River Level > **99.5'**	River Level < **80.0'**

THEN

Refer to Attachment 24
UNUSUAL EVENT (Common Site)

Action Required

11.0 Reportable Action Levels

11.1 Technical Specifications

Initiating Condition	INITIATION OF ANY UNIT SHUTDOWN REQUIRED BY THE TECHNICAL SPECIFICATIONS [10CFR50.72(b)(2)(i)]	EXCEEDING ANY TECHNICAL SPECIFICATION SAFETY LIMIT [10CFR50.36(c)(1), TS 6.7.1.a]	ANY DEVIATION FROM T/S OR LICENSE CONDITION PURSUANT TO 10CFR50.54(x) [10CFR50.72(b)(1)]
OPCON	1,2	1, 2, 3, 4, 5 (as applicable in T/S)	All
RAL #	**11.1.1.a**	**11.1.1.b**	**11.1.1.c**
REPORTING ACTION LEVELS	**IF** Unit shutdown is INITIATED to comply with Technical Specifications **THEN**	**IF** Exceeding ANY one of the following Technical Specification Safety Limits: • T/S 2.1.1, THERMAL POWER, Low Pressure or Low Flow • T/S 2.1.2, THERMAL POWER, High Pressure and High Flow • T/S 2.1.3, REACTOR COOLANT SYSTEM PRESSURE • T/S 2.1.4, REACTOR VESSEL WATER LEVEL **THEN**	**IF** Action required because no action consistent with Technical Specifications or license can provide adequate or equivalent protection in an emergency (See NC.NA-AP.ZZ-0005 (Q) for guidance on deviation from written procedures) **THEN**
Action Required	Refer to Attachment 14 4 Hour Report	Refer to Attachment 12 1 Hour Report	

11.0 Reportable Action Levels

11.1 Technical Specifications

Initiating Condition	VIOLATION OF THE REQUIREMENTS CONTAINED IN THE OPERATING LICENSE [HCGS Operating License, Sections 2.F]	ANY EVENT REQUIRING AN ENGINEERING EVALUATION BY TECHNICAL SPECIFICATIONS OR COMMITMENT [T/S 3.4.6.1, 3.4.4, 3.7.5]		
OPCON	All	All	All	All
RAL #	**11.1.3.a**	**11.1.3.b**	**11.1.3.c**	**11.1.3.d**
REPORTING ACTION LEVELS	**IF** Violation of ANY one of the requirements contained in Section 2.C of the Operating License EXCEPT as otherwise provided in the Technical Specifications or Environmental Protection Plan	**IF** Any of the T/S LCOs for RCS Pressure/ Temperature (T/S 3.4.6.1) are exceeded thereby requiring an Engineering Evaluation.	**IF** The conductivity, chloride concentration or pH in the RCS is in excess of its specified limits per T/S 3.4.4 Action Statement C.1 thereby requiring an Engineering Evaluation	**IF** One or more snubbers are found to be INOPERABLE and have been replaced or restored to an OPERABLE status, an Engineering Evaluation shall be performed per T/S 4.7.5.g
	THEN		**THEN**	
Action	Refer to Attachment 20 24 Hour Report		Refer to Attachment 22 OTHER Reports	

11.0 Reportable Action Levels

11.2 Degraded or Unanalyzed Condition

Initiating Condition	ANY EVENT OR CONDITION THAT RESULTS IN THE CONDITION OF THE PLANT BEING SERIOUSLY DEGRADED [10CFR50.72(b)(3)(ii)]	EVENT/CONDITION THAT COULD HAVE PREVENTED CERTAIN SAFETY FUNCTIONS [10CFR50.72 (b)(3)(v)]
OPCON	All	All
RAL #	**11.2.1**	**11.2.2.b**
REPORTING ACTION LEVELS	**IF** As judged by the OS/EDO, an event or condition that results in ANY one of the following: • The condition of the plant, including its principal safety barriers, being seriously degraded; • The plant being in an unanalyzed condition that significantly degrades plant safety.	**IF** Any event or condition that **at the time of discovery** could have prevented the fulfillment of the safety function of structures or systems that are needed to perform ANY one of the following: A. Shutdown the reactor and maintain it in a safe shutdown condition B. Remove residual heat. C. Control the release of radioactive material D. Mitigate the consequences of an accident
	THEN	**THEN**
Action Required	Refer to Attachment 26	

11.0 Reportable Action Levels

11.3 System Actuations

Initiating Condition	EVENT THAT RESULTS OR SHOULD HAVE RESULTED IN ECCS DISCHARGE INTO RCS AS RESULT OF A VALID SIGNAL EXCEPT PREPLANNED [10CFR50.72(b)(2)(iv)(A)]	ACTUATION OF REACTOR PROTECTION SYSTEM WHEN CRITICAL EXCEPT PREPLANNED [10CFR50.72(b)(2)(iv)(B)]	VALID ACTUATION OF LISTED SYSTEM EXCEPT PREPLANNED [10CFR50.72(b)(3)(iv)(A)]
OPCON	All	1,2	All
RAL #	**11.3.1**	**11.3.2**	**11.3.3**
REPORTING ACTION LEVELS	**IF** Valid ECCS Actuation, Manual or Automatic, has or should have occurred **AND** ECCS results, or should have resulted, in discharge to the vessel **AND** Actuation is NOT part of a pre-planned sequence during testing or reactor operation. **THEN**	**IF** Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. **THEN**	**IF** Any event or condition that results in valid actuation of any system listed in Technical Basis 11.3.3 except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. **THEN**
Action Required	Refer to Attachment 14 4 Hour Report		Refer to Attachment 26 8 Hour Report

11.0 Reportable Action Levels

11.4 Personnel Safety / Overexposure

Initiating Condition	ANY INCIDENT OR EVENT INVOLVING BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL CAUSING ANY OF THE LISTED RESULTS[10CFR20.2202(a)]	ANY INCIDENT OR EVENT INVOLVING LOSS OF CONTROL OF LICENSED MATERIAL CAUSING ANY OF THE LISTED RESULTS [10CFR20.2202(b)]	ONSITE FATALITY [10CFR50.72(b)(2)(xi)]
OPCON	All	All	All
RAL #	11.4.1	11.4.2.a	11.4.2.b
REPORTING ACTION LEVELS	IF	IF	IF
	PERSONNEL OVEREXPOSURE or potential for overexposure as indicated by ANY one of the following: • **TEDE exposure ≥ 25 Rem** • **LDE exposure ≥ 75 Rem** • **SDE exposure ≥ 250 Rem** • Release of radioactive material inside or outside of a Restricted Area so that had an individual been present for 24 hours the individual could have received ≥ **5** times the occupational ALI (Annual Limit of Intake) which would usually equate to **25 Rem CEDE**. This Does NOT apply to areas where personnel are NOT normally stationed during routine operations.	PERSONNEL OVEREXPOSURE or potential for overexposure within a 24 hour period, as indicated by ANY one of the following: • **TEDE exposure > 5 Rem** • **LDE exposure > 15 Rem** • **SDE exposure > 50 Rem** • Release of radioactive material inside or outside of a Restricted Area so that had an individual been present for 24 hours the individual could have received > **1** times the occupational ALI (Annual Limit of Intake) which would usually equate to **5 Rem CEDE**. This Does NOT apply to areas where personnel are NOT normally stationed during routine operations.	Any fatality has occurred within the Owner Controlled Area (OCA)
	THEN	THEN	THEN
Action Required	Refer to Attachment 12 1 Hour Report	Refer to Attachment 14 4 Hour Report	Refer to Attachment 17 4 Hour Report

11.0 Reportable Action Levels

11.4 Personnel Safety / Overexposure

REPORTING ACTION LEVELS

Initiating Condition	RADIOACTIVELY CONTAMINATED PERSON TRANSPORTED FROM THE SITE TO AN OFFSITE MEDICAL FACILITY FOR TREATMENT [10CFR50.72(b)(3)(xii)]	SIGNIFICANT FITNESS FOR DUTY EVENTS [10CFR26.73]	FITNESS FOR DUTY PROGRAM: FALSE POSITIVE DUE TO ADMINISTRATIVE ERROR (BLIND TEST BY LAB) [10CFR26, APP. A, 2.8(e)(5)]
OPCON	All	All	All
RAL #	**11.4.2.c**	**11.4.3.a**	**11.4.3.b**
	IF	**IF**	**IF**
	Transportation of a radioactively contaminated or **potentially contaminated individual** from the site to an offsite medical facility for treatment	Any event that is determined to be reportable by the Medical Review Officer (MRO) or designee IAW PSEG Nuclear's Fitness for Duty Program (NC.NA-AP.ZZ-0042(Q))	The occurrence of a false positive error on a blind lab performance test specimen under 10CFR26 as determined by the Medical Review Officer (MRO) IAW PSEG Nuclear's Fitness for duty Program (NC.NA-AP.ZZ-0042(Q)
	THEN	**AND**	**AND**
		The reportable details of the event are made available to the OS by the MRO or designee	The reportable details of the event are made available to the OS by the MRO or designee
		THEN	**THEN**
Action	Refer to Attachment 27 8 Hour Report	Refer to Attachment 19 24 Hour Report	Refer to Attachment 19 24 Hour Report

11.0 Reportable Action Levels

11.5 Environmental / State Notifications

Initiating Condition	SPILL/DISCHARGE OF ANY NON-RADIOACTIVE HAZARDOUS SUBSTANCE [10CFR50.72(b)(2)(xi); N.J.A.C. 7:1E]	SPILL/DISCHARGE OF ANY NON-RADIOACTIVE HAZARDOUS SUBSTANCE INTO OR UPON THE RIVER [10CFR50.72(b)(2) (xi); N.J.A.C.7:1E]	UNUSUAL OR IMPORTANT ENVIRONMENTAL EVENTS [E.P.P. SECTION 4.1]
OPCON	All	All	All
RAL #	**11.5.2.a**	**11.5.2.b**	**11.5.2.c**
REPORTING ACTION LEVELS	**IF**	**IF**	**IF**
	Spill/discharge of an industrial chemical or petroleum product outside of a plant structure within the Owner Controlled Area (OCA) that results in EITHER one of the following: • Spill / discharge that has passed through the engineered fill and into the ground water as confirmed by licensing • Spill / discharge that CANNOT be cleaned up within **24 hours** and no contact with groundwater is suspected	EITHER one of the following events occur: • Observation of a spill/discharge of an industrial chemical or petroleum product from on-site into the Delaware River or into a storm drain • Observation of an oil slick on the Delaware River from any source	As judged by the OS/EDO, ANY one of the following events has occurred: • Unusually large fish kill • Protected aquatic species impinge on Circulating or Service Water intake screens (eg.; sea turtle, sturgeon) as reported by Site personnel • Any occurrence of an unusual or important event that indicates or could result in significant environmental impact casually related to plant operation; such as the following: ◆ Onsite plant or animal disease outbreaks • Mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973 • Increase in nuisance organisms or conditions • Excessive bird impactation ◆ NJPDES Permit violations ◆ Excessive Opacity (smoke)
	THEN	**THEN**	**THEN**
	Note: This event MAY require IMMEDIATE (15 minute) notifications. DO NOT delay implementation of Attachment 16.		
Action Required	Refer to Attachment 16 Spill/Discharge Reporting	Refer to Attachment 16 Spill/Discharge Reporting	Refer to Attachment 15 Environmental Protection Plan

11.0 Reportable Action Levels

11.5 Environmental / State Notifications

Initiating Condition

OPCON

RAL #

REPORTING ACTION LEVELS

Action Required

BOILER OR PRESSURE VESSEL EXPLOSION OR PERSONAL INJURY [N.J.A.C. 5:11-3.11]

All

11.5.3

IF

EITHER one of the following events occur:

- Personal injury due to an **occurrence** to a boiler or pressure vessel
- A boiler or pressure vessel explosion

THEN

Refer to Attachment 28 B&PV Reporting

11.0 Reportable Action Levels

11.6 After-the-Fact

Initiating Condition

EMERGENCY CONDITIONS DISCOVERED AFTER-THE-FACT

MODE

All

RAL #

11.6.1

REPORTING ACTION LEVELS

IF

Discovery of events or conditions that had previously occurred (event was NOT ongoing at the time of discovery) which EXCEEDED an Emergency Action Level (EAL) and was NOT declared as an emergency

AND

More than ONE HOUR has elapsed since the condition occurred

AND

There are currently NO adverse consequences in progress as a result of the event

THEN

Action Required

Refer to Attachment 12
1 Hour Report

11.0 Reportable Action Levels

11.7 Security / Emergency Response Capabilities

REPORTING ACTION LEVELS

Initiating Condition	SAFEGUARDS EVENTS THAT ARE DETERMINED TO BE NON-EMERGENCIES, BUT ARE REPORTABLE TO THE NRC WITHIN ONE HOUR [10CFR73.71(b)(1)]	MAJOR LOSS OF EMERGENCY ASSESSMENT CAPABILITY, OFFSITE RESPONSE CAPABILITY, OR COMMUNICATIONS CAPABILITY [10CFR50.72(b)(3)(xiii)]	
OPCON	All	All	All
RAL #	**11.7.1.a**	**11.7.1.b**	**11.7.1.c**
	IF	**IF**	**IF**
	Any Non-Emergency safeguards event that is reportable in accordance with 10CFR73.71 as determined by Security (SCP-15)	OS/EC determines that an event(s) (excluding a scheduled test or preplanned maintenance activity) has occurred that would impair the ability to deal with an accident or emergency as indicated by the Loss of ANY one of the following:	
	THEN	• Nuclear Emergency Telecommunications System (NETS) for > **1 hr** • ENS for > **1 hr** in the Control Room, TSC, and EOF (N/A if reported by the NRC). • More than 17 Offsite Sirens for > **1 hr** • Use of the EOF for > **8 hrs** • All Meteorological data (Hope Creek AND Salem) for one parameter for > **8 hrs** • Site access due to Acts of Nature (snow, flood, etc.)	• Use of the TSC for > **8 hrs** • SPV, NPV, or FRVS vent radiation effluent monitors with no alternate method of monitoring for > **72 hrs** • SPDS OR CRIDS for > **8 hrs** • More than > 75% OHA's • Concurrent multiple accident or emergency condition indicators which in the judgement of the OS significantly impairs assessment capabilities Refer to Technical Basis 11.7.1.c for ERDS
		THEN	**THEN**
Action Required	Refer to Attachment 11 **1 Hour Report (Common Site)**	Refer to Attachment 25 **8 Hour Report (Common Site)**	Refer to Attachment 26 **8 Hour Report**

11.0 Reportable Action Levels

11.8 Public Interest

Initiating Condition

OPCON

RAL #

REPORTING ACTION LEVELS

Action Required

UNUSUAL CONDITIONS WARRANTING A NEWS RELEASE OR NOTIFICATION OF GOVERNMENT AGENCIES [10CFR50.72(b)(2)(xi)]

All

11.8.2.a

IF

OS/EDO judges that an event or situation has occurred that is related to ANY one of the following:

- The health and safety of the public
- The health and safety of onsite personnel
- Protection of the environment

AND

A news release is planned

AND

Notifications to a Local, State or Federal agency has been or will be made

THEN

Refer to Attachment 14

UNUSUAL CONDITIONS DIRECTLY AFFECTING LOWER ALLOWAYS CREEK TOWNSHIP (LACT) [LAC -MOU]

All

11.8.2.b

IF

As judged by the OS/EDO, events which are the responsibility of PSEG Nuclear which have or may result in EITHER one of the following:

- Anticipated unusual movement of equipment or personnel which may significantly affect local traffic patterns
- Onsite events which involve alarms, sirens or other noise which may be heard off-site

THEN

Refer to Attachment 21

11.0 Reportable Action Levels

11.9 Accidental Criticality / Special Nuclear Material / Rad Material Shipments - Releases

REPORTING ACTION LEVELS

Initiating Condition	UNPLANNED / ACCIDENTAL CRITICALITY [10CFR70.52(a)]	LOSS AND INVESTIGATION OF THE LOSS OF SPECIAL NUCLEAR MATERIALS/ SPENT FUEL [10CFR70.52(a); 10CFR70.52(b), 10CFR73.71(a)]	THEFT OR LOSS OF LICENSED MATERIAL [10CFR20.2201(a)(1)(i)]
OPCON	All	All	All
RAL #	**11.9.1.a**	**11.9.1.b**	**11.9.1.c**
	IF Any unplanned or accidental criticality **THEN**	**IF** ANY one of the following events occur involving **Special Nuclear Material** (SNM) or Spent Fuel: • Loss, other than normal operating loss, of **SNM** • Actual or attempted theft or unlawful diversion of **SNM** • Shipment of **SNM of low strategic significance** or Spent Fuel that is lost or unaccounted for after the estimated time of arrival • A lost or unaccounted for shipment of **SNM of low strategic significance** or Spent Fuel has been recovered or accounted for **THEN**	**IF** Lost, stolen or missing **licensed material > 1000** times the quantity specified in 10CFR20 Appendix C in such circumstances that an exposure could result to persons in **Unrestricted Areas**. **THEN**
Action Required	Refer to Attachment 12 1 Hour Report	Refer to Attachment 11 **1 Hour Report (Common Site)**	Refer to Attachment 11 **1 Hour Report (Common Site)**

11.0 Reportable Action Levels

11.9 Accidental Criticality / Special Nuclear Material / Rad Material Shipments - Releases

Initiating Condition

OPCON

RAL #

REPORTING ACTION LEVELS

Action Required

EXCESSIVE CONTAMINATION AND/OR RADIATION LEVELS ON A PACKAGE [10CFR20.1906(d)]

All

11.9.1.e

IF

Receipt survey indicates that package contamination/radiation levels equal or exceeds ANY one of the following:

- **2200 dpm/100 cm²**
- **200 mR/hr on contact**
- **10 mR/hr at 3 feet**

THEN

Refer to Attachment 10
1 Hour Report

ACCIDENT DURING TRANSPORT OF LICENSED MATERIAL [10CFR71.5(a)(1)(iv)]

All

11.9.2.a

IF

Accidents during the transportation of **radioactive material** which are reported to PSE&G as the shipper that involve (or potentially involve) damage to the cargo

THEN

Refer to Attachment 18
4 Hour Report

CONTAMINATION OUTSIDE OF THE RADIOLOGICALLY CONTROLLED AREA [10CFR50.72(b)(2)(xi)]

All

11.9.2.b

IF

Discovery of a Contaminated Area OUTSIDE of the RCA with removable activity due to **licensed material**

AND

Location of Contaminated Area is OUTSIDE of Plant Structures

AND

Size of Contaminated Area is LARGE (>100 FT²)

AND

Location of Contaminated Area is such that a contaminated person or material may have left the Protected Area

THEN

Refer to Attachment 13
4 Hour Report

11.0 Reportable Action Levels

11.10 Voluntary Notifications

Initiating Condition

OPCON

RAL #

REPORTING ACTION LEVELS

Events/conditions warrant voluntary/courtesy NRC notification [10CFR50.72 - Voluntary Report]

All

11.10.2

IF

In the judgement of the OS, notification to the NRC is warranted

AND

NO other EALs or RALs appear to be applicable

THEN

Refer to Attachment 14

Action

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PSEG NUCLEAR LLC
DOCKET NO. 50-354
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE

License No. NPF-57

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for a license filed by the Public Service Electric & Gas Company, acting on behalf of itself and Atlantic City Electric Company (the licensees), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Hope Creek Generating Station (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-120 and the application, as amended, the provisions of the Act and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

   E. PSEG Nuclear LLC is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

PSE&G
CONTROL
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H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-57, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings and approval by the Nuclear Regulatory Commission at a meeting on July 21, 1986, the License for Fuel Loading and Low Power Testing, License No. NPF-50, issued on April 11, 1986, is superseded by Facility Operating License NPF-57 hereby issued to PSEG Nuclear LLC (the licensee), to read as follows:

A. This license applies to the Hope Creek Generating Station, a boiling water nuclear reactor, and associated equipment (the facility) owned by PSEG Nuclear LLC. The facility is located on the licensee's site on the east bank of the Delaware River in Lower Alloways Creek Township, Salem County, New Jersey. The facility is located approximately eight miles southwest of Salem, New Jersey and is described in the PSEG Nuclear LLC Final Safety Analysis Report, as supplemented and amended, and in the Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) PSEG Nuclear LLC, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the above designated location in Salem County, New Jersey, in accordance with the procedures and limitations set forth in this license;

(2) Deleted

(3) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3339 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No. 4)*

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(4) Inservice Inspection (Section 6.6, SER; Sections 5.2.4.3 and 6.6.3, SSER No. 5)

a. PSE&G shall submit an inservice inspection program in accordance with 10 CFR 50.55a(g)(4) for staff review by October 11, 1986.

b. Pursuant to 10 CFR 50.55a(a)(3) and for the reasons set forth in Sections 5.2.4.3 and 6.6.3 of SSER No. 5, the relief identified in the PSE&G submittal dated November 18, 1985, as revised by the submittal dated January 20, 1986, requesting relief from certain requirements of 10 CFR 50.55a(g) for the preservice inspection program, is granted.

(5) Solid State Logic Modules

PSEG Nuclear LLC shall continue, for the life of the plant, a reliability program to monitor the performance of the Bailey 862 SSLMs installed at Hope Creek Generating Station. This program should obtain reliability data, failure characteristics, and root cause of failure of both safety-related and non-safety-related Bailey 862 SSLMs. The results of the reliability program shall be maintained on-site and made available to the NRC upon request.

(6) Fuel Storage and Handling (Section 9.1, SSER No. 5)

a. No more than a total of three (3) fuel assemblies shall be out of approved shipping containers or fuel assembly storage racks or the reactor at any one time.

b. The above three (3) fuel assemblies as a group shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and the storage rack array.

c. Fresh Fuel assemblies, when stored in their shipping containers, shall be stacked no more than three (3) containers high.

(7) Fire Protection (Section 9.5.1.8, SSER No. 5; Section 9.5.1, SSER No. 6)

PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 15 and as described in its submittal dated May 13, 1986, and as approved in the SER dated October 1984 (and Supplements 1 through 6) subject to the following provision:

PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(8) Solid Waste Process Control Program (Section 11.4.2, SER; Section 11.4, SSER No. 4)

PSEG Nuclear shall obtain NRC approval of the Class B and C solid waste process control program prior to processing Class B and C solid wastes.

(9) Emergency Planning (Section 13.3, SSER No. 5)

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

(10) Initial Startup Test Program (Section 14, SSER No. 5)

Any changes to the Initial Startup Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(11) Partial Feedwater Heating (Section 15.1, SER; Section 15.1, SSER No. 5; Section 15.1, SSER No. 6)

The facility shall not be operated with reduced feedwater temperature for the purpose of extending the normal fuel cycle. After the first operating cycle, the facility shall not be operated with a feedwater heating capacity that would result in a rated power feedwater temperature less than 400°F unless analyses supporting such operation are submitted by the licensee and approved by the staff.

(12) Detailed Control Room Design Review (Section 18.1, SSER No. 5)

a. PSE&G shall submit for staff review Detailed Control Room Design Review Summary Reports II and III on a schedule consistent with, and with contents as specified in, its letter of January 9, 1986.

b. Prior to exceeding five percent power, PSE&G shall provide temporary zone markings on safety-related instruments in the control room.

(13) Safety Parameter Display System (Section 18.2, SSER No. 5)

Prior to the earlier of 90 days after restart from the first refueling outage or July 12, 1988, PSE&G shall add the following parameters to the SPDS and have them operational:

a. Primary containment radiation
b. Primary containment isolation status
c. Combustible gas concentration in primary containment
d. Source range neutron flux

(14) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 135, are hereby incorporated into this license. PSEG Nuclear LLC shall operate the facility in accordance with the Additional Conditions.

(15) PSE&G to PSEG Nuclear LLC License Transfer Conditions

a. PSEG Nuclear LLC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application, the requirements of the Order Approving Transfer of License and Conforming Amendment, dated February 16, 2000, and the related Safety Evaluation dated February 16, 2000.

b. The decommissioning trust agreement shall provide that:

1) The use of assets in both the qualified and non-qualified funds shall be limited to expenses related to decommissioning of the unit as defined by the NRC in its regulations and issuances, and as provided in the unit's license and any amendments thereto. However, upon completion of decommissioning, as defined above, the assets may be used for any purpose authorized by law.

2) Investments in the securities or other obligations of PSE&G or affiliates thereof, or their successors or assigns, shall be prohibited. In addition, except for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants shall be prohibited.

3) No disbursements or payments from the trust shall be made by the trustee until the trustee has first given the NRC 30 days notice of the payment. In addition, no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the Director, Office of Nuclear Reactor Regulation.

4) The trust agreement shall not be modified in any material respect without prior written notification to the Director, Office of Nuclear Reactor Regulation.

5) The trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investor" standard, as specified in 18 CFR 35.32(3) of the Federal Energy Regulatory Commission's regulations.

c. PSEG Nuclear LLC shall not take any action that would cause PSEG Power LLC or its parent companies to void, cancel, or diminish the commitment to fund an extended plant shutdown as represented in the application for approval of the transfer of this license from PSE&G to PSEG Nuclear LLC.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. An exemption from the criticality alarm requirements of 10 CFR 70.24 was granted in Special Nuclear Material License No. 1953, dated August 21, 1985. This exemption is described in Section 9.1 of Supplement No. 5 to the SER. This previously granted exemption is continued in this operating license. An exemption from certain requirements of Appendix A to 10 CFR Part 50, is described in Supplement No. 5 to the SER. This exemption is a schedular exemption to the requirements of General Design Criterion 64, permitting delaying functionality of the Turbine Building Circulating Water System-Radiation Monitoring System until 5 percent power for local indication, and until 120 days after fuel load for control room indication (Appendix R of SSER 5). Exemptions from certain requirements of Appendix J to 10 CFR Part 50, are described in Supplement No. 5 to the SER. These include an exemption from the requirement of Appendix J, exempting main steam isolation valve leak-rate testing at 1.10 Pa (Section 6.2.6 of SSER 5); an exemption from Appendix J, exempting Type C testing on traversing incore probe system shear valves (Section 6.2.6 of SSER 5); an exemption from Appendix J, exempting Type C testing for instrument lines and lines containing excess flow check valves (Section 6.2.6 of SSER 5); and an exemption from Appendix J, exempting Type C testing of thermal relief valves (Section 6.2.6 of SSER 5). These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. These exemptions are hereby granted. The special circumstances regarding each exemption are identified in the referenced section of the safety evaluation report and the supplements thereto. These exemptions are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Salem-Hope Creek Nuclear Generating Station Security Plan," with revisions submitted through December 17, 2001; "Salem-Hope Creek Nuclear Generating Station Security Training and Qualification Plan," with revisions submitted through December 17, 2001; and "Salem-Hope Creek Nuclear Generating Station Security Contingency Plan," with revisions submitted through June 2, 1998. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, PSEG Nuclear LLC shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

G. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at midnight on April 11, 2026.

FOR THE NUCLEAR REGULATORY COMMISSION

- original signed by H.R. Denton -

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendix A - Technical Specifications (NUREG-1202)
2. Appendix B - Environmental Protection Plan

Date of Issuance: July 25, 1986

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days during POWER OPERATION.

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

   1. Within one hour:

      a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions.

      b) Disarm the associated directional control valves** hydraulically by closing the drive water and exhaust water isolation valves.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

   2. Restore the inoperable control rod, if withdrawn, to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above:

   1. If the inoperable control rod(s) is withdrawn, within one hour:

      a) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, and

      b) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range*

      Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves** either:

      a) Electrically, or
      b) Hydraulically by closing the drive water and exhaust water isolation valves.

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*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves** either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

3. The provisions of Specification 3.0.4 are not applicable.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

d. With one scram discharge volume vent valve and/or one scram discharge volume drain valve inoperable and open, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours.

e. With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore the inoperable valve(s) to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. At least once per 24 hours verifying each valve to be open,* and

b. At least once per 31 days cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed

*These valves may be closed intermittently for testing under administrative controls.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

SURVEILLANCE REQUIREMENTS (Continued)

electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. Within 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE at least once per 18 months, by verifying that the drain and vent valves:

1. Close within 30 seconds after receipt of a signal for control rods to scram, and

2. Open when the scram signal is reset.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

   1. Declare the control rod(s) with the slow insertion time inoperable, and

   2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days.

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.3 The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.43
39	0.86
25	1.93
05	3.49

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the average scram insertion times of control rods exceeding the above limits:

   1. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

   2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

3.1.3.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

----------NOTE----------

Separate condition entry is allowed for each control rod

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a. In OPERATIONAL CONDITIONS 1 or 2:

1. With one control rod scram accumulator inoperable and reactor pressure ≥ 900 psig, within 8 hours,

   a) Restore the inoperable accumulator to OPERABLE status, or

   b) Insert the associated control rod, declare the associated control rod inoperable and disarm the associated control valves either electrically or hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN with the next 12 hours.

2. With two or more control rod scram accumulators inoperable and reactor pressure ≥ 900 psig,

   a) Within 20 minutes of discovery of this condition concurrent with charging water pressure < 940 psig, restore charging water header pressure to ≥ 940 psig otherwise place the mode switch in the shutdown position**, and

   b) Within one hour insert the associated control rods, declare the associated control rods inoperable and disarm the associated control valves either electrically or hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

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* At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

LIMITING CONDITION FOR OPERATION (Continued)
ACTION (Continued)

3. With one or more control rod scram accumulators inoperable and reactor pressure < 900 psig,

   a) Immediately upon discovery of charging water header pressure < 940 psig, verify all control rods associated with inoperable accumulators are fully inserted otherwise place the mode switch in the shutdown position**, and

   b) Within one hour insert the associated control rod(s), declare the associated control rod(s) inoperable and disarm the associated control valves either electrically or hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

   1. With one or more withdrawn control rods inoperable, upon discovery immediately initiate action to fully insert inoperable withdrawn control rods.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed.

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* At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

LIMITING CONDITION FOR OPERATION

3.1.3.6 All control rods shall be coupled to their drive mechanisms.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 and 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

   1. If permitted by the RWM, insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

      a) Observing any indicated response of the nuclear instrumentation, and
      b) Demonstrating that the control rod will not go to the overtravel position.

   2. If recoupling is not accomplished on the first attempt or, if not permitted by the RWM, then until permitted by the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or
      b) Hydraulically by closing the drive water and exhaust water isolation valves.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours either:

   1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or

   2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or
      b) Hydraulically by closing the drive water and exhaust water isolation valves.

c. The provisions of Specification 3.0.4 are not applicable.

---

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.3.6 Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable, within 1 hour:

   1. Determine the position of the control rod by using an alternate method, or:

      a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,
      b) Returning the control rod, by single notch movement, to its original position, and
      c) Verifying no control rod drift alarm at least once per 12 hours, or

   2. Move the control rod to a position with an OPERABLE position indicator, or

   3. When THERMAL POWER is:

      a) Within the preset power level of the RWM, declare the control rod inoperable.
      b) Greater than the preset power level of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

         1) Electrically, or
         2) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

c. The provisions of Specification 3.0.4 are not applicable.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

a. At least once per 24 hours that the position of each control rod is indicated,

b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and

c. That the control rod position indicator corresponds to the control rod position indicated by the "Full Out" position indicator when performing Surveillance Requirement 4.1.3.6.b.

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 The control rod drive housing support shall be in place.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.8 The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The Rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#, when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER, minimum allowable low power setpoint.

ACTION:

a. With the RWM inoperable after the first 12 control rods are fully withdrawn, operation may continue provided that control rod movement and compliance with the prescribed control rod pattern are verified by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console.

b. With the RWM inoperable before the first twelve (12) control rods are fully withdrawn, one startup per calendar year may be performed provided that the control rod movement and compliance with the prescribed control rod pattern are verified by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console.

c. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 within 8 hours prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper indication of the selection error of at least one out-of-sequence control rod.

* Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.
# See Special Test Exception 3.10.2.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

SURVEILLANCE REQUIREMENTS (CONTINUED)

b. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

c. In OPERATIONAL CONDITION 1 within one hour after RWM automatic initiation when reducing THERMAL POWER, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

d. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

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ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

The material originally contained in Section 3/4.1.4.2 was deleted with the issuance of Amendment No. . However, to maintain numerical continuity between the succeeding sections and existing station procedural references to those Technical Specification sections, 3/4.1.4.2 has been intentionally left blank.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With one RBM channel inoperable:

   1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

   2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system consists of two redundant subsystems and shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one system subsystem inoperable, restore the subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With both system subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one system subsystem inoperable, restore subsystem to OPERABLE status within 30 days or insert all insertable control rods within the next hour.

2. With both standby liquid control system subsystems inoperable, insert all insertable control rods within one hour.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

1. The temperature of the sodium pentaborate solution in the storage tank is greater than or equal to 70°F.

2. The available volume of sodium pentaborate solution is within the limits of Figure 3.1.5-1.

3. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping to be greater than or equal to 70°F.

*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

2. Determining that the available weight of sodium pentaborate is greater than or equal to 5,776 lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

3. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm, per pump, at a pressure of greater than or equal to 1255 psig is met.

d. At least once per 18 months during shutdown by:

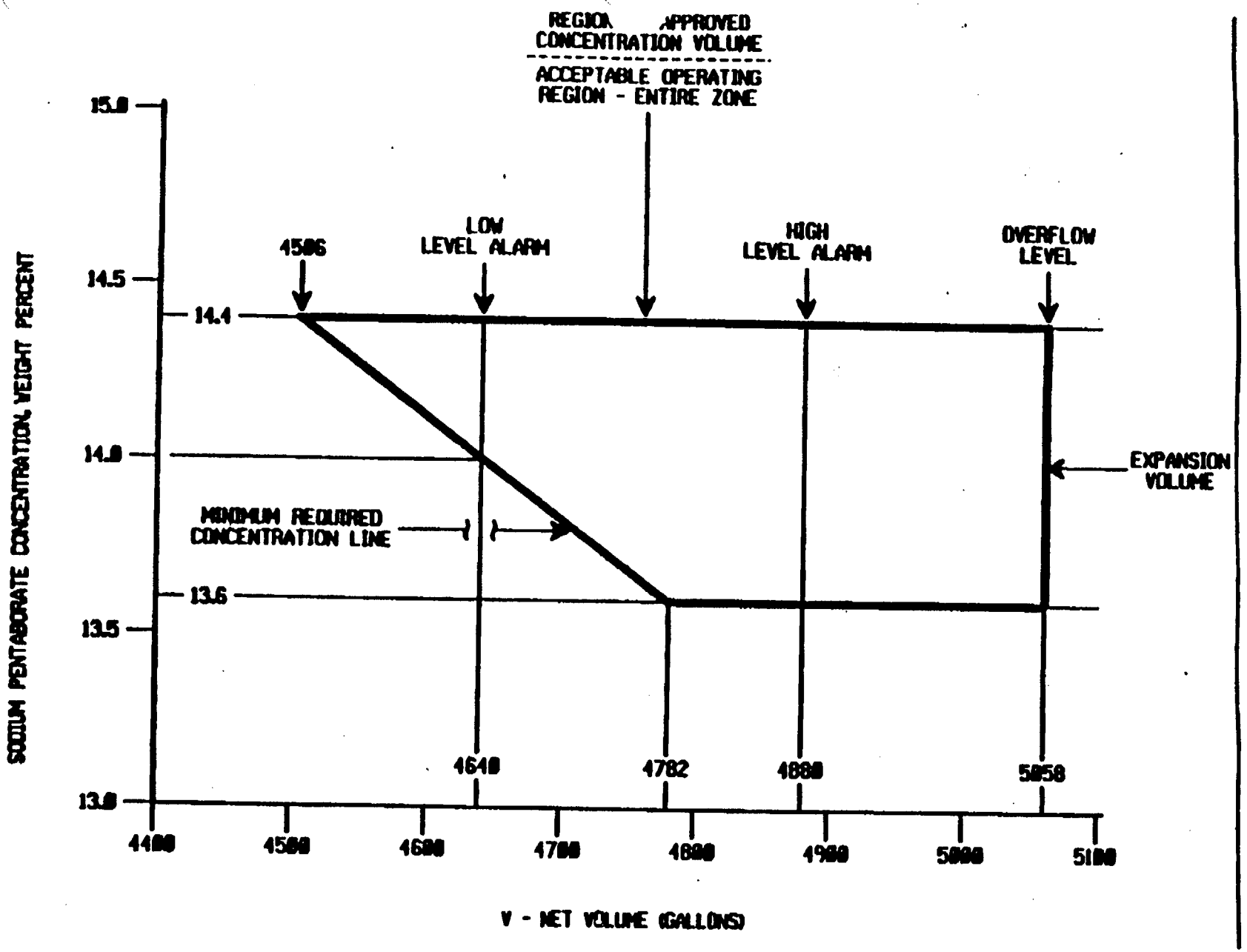
1. Initiating one of the standby liquid control system subsystem, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel and verifying that the relief valve does not actuate. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection subsystems shall be tested in 36 months.

2. **Demonstrating that all heat traced piping between the storage tank and the injection pumps is unblocked and then draining and flushing the piping with demineralized water.

3. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise of the sodium pentaborate solution in the storage tank after the heaters are energized.

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 70°F.

**This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.



SODIUM PENTABORATE SOLUTION
VOLUME/CONCENTRATION REQUIREMENTS

FIGURE 3.1.5-1

Q4, 23 remove note

3/4.3 INSTRUMENTATION
3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within twelve hours. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. For the Reactor Vessel Steam Dome Pressure - High Functional Unit and the Reactor Vessel Water Level - Low, Level 3 Functional Unit, the sensor is eliminated from response time testing for RPS circuits. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

4.3.1.4 The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 or 3 from OPERATIONAL CONDITION 1 for the Intermediate Range Monitors.

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors[b]:			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	5[c]	3[d]	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3[d]	3
2. Average Power Range Monitor[e]:			
a. Neutron Flux - Upscale, Setdown	2	2	1
	3, 4	2	2
	5[c]	2[d]	3
b. Flow Biased Simulated Thermal Power - Upscale	1	2	4
c. Fixed Neutron Flux - Upscale	1	2	4
d. Inoperative	1, 2	2	1
	3, 4	2	2
	5[c]	2[d]	3
3. Reactor Vessel Steam Dome Pressure - High	1, 2[f]	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1[g]	4	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. This item intentionally blank			
7. Drywell Pressure - High	1, 2[h]	2	1
8. Scram Discharge Volume Water Level - High			
a. Float Switch	1, 2	2	1
	5[i]	2	3
b. Level Transmitter/Trip Unit	1, 2	2	1
	5[i]	2	3
9. Turbine Stop Valve - Closure	1[j]	4[k]	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	1[j]	2[k]	6
11. Reactor Mode Switch Shutdown Position	1, 2	2	1
	3, 4	2	7
	5	2	3
12. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1	-	Be in at least HOT SHUTDOWN within 12 hours.
ACTION 2	-	Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.
ACTION 3	-	Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.
ACTION 4	-	Be in at least STARTUP within 6 hours.
ACTION 5	-	This ACTION is deleted
ACTION 6	-	Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.
ACTION 7	-	Verify all insertable control rods to be inserted within one hour.
ACTION 8	-	Lock the reactor mode switch in the Shutdown position within one hour.
ACTION 9	-	Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within one hour.

*Except replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn*.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per the Trip System are 4 APRMS, 6 IRMS and 2 SRMS.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is ≤ 159.7 psig equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration, and drift, a setpoint of ≤ 135.7 psig is used.

(k) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Q 29 No TS reference

ELECTRICAL POWER SYSTEMS

DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, 2 of the 4 channels, one of which shall be channel A or channel B, of the power distribution system shall be energized with:

a. A.C. power distribution:

1. Channel A, consisting of:

a)	4160 volt A.C. switchgear bus	10A401
b)	480 volt A.C. load centers	10B410
		10B450
c)	480 volt A.C. MCCs	10B212
		10B411
		10B451
		10B553
d)	208/120 volt A.C. distribution panels	10Y401(source:10B411)
		10Y411(source:10B451)
		10Y501(source:10B553)
e)	120 volt A.C. distribution panels	1AJ481
		1YF401(source:1AJ481)
		1AJ482

2. Channel B, consisting of:

a)	4160 volt A.C. switchgear bus	10A402
b)	480 volt A.C. load centers	10B420
		10B460
c)	480 volt A.C. MCCs	10B222
		10B421
		10B461
		10B563
d)	208/120 volt A.C. distribution panels	10Y402(source:10B421)
		10Y412(source:10B461)
		10Y502(source:10B563)
e)	120 volt A.C. distribution panels	1BJ481
		1YF402(source:1BJ481)
		1BJ482

3. Channel C, consisting of:

a)	4160 volt A.C. switchgear bus	10A403
b)	480 volt A.C. load centers	10B430
		10B470
c)	480 volt A.C. MCCs	10B232
		10B431
		10B471
		10B573
d)	208/120 volt A.C. distribution panels	10Y403(source:10B431)
		10Y413(source:10B471)
		10Y503(source:10B573)

e)	120 volt A.C. distribution panels	1CJ481 1YF403(source:1CJ481) 1CJ482

4. Channel D, consisting of:

a)	4160 volt A.C. switchgear bus	10A404
b)	480 volt A.C. load centers	10B440 10B480
c)	480 volt A.C. MCCs	10B242 10B441 10B481 10B583
d)	208/120 volt A.C. distribution panels	10Y404(source:10B441) 10Y414(source:10B481) 10Y504(source:10B583)
e)	120 volt A.C. distribution panels	1DJ481 1YF404(source:1DJ481) 1DJ482

b. D.C. power distribution:

1. Channel A, consisting of:

a)	125 volt D.C. switchgear	10D410
b)	125 volt D.C. fuse box	1AD412
c)	125 volt D.C. distribution panel	1AD417

2. Channel B, consisting of:

a)	125 volt D.C. switchgear	10D420
b)	125 volt D.C. fuse box	1BD412
c)	125 volt D.C. distribution panel	1BD417

3. Channel C, consisting of:

a)	125 volt D.C. switchgear	10D430 10D436
b)	125 volt D.C. fuse boxes	1CD412 1CD448
c)	125 volt D.C. distribution panel	1CD417

4. Channel D, consisting of:

a)	125 volt D.C. switchgear	10D440 10D446
b)	125 volt D.C. fuse box	1DD412 1DD448
c)	125 volt D.C. distribution panel	1DD417

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than two channels of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. With less than two channels of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system channels shall be determined energized at least once per 7 days by verifying correct breaker/switch alignment and voltage on the busses/MCCs/panels.

*When handling irradiated fuel in the secondary containment.

3/4 9 2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9 2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:**

a. Annunciation and continuous visual indication in the control room.

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn.#

d. During a SPIRAL UNLOAD, the count rate may drop below 3 cps when the number of assemblies remaining in the core drops to sixteen or less.

e. During a SPIRAL RELOAD, up to four fuel assemblies may be loaded in the four bundle locations immediately surrounding each of the four SRMs prior to obtaining 3 cps. Until these assemblies have been loaded, the 3 cps count rate is not required.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

#Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

**Three SRM channels shall be OPERABLE for critical shutdown margin demonstrations. An SRM detector may be retracted provided a channel indication of at least 100 cps is maintained.

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SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the detectors are inserted to the normal operating level, and

3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verifying that the channel count rate is at least 3 cps.

1. Prior to control rod withdrawal,

2. Prior to and at least once per 12 hours during CORE ALTERATIONS***, and

3. At least once per 24 hours***.

d. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, verifying that the RPS circuitry "shorting links" have been removed, within 8 hours prior to and at least once per 12 hours during the time any control rod is withdrawn.**

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.
***Except as noted in Specifications 3.9.2.d and 3.9.2.e.

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified:

a. Within 2 hours prior to:

   1. The start of CORE ALTERATIONS.

   2. The withdrawal of one control rod under the control of the reactor mode switch Refuel position one-rod-out interlock.

b. At least once per 12 hours.

* Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**See Special Test Exception 3.10.3.

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3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet 2 inches of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

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 inserted.

f. All fuel loading operations shall be suspended.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION.

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

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 inserted.

f. All fuel loading operations are suspended.

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors SRM are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations shall be suspended.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

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.

f. All fuel loading operations are suspended.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet 2 inches above the top of the reactor pressure vessel flange and heat losses to ambient** are not sufficient to maintain OPERATIONAL CONDITION 5.

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

**Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though REFUELING conditions are being maintained).

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each loop consisting of:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet 2 inches above the top of the reactor pressure vessel flange and heat losses to ambient** are not sufficient to maintain OPERATIONAL CONDITION 5.

ACTION:

a. With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the OPERABILITY of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

**Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though REFUELING conditions are being maintained).

PSEG Internal Use Only

PSEG NUCLEAR

FIRE DEPARTMENT

NC.FP-AP.ZZ-0025 (Q) – Rev. 0

PRECAUTIONS AGAINST FIRE

Sponsor Organization: Emergency Services

REVISION SUMMARY: Biennial Review performed Yes _X_ No ___ N/A ___

1. This is a new procedure. This change is in support of revision number 6 to administrative procedure NC.NA-AP.ZZ-0025(Q), Operational Fire Protection Program. **(80035483).**

   NAAP-25 is being revised to focus on high level process concepts, resulting in relocation of some process description detail and the associated forms, tables, and attachments to other lower tier documents.

2. This revision represents a significant editorial incorporation of the NAAP-25 Hot Work Program into a new stand alone procedure. The NAAP-25 sections utilized to create this procedure were primarily:
   - 5.5, Control of Ignition Sources - Hot Work
   - 5.8, Ignitable Metals

3. The associated NAAP-25 Hot Work forms, tables, and attachments have been relocated to this procedure.

4. Added term seismic joints to step 5.3.2.G to better define penetration in a fire barrier (70012801)

5. Added new section 5.11 Aerosol Products that deals with National Fire Protection Association (NFPA) Code 30B.

IMPLEMENTATION REQUIREMENTS: Effective Date: 1/28/03

APPROVED: _____ Manager - Emergency Services. 1/23/03 Date.

PRECAUTIONS AGAINST FIRE

TABLE OF CONTENTS

1.0 **PURPOSE**

1.1 This procedure is established to secure a reasonable level of safety to life and property from fire hazards incident to the occupancy and maintenance of PSEG Nuclear structures and facilities.

1.2 This procedure provides MINIMUM fire prevention measures and directs the user to procedures, which govern other fire protection processes in greater detail.

2.0 **SCOPE**

This procedure applies to all PSEG Nuclear buildings and facilities located on the owner controlled property.

3.0 **RESPONSIBILITIES**

3.1 **All Directors and Managers**

Ensure their organizations implement the applicable fire prevention requirements within facilities they maintain and in which they conduct work.

3.2 **Manager – Emergency Services**

- Coordinate and implement the Operational Fire Protection Program requirements for protecting plant, systems, components, and materials.
- Ensure General Employee Training (GET) discusses required employee actions in the event of fire.
- Ensure walk downs are performed to monitor control of transient combustibles.
- Ensure an inspection program is implemented to verify compliance with applicable regulations.
- Ensure Hot Work, Transient Combustible, and Fire Protection Impairment Permits (FPIPs) are properly prepared, authorized, and implemented.
- Maintain daily communication with the Operations Superintendent and/or Control Room Supervisors on the status of Salem and Hope Creek Station FPIPs. **[CD-434E]**.
- Ensure ignition source control and monitoring for fire prevention.
- Interface with fire insurance inspectors, state and local enforcement agencies
- Interface with offsite fire departments.
- Define non-FD responsibility for required testing and inspection of fire protection equipment.
- Ensure timely and effective preventive and corrective maintenance on fire protection systems and components assigned to the FD.

3.3 **Job Supervisors - Salem and Hope Creek Stations**

- Comply with fire prevention requirements of the Fire Protection Impairment, Hot Work, and Transient Combustible Control processes
- Ensure work practices support control of ignition sources, flammable liquids and gases, and ignitable metals.

3.4 **All Personnel**

- Comply with the requirements of this procedure.
- Ensure appropriate response to fires and fire alarms for the affected area.
- Ensure fire prevention requirements are observed.

4.0 **PROCESS**

4.1 This procedure identifies elements of the Operational Fire Protection Program categorized as **Fire Prevention** and **Readiness to Detect and Suppress Fires**.

4.2 This procedure is implemented in full detail by the following supplemental Fire Department administrative procedures.

- SC.FP.AP.ZZ-0003(Q), Actions for Inoperable Fire Protection - Salem
- HC.FP-AP.ZZ-0004(Q), Actions For Inoperable Fire Protection- Hope Creek
- NC.FP-AP.ZZ-0005(Q), Fire Protection Surveillance / Periodic Test Program
- NC.FP-AP.ZZ-0009Q), Fire Protection Training Program
- NC.FP-AP.ZZ-0010(Q), Fire Protection Impairment Program
- NC.FP-AP.ZZ-0012(Q), Safe Hot Work Practices
- NC.FP-AP.ZZ-0020(Q), Compensatory Measure Fire Watch Program

4.3 Users of this procedure are advised to refer to the above departmental procedures for the full detail and administrative controls which these procedures provide.

5.0 **PROCEDURE**

5.1 **Response to Fire Emergencies**

5.1.1 In the event a fire alarm sounds within an office facility, (e.g. Material Center, Nuclear Operations Services Facility (NOSF), PSEG Nuclear Administration Building (TB-2), Hope Creek Administration Building, Processing Center, or "B" Clean Facilities Building) evacuate the building in accordance with the established routes.

A. Accountability should be performed to ensure all personnel are out of the building and at a specified gathering place.

B. Evacuees should not crowd the entrances and should await further instructions from either the control room or FD personnel.

C. NEVER use the elevator to leave the building.

D. Always ensure fire boundary doors are closed.

5.1.2 For fire alarms within the Salem and Hope Creek Stations, listen to the instructions via the page system provided.

5.1.3 Upon discovering a fire, report it using the telephone via 3333, a manual pull box, or page system

A. Use page line 1 for Salem U/1, line 2 for Salem U/2

B. Use page line 5 at Hope Creek

5.1.4 When reporting an emergency, callers should provide information such as nature of the emergency, the location, and any equipment involved.

5.1.5 After reporting a fire, employees should fight the fire if its within their training and capabilities.

A. If not within the training and capabilities of the employee(s), to fight the fire, then the employee(s) should alert others in the area, evacuate to a safe distance and standby to direct FD personnel to the scene.

5.1.6 When the Control Room receives a report of a fire they should dispatch appropriate personnel in accordance with the applicable procedures.

5.2 Fire Prevention Requirements

5.2.1 General

The following conditions are prohibited and should be corrected immediately IAW N.J.A.C. 5:70-2-10 (New Jersey State Fire Code). These provisions should be applied in all buildings and facilities:

A. Dangerous conditions which are liable to cause or contribute to the spread of fire.

B. Conditions that would interfere with the use of any fire protection equipment.

C. Obstruction of egress pathways such as stairwells, exit doors and corridors.

D. Accumulation of dust or waste in HVAC systems or grease in kitchen or other exhaust ducts.

E. Accumulations of grease on kitchen cooking equipment, or oil, grease or dirt upon, under or around any mechanical equipment.

F. Accumulation of rubbish, waste, paper, boxes, or other combustible materials (trash) or excessive storage of any combustible materials in an area not designed for that purpose.

G. Hazardous conditions arising from defective or improperly used or installed electrical wiring, equipment or appliances.

H. Hazardous conditions arising from defective or improperly installed equipment for handling or use of combustible, explosive or otherwise hazardous materials.

I. Dangerous amounts of combustible, explosive or otherwise hazardous materials.

J. All equipment, materials, processes or operations that are in violation of the provisions and intent of this procedure, the NJ State Fire Code and any procedures for safe plant operation.

K. Fire Protection deficiencies observed should be immediately reported to the FD (Ext. 2800/2803) for evaluation

L. Smoking is prohibited in all buildings and areas unless specifically posted.

5.2.2 Storage

A. All storage areas should be maintained in a neat and orderly condition.

B. Storage of material should not affect access to or obstruct fire protection systems or fire fighting equipment.

C. Thirty inches (30") of clearance, free of combustible material, shall be maintained around energized electrical panels.

D. Materials should not be placed, stored or kept in any portion of an exit, elevator or at the bottom of a stairway or other means of escape.

5.2.3 **Fire Barriers**

A. Fire Doors shall be maintained closed at all times (unless provided with an approved hold-open device).

B. Blocking, propping or leaving a fire door open without proper authorization for any time period is a violation of our fire protection program and a serious fire safety hazard. **[CD-174B, CD-753X, CD-764E]**

C. It is imperative that **ALL PERSONNEL** make sure fire doors close behind them. For doors with hold open devices, the door opening and door swing area need to be free of obstructions.

D. All required fire resistance rated assemblies (ceiling tiles, walls, doors, etc.) shall be maintained, repaired, restored or replaced when damaged, altered, or penetrated

5.2.4 **Electrical**

A. Extension cords and flexible cords shall not be a substitute for permanent wiring.

B. Extension cords and flexible cords shall be a minimum of 12/3 gauge and not be affixed to structures; extended through walls, ceilings or floors, or under doors or floor coverings; nor shall such cords be subject to environmental damage or physical impact.

C. Multi-plug adaptors, such as cube adaptors, unfused plug strips or any other device not complying with NFPA 70, shall be prohibited.

5.2.5 **Fire Protection Systems**

A. Installed firefighting equipment, with the exception of portable fire extinguishers, may only be utilized for firefighting at the direction of the NFPS.

B. Use of fire protection water for non-fire protection purposes (not within the scope of a T-Mod), is prohibited without permission from the OS/CRS of the affected station and the NFPS, along with issuance of a permit for tracking.

C. Actuated detection or suppression systems should not be shut off until authorized by the NFPS or affected station OS/CRS. The FD should be notified immediately whenever a portable fire extinguisher is discharged.

D. Fire extinguishers mounted in the stations are for emergency use only.

E. Fire extinguisher location changes at Hope Creek are governed by the configuration control program and cannot be made until approved and documented via the design change process. In addition, use of certain type ABC dry chemical extinguishers are prohibited at Hope Creek station. Contact the FD for specific information. **[CR990312177]**

5.2.6 Portable Space Heaters

A. Any use of portable space heaters, whether in a safety related or non-safety related application, will require a review and approval by Fire Protection. Additionally, the following provisions also apply to the use of portable space heaters.

B. Portable space heaters use in office areas or trailers are prohibited.

C. Portable space heaters used to maintain operability of **Safety Related** structures, systems, or components (SSCs) are installed in accordance with NC.DE-AP.ZZ-0030(Q), Control of Temporary Modifications.**[C-0403].**

D. In **non-safety related areas** to support personnel comfort or equipment operation, use of a portable space heaters, will require a review by fire protection with the following restrictions:

   1. The department placing the heater is responsible for placing it in the plant, inspecting it to ensure the unit is in serviceable condition and all safety features are operational.
   2. The responsible department shall monitor the operation of the heater at all times.

E. Portable space heaters should be electrically powered where possible.

F. A clear combustible zone of 10' around the heater should be maintained at all times.

G. Portable heaters should not be placed in high traffic areas or other areas where personnel could come in contact with them.

H. Fuel fired portable heaters should not be used in a building unless all other alternatives are exhausted.

I. Fuel fired portable heaters may not be used without adequate ventilation

J. Fuel fired portable heaters may not be used in confined spaces.

K. Fuel fired portable heaters may not be fueled while in operation

L. A portable fire extinguisher must be present while a fuel fired portable heater is in operation.

5.3 **Fire Protection Impairment Program**

5.3.1 The Fire Department is the fire protection impairment authority, responsible for implementing the Fire Protection Impairment Program in accordance with administrative procedure: **ND.FP-AP.ZZ-0010(Q)**, Fire Protection Impairment Program. **[CD-434E]**

5.3.2 Work activities affecting the following fire protection systems, equipment and barriers shall be controlled through the Fire Protection Impairment Program **[CD-174B, CD-438A, CD-531A, CD-643A, CD-753X, CD-764E, CD-807X, CD-821X]:**

A. Fire water system (i.e. pumps, hydrants, piping, hose stations, deluge and sprinkler systems).

B. Carbon dioxide (CARDOX), Halon, and Foam systems.

C. Smoke and thermal detection systems.

D. Fire alarm and associated circuitry.

E. Manual pull box alarm stations.

F. Fire doors and fire dampers.

G. Penetrations, including seismic joints, in fire barriers. [70012801].

H. Marinite boards and structural steel fireproofing.

I. Fire Wrap on cable tray, conduits, cables, and vent ducts.

J. Radiant energy shields.

K. Appendix "R" 8-hour battery powered emergency lights.

L. Transient combustibles in non-safety related areas/rooms.

5.3.3 A Fire Protection Impairment Permit (FPIP) is also required when equipment taken out of service affects operability of a fire protection system (i.e. tagging a breaker that powers the detection system).

5.4 **Control of Ignition Sources - Hot Work**

5.4.1 The FD authorizes Hot Work Permits in accordance with procedure NC.FP-AP.ZZ-0012(Q), Safe Performance of Hot Work.

5.4.2 Work involving ignition sources, such as welding, cutting, burning, grinding or open flame soldering, is considered a fire watch required activity.

5.4.3 A Hot Work Permit (HWP) - Form 1 of this procedure, shall control all these processes. **[CD-226C, CD-300X, CD-317Y, CD-754X, CD-821X]**

5.4.4 Work involving wire wheeling, needle gun use, or hot iron soldering is **NOT** considered hot work and a Hot Work Permit is not required.

A. Precautions identified in this procedure for hot work activities should be applied to ensure a fire safe work area.

B. At the discretion of the NFPS, the use of certain equipment for pre-heating such as Cooper Heat (trade mark) resistance heaters, or similar devices, may be considered Hot Work.

5.5 **Combustible Material Control - General / All Areas**

5.5.1 Commonly encountered transient combustibles and their estimated heat content are identified in Table 2.

5.5.2 Flammable or combustible liquids introduced into any area/room within the stations shall be limited to daily usage. **[CD-755X]**

5.5.3 Use and storage of flammable liquids for decon within the stations shall be limited to five gallons. **[CD-791X]**

5.5.4 Items incidental to Radiation Protection such as catch-bags, step-off-pads, roping, and stanchions are acceptable in limited quantities as determined by the FD. PC and trash drums are evaluated as transient combustibles.

5.5.5 Accumulations of waste, debris, scrap, rags, and other combustibles resulting from work activities shall be removed from the work area immediately following job completion or at the end of a 24 hour period, whichever comes first.

5.5.6 For work in the RCA, combustibles shall be removed given proper consideration for ALARA or contained in a sealed container until removed. **[CD-426X]**

5.5.7 Combustible structures, (including trailers, sheds, etc.) are prohibited from being placed inside of or within 30 feet of permanent buildings, unless a Fire Protection Impairment Permit (FPIP) or TCP as appropriate is issued authorizing placement.

5.5.8 Introduction of combustibles into **Combustible Control Zones (CCZ)** at Salem is prohibited without a Transient Combustible Permit (TCP), see Form 1, and appropriate compensatory measures. CCZs are used to limit the potential for exposure of sensitive equipment to fire and/or combustion products, to limit the potential for fire propagation through plant equipment hatches, and to reduce overall fire risk. Fixed combustible material may be installed in CCZs with approval from the fire protection designated design authority or the fire department. Transient combustibles are combustible materials that are not part of permanent plant equipment, components or installations. Transient combustibles are intended to pass through or reside in the plant for a brief duration usually associated with, but not limited to, a maintenance or design change work activity (e.g., combustible liquids, wood & plastic products, waste materials, scrap, rags, trash bags, electrical extension cords, clothing hoop rings, or other combustible materials resulting from a work activity.)

5.5.9 Equipment or supplies (such as new fuel) shipped in untreated combustible containers can be unpacked (in preparation for immediate use) in safety-related areas if required for valid operating reasons. However, the combustible shipping materials shall be removed from the area immediately after unpacking, unless stored in metal containers with tight-fitting, self-closing metal covers or equivalent. Such material unless stored in the approved containers, shall NOT be left unattended at any time. **[CD-426X]**

5.5.10 All wood used for maintenance activities, refueling or modification operations, including wood carts and foreign material exclusion (FME) covers, etc. shall be fire retardant (performance rated Exterior Type per National Fire Protection Association Standard 703).

5.5.11 Combustible material for decoration in any building is prohibited, unless the materials have been made flame retardant with an approved flame retardant material or process and are approved by the NFPS prior to the installation, using the planned FPIP process.

5.5.12 Temporary buildings, enclosures, or fixtures within or on buildings should be constructed of noncombustible material. Use of combustible materials may affect compliance with Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards. Permanent items are prohibited unless evaluated and approved in accordance with NC.NA-AP.ZZ-0008 (Q) Configuration Control Program.

5.5.13 Storing combustible or flammable material in stairwells is prohibited.

5.5.14 Combustible or flammable materials in use shall not be stored on or against stations' fire rated barriers, where it will come in contact with such items as:

A. Metal floor hatches, pipes, conduits, cable trays, cable, and HVAC ducts that penetrate fire barriers.

B. Penetration seals installed in barriers or embedded sleeves used to form openings in fire barriers.

5.6 **Combustible Material Control - Safety Related Areas**

5.6.1 Storage of combustibles is NOT permitted in Salem and Hope Creek Station's safety related areas/rooms except when approved by Engineering. **[CD-317Y]**

5.6.2 When approved, storage of combustibles in safety related areas/rooms is permitted only in metal containers with tight-fitting, self-closing metal covers or equivalent.

5.6.3 Storage for Salem's RCA safety related areas is limited to the engineering approved locations and/or the "A" Controlled Facilities Building, while Hope Creek's RCA safety related areas is limited to the engineering approved locations and/or sprinklered areas of the Turbine Building.

5.6.4 Use of non-fire retardant wood (such as large dunnage, wooden carts, or FME) in any safety-related area/room, regardless of the total British Thermal Units (BTU) content, requires an evaluation by engineering.

5.6.5 Daily usage of combustible or flammable materials in safety related areas of Salem and Hope Creek Stations is considered transient combustibles and requires Section 5.7 of this procedure to be completed.

5.7 **Transient Combustibles**

5.7.1 All work activities requiring the introduction of **transient combustible** material into SAFETY RELATED AREAS or ROOMS shall be identified and administratively controlled. **[CD-226C, CD-300X, CD-463Y, CD-807X, CD-821X]**

5.7.2 Job Supervisors or designees, for work activities introducing combustible items into safety-related areas/rooms, are responsible for performing a calculation of the transient combustibles BTU content.

5.7.3 This calculation should be completed during the walk-down phase of the workweek process.

5.7.4 Use Form 1 "Transient Combustible Permit" (TCP) for performing the calculation.

5.7.5 The FD will issue an approved TCP for SAFETY RELATED AREAS or ROOMS and specify compensatory measures based on the following:

A. Total transient combustibles being utilized in a fire area/room.

B. Pre-existing quantities of transient combustibles found during roving patrols or the current TCP logs.

C. Valid hot work activities.

D. Sound Fire Protection Practices.

E. Presence/operability of installed detection or suppression equipment in area/room.

5.8 **Transient Combustible Fire Load Limits**

5.8.1 Transient combustible fire load limits have been established for safety related areas/rooms within the Salem and Hope Creek Stations and are specified in the Fire Hazards Analysis of each station. **[CD-300X, CD-939F]**

5.8.2 If it is determined the use of combustibles **WILL EXCEED** established limits, an engineering assessment is required to determine threshold limits and any additional controls **BEFORE** the combustibles are brought into the safety related areas/rooms.

5.8.3 Regardless of whether the transient combustibles do or do not exceed the limits, the FD will issue a TCP and specify compensatory measures.

5.8.4 Departments introducing the transient combustibles are responsible for the implementation, control and release of the fire watch compensatory measure.

5.8.5 Salem Station Transient Combustible Load Limit

A. **400,000 BTU** per fire area. **[84220]**

B. Table 1 contains the areas in both Salem Stations, which have restrictions to the established limits with regards to transient combustibles.

5.8.6 Hope Creek Station Transient Combustible Load Limit

A. **4,480,000 BTU** per room. **[CD-939F]**

B. Hope Creek rooms have restrictions with regards to transient combustible load limits: **[CD-463Y]**

1. Room 4326 (CRD Removal and Repair Area) and Room 4333 (CRD Storage Area): Transient combustibles are prohibited with the exception of CRD cleaning and rebuilding equipment.
2. Vestibules 5101, 5201 and Electrical Raceways 5216 and 5336: Transient combustibles are prohibited.
3. Main Control Room and Control Complex Peripheral Rooms: Transient combustibles are limited to Class "A" materials and small quantities of cleaning supplies incidental to Control Room Complex functions. This area shall not be used for any storage, as storage shall be limited to the peripheral rooms only. This does not include small quantities of items such as a ream of paper or stationary supplies like pens and notepads.

5.9 **Fire Department (FD) Transient Combustibles Inspections**

5.9.1 The FD performs a survey of transient combustibles in safety-related areas/rooms as part of their daily housekeeping inspections.

5.9.2 The FD performs a survey of transient combustibles in non-safety-related areas/rooms as part of their weekly housekeeping inspections.

5.9.3 Non-compliances discovered will be documented using the notification process. **[78105]**

5.9.4 For Non-Safety Related Areas Only:

A. No combustible or flammable material should be stored in any non-safety related area of either Salem or Hope Creek Stations, unless the area contains a fire suppression system, due to insurance restrictions (NEIL Property Loss Prevention Standards). Exceptions to this requirement are based on a FD evaluation with implementation of compensatory measures.

B. All wood used for maintenance activities should be fire retardant wood (performance rated Exterior Type per National Fire Protection Association [NFPA] Standard 703).

C. Introduction of combustibles considered to be excessive will be tracked by a Fire Protection Impairment Permit.

5.10 **Control of Flammable and Combustible Liquids and Gases**

5.10.1 Storage or dispensing of flammable or combustible liquids shall be in accordance with NFPA Standard 30, the "Flammable and Combustible Liquids Code", unless an authority having jurisdiction (such as the Nuclear Regulatory Commission (NRC) or the Nuclear Electric Insurance Limited (NEIL) grants exceptions or variances. All questions regarding compliance with NFPA 30 should be directed to the FD. **[CD-300X, CD-755X]**

5.10.2 Flammable and combustible liquids shall be handled carefully using the following guidelines: **[CD-755X]**

A. Kept in Factory Mutual (FM) or Underwriters Laboratory (UL) approved safety cans when in use, handled and/or dispensed. The only exception is when liquids remain in their original containers to maintain purity.

B. Flammable and combustible liquids when **NOT** in use are stored in FM or UL approved flammable liquid storage cabinets regardless of container.

C. Proper electrical grounding and bonding are provided when dispensing.

5.10.3 Portable Fuel Tanks

A. Portable fuel tanks, or fuel tanks used for storage, require the following measures:

1. A capacity > 660 gallons requires a permit by the State of N.J.
2. Contained within a dike capable of being drained frequently to remove accumulated rainwater or spills.
3. Located and arranged that access for fire fighting is not restricted.
4. Equipped with a lockable, automatic shutoff nozzle.
5. Portable ABC type fire extinguisher provided.

B. If any of the above fire prevention measures are not met, a fire protection impairment permit is required along with possible compensatory measures.

5.10.4 Engineering is to assess and approve all permanent or temporary placements of flammable liquid storage cabinets and gas cylinders in safety-related areas of either Salem or Hope Creek. **[CD-755X, CD-830X]**

A. Requests for permanent placement of a cabinet in a SAFETY RELATED area made via N1 Notification for engineering evaluation and approval.

B. Requests for temporary placement of a cabinet in a SAFETY RELATED area are made via the Transient Combustible Permit process through the Fire Department.

5.10.5 Placement of flammable liquid cabinets in non-safety related areas is in accordance with the following:

A. No more than 3 cabinets should be located in any one fire area.

B. In large plant areas (ie Turbine Building) more than 3 cabinets may be installed provided each group of 3 cabinets is separated by at least 100'.

C. If the area is fully sprinklered, up to 6 cabinets may be installed in a single area without 100' separation.

5.10.6 Flammable liquid cabinets should be maintained in a neat orderly condition with all latches, hardware and hinges properly functioning.

5.10.7 The maximum allowable quantity of Class I, II, and IIIA liquids should not exceed 120 gallons per cabinet.

5.10.8 NO storage of materials other than Class I, II and IIIA liquid is permitted in a Flammable liquid cabinet.

5.10.9 Flammable Gas Cylinders

A. Flammable gas cylinders in any safety-related area or room, regardless of the amount, shall NOT be left unattended for > 1 hour. **[78111]**

B. Flammable gas cylinders in any safety-related area or room, regardless of the amount requires a TCP.

1. An exception for the TCP would be when a Hot Work Permit (HWP) is issued for the gas cylinders on a daily basis.

2. The requirement to monitor flammable gas cylinders left for longer than one hour in a safety related area or room of the stations remains the same. **[CD-830X, TS990325186]**

5.10.10 Compressed Gas Cylinders

A. Personnel working with compressed gas cylinders shall comply with the precautions as specified in desk guide: NC.PM-DG.ZZ-0001 (Z) Storage and Handling of Compressed Gas Cylinders: **[CD-754X, CD-830X]**

B. Compressed Gas Cylinder Control Tags must be affixed to cylinder(s). These tags should be maintained to match cylinder level (EMPTY, IN-USE, and FULL).

C. Compressed gas cylinders used for hot work with a valid daily hot work permit issued, are considered "in use". Those not in use will be treated as "storage" and shall <u>NOT</u> be left unattended for longer than one hour in safety related areas of the stations. **[78111]**

D. Cylinders should contain a job information tag attached for identifying the owners. **[TS990503190]**

E. Cylinders should be kept in an upright position, secured in specially manufactured holding devices, or to a cart or permanent structure by chain or double wrapped wire (minimum No. 9 gauge), to prevent them from falling or being knocked over.

F. Open cylinder valves slowly and close valves when work is complete or when personnel leave the area.

G. Keep cylinders capped whenever regulator is removed, while moving a cylinder, or when cylinder is in storage.

H. Promptly return cylinders to the appropriate gas bottle storage areas upon completion of the work.

I. All storage and issuance of compressed gas cylinders in the gas bottle storage areas should be in accordance with Compressed Gas Association Pamphlet P-1.

J. Warning signs are required in areas where the possibility of hydrogen leakage exists (Re-combiners, Waste Gas Compressors, Decay Tanks, and Turbine Generators) to prevent open flames or other sources of ignition within 35' feet of the hazard. **[CD-173B]**

5.11 **Aerosol Products**

5.11.1 An aerosol is a product that is dispensed from an aerosol container by a propellant.

5.11.2 The definition of aerosol does not apply to the storage or display of containers whose contents are composed entirely of liquid petroleum (LP) gas products.

A. Examples include MAP (methane, acetylene, propane) gas cylinders, propane gas cylinders, and butane gas cylinders.

B. Uses for these LP-gases might include small hand torches for brazing and soldering.

5.11.3 An aerosol container is a metal can up to a maximum size of 33.8 fluid ounces (1000 ml) or a glass or plastic bottle up to a maximum size of 4 fluid ounces (118 ml).

5.11.4 Storage and display of aerosols shall be in accordance with NFPA standard 30B, Manufacture and Storage of Aerosol Products Code.

5.11.5 Aerosols introduced into any safety related area/room within the stations shall be limited to daily usage.

5.11.6 Daily usage of aerosols in safety related areas of Salem and Hope Creek Stations is considered transient combustibles and is required to be controlled accordingly.

6.0 **Records**

6.1 Transient Combustible Permits shall be retained with the work package in accordance with NC.NA-AP.ZZ-0011(Q), Records Management Program (NAP-11). **[CD-434E]**

6.2 The FD, in accordance with NAP-11, shall retain transient combustible permits and logs pertaining to hot work permits and fire protection impairment permits. **[CD-434E]**

7.0 **DEFINITIONS**

7.1 **Aerosol** - A product that is dispensed from an aersol container by a propellant.(NFPA 30B).

7.2 **Combustible Control Zone (CCZ)** - An area of the plant in which transient combustible material(s) is prohibited without a Temporary Combustible Permit (TCP) and appropriate compensatory measures.

7.3 **Combustible Liquid** - A liquid having a flash point at or above 100° F (37.8° C).

7.4 **Combustible Material** – Material, which in the form it is used and under the conditions anticipated, **WILL** ignite, burn, support combustion, or release flammable vapors when subjected to fire and heat.

7.5 **Fines** – very small particles in a mixture of various sizes.

7.6 **Fire Barrier** - Those construction elements (walls, floors, and their supports), including beams, columns, penetration seals or closures, fire doors and dampers, that are rated by approving laboratories in hours of resistance to fire and used to prevent the spread of fire.

7.7 **Fire Retardant Wood** - Wood that has been treated with fire retardant chemicals and is performance rated Exterior Type per National Fire Protection Association Standard 703).

7.8 **Fire Watch** - An individual who has satisfactorily completed fire watch training and is designated by the FD for compensatory measure, hot work, or both fire watch type duties.

7.9 **Flammable Liquid** - A liquid with a flash point below 100° F and a vapor pressure not exceeding 40 pounds per square inch absolute at 100° F. (Also known as a Class I liquid.)

7.10 **Hot Work** - Work that involves ignition sources.

7.11 **Ignition Source** - Heat or flame from any source capable of igniting combustible or flammable material.

7.12 **Impairment** - Any condition affecting the intended operation of a fire protection system or barrier or posing as a fire hazard within the stations.

7.13 **Liquified Petroleum Gas (LP-Gas)** - Material having a vapor pressure not exceeding that allowed for commercial propane composed predominantly of the following hydrocarbons, either by themselves or as mixtures: propane, propylene, butane (normal butane or isobutene), butylenes (NFPA 58).

7.14 **Noncombustible Material** - Material which, in the form it is used and under the conditions anticipated, will **NOT** ignite, burn, support combustion, or release flammable vapors when subjected to fire and heat.

7.15 **Permanent** - A condition with an indefinite period of time.

7.16 **Safety Related Area** - An area that contains systems and components required to shut down the reactor, mitigate the consequences of postulated accidents or maintain the reactor in a safe cold shutdown condition.

7.17 **Sealed Container** - A metal container sealed by means of a tightly fitting lid or other device so neither liquid nor vapors can escape at room temperature.

7.18 **Storage** - The placing of items in a location and leaving them unattended until later use or disposal.

7.19 **Temporary** - planned time duration of no more than 60 days.

7.20 **Temporary Buildings** - Any temporary structure or supported protective covering where personnel can move freely in and out of the structure.

7.21 **Temporary Enclosure** - Any supported protective covering where personnel cannot move in and out of the structure or where the protection is accomplished by draping the protective cover over the material and/or equipment. This definition also applies to temporary enclosures inside of permanent buildings.

7.22 **Temporary Fixture** - Something securely placed or attached as an appendage or as a structural part of a building (i.e. door, portable steps, storage rack), constructed of a combustible material. This definition does not apply to furniture permanently installed in Station facilities.

7.23 **Transient Combustible** - Any combustible material introduced into a Safety-Related area/room which will be used to complete an assigned task or left unattended for any period of time. It does not include items moved through areas in route to the final destination, unless the items are left un-monitored during transit.

8.0 REFERENCES

8.1 NC.NA-AP.ZZ-0025 (Q), Operational Fire Protection Program

8.2 N.J.A.C. 5:70-2-10 – New Jersey State Fire Code

8.3 Salem Updated Final Safety Analysis Report (UFSAR), Section 9.5, - Fire Protection System

8.4 Salem Fire Protection Report - Fire Hazards Analysis

8.5 HCGS Updated Final Safety Analysis Report, (UFSAR) Section 9.5.1, - Fire Protection Program

8.6 HCGS UFSAR, Appendix 9A, - Appendix R Comparison

8.7 Appendix A to Branch Technical Position APCSB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976

8.8 Branch Technical Position CMEB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants

8.9 NRC Supplemental Guidance, Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance, August 1977

8.10 Occupational Safety and Health Standards Title 29, Chapter XVII, Part 1910.36FR 10466

8.11 10CFR50 Appendix R, Fire Prot. Program for Nuclear Power Facilities Operating Prior to Jan. 1, 1979

8.12 NC.DE-PS.ZZ-0001(Q), Fire Protection Programmatic Standard

8.13 IEEE Standard 634 - 1978

8.14 Hope Creek Station Generic Letter 86-10 Submittal, dated May 13, 1986

8.15 HC.DE-PS.ZZ-0021(F), Hope Creek Penetration Seal Program

8.16 SC.DE-PS.ZZ-0035(Q), Salem Penetration Seal Program

8.17 National Fire Protection Association Standard 701, Standard Methods for Fire Tests for Flame Propagation of Textiles and Films

8.18 **Cross References**

8.18.1 NFPA 30 - Flammable Liquids Code, National Fire Protection Association (NFPA)

8.18.2 NFPA 30B - Code for the Manufacturer and Storage of Aerosol Products

8.18.3 NFPA 58 - Liquefied Petroleum Gas Code

8.18.4 NFPA 703 - Standard for Fire Retardant Impregnated Wood and Fire Retardant Coatings for Building Materials

8.18.5 Compressed Gas Association Pamphlet P-1, Safe Handling of Compressed Gases in Containers.

8.18.6 Property Loss Prevention Standards of Nuclear Generating Stations - Nuclear Services Organization (NSO).

8.18.7 NC.NA-AP.ZZ-0008(Q), Configuration Control Program

8.18.8 NC.NA-AP.ZZ-0011(Q), Records Management Program (NAP-11)

8.18.9 NC.DE-AP.ZZ-0030(Q), Control of Temporary Modifications

8.18.10 SC.FP.AP.ZZ-0003(Q), Actions for Inoperable Fire Protection - Salem

8.18.11 HC.FP-AP.ZZ-0004(Q), Actions For Inoperable Fire Protection- Hope Creek

8.18.12 NC.FP-AP.ZZ-0005(Q), Fire Surveillance and Periodic Test Program

8.18.13 NC.FP-AP.ZZ-0009(Q), Fire Department Training Program

8.18.14 NC.FP-AP.ZZ-0010(Q), Fire Protection Impairment Program

8.18.15 NC.FP-AP.ZZ-0012(Q), Safe Performance of Hot Work

8.18.16 NC.FP-AP.ZZ-0020(Q), Compensatory Measure Fire Watch Program

8.18.17 NC.PM-DG.ZZ-0001 (Z) Storage and Handling of Compressed Gas Cylinders

8.19 **Commitment Documents:** - Hope Creek

8.19.1 CD-156A (NRC EI Circular 80-09) (See FPAP-12)

8.19.2 CD-173B (INPO SOER 82-09)

8.19.3 CD-174B (SOER 82-10R02, 04, 07)

8.19.4 CD-226C (NRC IE Bulletin 75-04A)

8.19.5 CD-300X (FSAR 8.1.4.14.3.1)

8.19.6 CD-317Y (FSAR Q 430.85(B))

8.19.7 CD-426X (FSAR 9.1.1.3.2)

8.19.8 CD-434E (IR 86-062) (LER 86-062)

8.19.9 CD-438A (INPO SER 055-81)

8.19.10 CD-463Y (FSAR Appendix 9A.6.5)

8.19.11 CD-531A (INPO SER 46-82)

8.19.12 CD-643A (INPO O&MR 64)

8.19.13 CD-652A (INPO O&MR 73) (See FPAP-12)

8.19.14 CD-682D (INPO SER 01-85) (See FPAP-12)

8.19.15 CD-753X (FSAR 9.5.1.1.10)

8.19.16 CD-754X (FSAR 9.5.1.1.11)

8.19.17 CD-755X (FSAR 9.5.1.1.13)

8.19.18 CD-764E (NHO IR 86-274)

8.19.19 CD-791X (FSAR 9.5.1.2.32)

8.19.20 CD-807X (FSAR Appendix 9A.III.K)

8.19.21 CD-821X (FSAR 9.5.1.5.3)

8.19.22 CD-830X (BTP CMEB 9.5-1.c.8.1)

8.19.23 CD-939F (NRC OPEN 354/88-18-F1)

8.20 **Closing Documents** - Salem

8.20.1 C0403 (NRC-INFO 89-04)

8.20.2 C0549 (SR 272/92-8) (See FPAP-10)

8.21 **Commitment Documents** - Engineering

8.21.1 78105 (Appendix A to BTP APSCB 9.5-1, Section B.2)

8.21.2 78111 (Letter from PSE&G to NRC, 07/26/78)

8.21.3 84220 (Memo from Vargas to PSE&G, 03/23/84)

TABLE 1
SALEM STATION
COMBUSTIBLE CONTROL ZONES SALEM 1 & 2

Unit	CCZ No	Elev	Type	Description
1	CCZ-1	55' AUX	Hatch	Area beneath Hatch opening between 1FA-AB-55B/1FA-AB-64B
1	CCZ-1		Wall Opening	Door opening between 1FA-AB-55A/1FA-AB-55B
C	CCZ-2	64' AUX	Hatch	Area beneath Hatch opening between 1FA-AB-64A/12FA-AB-84A
1	CCZ-3		Hatch	Hatch opening between 1FA-AB-55B/1FA-AB-64B
1	CCZ-4		Lack of Detector	Room 15310 # 1 Waste Monitor Hold-up Tank Pump (WMHUT Pp)
1	CCZ-5		Hatch	Hatch opening between 1FA-AB-64B/1FA-AB-84B
C	CCZ-6	84' AUX	Hatch	Hatch opening between 1FA-AB-64A/12FA-AB-84A
1	CCZ-7		Hatch	Area beneath Hatch opening between 1FA-EP-78C/1FA-PP-100H
1	CCZ-8		Hatch	Hatch opening between 1FA-AB-64B/1FA-AB-84B
			Cable Separation	Separation between Service Water Cables Train A& B
C	CCZ-9		Trench	Service Water Pipe Trench
C	CCZ-10		Tunnel	Service Water Pipe Trench
C	CCZ-11	100' AUX	Hatch	Hatch opening between 12FA-AB-84A/12FA-AB-100A
1	CCZ-12		Hatch	Hatch opening between 1FA-EP-78C/1FA-PP-100H
1	CCZ-13		Hatch	Hatch opening between 1FA-AB-100C/1FA-AB-84B
C	CCZ-14		Wall Opening	Roll up door opening at Truck Bay - IPEEE
1	CCZ-15	122' AUX	Hatch	Hatch opening between 1FA-AB-122B/1FA-AB-100C
2	CCZ-16	55 AUX	Hatch	Area beneath Hatch opening between 2FA-AB-55B/2FA-AB-64B
			Wall Opening	Door opening between 2FA-AB-55A/2FA-AB-55B
2	CCZ-17	64 AUX	Hatch	Hatch opening between 2FA-AB-55B/2FA-AB-64B
2	CCZ-18		Lack of Detection	Room 25310 # 2 Waste Monitor Hold-up Tank Pump (WMHUT Pp)
	CCZ-19		Hatch	Hatch Opening between 2FA-AB-64B/2FA-AB-84B
2	CCZ-20	84' AUX	Hatch	Area beneath hatch opening between 2FA-EP-78C/2FA-PP-100H
2	CCZ-21		Hatch	Hatch opening between 2FA-AB-64B/2FA-AB-84B
			Cable Separation	Separation between Service Water Cables Train A& B
2	CCZ-22	100 AUX	Hatch	Hatch opening between 2FA-EP-78C/2FA-PP-100H
2	CCZ-23		Hatch	Hatch opening between 2FA-AB-100C/2FA-AB-84B
2	CCZ-24	122' AUX	Hatch	Hatch opening between 2FA-AB-122B/2FA-AB-100C
1	CCZ-25	88' TG	Off Site Power	Off Site Power from No 13 & 14 Power Transformer
2	CCZ-26	88' TG	Off Site Power	Off Site Power from No 23 & 24 Power Transformer
1	CCZ-27	100' RC	335 Panel	Reactor Containment around the 335 Panel
1	CCCZ-28	81' RC	Pressurizer	Reactor Containment around the Pressurizer
1	CCCZ-29	103' RC	Pressurizer	Reactor Containment around the Pressurizer Platform
2	CCCZ-31	81' RC	Pressurizer	Reactor Containment around the Pressurizer
2	CCCZ-32	103' RC	Pressurizer	Reactor Containment around the Pressurizer Platform

Reference Drawings: 605810, 605811, 605812, 605813, 605815, 605816, & 605817

TABLE 2

COMMON TRANSIENT COMBUSTIBLES / ESTIMATED HEAT CONTENT

ITEM	ESTIMATED HEAT CONTENT
Acetylene	**20,800 BTU/lb**
Common size acetylene cylinders found on site:	*MC Size Cylinder (4" D x 12" H) = ¾ lbs.* *B Size Cylinder (6" D x 19" H) = 2 lbs.* *R Size Cylinder (7" D x 25" H) = 3 lbs.* *S Size Cylinder (10" D x 30" H) = 15 lbs.* *M Size Cylinder (12" D x 36" H) = 18 lbs.* *L Size Cylinder (12" D x 39" H) = 24 lbs.*
Cable insulation	12,000 BTU/lb
Cardboard	6,000 BTU/lb
Charcoal	13,000 BTU/lb
Cloth/cloth pcs	16,000 BTU/lb *(1 PC = ½ LB or 8,000 BTUs)*
Combustible liquid	160,000 BTU/gal
Dry ion resins	12,000 BTU/lb
Fiberglass ladder (6')	32,000 BTU/ea *(1 foot = 5,333 BTUs)*
Flammable liquid	90,000 BTU/gal
Paper	8,000 BTU/lb or 453,000 BTU/cu. ft. *(1 case of copier paper = 416,000 BTUs)*
Plastic	20,000 BTU/lb *(1 sheet of herculite - 50' x 50' = 50,000 BTUs)*
Rubber	10,000 BTU/lb *(1 - ¾" hose, 50' = 50,000 BTUs)*
Titanium	8,500 BTU/LB
Wood	9,000 BTU/lb or 414,000 BTU/cu. ft. *(1 plank - 2" x 6" x 10' = 277,500 BTUs)*

FORM 1
TRANSIENT COMBUSTIBLE PERMIT

LOCATION	FIRE AREA	
START DATE	DURATION	ORDER #
JOB SUPERVISOR		EXT

EVALUATE TRANSIENT COMBUSTIBLES WHICH YOU ANTICIPATE BEING USED

COMBUSTIBLES	ESTIMATED HEAT CONTENT	TOTAL BTU's
FLAMMABLE LIQUID	90,000 BTU/GALLON	
COMBUSTIBLE LIQUID or GREASE	160,000 BTU/GALLON	
CHARCOAL	13,000 BTU/POUND	
FIBERGLASS LADDER	5,333 BTU/FOOT OF LADDER	
CABLE INSULATION	12,000 BTU/POUND	
CARDBOARD	6,000 BTU/POUND	
PAPER	8,000 BTU/POUND (453,000 BTU/CUBIC FOOT)	
CLOTH and CLOTH PCs	16,000 BTU/POUND 1PC = ½ POUND	
PLASTIC	20,000 BTU/POUND	
WOOD	9,000 BTU/Pound (414,000 BTU/Cubic Foot)	
DRY ION RESINS	12,000 BTU/Pound	
TITANIUM	8,500 BTU/Pound	
ACETYLENE	20,800 BTU/Pound	
OTHER	Contact FIRE DEPT.	

Engineering Assessment Needed ☐ Yes ☐ No	Approved ☐ Yes ☐ No
Completed By	Date
TCP #	Total BTUs

REMOVE ALL TRANSIENT COMBUSTIBLES UPON JOB COMPLETION

PSEG NUCLEAR FIRE DEPARTMENT

NC.FP-AP.ZZ-0012(Q) – Revision 0

SAFE PERFORMANCE OF HOT WORK

Sponsor Organization: Emergency Services – Fire Department

REVISION SUMMARY:

1. This is a new procedure. This change is in support of revision number 6 to administrative procedure NC.NA-AP.ZZ-0025(Q), Operational Fire Protection Program. **(80035483)**. NAAP-25 is being revised to focus on high level process concepts, resulting in relocation of some process description detail and the associated forms, tables, and attachments to other lower tier documents.

2. This revision represents a significant editorial incorporation of the NAAP-25 Hot Work Program into a new stand alone procedure. The NAAP-25 sections utilized to create this procedure were primarily:
   - 5.5, Control of Ignition Sources - Hot Work
   - 5.8, Ignitable Metals

3. The associated NAAP-25 Hot Work forms, tables, and attachments have been relocated to this procedure.

4. This revision meets the requirements of the biennial review.

IMPLEMENTING REQUIREMENTS Effective On: 1/28/03

Approved By: _____ 1/23/03
Manager – Emergency Services Date

SAFE PERFORMANCE OF HOT WORK

TABLE OF CONTENTS

1.0 **PURPOSE**

Provide a method and instruction for preparing, authorizing, and tracking Hot Work Permits, to control and limit the introduction of potential fire hazards into plant areas. **[CD-300X]**

2.0 **SCOPE**

2.1 This procedure applies to the performance of hot work in the Salem 1 and 2 Stations, the Hope Creek Generating Station, and associated out buildings.

2.2 Implementation of this procedure provides a log of the areas where hot work has been authorized including High Risk jobs and hot work in approved fab shops.

2.3 This procedure ensures that the Emergency Services - Fire Department (FD) duty Nuclear Fire Protection Supervisor (NFPS) is aware of all hot work jobs, possible fire hazards and their effect on station safety.

2.4 This procedure also allows for the tracking of flammable compressed gas cylinders utilized for hot work.

3.0 **RESPONSIBILITIES**

3.1 **Nuclear Fire Protection Supervisor (NFPS) or designee**

3.1.1 Compile information on the hot work permit (HWP) log prior to authorizing a permit.

3.1.2 Authorize Hot Work Permit.

3.1.3 Maintain log of authorized hot work permits.

3.1.4 Recognize the extent of hot work jobs in progress during his shift.

3.1.5 Inspect and authorize high risk hot work jobs.

3.2 **Job Supervisor**

3.2.1 Ensure their personnel are trained and qualified to perform hot work.

3.2.2 Prepare their jobsite in accordance with the HWP.

3.2.3 Inspect their jobsite prior to requesting a HWP number.

3.2.4 Maintain their jobsite in accordance with the HWP.

3.2.5 Ensure precautions are taken regarding work practices that have the potential to generate electromagnetic interference and radio frequency interference

3.2.6 Notify the OS/CRS of the affected station prior to welding in the vicinity of a Radiation Monitor System (RMS) component

- Spiking of the RMS may result, causing unexpected alarms of the RMS. **[TS980622174].**

3.2.7 Assign qualified fire watch to the hot work job site.

- Additional firewatch may be assigned due to sparks or slag streams affecting adjacent fire areas or elevations.

3.2.8 Terminate work activities properly to ensure that flammable compressed gas cylinders are not left unattended.

3.2.9 Ensure fire watch remains for 30 minutes after hot work is completed.

3.3 **Hot Work Fire Watchers**:

3.3.1 Present and observant at all times of the work being performed and the surrounding area.

3.3.2 Ensure that the work area is maintained as described on the HWP Inspection at all times while hot work is being performed.

3.3.3 Notify the control room immediately of any fire occurring after ensuring the safety of their co-workers.

3.3.4 Extinguish fires within the capabilities of their training after the fire has been reported.

3.3.5 Remain at their post for 30 minutes following the last hot work.

4.0 HOT WORK PROCESS OVERVIEW

4.1 Hot Work

4.1.1 Work involving ignition sources, such as welding, cutting, burning, grinding or open flame soldering, is considered Hot Work.

4.1.2 Hot work is a fire watch required activity.

4.1.3 A Hot Work Permit (HWP) shall control work involving ignition sources, such as welding, cutting, burning, grinding or open flame soldering. **[CD-226C, CD-300X, CD-317Y, CD-754X, CD-821X]**

4.2 General

4.2.1 Daily Hot Work permits are valid for the one calendar day in which they are issued.

4.2.2 Annual hot work permits, with specific numbers, are issued individually for each approved shop facility on site for compliance with the New Jersey State Uniform Fire Code.

4.2.3 Fire watch are not required in shops where welding and work involving ignition sources are routinely performed or in yard areas where no flammable or combustible materials are located within 35 feet of the hot work.

4.2.4 Separate Hot Work tracking logs should be established for the Hope Creek and Salem stations.

4.2.5 The NFPS should review the logs at the beginning of each operating shift to become familiar with hot work that has been authorized for the shift and what detection zones may be affected.

4.2.6 Hot work requires the fire watch to be maintained for 30 minutes after hot work is completed or suspended, to ensure the work area is safe from fire.

4.2.7 Fire watch is responsible for monitoring the work area, including levels below floor grating and floor openings.

4.2.8 A portable fire extinguisher, inspected within the previous 30 days, is required for the hot work job.

4.3 **High Risk Hot Work**

4.3.1 Hot Work activities will be defined as **HIGH RISK** if certain criteria are met.

4.3.2 **HIGH RISK** hot work activities will require additional administrative controls and supervisory oversight, beyond the normal hot work controls.

4.3.3 The following hot work activities are considered **HIGH RISK**:

- Work on tanks, vessels or piping that contained, or had contained combustible or flammable chemicals, solvents, gases, etc.
- Work in areas identified as having hydrogen as part of equipment or processes.
- Use of arc gouging or plasma arc welding which produce copious amounts of heat, with molten slag and sparks.
- Work in cable shafts, filter housings, or ventilation ductwork
- Work in Hope Creek Drywell **[CD-682-D].**
- Work on equipment containing combustible (ignitable) metals, which are listed in Table 1 and 2 of this procedure

4.4 **Ignitable Metals**

4.4.1 Ignitable metals utilized at Salem and Hope Creek Stations are identified in Table 1 and Table 2.

4.4.2 The following precautions apply when working on or in proximity of equipment containing ignitable metals:

- Metals are considered potentially combustible whenever exposed.
- Notify the Fire Department (FD) of work activities involving ignitable metals prior to the exposure of the metal.
- Class D fire extinguisher (identified by a yellow star with a "D", located on the extinguisher body), is readily available.
- Ignition mechanisms consists of oxy-acetylene or carbon-arc welding, band saw friction cutting, abrasive cutting wheels and grinding operations.
- Heat sink effect dissipates ignition source heating when in bulk form.
- The ignitable metal hazard increases significantly when shavings, chips, dust, etc., are involved.
- Metal fines produced from combustible metals, increase the hazard potential because they act like kindling.
- Proper cleanup and removal of accumulations of metal fines, produced in such operations as band saw cutting, are essential for fire prevention and safety.

4.5 **Electromagnetic / Radio Frequency Interference**

4.5.1 Plant areas susceptible to electromagnetic interference (EMI) and radio frequency interference are listed in NC.NA-AP.ZZ-0005(Q) (NAAP-5), Station Operating Practices.

4.5.2 Prior to commencing electric arc welding in plant areas susceptible to EMI or RFI, the job supervisor should ensure work practice precautions and Main Control Room notifications have been addressed in accordance with NAAP-5.

4.5.3 Hot Work Permit (Form 1) provides the job supervisor with the direction to inspect the work area against the precautions for EMI and RFI.

4.6 **Exceptions**

4.6.1 Work involving wire wheeling, needle gun use, or hot iron soldering is **NOT** a fire watch required activity and a HWP is not required. However precautions identified in this procedure for hot work activities should be taken to ensure a fire safe work area.

4.6.2 Equipment for pre-heating such as Cooper Heat (trade mark) resistance heaters, or similar devices, is not ordinarily considered Hot Work.

- At the discretion of the NFPS, formal Hot Work administrative controls may be required.
- The job supervisor is not exempt from enforcing the intent of maintaining any work area in good housekeeping condition.
- At the discretion of the job supervisor, formal Hot Work administrative controls may be requested. If any uncertainty exists, regarding the nature of the work activity, the job supervisor should review the activity with the NFPS.

5.0 **APPROVAL PROCESS**

5.1 Prior to requesting a HWP, the job supervisor should ensure that all of the inspection criteria on the HWP have been satisfied and the job site inspected.

5.2 FD personnel develop and record the following baseline information fields in the Hot Work Authorization (HWA) Log:

- Job location.
- Work order number.
- Type of work (including use of flammable compressed gas cylinders).
- Responsible supervisor and telephone extension.
- Affected detection zones.
- Time and date of authorization.

5.3 **Evaluate For HIGH RISK**

5.3.1 FD personnel evaluate the hot work activity request against the following **HIGH RISK** criteria, prior to providing the hot work authorization number:

- Work on tanks, vessels or piping that contained, or had contained combustible or flammable chemicals, solvents, gases, etc.
- Work in areas identified as having hydrogen as part of equipment or processes.
- Use of arc gouging or plasma arc welding which produce copious amounts of heat, with molten slag and sparks.
- Work in cable shafts, filter housings, or ventilation ductwork
- Work in Hope Creek Drywell **[CD-682-D].**
- Work on equipment containing combustible metals, which are listed in Table 1 and 2 of this procedure.

5.3.2 IF the job is **HIGH RISK**, THEN no authorization number will be provided until after the job site is inspected by the Fire Protection Supervisor or his designee.

- The inspection will determine whether the jobsite complies with the minimum standards set forth on the HWP or whether more stringent requirements need to be implemented.
- A FD field inspection is REQUIRED to be performed prior to issue of a **HIGH RISK** HWP.

5.3.3 The NFPS should notify the Operations Superintendent (OS) of High Risk work being conducted.

5.3.4 FD reviews the Impairment Log for any fire protection components in the work area that are inoperable and identifies additional compensatory measures as necessary.

5.4 **Hot Work Authorization**

5.4.1 Hot Work may be authorized by FD after completion of the following:

- FD completes baseline information fields in the HWA Log (Form 2)
- FD completes a **HIGH RISK** activity criteria evaluation.
- FD completes a work area inspection and signs/dates the field HWP, for activities designated HIGH RISK.

5.4.2 Authorization to perform hot work is indicated by an authorization number. The authorization number is developed using the following format and associated legend:

- Format:
  - X-YY-MM-DD-ZZ
- Legend:
  - X – Station where work is performed. (S=Salem, H=Hope Creek)
  - YY – Last 2 digits of current year
  - MM – Month
  - DD – Day that work is being performed
  - ZZ – Sequential number beginning with 01 each day.

5.4.3 FD enters the authorization number for each activity into the HWA Log.

5.4.4 FD initials the HWA Log and records the time the HWP was issued.

5.5 **Work Area Hot Work Permit**

5.5.1 Upon issuance of a hot work authorization number, the job supervisor should complete the following fields of the HWP:

- HWP Number
- Job Supervisor Name
- Date

5.5.2 The job supervisor should ensure the HWP is posted in the work area.

5.5.3 The job supervisor should ensure that the work area is maintained in accordance with the HWP Work Area Inspection criteria.

5.5.4 IF hot work is discontinued for longer than 8 hrs, THEN re-inspection of the area in accordance with Part 2 of the HWP (Form 1) is required prior to the commencement of work.

5.5.5 The FD will verify HWP criteria/conditions on a random basis after issuance of hot work permits.

6.0 **RECORDS**

6.1 Hot work authorization logs should be maintained for a minimum of 30 days.

6.2 Hot Work Permits and Transient Combustible Permits shall be retained with the work package in accordance with NC.NA-AP.ZZ-0011(Q), Records Management Program (NAP-11). **[CD-434E]**

6.3 The FD, in accordance with NAP-11 shall retain logs pertaining to hot work permits. **[CD-434E]**

7.0 **REFERENCES**

7.1 NC.NA-AP.ZZ-0025 (Q) Operational Fire Protection Program

7.2 New Jersey Administrative Code - NJAC 5:70-2-10 - NJ State Fire Code

7.3 NC.PM-DG.ZZ-0001 (Z) Storage and Handling of Compressed Gas Cylinders

7.4 Compressed Gas Association Pamphlet P-1

7.5 Property Loss Prevention Standards of Nuclear Generating Stations, - Nuclear Electric Insurance Limited (NEIL).NFPA 30 - Flammable Liquids Code, National Fire Protection Association

7.6 NC.IS-TM.ZZ-0001(Z) PSEG Nuclear Industrial Health and Safety Manual

7.7 Salem Updated Final Safety Analysis Report (UFSAR), Section 9.5, - Fire Protection System

7.8 Salem Fire Protection Report - Fire Hazards Analysis

7.9 HCGS Updated Final Safety Analysis Report, (UFSAR) Section 9.5.1, - Fire Protection Program

7.10 HCGS UFSAR, Appendix 9A, - Appendix R Comparison

7.11 Appendix A to Branch Technical Position APCSB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976

7.12 Branch Technical Position CMEB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants

7.13 NRC Supplemental Guidance, Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance, August 1977

7.14 Occupational Safety and Health Standards Title 29, Chapter XVII, Part 1910.36FR 10466

7.15 10CFR50 Appendix R, Fire Prot. Program for Nuclear Power Facilities Operating Prior to Jan. 1, 1979

7.16 **Commitment Documents:** - Hope Creek

7.16.1 CD-156A (NRC EI Circular 80-09)

7.16.2 CD-226C (NRC IE Bulletin 75-04A)

7.16.3 CD-300X (FSAR 8.1.4.14.3.1)

7.16.4 CD-317Y (FSAR Q 430.85(B))

7.16.5 CD-434E (IR 86-062) (LER 86-062)

7.16.6 CD-652A (INPO O&MR 73)

7.16.7 CD-682D (INPO SER 01-85)

7.16.8 CD-754X (FSAR 9.5.1.1.11)

7.16.9 CD-821X (FSAR 9.5.1.5.3)

TABLE 1
Ignitable Metals - Salem
Page 1 of 2

Location	System(s)	Component	Ignitable Metal
★ Units 1 & 2 Turbine Bldg. El. 100'	Turbine Auxiliary Cooling (TAC)	Heat Exchangers Nos. 11,12,21,22	Titanium Tubes
★ Units 1 & 2 Auxiliary Bldg. El. 84'	Component Cooling	Heat Exchangers Nos. 11,21,22	Titanium Tubes
★ Unit 1 Auxiliary Bldg. El. 84'	Component Cooling	Heat Exchanger No. 12	Titanium Plates
★ Unit 1 Auxiliary Bldg. El. 100'	Service Water	Diesel Generator Lube Oil Coolers & Jacket Water Heat Exchangers (Nos. 1A, 1B, 1C)	Titanium Tubes & Heads
★ Unit 2 Auxiliary Bldg. El. 100'	Service Water	Diesel Generator Lube Oil Coolers & Jacket Water Heat Exchangers (Nos. 2A, 2B, 2C)	Titanium Tubes & Heads
★ Units 1 & 2 Inner Piping Pen Area, El. 100'	Service Water	Chiller Condensers	Titanium Plates Nozzles & Flanges Water Box Housings Turn Around Heads
★Units 1 & 2 Service Water Intake Structure	Service Water	Service Water Pump Upper Motor Bearing Coolers Nos. 11-16& 21-23, 25 & 26	Titanium Tube Coils

TABLE 1
Ignitable Metals - Salem

Location	System(s)	Component	Ignitable Metal
★ Units 1 & 2 Turbine Bldg. El. 100'	Turbine Auxiliary Cooling (TAC)	Condensate Pump Upper Motor Oil Finned Coil Coolers	Titanium Tubes
★ Units 1 & 2 Auxiliary Bldg. El. 84'	Service Water	Charging Pump Bearing Oil Coolers Nos. 11, 12, 21, 22	Titanium Tubes & Tube Sheets
★ Units 1 & 2 Auxiliary Bldg. El. 84'	Service Water	Safety Injection Pump Bearing Oil Coolers Nos. 11, 12, 21, 22	Titanium Tubes & Tube Sheets
★ Unit 1 Auxiliary Bldg. El. 84'	Service Water	Chg/Safety Inj. Pump Speed Increaser Gear Lube Oil Cooler Nos. 11 & 12	Titanium Tubes & Tube Sheets
★ Unit 2 Auxiliary Bldg. El. 84'	Service Water	Chg/Safety Inj. Pump Speed Increaser Gear Lube Oil Cooler Nos. 21 & 22	Titanium Tubes & Tube Sheets
★ Circulating Water Intake Structure	Circ Water	Circ. Water Pump Motor Coolers	Titanium Tubes

TABLE 2
Ignitable Metals - Hope Creek

Equipment	Description	Ignitable Metal	Locations
1A1E-201 1A2E-201 1B1E-201 1B2E-201	SACS Heat Exchangers Tubing	Titanium	Reactor Bldg., El. 102' Room 4309 Room 4307
1A-E-217 1B-E-217	RACS Heat Exchangers Tubing	Titanium	Reactor Bldg., El. 77' Room 4211
1A-E-108 1B-E-108 1C-E-108	Main Condensers Tubing	Titanium	Turb Bldg., El. 54' - 102' Rooms 1310 to 1312
	Rx Fuel Channels Rx Fuel Spacers	Zirconium-4	Reactor Bldg El. 201'
	Rx Fuel Elements Tubing End Plugs	Zirconium-2	
	Limitorque Actuators Motor Core Rotor	Magnesium	Drywell

FORM 1

HOT WORK PERMIT
Page 1 of 2

PART 1: GENERAL INFORMATION

TYPE OF WORK:.....☐ W ☐ C ☐ B ☐ G ☐ OFS

OXY-ACETYLENE RIG TO BE USED:....☐ YES ☐ NO

LOCATION:.....☐ SGS U/1 ☐ SGS U/2 ☐ HCGS ☐ YARD

LOCATION DESCRIPTION:_____

ANNUAL PERMIT:.....☐ YES ☐ NO

ORDER NUMBER: _____

JOB SUPERVISOR (PRINT): _____

PART 2: WORK AREA INSPECTION

☐ Burning, cutting, and welding equipment is in good repair.

☐ Ignitable metals have been considered (i.e. titanium).

☐ Floor has been swept clean of loose combustibles within 35 feet.

☐ Wall, floor openings, cable trays and readily ignitable combustibles within a 35 foot radius which may be subjected to ignition sources have been covered.

☐ Combustibles have been moved away from opposite sides of walls or ceilings.

☐ Enclosed space/equipment has been cleaned and purged of combustibles. **[CD-682D]**

☐ Fire extinguisher is available (obtain extinguisher from Firehouse/Storeroom).

☐ Fire Watch will be provided during and 30 minutes after work completion to ensure work area is safe from fire danger.

☐ Precautions required for protection of filtration/adsorption ventilation systems have been taken (per NAP-9). **[CD-652A]**

☐ For electric arc welding, electromagnetic interference (EMI) precautions have been taken. (per NAP-5) **[CD-156A]**

Signature on page 2 indicates inspection was satisfactory and hot work activity may proceed following the acquisition of the DAILY hot work permit number.

FORM 1

HOT WORK PERMIT
Page 2 of 2

Hot Work Permit No.	Job Supervisor	Date	High Risk Only Fire Protection Supervisor	Date/Time

PART 3: FINAL CHECK-JOB COMPLETION

Work areas and all adjacent areas have been inspected following job completion and no evidence of fire is present. All equipment has been returned to its proper location.

VERIFIED BY: _____ _____

Job Supervisor Date/ Time

FORM 2
HOT WORK AUTHORIZATION LOG

KEY: W = Welding
C = Cutting
G= Grinding

B = Burning
OFS = Open Flame Soldering
OA = Oxygen Acetylene Unit

STATION: _____

Hot Work Number	Work Location / Equip Involved	Type of Work	Order Number	Job Supv. Ext.	Hi Risk	Det. Zone	FD Initials / Time
		W, C ,G, B, OFS, OA		x-	Yes No		
		W ,C ,G, B, OFS, OA		x-	Yes No		
		W ,C ,G, B, OFS, OA		x-	Yes No		
		W ,C ,G, B, OFS, OA		x-	Yes No		
		W ,C ,G, B, OFS, OA		x-	Yes No		
		W, C ,G, B, OFS, OA		x-	Yes No		
		W, C ,G, B, OFS, OA		x-	Yes No		
		W ,C ,G, B, OFS, OA		x-	Yes No		

ALL ACTIVE ON-THE-SPOT CHANGES MUST BE ATTACHED FOR FIELD USE
20030221

PSEG NUCLEAR

FIRE DEPARTMENT

NC.NA-AP.ZZ-0025 (Q) – Rev. 6

OPERATIONAL FIRE PROTECTION PROGRAM

Sponsor Organization: Emergency Services

REVISION SUMMARY: Biennial Review performed Yes _X_ No ___.

1. This is a full revision to NC.NA-AP.ZZ-0025 (Q) – Rev. 5. **(80035483).**
2. Due to the extensive nature of the revision, change bars have not been used. This revision represents a significant editorial change with regard to content, formatting, process descriptions, and section numbering.
3. This procedure has been revised to focus on high level process concepts, resulting in relocation of process description detail and associated forms, tables, and attachments to other lower tier administrative procedures.
4. These changes are editorial. There is no reduction in program content or control, as a result of this revision.
5. This revision relocates all Commitment Document numbers to the departmental administrative procedures implementing those commitments.
6. Revised titles to reflect the current organizational assignments.
7. Replaced Nuclear Electric Insurance Limited (NEIL) with Nuclear Services Organization (NSO) throughout the procedure.

IMPLEMENTATION REQUIREMENTS: Effective on: 1/28/03

1. Fire Protection Impairment, Hot Work, and Transient Combustible Permits processed by the Fire Department under Revision 5 as of the issue date, do not have to be changed in the field to meet Revision 6 requirements.
2. Nuclear Outlook article and PSEG Nuclear L.L.C. Intranet website announcement describing the revision.
3. Provide a global electronic Power Point presentation to all PSEG Nuclear personnel.

APPROVED: _____ 1/23/03
Manager - Emergency Services Date

APPROVED: _____ 1/24/03
Director - Business Support Date

Revision Summary (Continued)

8. Revised company name from PSE&G Nuclear Business Unit (NBU) to PSEG Nuclear.
9. Removed detailed information on the Combustible Control Program. All detailed information is now found in NC.FP-AP.ZZ-0025(Q).
10. Removed detailed information and forms for fire protection impairments. This information is now located in NC.FP-AP.ZZ-0025 (Q) and NC FP-AP.ZZ-0010 (Q).
11. Removed detailed information and forms for hot work operation and the control of ignition sources. This information is now located in NC.FP-AP.ZZ-0025 (Q) and NC.FP-AP.ZZ-0012(Q).

OPERATIONAL FIRE PROTECTION PROGRAM

TABLE OF CONTENTS

1.0 **PURPOSE**

This procedure addresses the Operational Fire Protection Program (OFPP) for PSEG Nuclear.

2.0 **SCOPE**

2.1 This procedure applies to all organizations and personnel engaged in activities which may affect the Operational Fire Protection Program (OFPP) for PSEG Nuclear.

3.0 **RESPONSIBILITIES**

3.1 **All Personnel**

Comply with the requirements of this procedure.

3.2 **Job Supervisors - Salem and Hope Creek Stations**

- Comply with fire prevention requirements of the Fire Protection Impairment, Hot Work, and Transient Combustible Control processes
- Adhere to controls for ignition sources and flammable liquids and gases.

3.3 **All Directors and Managers**

Ensure their organizations implement the applicable fire prevention requirements within facilities they maintain and in which they conduct work.

3.4 **Chemistry Manager**

Ensure the performance of the Hope Creek Diesel Driven Fire Pump Fuel Oil Storage Tank sampling surveillance.

3.5 **Operations Department Manager - Salem**

- Ensure the performance of the Diesel Driven Fire Pump Fuel Oil Storage Tank sampling surveillance
- Ensure that personnel within their organization, designated as the Operation's liaison to the Fire Department (FD), are available as required, during fire fighting activities.

3.6 **Operations Department Manager - Hope Creek**

Ensure that personnel within their organization designated as the Operation's liaison to the FD, are available as required, during fire fighting activities.

3.7 **Radiation Protection Manager**

Provide required radiological monitoring when their personnel respond with FD personnel to fires located within the Radiological Controlled Areas (RCAs).

3.8 **Work Control Center Personnel - Salem and Hope Creek Operations**

Notify the FD when a fire protection system/barrier is being affected and when it is being returned to service.

3.9 **Manager - Emergency Services**

- Ensure General Employee Training (GET) discusses required employee actions in the event of fire.
- Ensure a qualified, trained fire department is present on-site in accordance with NC.FP-AP.ZZ-0009(Q) Fire Protection Training Program.
- Develop and maintain FD surveillance and test procedures.
- Perform walk downs to monitor control of transient combustibles.
- Authorize Hot Work, Transient Combustible, and Fire Protection Impairment Permits (FPIPs) and ensuring the aggregate impact is considered.
- Maintain daily communication with the Operations Superintendent and/or Control Room Supervisors on the status of Salem and Hope Creek Station FPIPs.
- Ignition source control and monitoring for fire prevention.
- Fire incident reports and investigations.
- Interface with fire insurance inspectors, state and local enforcement agencies, and with offsite fire departments.
- Define non-FD responsibility for required testing and inspection of fire protection equipment.
- Performance of preventive and corrective maintenance on fire protection systems and components assigned to the FD.
- Administration and control of the Fire Watch Program.

3.10 **Vice President - Engineering**

- Establish programmatic requirements for implementing design related aspects of the Fire Protection and Fire Barrier Penetration Seal Programs.
- Development and review of Design Change Packages for controlling configuration of both the Fire Protection and Fire Barrier Penetration Seal Programs.
- Ensure a member grade Society of Fire Protection Engineer (SFPE) person exists within the organization. [CR960416211]

3.11 **Vice President - Operations**

- Ensure trained maintenance personnel perform preventive or corrective maintenance of Fire Protection systems in a prompt and effective manner.
- Ensure performance of preventive maintenance surveillances for fire protection components at both the Salem and Hope Creek Stations, as agreed upon with the Manager - Emergency Services.
- Ensure PSEG Nuclear facilities are maintained in compliance with the applicable state fire codes and insurance regulations.

4.0 **PROCESS DESCRIPTION**

4.1 This procedure identifies elements of the Operational Fire Protection Program categorized as **Fire Prevention** and **Readiness to Detect and Suppress Fires**.

4.2 This procedure is implemented in full detail by the following Fire Department administrative procedures:

4.2.1 NC.FP.AP.ZZ-0025(Q); Precautions Against Fire

4.2.2 SC.FP.AP.ZZ-0003(Q), Actions for Inoperable Fire Protection - Salem

4.2.3 HC.FP-AP.ZZ-0004(Q), Actions For Inoperable Fire Protection- Hope Creek

4.2.4 NC.FP-AP.ZZ-0005(Q), Fire Protection Surveillance / Periodic Test Program

4.2.5 NC.FP-AP.ZZ-0009Q), Fire Protection Training Program

4.2.6 NC.FP-AP.ZZ-0010(Q), Fire Protection Impairment Program

4.2.7 NC.FP-AP.ZZ-0012(Q), Safe Hot Work Practices

4.2.8 NC.FP-AP.ZZ-0020(Q), Compensatory Measure Fire Watch Program

5.0 **PROCEDURE**

5.1 **Response to Fire Emergencies**

5.1.1 Refer to NC.FP-AP.ZZ-0025(Q) (FPAP-25) for detailed instructions on response to fire emergencies. Key concepts are reviewed here.

- In the event a fire alarm sounds within an office facility, evacuate the building in accordance with the established routes. Accountability should be performed.
- NEVER use the elevator during a fire emergency building evacuation.
- For fire alarms within the Salem and Hope Creek Stations, listen to the instructions via the page system provided.

5.1.1 (Continued)

- Upon discovering a fire, your first duty is to report the fire. The Fire Department will respond with personnel and equipment to combat the emergency.
- The communication systems vary between the Salem and Hope Creek stations. Know the differences between the plants. Refer to FPAP-25.
- After reporting a fire, employees should fight the fire if its within their training and capabilities. If not, they should alert others in the area, evacuate to a safe distance and standby to direct FD personnel to the scene.

5.2 **Fire Prevention Requirements**

5.2.1 Refer to NC.FP-AP.ZZ-0025(Q) (FPAP-25) for detailed instructions on fire prevention work practices and requirements.

5.2.2 Some key concepts are briefly reviewed here.

- Fire Doors shall be maintained closed, except for normal passage, unless provided with an approved permit.
- It is imperative that **ALL PERSONNEL** make sure fire doors close behind them.
- Fire Protection deficiencies observed should be immediately reported to the FD (Ext. 2800/2803) for evaluation.
- Restrictions apply to combustible materials introduced into any safety-related area/room.
- Restrictions apply to locating combustible materials in the vicinity of permanent buildings.
- Restrictions apply to use of fire protection water for non-fire protection purposes.
- Storage of material should not affect access to, or obstruct, fire protection systems or fire fighting equipment.
- Fire extinguishers mounted in the stations are for emergency use only.
- Clearance, free of combustible material, shall be maintained around energized electrical panels.
- Extension cords and flexible cords shall not be a substitute for permanent wiring.

5.3 **Control of Ignition Sources - Hot Work**

5.3.1 Work involving ignition sources, such as welding, cutting, burning, grinding or open flame soldering, is considered hot work.

5.3.2 A Hot Work Permit will be required to control these ignition sources.

5.3.3 Plant work involving ignition sources, such as welding, cutting, burning, grinding or open flame soldering is considered a fire watch required activity.

- An exception to this requirement is allowed for work performed in shop areas.
- Hot work permits are required for all hot work activities.

5.4 **High Risk Hot Work**

5.4.1 High risk hot work is a PSEG Fire Department term applied to work with well defined and special hazard criteria.

- This type of hot work is distinguished from non-high risk hot work.
- High Risk hot work still involves ignition sources, such as welding, cutting, burning, grinding or open flame soldering.

5.4.2 High risk work involves:

- Vessels and systems that contain, or had contained, combustible or flammable chemicals, solvents, or gases.
- Work involving ignitable metals.
- Work involving processes that produce copious amounts of heat with molten slag and sparks. For example, current industrial technology may employ equipment such as "plasma arc" cutting machines. This technology produces copious amounts of energetic slag and sparks.

5.4.3 Special and additional controls apply to high risk hot work. These controls are covered under NC.FP-AP.ZZ-0012(Q), Safe Hot Work Practices.

5.5 **Combustible Material Control**

5.5.1 General Use and Controls - All areas:

- Combustibles Material will be controlled IAW NC.FP-AP.ZZ-0025 (Q).

5.5.2 For Safety Related Areas Only:

- Storage of combustibles is **NOT** permitted in Salem and Hope Creek Station's safety related areas/rooms except when approved by Engineering.

5.6 **Control of Flammable and Combustible Liquids and Gases**

5.6.1 Detailed guidance on the handling and control of flammable and combustible liquids and gases is provided in NC.FP-AP.ZZ-0025(Q).

5.7 **Control of Portable Space Heaters**

5.7.1 The use of portable space heaters is prohibited in safety related areas except as indicated IAW NC.DE-AP.ZZ-0030 Q)., Control of Temporary Modifications.

5.7.2 Space heaters in non- safety related areas are controlled IAW NC.FP-AP.ZZ-0025 (Q).

5.8 **Fire Protection Impairment Program**

5.8.1 FD is the fire protection impairment authority.

5.8.2 Impairments are formal controls that recognize, evaluate, and administratively approve and track either planned actions or emergent conditions that impact the operability of the fire protection system or a fire protection component.

5.8.3 Job supervisors prepare and submit impairment requests for work activities that will place fire protection equipment or system in an inoperable condition.

5.8.4 Nuclear fire protection supervisors (NFPS) evaluate impairment requests and emergent system and equipment issues to identify the full scope, or aggregate impact, of the issue.

- NFPS identifies the appropriate compensatory measures and generates an impairment, accordingly.

5.8.5 Prior to issuing impairments, the NFPS notifies and reviews the impairment issue with the Operations Superintendent or Control Room Supervisor (OS/CRS), of the affected station, and obtains OS/CRS approval to issue the impairment..

5.9 **Compensatory Measure Fire Watch Program**

5.9.1 Adequate compensatory measures are required to be implemented to supplement inoperable fire protection systems and equipment

5.9.2 Procedures SC.FP-AP.ZZ-0003(Q) and HC.FP-AP.ZZ-0004(Q) identify those systems and components requiring compensatory actions to be taken when a fire protection system or component is determined to be inoperable.

5.10 **Ignitable Metals**

5.10.1 The metals titanium, zirconium, and magnesium represent a special combustible material hazard class that warrants extra care and control.

5.10.2 These metals are classified as ignitable metals. These are metals that can burn readily, vigorously, and with a significant release of heat.

5.10.3 Controls relative to working on or in the vicinity of ignitable metals are provided in procedure NC.FP-AP.ZZ-0012(Q), Safe Performance of Hot Work.

6.0 **RECORDS**

6.1 Hot Work Permits and Transient Combustible Permits shall be retained with the work package in accordance with NC.NA-AP.ZZ-0011(Q), Records Management Program (NAP-11).

6.2 The FD, in accordance with NAP-11 shall retain transient combustible permits and logs pertaining to hot work permits and fire protection impairment permits for a minimum of six months.

6.3 FPIRs that were used to create FPIPs should be retained with work packages. The FD maintains a hard copy log of all FPIPs generated from FPIRs.

7.0 **DEFINITIONS**

7.1 **Combustible Liquid** - Liquid having a flash point at or above 100 deg-F (37.8° C).

7.2 **Combustible Material** - Material, which in the form it is used and under the conditions anticipated, **WILL** ignite, burn, support combustion, or release flammable vapors when subjected to fire and heat.

7.3 **Fire Barrier** - Those construction elements (walls, floors, and their supports), including beams, columns, penetration seals or closures, fire doors and dampers, that are rated by approving laboratories in hours of resistance to fire and used to prevent the spread of fire.

7.4 **Fire Watch** - An individual who has satisfactorily completed fire watch training and is designated by the FD for compensatory measure, hot work, or both fire watch type duties.

7.5 **Flammable Liquid** - A liquid with a flash point below 100 deg-F and a vapor pressure not exceeding 40 pounds per square inch absolute at 100 deg-F.

- Also known as a Class I liquid.

7.6 **Hot Work** - Work that involves ignition sources.

7.7 **Ignition Source** - Heat or flame from any source capable of igniting combustible or flammable material.

7.8 **Impairment** - Any condition affecting the intended operation of a fire protection system or barrier or posing as a fire hazard within the stations.

7.9 **Noncombustible Material** - Material which, in the form it is used and under the conditions anticipated, will **NOT** ignite, burn, support combustion, or release flammable vapors when subjected to fire and heat.

7.10 **Permanent** - A condition with an indefinite period of time.

7.11 **Safety Related Area** - An area that contains systems and components required to shut down the reactor, mitigate the consequences of postulated accidents or maintain the reactor in a safe cold shutdown condition.

7.12 **Storage** - The placing of items in a location and leaving them unattended until later use or disposal.

7.13 **Temporary** - planned time duration of no more than 60 days.

7.14 **Transient Combustible** - Any combustible material introduced into a Safety-Related area/room which will be used to complete an assigned task or left unattended for any period of time. It does not include items moved through areas in route to the final destination, unless the items are left un-monitored during transit.

8.0 **REFERENCES**

8.1 **General Documents**

8.1.1 Salem Updated Final Safety Analysis Report (UFSAR), Section 9.5, - Fire Protection System

8.1.2 Salem Fire Protection Report - Fire Hazards Analysis

8.1.3 HCGS Updated Final Safety Analysis Report, (UFSAR) Section 9.5.1, - Fire Protection Program

8.1.4 HCGS UFSAR, Appendix 9A, - Appendix R Comparison

8.1.5 Appendix A to Branch Technical Position APCSB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976

8.1.6 Branch Technical Position CMEB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants

8.1.7 NRC Supplemental Guidance, Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance, August 1977

8.1.8 Occupational Safety and Health Standards Title 29, Chapter XVII, Part 1910.36FR 10466

8.1.9 10CFR50 Appendix R, Fire Prot. Program for Nuclear Power Facilities Operating Prior to Jan. 1, 1979

8.1.10 NC.DE-PS.ZZ-0001(Q), Fire Protection Programmatic Standard

8.1.11 IEEE Standard 634 - 1978

8.1.12 Hope Creek Station Generic Letter 86-10 Submittal, dated May 13, 1986

8.1.13 HC.DE-PS.ZZ-0021(F), Hope Creek Penetration Seal Program

8.1.14 SC.DE-PS.ZZ-0035(Q), Salem Penetration Seal Program

8.1.15 HC.FP-EO.ZZ-0001(Z), Control Room Fire Response - HCGS

8.1.16 SC.FP-EO.ZZ-0001(Z), Control Room Fire Response - Salem

8.2 **Cross References:**

8.2.1 NC.NA-AP.ZZ-0011(Q), Records Management Program

8.2.2 NC.DE-AP.ZZ-0030(Q), Control of Temporary Modifications

8.2.3 SC.FP.AP.ZZ-0003(Q), Actions for Inoperable Fire Protection - Salem

8.2.4 HC.FP-AP.ZZ-0004(Q), Actions For Inoperable Fire Protection- Hope Creek

8.2.5 NC.FP-AP.ZZ-0005(Q), Fire Surveillance and Periodic Test Program

8.2.6 NC.FP-AP.ZZ-0009(Q), Fire Department Training Program

8.2.7 NC.FP-AP.ZZ-0010(Q), Fire Protection Impairment Program

8.2.8 NC.FP-AP.ZZ-0012(Q), Safe Performance of Hot Work

8.2.9 NC.FP-AP.ZZ-0020(Q), Compensatory Measure Fire Watch Program

8.2.10 NC.FP.AP.ZZ-0025(Q); Precautions Against Fire