ECG SECTIONS

HOPE CREEK ECG

SECTIONS 1 - 11
EMERGENCY ACTION LEVELS (EALs)
&
REPORTING ACTION LEVELS (RALs)

ECG SECTIONS

NOTE: THESE 11 SECTIONS WILL REPLACE THE 18 SECTIONS CURRENTLY IN THE ECG.

HOPE CREEK EVENT CLASSIFICATION GUIDE
TABLE OF CONTENTS/SIGNATURE PAGE
October 15, 1995

HOPE CREEK EVENT CLASSIFICATION GUIDE
TABLE OF CONTENTS/SIGNATURE PAGE
October 15, 1995

HOPE CRREK EVENT CLASSIFICATION GUIDE
TABLE OF CONTENTS/SIGNATURE PAGE
October 15, 1995

SIGNATURE PAGE

Prepared By: ______________________________ ________
(Editorial Revisions Only, Last Approved Revision) Date

Section/Attachments Revised ______________________________ ________
(List Non Editorial Only - Section/Attachments) Date

Reviewed By: ______________________________ ________
Station Qualified Reviewer Date

Reviewed By: ______________________________ ________
Department Manager Date

Reviewed By: ______________________________ ________
Manager - Emergency Preparedness & Radiological Support Date

Reviewed By: ______________________________ ________
Director - Quality Assurance/Safety Review Date
(If Applicable)

SORC Review and Station Approvals

______________________________ ______________________________
Mtg. No. Hope Creek Chairman General Manager - Hope Creek Operations

________ ________
Date Date

HOPE CREEK
EVENT CLASSIFICATION GUIDE
INTRODUCTION & USAGE
Section i

I. PURPOSE OF THE EVENT CLASSIFICATION GUIDE (ECG)

A. To provide a central reference document which enables the Senior Nuclear Shift Supervisor (SNSS) or the Emergency Coordinator (EC) to classify emergency or non-emergency events and conditions.

B. To provide the required procedures for immediate and prompt notifications and direction to other required written reports.

C. To direct the Emergency Coordinator to implement procedures which will ensure appropriate response as required by the classified emergency level.

II. EMERGENCY CLASSIFICATION DESCRIPTIONS

A. Emergency Classes:

1. The NRC/FEMA established four emergency classes for fixed nuclear facilities.

2. An emergency class is used for grouping off-normal nuclear power plant conditions according to their relative radiological seriousness and the time sensitive onsite and offsite actions needed to respond to such conditions.

3. The four emergency classes in ascending order are:

Unusual Event (UE)	Least Severe
Alert (A)	|
Site Area Emergency (SAE)	|
General Emergency (GE)	Most Severe

B. Unusual Event:

1. Plant events which are in progress or have occurred which indicate a potential degradation of the plant safety level.

2. The lowest level of emergency at the plant, which can usually be handled by the normal operating shift.

3. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. Dose consequences would not exceed 5 mRem TEDE.

C. Alert:

1. Plant events which are in progress or have occurred that are more serious than an Unusual Event which involve an actual or potential substantial degradation of the plant safety level.

2. Emergency Response personnel are required in addition to the normal operating shift. The entire emergency response organization is called in. The TSC is activated, and the EOF and ENC are manned and may activate if needed for support.

3. Any release of radioactive material is expected to be limited to a small fraction of the EPA Protective Action Guideline exposure levels. Dose consequences not to exceed 100 mRem TEDE.

D. Site Area Emergency:

1. Serious plant events are in progress or have occurred which involve actual or likely major failure of plant functions required for protection of the public.

2. The entire emergency response organization is activated.

3. Any release of radioactive material is not expected to exceed EPA Protective Action Guideline exposure levels beyond the plant boundary. Dose consequences not to exceed 1000 mRem TEDE.

E. General Emergency:

1. Serious plant events are in progress or have occurred which involve actual or imminent core degradation or core melting with potential for loss of containment integrity.

2. The entire emergency response organization is activated.

3. Release of radioactive material can be expected to exceed

EPA Protective Action Guideline exposure levels of 1000 mRem TEDE offsite.

III. **EVENT CLASSIFICATION GUIDE STRUCTURE**

A. The ECG is divided into 3 segments which are:

1. ECG Front Matter: Information which include the Table of Contents, Introduction/Reference, and a Glossary of acronyms.

2. ECG Sections: Flow charts diagrams used to classify events/conditions as emergencies or non-emergencies.

3. ECG Attachments: Implementing documents that provide direction for emergency and non-emergency classification, notification, reporting requirements, references and forms required to facilitate event communications.

B. ECG Sections Format

With the exception of ECG Section 3, the ECG section flowcharts are comprised of the following segments:

1. Initiating Condition (IC): A generic nuclear power plant condition or event where either the potential exist for a radiological emergency OR non-emergency reportable event OR such an emergency OR non-emergency reportable event has occurred.

2. OPCON: Refer to the Operating Condition at Hope Creek during which a particular IC/EAL is applicable. The OPCON that the plant was in when the event started, prior to any protective system or operator actions, should be utilized when classifying events.

3. EAL Number (EAL#): Each Emergency Action Level (EAL) has been assigned a unique numeric identifier called the EAL#. This EAL# is used in communication within PSE&G's Emergency Response Organization as well as when communicating with offsite officials who use an offsite reference manual which is indexed in accordance with the EAL#'s. Each digit of the EAL# has a specific meaning that is not important to the users but is important to the personnel who develop and maintain the ECGs. The digit and EAL# are defined below.

Sample EAL# = 9.4.1.a

First Digit = Identifies which section of the ECG that a particular EAL is contained in. In the example the Digit 9 identifies that the EAL is from Section 9, Hazards.

Second Digit= Identifies the subsection that the EAL is contained in. In the above example the Digit 4 identifies that the EAL is found in subsection 4 of Section 9 thus 9.4, Toxic Gases.

Third Digit = The third digit identifies the emergency class associated with that particular EAL as follows:

If 3rd Digit is a 1 THEN EAL results in UE
If 3rd Digit is a 2 THEN EAL results in A
If 3rd Digit is a 3 THEN EAL results in SAE
If 3rd Digit is a 4 THEN EAL results in GE

If looking at a RAL in Section 11 ONLY, the Third Digit identified the type of non-emergency event report to be made as follows.

If 3rd Digit is a 1 THEN RAL is 1hr report
If 3rd Digit is a 2 THEN RAL is 4hr report
If 3rd Digit is a 3 THEN RAL is 24hr report OR GREATER

Fourth Digit= If a fourth digit is used, it is always a lower case letter and delineate one of multiple events which lead to similar emergency or non-emergency class levels. In the above example the "a" delineate 1 of 2 EALs that result in an Unusual Event and fall under a common initiating condition.

4. Emergency Action Level (EAL) or Reporting Action level (RAL): A predetermined, site-specific, or observable threshold used to define a generic initiating condition that places the plant in a given emergency class or non-emergency report. An EAL/RAL can be an instrument reading, an equipment status indicator, a measurable parameter, a discrete observable event, analysis results, entry into specific EOPs, or another phenomenon which indicates the need for classification of an emergency or non-emergency.

5. <u>Action Required:</u> Identifies the specific emergency class or non-emergency report that is required and refers the user to a specific ECG Attachment for implementation direction for the emergency or non-emergency event declared.

C. ECG Attachments:

1. The ECG attachments are comprised of various formats since the attachments are used for implementing directions, phone listings, and informational data.

IV. <u>EVENT CLASSIFICATION GUIDE (ECG) USE</u>

A. The Sections of the ECG are a guide. The EALs described in the ECG are not all inclusive and will not identify each and every condition, parameter or event which could lead to an event classification. If the Emergency Coordinator, using his best judgment, determines an Initiating Condition has been satisfied but the specific EAL <u>is in question</u>, he/she should promptly classify the event in accordance with the Initiating Condition. If it is clear that the EAL has not been satisfied, then the Emergency Coordinator should not classify the event based on the Initiating Condition (IC). In any event, if the plant conditions are equivalent to one of the four emergency classes as described in Section II above, that classification should be declared.

Assessment of an Emergency Condition should be completed in a timely manner which is considered to be within about 15 minutes of recognition of an event. If an EAL specifies a duration (e.g. loss of annunciations for >15 min), then the assessment time runs concurrently with the EAL duration time and is the same length. If an event is recognized or reported and the required duration is <u>known</u> to have already been exceeded then the duration portion of the EAL should be considered as being satisfied and the assessment time for the remaining portions of the EAL should be within about 15 minutes from the time of recognition.

B. The ECG is not a stand alone document. At times, the ECG will refer the user to other attachments or procedures for accomplishment of specific evolutions such as: accountability, recovery, development of PARs, etc. They should be followed in a step-by-step fashion.

The ECG should be considered an "Implementing Procedure" and used in accordance with the requirements of a "Category II" procedure as defined in NC.NA-AP.ZZ-0001(Q) (see definition of Category II below). The ECG's classification Sections allow for judgement and decision making as to whether or not an Emergency Action Level (EAL) is exceeded.

> NOTE
>
> The word user (person assigned to implement a specific procedure), has been substituted for OPERATOR/TECHNICIAN since EPIPs and ECG Attachments are implemented by personnel in various job classification.

CATEGORY II - PROCEDURE-AT-THE-JOB

The procedure shall be at the job site. The user shall refer to the procedure at the beginning and end of the job, and as frequently as necessary (based on the task, experience of the user, and familiarization with the task) to complete the job in accordance with the procedure. The user is responsible for completing the procedure correctly regardless of how often he refers to it. Data, hold print, and notifications shall be recorded before proceeding to the next step. Place-keeping checkmark points may be provided, for example, by parentheses "()", and should be completed each time the procedure is referred to and at the end of the procedure. The job supervisor may require the user to perform these procedures with "procedure-in-hand", if he/she feels it is appropriate.

C. To use this ECG volume, follow this sequence:

> NOTE:
>
> Confirmation of actual plant conditions should be made by comparing redundant instrumentation, indications, and/or alarms.

1. Assess the event and/or plant conditions and determine which ECG section(s) is most appropriate.

2. Refer to Section EAL Flowchart diagram(s), review and

identify the initiating condition(s) that are related to the event/condition that has occurred or is ongoing.

(ECG Section 3 has its own unique usage instruction as part of the Fission Product Barrier Table 3.0)

NOTE:
The Emergency Coordinator should classify and declare an emergency before an Emergency Action Level (EAL) is exceeded if, using his best judgement, it is determined that the EAL will be exceeded.

3. Review and assess the associated EALs or RALs as compared to the event and select the highest appropriate emergency or reportable action level. If identification of an EAL is questionable refer to paragraph IV.A above. If there is any doubt with regard to assessment of a particular EAL or RAL, the ECG Basis Document can be reviewed. Words contained in an EAL OR RAL that are bold face are either threshold values associated with that action level OR are words that are defined in the basis for that specific EAL/RAL.

4. Identify and implement the referenced Attachment.

5. After classification and Attachment initiation, return to the ECG Section to review action levels that may result in escalation/deescalation of the emergency level.

D. Guidance for EMERGENCY/NON-EMERGENCY conditions discovered after-the-fact.

NOTE:
Plant emergency events that are in progress or that have occurred with ongoing consequences, effects, or corrective actions should not be considered "After-The-Fact" events and should therefore be classified and declared as an ongoing emergency event.

1. EMERGENCY CONDITIONS - if "After-The-Fact" (not ongoing at the time of discovery) it is discovered that an event

or condition had occurred that exceeded an Emergency Action Level (EAL) but was not declared as an emergency, then an emergency declaration is NOT required. A non-emergency, One-Hour Report should be initiated in accordance with ECG Section 11.6, After-The-Fact.

2. NON-EMERGENCY CONDITIONS - if "After-The-FACT" (regardless of whether the event is on-going at the time of discovery) it is discovered that an event or condition had occurred that should have resulted in the classification and implementation of a non-emergency report (1 hour, 4 hour, 24 hour), the applicable non-emergency report Attachment in the ECG should be implemented.

E. Guidance concerning NRC communications during an emergency.

1. Complete and accurate communications with the NRC Operations Center during emergencies is required and expected. The purpose of notifying the NRC within one-hour of an emergency, is to provide event information when immediate NRC action may be required to protect the public health and safety <u>OR</u> when the NRC needs accurate and timely information to respond to heightened public concern. If the information we provide is not accurate or does not contain sufficient detail, then we hamper the NRC from doing their job.

2. The NRC Data Sheet, along with the Initial Contact Message Form, is the primary vehicle to ensure the NRC is kept informed. General Guidance on completing the event description portion of the NRC Data Sheet is provided in Attachment 5 of the ECG.

HOPE CREEK GENERATING STATION
Emergency Action Levels and Reportable Action Levels
Glossary of Acronyms
Section ii

AC	-	Alternating Current
ADS	-	Automatic Depressurization System
APRM	-	Average Power Range Monitor
ARI	-	Alternate Rod Insertion
ATWS	-	Anticipated Transient Without Scram
BNE	-	Bureau of Nuclear Engineering (NJDEPE)
CACS	-	Containment Atmosphere Control System
CEDE	-	Committed Effective Dose Equivalent
CDE	-	Committed Dose Equivalent
CFR	-	Code of Federal Regulations
CIS	-	Containment Isolation System
CNTMT	-	Containment
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)
DEI	-	Dose Equivalent Iodine
DEMA	-	Delaware Emergency Management Agency
DEPE	-	NJ Department of Environmental Protection & Energy
DID	-	Direct Inward Dial (phone system)
EACS	-	ESF Equipment Area Cooling System
EAL	-	Emergency Action Level
EC	-	Emergency Coordinator
ECCS	-	Emergency Core Cooling Systems
EDG	-	Emergency Diesel Generator
EDO	-	Emergency Duty Officer
EMRAD	-	Emergency Radio (NJ)

ENC	-	Emergency News Center
ENS	-	Emergency Notification System (NRC)
EOF	-	Emergency Operations Facility
EOP	-	Emergency Operations Procedures
EPA	-	Environmental Protection Agency
ERM	-	Emergency Response Manager
FC	-	Fuel Clad (Barrier)
FRVS	-	Filtration, Recirculation, and Ventilation System
FTS	-	Federal Tele Communications System (NRC)
GE	-	General Electric
GE	-	General Emergency
GPM	-	Gallons Per Minute
HCLL	-	Heat Capacity Level Limit
HCGS	-	Hope Creek Generating Station
HCTL	-	Heat Capacity Temperature Limit
HPCI	-	High Pressure Coolant Injection
HTV	-	Hardened Torus Vent
HWCI	-	Hydrogen Water Chemical Injection
IC	-	Initiating Condition
ICMF	-	Initial Contact Message Form
IRM	-	Intermediate Range Monitor
KV	-	KiloVolt
LAC	-	Lower Alloways Creek
LCO	-	Limiting Condition for Operation
LDE	-	Lens Dose Equivalent
LOCA	-	Loss of Coolant Accident
LPCI	-	Low Pressure Coolant Injection
LPZ	-	Low Population Zone
MEA	-	Minimum Exclusion Area
MET	-	Meteorological
MPH	-	Miles Per Hour
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
NETS	-	Nuclear Emergency Telecommunications System

NFPB	-	Normal Full Power Background
NJSP	-	New Jersey State Police
NPV	-	North Plant Vent
NRC	-	Nuclear Regulatory Commission
NSSSS	-	Nuclear Steam Supply Shutoff System
ODCM	-	Offsite Dose Calculation Manual
OEM	-	Office of Emergency Management (NJ)
OPCON	-	Operating Condition
OSC	-	Operations Support Center
PAG	-	Protective Action Guidelines
PC	-	Primary Containment (Barrier)
PCIG	-	Primary Containment Instrument Gas System
PCIS	-	Primary Containment Isolation System
PSIG	-	Pounds Square Inch Gauge
RAL	-	Reporting Action Level
RC	-	Reactor Coolant
RCIC	-	Reactor Core Isolation Cooling
RCS	-	Reactor Coolant System (Barrier)
RHR	-	Residual Heat Removal (Containment Heat Removal)
RMS	-	Radiation Monitoring System
RPS	-	Reactor Protection System
RPV	-	Reactor Pressure Vessel
RRCS	-	Redundant Reactivity Control System
SACS	-	Safety Auxiliaries Cooling System
SAE	-	Site Area Emergency
SBO	-	Station Blackout
SCP	-	Security Contingency Procedure
SDE	-	Skin Dose Equivalent
SDM	-	Shutdown Margin
SLC	-	Standby Liquid Control
SJAE	-	Steam Jet Air Ejector
SNM	-	Special Nuclear Material
SNSS	-	Senior Nuclear Shift Supervisor
SPDS	-	Safety Parameter Display System
SPV	-	South Plant Vent
SRM	-	Source Range Monitor
SRV	-	Safety Relief Valve

SSCL	-	Station Status Checklist
SSWS	-	Station Service Water System
TAF	-	Top of Active Fuel
TEDE	-	Total Effective Dose Equivalent
TSC	-	Technical Support Center
UE	-	Unusual Event
UFSAR	-	Updated Final Safety Analysis Report
UHS	-	Ultimate Heat Sink
USCG	-	United States Coast Guard
VDC	-	Volts Direct Current

FILE: acronyms.hc
Rev date 5/19/95

1.0 Fuel Clad Challenge

1.1 RCS Activity

Initiating Condition: Fuel Clad Degradation

EMERGENCY ACTION LEVELS

OPCON	1,2,3,4,5	1,2,3,4	1,2,3,4	1,2,3
EAL #	**1.1.1.a**	**1.1.1.b**	**1.1.1.c**	**1.1.2**

1.1.1.a IF Reactor Coolant Sample Activity **> 4 µCi/gm Dose Equivalent I-131**

1.1.1.b IF **Valid** Offgas Pretreatment Radiation Monitor (9RX621 / 9RX622) High Alarm Condition **(≥ 2.2E+04 mRem/hr)**

1.1.1.c IF **Valid** Main Steam Line Radiation Monitor High High Alarm Condition **(≥ 3 times Normal Full Power Background)**

THEN → **Action Required:** Refer to Attachment 1 **UNUSUAL EVENT**

1.1.2 (any of the above) AND <u>ANY</u> SRV is determined to be **Stuck Open**

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

<u>NOTE</u>:
Refer to Section 3.0, Fission Product Barrier Table prior to Event Classification

2.0 RCS Challenge

2.1 RCS Leakage

Initiating Condition: RCS Leakage

EMERGENCY ACTION LEVELS

	2.1.1.a	2.1.1.b	2.1.1.c	2.1.1.d
OPCON	1, 2, 3	1, 2, 3	1, 2, 3	1, 2, 3
IF	Reactor Coolant System Pressure Boundary Leakage **> 10 gpm** (Using 10 minute average)	Reactor Coolant System Unidentified Leakage **> 10 gpm** (Using 10 minute average)	Reactor Coolant System Identified Leakage **> 25 gpm** averaged over any 24 hour period	**Successful** Isolation of a Reactor Recirc Pump Dual Seal Failure within **10 minutes** of **recognition**

THEN

NOTE:
Refer to Section 3.0, Fission Product Barrier Table prior to Event Classification

Action Required:
Refer to Attachment 1
UNUSUAL EVENT

3.1 Fuel Clad Barrier

3.1.1 REACTOR WATER LEVEL

POTENTIAL LOSS = 3 PTs	LOSS = 4 PTs
EAL # 3.1.1.a Reactor Water Level REACHES **- 161" (Top of Active Fuel)** EXCLUDING intentional lowering of Reactor Water Level during an ATWS	**EAL # 3.1.1.b** Reactor Water Level CANNOT BE RESTORED AND MAINTAINED above **-200"** (Minimum Zero Injection RPV Water Level)

- OR -

3.1.2 DRYWELL ATMOSPHERE POST ACCIDENT (DAPA) RADIATION LEVEL

POTENTIAL LOSS = 0 PTs	LOSS = 4 PTs
Not Applicable	**EAL # 3.1.2** DAPA Radiation Monitor reading **≥ 5000 R/hr**

- OR -

3.1.3 RCS IODINE CONCENTRATION

POTENTIAL LOSS = 0 PTs	LOSS = 4 PTs
Not Applicable	**EAL # 3.1.3** Reactor Coolant Sample Activity **≥ 300 µCi/gm Dose Equivalent I-131**

- OR -

3.1.4 EMERGENCY COORDINATOR JUDGEMENT

POTENTIAL LOSS = 3 PTs	LOSS = 4 PTs
EAL # 3.1.4 ANY condition, in the opinion of the EC, that indicates a Potential Loss (3 pts) or Loss (4 pts) of the Fuel Clad Barrier	

3.2 RCS Barrier

3.2.1 REACTOR WATER LEVEL

POTENTIAL LOSS = 3 PTs	LOSS = 4
EAL # 3.2.1.a Reactor Water Level REACHES **-129"** EXCLUDING intentional lowering of Reactor Water Level during an ATWS	**EAL # 3.2.1.b** Reactor Water Level REACHES **-161" (Top of Active Fuel)** EXCLUDIN intentional lowering of Rea Water Level during an ATV

- OR -

3.2.2 RCS LEAK RATE/ DRYWELL PRESSURE

POTENTIAL LOSS = 3 PTs	LOSS = 4
EAL # 3.2.2.a **Unisolable** RCS Leak Rate **≥ 50 GPM** INSIDE Primary Containment	**EAL # 3.2.2.b** **Valid** High Drywell Pressu Condition (**≥ 1.68 psig**)

- OR -

3.2.3 RCS LINE BREAK/CONTAINMENT BYPASS

POTENTIAL LOSS = 3 PTs	LOSS = 4
EAL # 3.2.3.a **Main Steam Line Break** OUTSIDE Primary Containment, resulting in an AUTOMATIC MSIV Isolation Signal **AND** ALL 4 Main Steam Lines have been successfully isolated based on NO indication of CONTINUING FLOW / LEAKAGE OUTSIDE the Primary Containment AFTER **valve closure** from the Main Control Room has been attempted	**EAL # 3.2.3.b** **RCS Line Break** OUTSID Primary Containment, resulting in a **Valid** Isolatio Signal for ANY one of the following systems: • NSSSS • HPCI • RCIC **AND** Indication of CONTINUINC FLOW / LEAKAGE OUTSIDE the Primary Containment through the **effected system** AFTER **valve closure** from the Main Control Room has been attempted

- OR -

3.2.4 EMERGENCY COORDINATOR JUDGEMENT

POTENTIAL LOSS = 3 PTs	LOSS = 4
EAL # 3.2.4 ANY condition, in the opinion of the EC, that indicates a Potential Loss (3 pts) or Loss (4 pts) of the RCS Barrier	

FUEL CLAD BARRIER EAL# ________

POINT VALUE 0 / 3 / 4 (circle one) +

RCS BARRIER EAL# ________

POINT VALUE 0 / 3 / 4 (circle one)

3.3 CNTMT Barrier

3.3.1 REACTOR WATER LEVEL

POTENTIAL LOSS = 1 PT	LOSS = 0 PTs
EAL # 3.3.1 Reactor Water Level CANNOT BE RESTORED AND MAINTAINED above **-200"** (Minimum Zero Injection RPV Water Level)	**Not Applicable**

- OR -

3.3.2 DRYWELL PRESSURE

POTENTIAL LOSS = 1 PT	LOSS = 2 PTs
EAL # 3.3.2.a **Containment Venting is Required** by the Emergency Operating Procedures (EOPs) EXCLUDING Containment Venting due to an ATWS	**EAL # 3.3.2.b** **Containment Failure** as indicated by a rapid decrease in Drywell pressure following an increase in pressure above **1.68 psig**

- OR -

3.3.3 DRYWELL ATMOSPHERE POST ACCIDENT (DAPA) RADIATION LEVEL

POTENTIAL LOSS = 1 PT	LOSS = 0 PTs
EAL # 3.3.3 DAPA Radiation Monitor reading ≥ **28000 R/hr**	**Not Applicable**

- OR -

3.3.4 RCS LINE BREAK/CONTAINMENT BYPASS

POTENTIAL LOSS = 1 PT	LOSS = 2 PTs
EAL # 3.3.4.a **RCS Line Break** OUTSIDE Primary Containment, resulting in a **Valid** Isolation Signal for ANY one of the following systems: • NSSSS (excluding Main Steam Lines) • HPCI • RCIC **AND** NO indication of CONTINUING FLOW / LEAKAGE OUTSIDE the Primary Containment through the **effected system** AFTER **valve closure** from the Main Control Room has been attempted	**EAL # 3.3.4.b** Isolation Signal for ANY one of the following systems: • NSSSS • PCIS • HPCI • RCIC **AND** Indication of CONTINUING FLOW / LEAKAGE OUTSIDE the Primary Containment through the **effected system** AFTER **valve closure** from the Main Control Room has been attempted

- OR -

3.3.5 EMERGENCY COORDINATOR JUDGEMENT

POTENTIAL LOSS = 1 PT	LOSS = 2 PTs
EAL # 3.3.5 ANY condition, in the opinion of the EC, that indicates a Potential Loss (1 pt) or Loss (2 pts) of the Containment Barrier	

TABLE 3.0 FISSION PRODUCT BARRIERS

APPLICABLE OPERATIONAL CONDITIONS ARE 1, 2, 3 ONLY

NOTE
If the Loss or Potential Loss is considered IMMINENT (may occur within 2 hours), use judgement and classify as if the threshold is exceeded.

Usage Instructions:

1. In the table to the left, review the Emergency Action Levels of all columns and identify which need further review.
2. For each of the three barriers, determine the EAL with the highest point value; enter that EAL # in the space provided at the bottom of the column, and circle the corresponding point value. No more then one EAL should be selected for each barrier.
3. Use the tabulation section at the bottom of the table and add the point values circled for the three barriers and enter the sum below:
4. Classify based on the point value sum as follows:

If the sum is:	Classify as:	Refer to Attachment:
1, 2	UNUSUAL EVENT	1
3, 4	ALERT	2
5, 6, 7, 8	SITE AREA EMERGENCY	3
9,10	GENERAL EMERGENCY	4

____
CLASSIFICATION

5. Implement the appropriate ECG attachment per above chart.
6. Continue to review the EALs on this Table for changes that could result in emergency escalation or deescalation.

CNTMT BARRIER EAL # _______ Total (All 3 barriers)

POINT VALUE 0 / 1 / 2 (circle one) = _________ Emergency Classification Points

9508300383-01

4.0 Miscellaneous

4.1 Emergency Coordinator Discretion

Initiating Condition	Other Conditions Exist Which In the Judgement of the Emergency Coordinator Warrant Declaration of an Unusual Event	Other Conditions Exist Which In the Judgement of the Emergency Coordinator Warrant Declaration of an Alert	Other Conditions Exist Which In the Judgement of the Emergency Coordinator Warrant Declaration of a Site Area Emergency	Other Conditions Exist Which In the Judgement of the Emergency Coordinator Warrant Declaration of a General Emergency
OPCON	All	All	All	All
<u>EAL #</u>	<u>4.1.1</u>	<u>4.1.2</u>	<u>4.1.3</u>	<u>4.1.4</u>
EMERGENCY ACTION LEVELS	IF Events are in progress or have occurred which, in the judgement of the Emergency Coordinator, indicate a **Potential Degradation of Plant Safety** THEN	IF Events are in progress or have occurred which, in the judgement of the Emergency Coordinator, indicate plant safety systems **(more than one)** are, or may be degraded AND Increased monitoring of plant functions is warranted THEN	IF Events are in progress or have occurred which, in the judgement of the Emergency Coordinator, indicate <u>EITHER</u> one of the following: • The potential for an uncontrolled radiological release or the source term available in the Containment atmosphere, could result in Site Boundary dose rates in **excess of 100 mRem/hr** • Criteria for declaration of a Site Area Emergency per the ECG Introduction Section exists THEN	IF Events are in progress or have occurred which, in the judgement of the Emergency Coordinator, indicate <u>EITHER</u> one of the following: • The potential for uncontrolled radiological releases expected to **exceed** Protective Action Guidelines levels per EAL 6.1.4.a • Criteria for declaration of a General Emergency per the ECG Introduction Section exists 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

	5.1.2.a	5.1.2.b	5.1.3	5.1.4
Initiating Condition	Failure of the Reactor Protection System (RPS) to Successfully Complete a Reactor Scram (Automatic or Manual)		Failure of the Reactor Protection System (RPS) to Successfully Complete a Reactor Scram (Automatic and Manual) and Reactor Power is above 4%	Failure of the Reactor Protection System (RPS) to Successfull; Complete a Reactor Scram (Automatic and Manual) and there is indication of an Extreme Challenge to the Ability to Cool the Core
OPCON	1, 2	1, 2	1, 2	1, 2
EAL #	**5.1.2.a**	**5.1.2.b**	**5.1.3**	**5.1.4**
EMERGENCY ACTION LEVELS	IF An Automatic Reactor Scram Condition exists AND An Automatic Reactor Scram (RPS) IS NOT **successful** THEN	IF ANY Manually Initiated Reactor Scram (RPS) from the Control Room IS NOT **successful** THEN	AND ALL Reactor Scram attempts from the Control Room (RPS and ARI) DID NOT REDUCE and MAINTAIN Reactor Power to $\leq$ **4%** THEN	AND EITHER one of the following: • Reactor Water Level CANNOT BE MAINTAINED > **-190"** • The combination of Suppression Pool Temperature and RPV Pressure CANNOT BE MAINTAINED below the HCTL Curve THEN
Action Required	Refer to Attachment 2 **ALERT**		Refer to Attachment 3 **SITE AREA EMERGENCY**	Refer to Attachment 4 **GENERAL EMERGENCY**

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

Initiating Condition: Any **Unplanned** Release of Gaseous Radioactivity to the Environment that Exceeds 2 Times the Radiological Technical Specifications for 60 minutes or longer

EMERGENCY ACTION LEVELS

EAL # 6.1.1.a
OPCON: All

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

EAL # 6.1.1.b
OPCON: All

Measured Dose Rate

IF

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

EAL # 6.1.1.c
OPCON: All

Sample Analysis

IF

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

- **FRVS:**
  - **≥ 5.65E-03 µCi/cc Total Noble Gas**
  - **≥ 8.00E-06 µCi/cc I-131**
- **NPV:**
  - **≥ 1.21E-03 µCi/cc Total Noble Gas**
  - **≥ 1.72E-06 µCi/cc I-131**
- **SPV:**
  - **≥ 1.13E-04 µCi/cc Total Noble Gas**
  - **≥ 1.61E-07 µCi/cc I-131**

EAL # 6.1.1.d
OPCON: All

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 1/3; 9RX518)

AND

Total Plant Vent release rate EXCEEDS one of the following limits:

- **2.40E+04 µCi/sec Total Noble Gas**
- **3.40E+01 µCi/sec I-131** (USE FOR NPV & SPV ONLY)

AND

Dose Assessment is NOT available

AND

Release is ongoing for **≥ 60 minutes**

THEN

Action Required: Refer to Attachment 1 **UNUSUAL EVENT**

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

Initiating Condition: Any **Unplanned** Release of Gaseous Radioactivity to the Environment that Exceeds 200 Times the Radiological Technical Specifications for 15 minutes or longer

EMERGENCY ACTION LEVELS

EAL #	6.1.2.a	6.1.2.b	6.1.2.c	6.1.2.d
OPCON	All	All	All	All
	Dose Assessment	Measured Dose Rate	Sample Analysis	Alarm Indications
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**	Dose Rate measured at the Protected Area Boundary or beyond EXCEEDS **5 mRem/hr**	Total gaseous effluent release sample analysis for ANY one of the following indicates a concentration of: • **FRVS:** **≥ 5.65E-01 µCi/cc Total Noble Gas** **≥ 8.00E-04 µCi/cc I-131** • **NPV:** **≥ 1.21E-01 µCi/cc Total Noble Gas** **≥ 1.72E-04 µCi/cc I-131** • **SPV:** **≥ 1.13E-02 µCi/cc Total Noble Gas** **≥ 1.61E-05 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 1/3; 9RX518) **AND** Total Plant Vent release rate EXCEEDS **2.40E+06 µCi/sec Total Noble Gas** **AND** Dose Assessment is NOT available.

AND

Release is ongoing for **≥ 15 minutes**

THEN

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

EMERGENCY ACTION LEVELS

OPCON	All	All	All
EAL #	**6.1.3.a**	**6.1.3.b**	**6.1.3.c**
	Dose Assessment **IF**	Measured Dose Rate **IF**	Sample Analysis **IF**
	Dose Assessment indicates <u>EITHER</u> one of the following at the MEA or beyond as calculated on the SSCL: • TEDE 4-Day Dose **≥ 1.0E+02 mRem** • Thyroid-CDE Dose **≥ 5.0E+02 mRem**	Dose Rate measured at the Protected Area Boundary or beyond <u>EXCEEDS</u> **100 mRem/hr** **AND** Release is ongoing for **≥ 15 minutes**	Analysis of field survey samples at the Protected Area Boundary indicates <u>EITHER</u> one of the following: • **≥ 5.24E+02 CCPM** • **≥ 4.63E-07 µCi/cc I-131**

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

OPCON	All	All	All
EAL #	**6.1.4.a**	**6.1.4.b**	**6.1.4.c**
EMERGENCY ACTION LEVELS	Dose Assessment **IF** Dose Assessment indicates <u>EITHER</u> one of the following at the MEA or beyond as calculated on the SSCL: • TEDE 4-Day Dose ≥ **1.0E+03 mRem** • Thyroid-CDE Dose ≥ **5.0E+03 mRem**	Measured Dose Rate **IF** Dose Rate measured at the Protected Area Boundary or beyond <u>EXCEEDS</u> **1000 mRem/hr** **AND** Release is ongoing for ≥ **15 minutes**	Sample Analysis **IF** Analysis of field survey samples at the Protected Area Boundary indicates <u>EITHER</u> one of the following: • ≥ **5.24E+03 CCPM** • ≥ **4.63E-06 µCi/cc I-131**

THEN

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

6.0 Radiological Releases/Occurrences

6.2 Liquid Effluent Release

	EAL # 6.2.1	EAL # 6.2.2
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

EMERGENCY ACTION LEVELS

EAL # 6.2.1

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 is ongoing for $\geq$ **60 minutes** after the alarm occurs

THEN

Action Required: Refer to Attachment 1 **UNUSUAL EVENT**

EAL # 6.2.2

AND

Sample analysis of liquid effluent indicates concentration in excess of **200 times Technical Specification limits**

AND

Release is ongoing for $\geq$ **15 minutes** after the alarm occurs

THEN

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

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

Initiating Condition: Unplanned Increase in Plant Radiation

OPCON: All

EAL # **6.3.1.a**

IF

Unplanned increase 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

THEN

Action Required: Refer to Attachment 1 **UNUSUAL EVENT**

Initiating Condition: 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

EAL # **6.3.2.a**

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)

OPCON: All

EAL # **6.3.2.b**

IF

Unplanned Dose Rates **≥ 15 mRem/hr** in EITHER one of the following:

- Main Control Room
- Security Central Alarm Station (CAS)

THEN

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

AND

EMERGENCY ACTION LEVELS

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

Initiating Condition	Unplanned Increase in Plant Radiation		Major Damage to Irradiated Fuel	Events that have or may result in uncovering Irradiated Fuel outside the Reactor Vessel
OPCON	5	All	All	All
EAL #	<u>6.3.1.b</u>	<u>6.3.1.c</u>	<u>6.3.2.c</u>	<u>6.3.2.d</u>
EMERGENCY ACTION LEVELS	IF **Uncontrolled** water level decrease in the Reactor Cavity as indicated by <u>EITHER</u> one of the following: • Visual Observation • Reactor Water Level Shutdown Range Level Indicator 1BBLI-R605	IF **Uncontrolled** water level decrease in the Spent Fuel Pool as indicated by <u>EITHER</u> one of the following: • Visual Observation • **Valid** Fuel Pool Low Level Alarm Condition	IF **Major Damage** to Irradiated Fuel has occurred AND **Valid** High Alarm received from <u>ANY</u> one of the following RMS channels: • Refuel Floor Exhaust Channel A (9RX627) • Refuel Floor Exhaust Channel B (9RX628) • Refuel Floor Exhaust Channel C (9RX629)	IF **Unplanned** increase on <u>ANY</u> one of the following Area Rad Monitors or by general area rad survey indicates **≥ 2000 mRem/hr**: • Spent Fuel Storage Pool Area (9RX707) • New Fuel Criticality Storage Channel A (9RX612) • New Fuel Criticality Storage Channel B (9RX613)
	THEN		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

	7.1.1	7.1.2.a	7.1.2.b
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 Vital 4.16 KV 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 (OPCON 4, 5, and Defueled ONLY)
		AND **> 15 minutes** have elapsed THEN (7.1.2.a and 7.1.2.b)	
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**	Refer to Attachment 2 **ALERT** (7.1.2.a and 7.1.2.b)	

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 Power 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

EAL 7.1.3

IF

ALL 4.16 KV Vital Buses are deenergized

(OPCON 1, 2, 3 ONLY)

THEN

AND

> 15 minutes have elapsed

THEN

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

EAL 7.1.4.a

AND

Restoration of Power to **at least one** 4.16 KV Vitial Bus within **4 hours** is NOT likely

THEN

EAL 7.1.4.b

AND

A Loss of **any 2** Fission Product Barriers with the Potential Loss of the **third** Barrier

THEN

Action Required: 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** (OPCON 4, 5, and Defueled ONLY) AND **> 15 minutes** have elapsed THEN	IF **Unplanned** degraded voltage condition for **ALL Vital 125 VDC Buses**, such that voltage is **< 108 VDC** (OPCON 1, 2, 3 ONLY) 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

8.1.2

IF

Unplanned, Complete Loss of ALL Technical Specification required systems **available to provide Decay Heat Removal functions**

AND

RCS Temperature has increased to **> 200°F** (Excluding a **momentary** increase >200°F with a **heat removal function** restored)

AND

An UNCONTROLLED temperature increase is RAPIDLY approaching **200°F** (with NO **heat removal function** restored)

THEN

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

8.1.3.a

IF

Reactor Water Level REACHES **-161" (Top of Active Fuel)**

8.1.3.b

IF

Loss of Main Condenser capabilities, as evidenced by an inability to remove Decay Heat from the Reactor

AND

Loss of Torus capabilities as evidenced by EITHER one of the following:

- Entry into an Unsafe region of ANY 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

THEN (8.1.3.a or 8.1.3.b)

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

8.0 System Malfunctions

8.2 Loss of Assessment Capabililty

EMERGENCY ACTION LEVELS

Initiating Condition	Unplanned Loss of All Onsite or Offsite Communications Capabilities		Unplanned Loss of Most or All Safety System Annunciation or Indication in the Control Room for Greater Than 15 Minutes	Unplanned Loss of Control Room Annunciators and a Significant Transient is in Progress or Compensatory Indicators are Unavailable		Inability to Monitor a Significant Transient in Progress
OPCON	All	All	1, 2, 3	1, 2, 3	1, 2, 3	1, 2, 3
EAL #	8.2.1.a	8.2.1.b	8.2.1.c	8.2.2.a	8.2.2.b	8.2.3

8.2.1.a

IF

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

- Station Page (Gaitronics)
- Station Radio
- Direct Inward Dial (DID)
- Essex (Centrex) Phone
- Nuclear Emergency Telephone (NETS)

THEN

8.2.1.b

IF

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

- Direct Inward Dial (DID)
- Nuclear Emergency Telephone (NETS)
- Essex (Centrex) Phone
- NAWAS
- EMRAD
- FTS 2000

THEN

8.2.1.c

IF

Unplanned Loss of > 75% of Main Control Room Overhead Annunciators for ≥ **15 minutes**

THEN

Action Required: Refer to Attachment 1 **UNUSUAL EVENT**

8.2.2.a

AND

BOTH of the following:

- CRIDS
- SPDS

are NOT AVAILABLE

8.2.2.b

AND

A **significant transient** is in progress

THEN

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

8.2.3

AND

Main Control Room Indications are NOT AVAILABLE to monitor ANY one of the following:

- RCS Status
- Reactivity Control
- ECCS
- Containment Parameters

AND

BOTH of the following:

- CRIDS
- SPDS

are NOT AVAILABLE

THEN

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

8.0 System Malfunctions

8.3 Loss of Control Room Habitibililty

	8.3.2	8.3.3
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 #

EMERGENCY ACTION LEVELS

8.3.2

IF

Main Control Room Evacuation has been initiated

THEN

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

8.3.3

Main Control Room Evacuation has been initiated

AND

Control of the plant CANNOT be established from outside the Main Control Room within **15 minutes**

THEN

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

8.0 System Malfunctions

8.4 Technical Specifications

Initiating Condition

Inability to Reach Required Operational Condition within Technical Specification Limits

OPCON

1, 2, 3

EAL #

EMERGENCY ACTION LEVELS

8.4.1

IF

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

THEN

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 or 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** Confirmed security threat directed toward the station as evidenced by <u>ANY</u> 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 **THEN**	**IF** Confirmed hostile intrusion or malicious acts as evidenced by <u>ANY</u> 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 or destructive device discovered in a Vital Area, resulting in **SCP-10** implementation **THEN**	**IF** Confirmed hostile intrusion or malicious acts in Plant Vital Areas as evidenced by : • Discovery of an intruder(s), armed and violent, within the Vital Area, resulting in **SCP-6** implementation **THEN**	**IF** Security event resulting in the **actual loss of physical control** of <u>EITHER</u> one of the following: • Main Control Room • Remote Shutdown Panel **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**

9.0 Hazards - Internal/External

9.2 Fire

Initiating Condition

OPCON

EAL #

EMERGENCY ACTION LEVELS

9.2.1

Initiating Condition: Fire within the Protected Area Boundary Not Extinguishedwithin 15 minutes of Detection

OPCON: All

IF

9.2.1 — **Valid Fire** Alarm is received in the Main Controı Room

9.2.1 — Report of a **fire** from personnel at the scene

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 Faciiity

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

THEN

Action Required: Refer to Attachment 1
UNUSUAL EVENT

9.2.2

Initiating Condition: Fire Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown

OPCON: All

IF

Fire within ANY one of the following Plant Vital Structures:

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

AND

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

Damaged Safety System(s) or Plant Vital Structure is required for the present Operational Condition

THEN

Action Required: Refer to Attachment 2
ALERT

9.0 Hazards - Internal/External

9.3 Explosion

EMERGENCY ACTION LEVELS

	9.3.1	9.3.2
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
IF	**Confirmed Explosion** within the Protected Area	**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	Report of visible damage to Plant equipment or Protected Area Permanent Plant Structures	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 Structure which renders the structure incapable of performing its Design Function
AND		Damaged Safety System(s)or Plant 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 Gases

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

EMERGENCY ACTION LEVELS

EAL #	9.4.1.a	9.4.1.b	9.4.1.c
OPCON	All	All	All
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	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	(same as 9.4.1.b)

THEN

Action Required: Refer to Attachment 1 **UNUSUAL EVENT**

9.0 Hazards - Internal/External

9.4 Toxic 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 Estab[illegible] or Maintain Cold Shutdown Conditions

EMERGENCY ACTION LEVELS

OPCON	All	All
EAL #	9.4.2.a	9.4.2.b

9.4.2.a (OPCON: All)

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 Lower Toxicity Limit being EXCEEDED

9.4.2.b (OPCON: All)

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

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
<u>EAL #</u>	<u>9.5.1</u>	<u>9.5.2</u>

EMERGENCY ACTION LEVELS

9.5.1

IF

Seismic Event felt by personnel within the Protected Area

AND

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

THEN

Refer to Attachment 1
UNUSUAL EVENT

9.5.2

(9.5.1 conditions) **AND**

Valid Actuation of the Seismic Switch **(> 0.1g)** has occurred as verified by <u>EITHER</u> 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

Refer to Attachment 2
ALERT

Action Required

9.0 Hazards - Internal/External

9.6 High Winds

Initiating Condition

Natural and Destructive Phenomena Affecting the Protected Area

- **OPCON:** All — **EAL # 9.6.1.a**
  - IF: Report of a Tornado TOUCHING DOWN within the Protected Area
- **OPCON:** All — **EAL # 9.6.1.b**
  - IF: **Sustained** wind speeds **≥ 75 MPH for 15 minutes**, measured at ANY elevation of the Met Tower

THEN

Action Required: Refer to Attachment 1 **UNUSUAL EVENT**

Initiating Condition

Natural and Destructive Phenomena Affecting the Plant Vital Area

- **OPCON:** All — **EAL # 9.6.2**

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 2 **ALERT**

EMERGENCY ACTION LEVELS

9.0 Hazards - Internal/External

9.7 Abnormal River Level

Initiating Condition	Natural and Destructive Phenomena Affecting the Protected Area		Natural and Destructive Phenomena Affecting the Plant Vital Area	
OPCON	All	All	All	All
EAL #	**9.7.1.a**	**9.7.1.b**	**9.7.2.a**	**9.7.2.b**
EMERGENCY ACTION LEVELS	IF River Level ≥ 98.0'	IF River Level ≤ 80.0'	IF River Level ≥ 101.0'	IF River Level ≤ 76.0'
	THEN		THEN	
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**		Refer to Attachment 2 **ALERT**	

9.0 Hazards - Internal/External

9.8 Flooding

	9.8.1	9.8.2
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 #	9.8.1	9.8.2
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

9.9 Turbine Failure / Vehicle - Missile Impact

EMERGENCY ACTION LEVELS

Initiating Condition: Natural and Destructive Phenomena Affecting the Protected Area

OPCON: 1

EAL # 9.9.1.a

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

OPCON: All

EAL # 9.9.1.b

IF

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
- Low Level Radwaste Interim Storage Facility

THEN

Action Required: Refer to Attachment 1 **UNUSUAL EVENT**

Initiating Condition: Natural and Destructive Phenomena Affecting the Plant Vital Area

OPCON: All

EAL # 9.9.2

IF

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 Operatioal Condition

THEN

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

11.0 Reportable Action Levels

11.1 Technical Specifications

	11.1.1.a	11.1.1.b	11.1.1.c
Initiating Condition	INITIATION OF ANY UNIT SHUTDOWN REQUIRED BY THE TECHNICAL SPECIFICATIONS [10CFR50.72(b)(1)(i)(A)	EXCEEDING ANY TECHNICAL SPECIFICATION SAFETY LIMIT [10CFR50.72(b)(1)(i)(A); 10CFR50.36(c)(1)]	ANY DEVIATION FROM T/S OR LICENSE CONDITION PURSUANT TO 10CFR50.54(x) [10CFR50.72(b)(1)(i)(B)]
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** <u>INITIATED</u> to comply with Technical Specifications	IF Exceeding <u>ANY</u> 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	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 procedures) NOTE: Such action must be approved by at least a licensed SRO

THEN

Action Required: 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 #	**11.1.2.a**	**11.1.2.b**	**11.1.2.c**	**11.1.2.d**
REPORTING ACTION LEVELS	**IF** Violation of the requirements contained in Section 2.C (Items 3 through 13) 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 heatup or cooldown rates are exceeded (T/S 3.4.6.1)	**IF** The conductivity, chloride concentration or pH in the RCS is in excess of its specified limits per T/S 3.4.4 Action Statements C.1 thereby requiring an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the RCS	**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 4.7.5g
	THEN	**THEN**		
Action Required	Refer to Attachment 20 24 Hour Report	Refer to Attachment 22 OTHER Reports		

11.0 Reportable Action Levels

11.2 Design Basis / Unanalyzed Condition

	11.2.1	11.2.2.a	11.2.2.b
Initiating Condition	ANY EVENT OR CONDITION DURING OPERATION THAT RESULTS IN THE CONDITION OF THE PLANT BEING SERIOUSLY DEGRADED [10CFR50.72(b)(1)(ii)]	ANY EVENT FOUND WHILE SHUTDOWN THAT WOULD HAVE SERIOUSLY DEGRADED THE PLANT OR RESULTED IN BEING IN AN UNANALYZED CONDITION [10CFR50.72(b)(2)(i)]	EVENT/CONDITION THAT <u>ALONE</u> COULD HAVE PREVENTED CERTAIN SAFETY FUNCTIONS [10CFR50.72 (b)(2) (iii)]
OPCON	1,2	3,4,5	All
<u>RAL #</u>	<u>11.2.1</u>	<u>11.2.2.a</u>	<u>11.2.2.b</u>
REPORTING ACTION LEVELS	IF As judged by the SNSS/EDO, an event or condition found during plant operations that results in <u>ANY</u> one of the following: • The condition of the plant, including its principle safety barriers, being seriously degraded. • The plant being in an unanalyzed condition that significantly compromises plant safety. • The plant being in a condition outside the design basis of the plant. • The plant being in a condition not covered by normal/abnormal or emergency operating procedures. THEN	IF Any event, found while the Reactor is shutdown, that, if had it been found during operation, would have resulted in the plant, including it principle safety barriers being in <u>EITHER</u> one of the following conditions: • Seriously degraded • In an unanalyzed condition that significantly compromises Plant safety. THEN	IF Any event or condition that **alone could have prevented** the fulfillment of the safety function of structures or systems that are needed to perform <u>ANY</u> one of the following: • Control the release of radioactive material • Shutdown the reactor and maintain it in a safe shutdown condition • Remove residual heat • Mitigate the consequences of an accident THEN
Action Required	Refer to Attachment 12 1 Hour Report	Refer to Attachment 14 4 Hour Report	Refer to Attachment 14 4 Hour Report

11.0 Reportable Action Levels

11.2 Design Basis / Unanalyzed Condition

Initiating Condition	PRESENCE OF A LOOSE PART IN THE REACTOR COOLANT SYSTEM [Reg. Guide 1.133]
OPCON	All
RAL #	11.2.2.c
REPORTING ACTION LEVELS	IF Presence of a Loose Part in the RCS is Confirmed THEN
Action Required	Refer to Attachment 14 4 Hour Report

11.0 Reportable Action Levels

11.3 Engineered Safety Features (ESF)

	RAL 11.3.1	RAL 11.3.2
Initiating Condition	Any Event that results or should have resulted in ECCS Discharge into the RCS as the result of a **Valid** signal [10CFR50.72(b)(1)(iv)]	ACTUATION OF ENGINEERED SAFETY FEATURE (INCLUDING THE REACTOR PROTECTION SYSTEM) EXCEPT PREPLANNED [10CFR50.72(b)(2)(ii)]
OPCON	All	All
RAL #	11.3.1	11.3.2
REPORTING ACTION LEVELS	IF **Valid** ECCS Actuation, Manual or Automatic, has or should have occurred AND ECCS Actuation resulted in or should have resulted in, discharged to the vessel THEN	IF Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), except as part of a preplanned sequence during operation or testing, including the Reactor Protection System (RPS) (Manual or Automatic Scram) AND ESF / RPS Actuation is determined to be reportable in accordance with NC.NA-AP-0006(Q) THEN
Action Required	Refer to Attachment 12 1 Hour Report	Refer to Attachment 14 4 Hour Report

11.0 Reportable Action Levels

11.4 Personnel Safety / Overexposure

Initiating Condition	ANY INCIDENT OR EVENT INVOLVING BYPRODUCT, SOURCE, OR SPECIAL NUCLEARMATERIAL CAUSING ANY OF THE LISTED RESULTS[10CFR20.2202(a)(1)]	ANY INCIDENT OR EVENT INVOLVING BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL CAUSING ANY OF THE LISTED RESULTS [10CFR20.2202(b)(1)]	ONSITE FATALITY [10CFR50.72(b)(2)(vi)]
OPCON	All	All	All
RAL #	11.4.1	11.4.2.a	11.4.2.b
REPORTING ACTION LEVELS	IF PERSONNEL OVEREXPOSURE or potential for overexposure as indicated by ANY one of the following: • **TEDE exposure $\geq$ 25 REM** • **LDE exposure $\geq$ 75 REM** • **SDE exposure $\geq$ 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 $\geq$ **5** times the occupational ALI (annual limit of intake) which would usually equate to **25 Rem CEDE**. This applies to areas where personnel are not normally stationed during routine operations. THEN	IF PERSONNEL OVEREXPOSURE or potential for overexposure as indicated by ANY one of the following: • **TEDE exposure $\geq$ 5 REM** • **LDE exposure $\geq$ 15 REM** • **SDE exposure $\geq$ 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 $\geq$ **1** times the occupational ALI (annual limit of intake) which would usually equate to **5 Rem CEDE**. This applies to areas where personnel are not normally stationed during routine operations. THEN	IF Any fatality has occurred onsite (within the owner controlled area) 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

Initiating Condition	CONTAMINATED INJURED PERSON TRANSPORTED FROM THESITE TO AN OFFSITE MEDICAL FACILITY [10CFR50.72(b)(2)(v)]	SIGNIFICANT FITNESS FOR DUTY EVENTS [10CFR26.73(a)]
OPCON	All	All
RAL #	11.4.2.c	11.4.3
REPORTING ACTION LEVELS	IF Transportation of a contaminated or **potentially contaminated individual** from the site to an offsite medical facility THEN	IF Any event that is determined to be reportable by the Medical Review Officer (MRO) or designee i.a.w. PSE&G's Fitness for Duty Program (NC-NA-AP.ZZ-0042(Q)) AND The Reportable details of the event are made available to the SNSS by the Medical Review Officer (MRO) or designee THEN
Action Required	Refer to Attachment 14 4 Hour Report	Refer to Attachment 19 24 Hour Report

11.0 Reportable Action Levels
11.5 Environmental

	11.5.2.a	11.5.2.b	11.5.2.c
Initiating Condition	SPILL/DISCHARGE OF ANY NON-RADIOACTIVE HAZARDOUS SUBSTANCE [10CFR50.72(b)(2)(vi); N.J.A.C. 7:1E]	SPILL/DISCHARGE OF ANY NON-RADIOACTIVE HAZARDOUS SUBSTANCE INTO OR UPON THE RIVER [10CFR50.72(b)(2) (vi); 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 Spill/discharge of an industrial chemical or petroleum product outside of a plant structure within the owner controlled area 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 1 hour and no contact with groundwater is suspected. NOTES: This event may require 15 minute notifications. Do not delay implementation of Attachment 16. Contact licensing per ECG Attachment 9 for guidance concerning reportability as necessary THEN	IF 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 oil slick on the Delaware River which may have originated from Salem or Hope Creek Station. NOTES: This event will require 15 minute notifications. Do not delay implementation of Attachment 16. Contact licensing per ECG Attachment 9 for guidance concerning reportability as necessary THEN	IF As judged by the SNSS/EDO, ANY one of the following events has occurred: • Unusually large fish kill; • Protected aquatic species impinge on Circulating or Service Water intake screens (ex.: 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 THEN
Action Required	Refer to Attachment 16 Spill/Discharge Reporting	Refer to Attachment 16 Spill/Discharge Reporting	Refer to Attachment 15 Spill/Discharge Reporting

11.0 Reportable Action Levels
11.5 Environmental

Initiating Condition: OVERFLOW ALARM FAILURE ON ABOVE GROUND STORAGE TANKS [10CFR50.72(b)(2) (vi), N.J.AC.7:1E]

OPCON: All

RAL # 11.5.2.d

REPORTING ACTION LEVELS

IF

Complete failure of ANY one of the below listed storage tank level alarms:

TANK IDENTIFICATION	ALARM ID#
Circ Wtr Caustic Storage Tank (0A-T-500)	0DELAH-3552A
Circ Wtr Sodium Hypochlorite Storage Tk (0B-T-501)	0DDLAHL-3550B
Circ Wtr Sodium Hypochlorite Storage Tk (0C-T-501)	0DDLAHL-3550C
S.W. Sodium Hypochlorite Storage Tk (0E-T-501)	0EQLAHL-7800B
S.W. Sodium Hypochlorite Storage Tk (0F-T-501)	0EQLAHL-7800C
Million Gallon Fuel Oil Storage Tk (00-T-516)	0JALAH-3206A
Aux Boiler Fuel Oil Day Tk (00-T-527)	0JALAHH-3215

NOTES:
This event may require 15 minute notifications. Do not delay implementation of Attachment 16.

Contact licensing per ECG Attachment 9 for guidance concerning reportability as necessary

THEN

Action Required

Refer to Attachment 16 Spill/Discharge Reporting

11.0 Reportable Action Levels

11.6 After-the-Fact

Initiating Condition

EMERGENCY CONDITIONS DISCOVERED AFTER-THE-FACT

OPCON

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

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

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)(1)(v)]

OPCON: All | All

RAL #

REPORTING ACTION LEVELS

11.7.1.a

IF

Any Non-Emergency safeguards event that is reportable in accordance with 10CFR73.71 as determined by Security (SCP-15)

THEN

Action Required

Refer to Attachment 11
1 Hour Report

11.7.1.b

IF

SNSS/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:

- Emergency Phone System (NETS) for **> 1 hour**
- ENS for **> 1 Hour** in the Control Room, TSC, or EOF (N/A if reported by the NRC).
- Greater than or equal to **8** Offsite sirens for **> 1 Hour**
- Use of the TSC or EOF for **> 8 Hours**
- All Meteorological data (Hope Creek AND Salem) for **> 8 Hours**
- Site access due to Acts of Nature (snow, flood, etc.)
- SPV & NPV & FRVS plant vent radiation effluent monitors for **> 8 Hours**
- SPDS OR CRIDS for **> 8 Hours**
- All or most (> 75%) OHA's for **< 15 minutes**
- Concurrent loss of multiple accident or emergency condition indicators which in the judgement of the SNSS significantly impairs assessment capabilities

THEN

Action Required

Refer to Attachment 12
1 Hour Report

11.0 Reportable Action Levels

11.8 Public Interest

	11.8.2.a	11.8.2.b
Initiating Condition	UNUSUAL CONDITIONS WARRANTING A NEWS RELEASE OR NOTIFICATION OF GOVERNMENT AGENCIES [10CFR50.72(b)(2)(vi)]	UNUSUAL CONDITIONS DIRECTLY AFFECTING LOWER ALLOWAYS CREEK TOWNSHIP (LACT) [LAC -MOU]
OPCON	All	All
RAL #	11.8.2.a	11.8.2.b
REPORTING ACTION LEVELS	IF SNSS/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	IF As judged by the SNSS/EDO, events which are the responsibility of PSE&G 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
Action Required	Refer to Attachment 14 4 Hour Report	Refer to Attachment 21 LACT / MOU Report

11.0 Reportable Action Levels

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

Initiating Condition	UNPLANNED / ACCIDENTAL CRITICALITY [10CFR70.52(a)]	LOSS AND INVESTIGATION OF THE LOSS OF SPECIAL NUCLEAR MATERIALS/ SPENT FUEL [10CFR73.27(c), 10CFR73.71(a)]	THEFT OR LOSS OF LICENSED MATERIAL [10CFR20.2201(a)(1)(i)]
OPCON	2, 3, 4, 5	All	All
RAL #	**11.9.1.a**	**11.9.1.b**	**11.9.1.c**
REPORTING ACTION LEVELS	**IF** Any unplanned or accidental criticality **THEN**	**IF** ANY one of the following events occur involving SNM or Spent Fuel: • Shipment of Special Nuclear Material (SNM) or Spent Fuel that is lost or Unaccounted for after the estimated time of arrival • A lost or unaccounted for shipment of SNM or Spent Fuel has been recovered or accounted for • Results of a trace investigation of lost or unaccounted for SNM shipment are received **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	Refer to Attachment 11 1 Hour Report

11.0 Report[illegible]e Action Levels

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

Initiating Condition	RECEIPT OF SNM MATERIAL [10CFR73.27(b)]	EXCESSIVE CONTAMINATION AND/OR RADIATION LEVELS ON A PACKAGE [10CFR20.1906(d)]	ACCIDENT DURING TRANSPORT OF LICENSED MATERIAL [10CFR71.5(a)(1)(v)]	INADVERTANT RELEASE OF RADIOACTIVE CONTAININATED MATERIAL [10CFR50.72(b)(2)(vi)]
OPCON	All	All	All	All
RAL #	**11.9.1.d**	**11.9.1.e**	**11.9.2.a**	**11.9.2.b**
REPORTING ACTION LEVELS	IF Receipt of shipment of **strategic Special Nuclear Material** (SNM) THEN	IF Receipt survey indicates that package contamination/radiation levels equal or exceeds ANY one of the following: • **2200 dpm/100cm2** • **200 mR/hr on contact** • **10 mR/hr at 3 feet** THEN	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	IF As judged by the SNSS/EDO, EITHER of the following events has occurred: • Unusual or abnormal release of radiological effluents • Release of radiologically contaminated tools or equipment to public areas AND A news release is planned AND Notification to Local, State or Federal Agencies has been or will be made THEN
Action Required	Refer to Attachment 10 1 Hour Report	Refer to Attachment 10 1 Hour Report	Refer to Attachment 18 4 Hour Report	Refer to Attachment 14 4 Hour Report

NUCLEAR
BUSINESS
UNIT

SALEM HOPE CREEK

EVENT CLASSIFICATION GUIDE

TECHNICAL BASIS

HOPE CREEK GENERATING STATION

PSE&G
The Energy People

BASIS DOCUMENT

HOPE CREEK ECG

SECTIONS 1 - 11

EMERGENCY ACTION LEVELS (EALs)
&
REPORTING ACTION LEVELS (RALs)

NOTE: THIS IS A NEW DOCUMENT WHICH WILL BE SEPARATE FROM THE ECG AND USED AS A REFERENCE DOCUMENT.

BASIS DOCUMENT

FILE: COVERS.ECG

1.0 Fuel Clad Challenge

1.1 RCS Activity

UNUSUAL EVENT - 1.1.1.a

IC Fuel Clad Degradation

EAL

Reactor Coolant Sample Activity > **4 µCi/gm Dose Equivalent I-131**

OPERATIONAL CONDITION - 1,2,3,4,5

BASIS

A Reactor Coolant sample analysis with specific activity in excess of the Technical Specification limit of 4 µCi/gm Dose Equivalent Iodine-131 (DEI-131) is indicative of a degradation of the fuel clad, and is a precursor of more serious problems. This activity level is chosen instead of the 0.2 µCi/gm DEI-131 Technical Specification limit, under which operation is allowed to continue for up to 48 hours to accommodate short duration Iodine spikes following changes in thermal power. This EAL threshold <u>does not</u> use the term "Valid", since Reactor Coolant Sample Activity of greater than 4 uCi/gm DEI- 131 can only occur as the result of fuel clad degradation and not as the result of a resin / chemical intrusion transient or HWCI System malfunction. Unusual Event declaration is warranted <u>only</u> when actual fuel clad degradation has occurred.

Barrier Analysis

This event does not reach the threshold for the loss of the Fuel Clad Barrier, but does affect that barrier.

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when a sample analysis of Reactor Coolant activity exceeds 300 µCi/gm DEI-131 per EAL Section 3.1.3.

DISCUSSION

The Technical Specification limit on Reactor Coolant activity ensures that the 2 hour thyroid and whole body doses resulting from a Main Steam Line failure outside the containment during steady state operation will not exceed a small fraction of the 10CFR100 limits. This limit accommodates Iodine Spiking, which frequently occurs following shutdowns, startups rapid power changes and coolant depressurization. Iodine spikes are characterized by a rapid increase in Reactor Coolant Iodine concentration by as much as three orders of magnitude followed by a return to prespike concentrations. This spiking is a temporary excursion and is not caused by a sudden fuel failure. The Technical Specification limit of > 100/Ē is excluded from this EAL because this limit does not include Iodine Activity.

DEVIATION

NUMARC EAL SU 4.2 suggests that the Operating Mode Applicability for this EAL is ALL. When the Reactor is defueled, the source term needed to achieve an RCS Activity of 4 uCi/gm Dose Equivalent I-131 is not available. Hence, this EAL is applicable in Operational Conditions 1,2,3,4 and 5.

REFERENCES

NUMARC NESP-007, SU4.2
Technical Specification LCO 3.4.5
HC.OP-AB.ZZ-0100(Q), High Reactor Coolant Activity
HC.OP-AB.ZZ-0203(Q) Main Steam Line High Radiation
10 CFR100

1.0 Fuel Clad Challenge

1.1 RCS Activity

UNUSUAL EVENT - 1.1.1.b

IC Fuel Clad Degradation

EAL

Valid Offgas Pretreatment Radiation Monitor (9RX621 / 9RX622) High Alarm Condition (**≥ 2.2E+04 mRem/hr**)

OPERATIONAL CONDITION - 1,2,3,4

BASIS

A **Valid** Offgas Pretreatment Radiation Monitor High alarm is indicative of a degradation of the fuel clad, and is a precursor of a more serious problems. The alarm is set at 2.2E+04 mR/hr, which ensures that the alarm will actuate prior to exceeding the Technical Specification Offgas System Noble Gas Effluent Limit of 3.3E5 µCi/s. **Valid** is defined as the Offgas Pretreatment Radiation Monitor High Alarm actuating specifically due to fuel clad degradation, thus precluding unwarranted Unusual Event declaration as the result of a resin / chemical intrusion transient, or HWCI System malfunction. Unusual Event declaration is warranted only when actual fuel clad damage has occurred.

Barrier Analysis

This event does not reach the threshold for the loss of the Fuel Clad Barrier, but does affect that barrier.

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when a sample analysis of Reactor Coolant activity exceeds 300 µCi/gm DEI-131 per EAL Section 3.1.3.

DISCUSSION

The Offgas Pretreatment Radiation Monitors (9RX621 / 9RX622) monitor gamma radiation levels attributable to the non-condensible fission product gases produced in the reactor and transported with steam through the turbine to the condenser. This instrument takes a sample from the sample tap between the fourth and fifth holdup pipe of the Offgas system.

Restricting the gross radioactivity from the Main Condenser provides reasonable assurance that the Total Body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment.

Operating Experience at HCGS has demonstrated that Reactor coolant activity changes for reasons other than fuel clad degradation can result in increasing Main Steam Line Radiation Monitors and Offgas Pretreatment Radiation Monitor. Such events (e.g. - resin intrusion) do not require classification under this EAL.

DEVIATION

NUMARC EAL SU 4.1 suggests that the Operating Mode Applicability for this EAL is ALL. In Operational Condition 5 and Defueled, the MSIVS will be closed, thus rendering the Offgas Pretreatment Radiation Monitors unavailable for detection of increased RCS Activity. Hence, this EAL is applicable in Operational Conditions 1,2,3 and 4.

REFERENCES

NUMARC NESP-007, SU4.1
Technical Specifications; Table 3.3.7.1 (5); LCO 3.11.2.7
HC.OP-AB.ZZ-0100(Q), High Reactor Coolant Activity
HC.OP-AB.ZZ-0203(Q) Main Steam Line High Radiation
HC.RP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response
OE-6144, Resin Intrusion
10 CFR100

1.0 Fuel Clad Challenge

1.1 RCS Activity

UNUSUAL EVENT - 1.1.1.c

IC Fuel Clad Degradation

EAL

Valid Main Steam Line Radiation Monitor High High Alarm Condition **($\geq$ 3 times Normal Full Power Background)**

OPERATIONAL CONDITION - 1,2,3,4

BASIS

A **Valid** Main Steam Line Radiation Monitor High High alarm ($\geq$ 3 times normal full power background) is indicative of degradation of the fuel clad and may be a precursor of more serious problems. **Valid** is defined as the Main Steam Line Radiation Monitor High High Alarm actuating specifically due to fuel clad degragation, thus precluding unwarranted Unusual Event declaration as the result of a resin / chemical intrusion transient, or HWCI System malfunction. Unusual Event declaration is warranted only when actual fuel clad degradation has occurred. Reaching the High High Alarm on ANY of the 4 Main Steam Line Radiation Monitor channels, as determined by receipt of ANY one of the following, due to fuel clad degradation, warrants Unusual Event declaration.

- Overhead Annunciator C6-B2, MN STM LINE RAD HI HI OR INOP
- CRIDS Point D2121, MN STM LINE HI HI RAD / INOP - W
- CRIDS Point D2122, MN STM LINE HI HI RAD / INOP - X
- CRIDS Point D2123, MN STM LINE HI HI RAD / INOP - Y
- CRIDS Point D2124, MN STM LINE HI HI RAD / INOP - Z

Barrier Analysis

This event does not reach the threshold for the loss of the Fuel Clad Barrier, but does affect that barrier.

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when a sample analysis of Reactor Coolant activity exceeds 300 uCi/gm DEI-131 per EAL Section 3.1.3.

DISCUSSION

The Main Steam Line Radiation Monitor Channels (9RX509, 9RX510, 9RX511, 9RX512) monitor gamma radiation levels at the Main Steam Lines. A High High alarm is indicative of a release of gap activity to the coolant but is not indication of a major failure of the fuel clad. A **Valid** Main Steam Line Radiation High High alarm condition requires a manual Reactor Scram and Main Steam Isolation Valve closure to reduce and isolate the potential source of the radioactivity release.

The terminology used for the 3 times Normal Full Power Background threshold differs between the Main Control Room Overhead Annunciators and the Radiation Monitoring System (RM-11). As a result, specific monitor channels are not included in the EAL. Overhead Annunciators use the terminology of "High High" for this threshold, where the RM-11 uses the terminology of "High" for the same threshold. For the purpose of this EAL, the High High setpoint terminology used by the Overhead Annunciators is used, though the same indications are available on the following RM-11 Channels:

- Main Steam Line "Channel" A (Grid 1/4; 9RX509)
- Main Steam Line "Channel" B (Grid 1/4; 9RX510)
- Main Steam Line "Channel" C (Grid 1/4; 9RX511)
- Main Steam Line "Channel" D (Grid 1/4; 9RX512)

In addition, the Main Steam Line Radiation Monitor Numac Drawers can be used to trend changes in Main Steam Line Radiation Levels.

A rapid power reduction from full power may cause the Main Steam line Radiation Monitors to momentarily increase to 1.5 times normal full background readings. This is due to the response time of the HWCI Hydrogen Flow Controller and the transport time from the Hydrogen Injection point (Secondary Condensate Pumps).

Operating Experience at HCGS has demonstrated that Reactor coolant activity changes for reasons other than fuel clad degradation can result in increases in Main Steam Line Radiation Monitors and Offgas Pretreatment Radiation Monitors. Such events (e.g. - resin intrusion) do not require classification under this EAL.

DEVIATION

NUMARC EAL SU 4.1 suggests that the Operating Mode Applicability for this EAL is ALL. In Operational Condition 5 and Defueled, the MSIVS will be closed, thus rendering the Main Steam Line Rad Monitors unavailable for detection of increased RCS Activity. Hence, this EAL is applicable in Operational Conditions 1,2,3 and 4.

REFERENCES

NUMARC NESP-007, SU4.1
HC.RP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response
HC.OP-AR.ZZ-0011(Q), Annunciator Response Procedures, Window C6-B2
HC.OP-AB.ZZ-0203(Q), Main Steam Line High Radiation
HCGS Technical Specifications 3/4.3, Instrumentation
Technical Specifications, LCO 3.11.2.7
OE-6144, Resin Intrusion
10CFR100

1.0 Fuel Clad Challenge

1.1 RCS Activity

ALERT - 1.1.2

IC Fuel Clad Degradation

EAL

ANY one of the following:

- Reactor Coolant Sample Activity > **4 µCi/gm Dose Equivalent I-131**
- **Valid** Offgas Pretreatment Radiation Monitor (9RX621 / 9RX622) High Alarm Condition **(≥ 2.2E+04 mRem/hr)**
- **Valid** Main Steam Line Radiation Monitor High High Alarm Condition **(≥ 3 times Normal Full Power Background)**

AND

ANY SRV is determined to be **Stuck Open**

OPERATIONAL CONDITION - 1,2,3

BASIS

Indication of Fuel Clad Degradation coincident with ANY SRV determined to be Stuck Open is indicative of a Loss of the RCS Barrier, as fission products are being transported directly to the Suppression Pool, thus compromising the integrity of the RCS Barrier. Hence, Alert declaration is warranted. In the event an SRV is Stuck Open with NO indications of Fuel Clad degradation, an emergency declaration is NOT warranted, since an open SRV is within the analyzed design envelope of the plant and does not, by itself, represent a degradation in the level of plant safety. An SRV is considered to be **Stuck Open** when the SRV can not be reclosed by operator action within 2 minutes of ANY spurious, automatic or manual actuation. A Stuck Open SRV SHOULD NOT be considered as an Unisolable RCS Leak > 50 GPM, as the consequences of a Stuck Open SRV discharging to the Suppression Pool are different than an Unisolable RCS Leak exceeding 50 GPM that is discharging into the Drywell Air Space.

A Stuck Open SRV by itself requires a 1 Hour Report if a Unit Shutdown (Manual Reactor Scram) is initiated to comply with Technical Specification or a 4 Hour Report if the SRV is reclosed within the Technical Specification limits, due to the ESF actuation.

Barrier Analysis

RCS Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based upon the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0

DISCUSSION

A Stuck Open SRV discharging Reactor Coolant to the Suppression Pool does not represent the same challenge to the RCS and Primary Containment as an Unisolable RCS Leak discharging into the Drywell. The consequences of a Stuck Open SRV do not represent a significant precursor to further plant degadation, as plant design (Pressure Suppression ability of the Torus) and the Abnormal Operating Procedure for a Stuck Open SRV (directing a Manual Reactor Scram within 2 minutes if the SRV can not be closed), minimize the consequences of the event. In contrast, an Unisolable RCS Leak represents a situation where there is concern for "break propagation", which could lead to a significantly larger uncontrolled for loss of RCS inventory. Hence, a Stuck Open SRV must be coincident with Fuel Clad Degradation for the RCS Barrier to be considered lost.

DEVIATION

None

REFERENCES

NUMARC Questions and Answers, June 1993, "Fission Product Barrier Question #7"
HC.OP-AB.ZZ-0121(Q), Failed Open Safety/Relief Valve

2.0 RCS Challenge

2.1 RCS Leakage

UNUSUAL EVENT - 2.1.1.a / 2.1.1.b

IC RCS Leakage

EAL

<u>EITHER</u> one of the following:

- Pressure Boundary Leakage > **10 gpm**
  (Using 10 minute average)
- Reactor Coolant System Unidentified Leakage > **10 gpm**
  (Using 10 minute average)

OPERATIONAL CONDITION - 1, 2, 3

BASIS

RCS Pressure Boundary and Unidentified Leakage exceeding 10 GPM is indicative of possible degradation of the RCS and may be a precursor of a more serious condition. RCS Operational Leakage addressed by these 2 EALs is <u>specifically</u> RCS leakage into the Drywell. Leakage into the Drywell that is confirmed to <u>not</u> be RCS Leakage, i.e. a leaking Drywell Cooling Coil, does not warrant classification under this EAL. These types of RCS Operational Leakage, exceeding their respective EAL thresholds, should be classified as an Unusual Event, regardless of whether or not the leak has been isolated, since the EAL thresholds exceed the Technical Specification limit. Classification should be based on the 10 minute average and not an instantaneous value, to assure accurate event classification.

The value of 10 gpm for RCS Pressure Boundary and Unidentified Leakage was set higher than the Technical Specification limit of 0 and 5 GPM respectively, to allow time to implement corrective actions (including plant shutdown) prior to exceeding the threshold.

Barrier Analysis

This event does not reach the threshold for the loss of the RCS Barrier, but does affect that barrier.

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when either Unisolable RCS Leak Rate exceeds 50 GPM or Drywell Pressure exceeds 1.68 PSIG per EAL Section 3.2.2

DISCUSSION

Allowable leakage rates from the Reactor Coolant System are based on predicted and experimentally observed behavior of cracks in pipes. Utilizing the leak before break methodology, it is anticipated that there will be indication(s) of minor reactor coolant system boundary leakage prior to a fault escalating to a major leak or a system rupture. Detection of low levels of leakage while pressurized allows for implementation of mitigative actions and permits monitoring for catastrophic failure or rupture precursors.

The limit for Unidentified and Pressure Boundary Leakage is set to a lower value, than Identified Leakage due to concern over "break propagation" resulting from an Unidentified or Pressure Boundary Leak (Small Break), that could potentially lead to a significantly larger loss of inventory. Identified leakage occurs when there is degradation or failure of a mechanical joint. Pipe "break propagation" is thus not an issue.

Instrumentation available via the Radiation Monitoring System (RM-11) to determine RCS Leakage into the Drywell includes:

- Drywell Equipment Drain Sump (DLD EQPT) Monitor (9AX313)
- Drywell Floor Drain Sump (DLD FLR) Monitor (9AX314)
- Lower Drywell Air Condensate Coolers (DLD CCM LOW) Monitor (9AX317)
- Upper Drywell Air Condensate Coolers (DLD CCM UP) Monitor (9AX318)
- Drywell Sumps (DLD SMS) Monitor (9AX319)
- Drywell Air Condensate Coolers Summation (DLD CCM SUM) Monitor (9AX320)

Redundant Instrumentation for Drywell Leak Detection is available on panel 10-C-604 located in the back of the Main Control Room.

Technical Specification required actions based on this leak rate may require a plant shutdown and subsequent depressurization, unless the source of the leak can be located, identified, and/or stopped.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SU5
NUMARC Questions and Answers, June 1993, "General" Question #12
NUMARC Questions and Answers, June 1993, "Fission Product Barrier Question #11"
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0116 (Q), Containment Isolation and Recovery From An Isolation
HC.OP-AB.ZZ-0201 (Q), Drywell High Pressure/Loss of Drywell Cooling
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-GP.ZZ-0005 (Q), Drywell Leakage Source Detection
HCGS Technical Specifications, LCO 3.4.3.2

2.0 RCS Challenge

2.1 RCS Leakage

UNUSUAL EVENT - 2.1.1.c

IC RCS Leakage

EAL

Reactor Coolant System Identified Leakage > **25 gpm** averaged over any 24 hour period

OPERATIONAL CONDITION - 1, 2, 3

BASIS

RCS Identified Leakage exceeding 25 GPM is indicative of possible degradation of the RCS and may be a precursor of a more serious condition. RCS Operational Leakage addressed by this EAL is specifically RCS leakage into the Drywell. Leakage into the Drywell that is confirmed to not be RCS Leakage, i.e. a leaking Dry .ell Cooling Coil, does not warrant classification under this EAL. Identified Leakage should ONLY be classified as an Unusual Event, when the leak rate exceeds 25 GPM when averaged over any 24 Hour period, regardless of whether or not the leak has been isolated. The 24 Hour average is included as part of the EAL threshold to provide consistency with the Technical Specification limt for Identified Leakage.

Barrier Analysis

This event does not reach the threshold for the loss of the RCS Barrier, but does affect that barrier.

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when either Unisolable RCS Leak Rate exceeds 50 GPM or Drywell Pressure exceeds 1.68 PSIG per EAL Section 3.2.2

DISCUSSION

Allowable leakage rates from the Reactor Coolant System are based on predicted and experimentally observed behavior of cracks in pipes. Utilizing the leak before break methodology, it is anticipated that there will be indication(s) of minor reactor coolant system boundary leakage prior to a fault escalating to a major leak or a system rupture. Detection of low levels of leakage while pressurized allows for implementation of mitigative actions and permits monitoring for catastrophic failure or rupture precursors.

The limit for Unidentified and Pressure Boundary Leakage is set to a lower value, then Identified Leakage due to concern over "break propagation" resulting from an Unidentified or Pressure Boundary Leak (Small Break), that could potentially lead to a significantly larger loss of inventory. Identified leakage occurs when there is degradation or failure of a mechanical joint. Pipe "break propagation" is thus not an issue.

Instrumentation available via the Radiation Monitoring System (RM-11) to determine RCS Leakage into the Drywell includes:

- Drywell Equipment Drain Sump (DLD EQPT) Monitor (9AX313)
- Drywell Floor Drain Sump (DLD FLR) Monitor (9AX314)
- Lower Drywell Air Condensate Coolers (DLD CCM LOW) Monitor (9AX317)
- Upper Drywell Air Condensate Coolers (DLD CCM UP) Monitor (9AX318)
- Drywell Sumps (DLD SMS) Monitor (9AX319)
- Drywell Air Condensate Coolers Summation (DLD CCM SUM) Monitor (9AX320)

Redundant Instrumentation for Drywell Leak Detection is available on panel 10-C-604 located in the back of the Main Control Room.

Technical Specification required actions based on this leak rate may require a plant shutdown and subsequent depressurization, unless the source of the leak can be located, identified, and/or stopped.

DEVIATION

NUMARC EAL SU5 suggests that exceeding an RCS Identified Leakage limit of 25 gpm warrants the declaration of an Unusual Event because it may be a precursor to a more serious condition. The Hope Creek Technical Specifiction limit for RCS Identified Leakage is 25 GPM averaged over any 24 hour period. The plant is within the Safety Envelope of the Technical Specification as long as this limit is not exceeded and hence an Unusual Event is not warranted until the limit is exceeded. This philosophy is consistent with that contained in NUMARC EAL SU2, which only requires declaration of an Unusual Event when the plant is outside the Technical Specification Safety Envelope. RCS Pressure Boundary and Unidentified Leakage that exceed the NUMARC EAL threshold will be classified as an Unusual Event, as this leakage exceeds the Technical Specification limit.

In addition, NUMARC EAL SU5 appears to apply specifically to those plants that do not allow for averaging of RCS Identified Leakage over a 24 hour period. Furthermore, NUMARC Questions and Answers Document , June 1993, "General" Question #12 addresses those cases where the Technical Specification LCO has been exceeded and the required Action section has been entered (i.e. 4 Hours to identify and reduce the leakage below the limit). The EAL threshold for RCS Identified Leakage does not consider this time for Unusual Event declaration. The Q&A also states that the EAL for RCS Identified Leakage has been significantly raised from 10 to 25 gpm at some plants. Since the Hope Creek Technical Specification limit is already set at 25 gpm averaged over any 24 hour period, the EAL should not be more limiting than the Technical Specifications.

REFERENCES

NUMARC NESP-0007, SU5
NUMARC Questions and Answers, June 1993, "General" Question #12
NUMARC Questions and Answers, June 1993, "Fission Product Barrier Question #11"
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0116 (Q), Containment Isolation and Recovery From An Isolation
HC.OP-AB.ZZ-0201 (Q), Drywell High Pressure/Loss of Drywell Cooling
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-GP.ZZ-0005 (Q), Drywell Leakage Source Detection
HCGS Technical Specifications, LCO 3.4.3.2

2.0 RCS Challenge

2.1 RCS Leakage

UNUSUAL EVENT - 2.1.1.d

IC RCS Leakage

EAL

Successful Isolation of a Reactor Recirc Pump Dual Seal Failure within **10 minutes** of **recognition**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Successful Isolation of a Reactor Recirc Pump Dual Seal Failure within 10 minutes of recognition is classified as an Unusual Event, due to the significance of the event. Even though the ramifications from a successfully isolated Dual Recirc Pump seal failure are minor, with no possibility for "break propagation", an Unusual Event is warranted due to the multiple failures of mechanical joints that allowed the discharge of a significant quantity of Reactor Coolant (>50 GPM) directly into the Drywell Air Space.

Successful is defined as indication of ALL of the following within **10 minutes** of **recognition** of the Recirc Pump Dual Seal failure.

- Recirc Pump Suction and Discharge Valves have closed
- RWCU Suction Valve from the Recirc Loop has closed
- Recirc Pump Seal Purge Water Valve has closed
- Drywell Pressure and Temperature has begun to decrease
- RCS Leakage has begun to decrease

10 minutes was determined to be a reasonable amount of time to isolate the pump and monitor for the effectiveness of the actions.

Barrier Analysis

This event does not reach the threshold for the loss of the RCS Barrier, but does affect that barrier.

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when either Unisolable RCS Leak Rate exceeds 50 GPM or Drywell Pressure exceeds 1.68 PSIG per EAL Section 3.2.2

DISCUSSION

Prompt recognition of a Dual Recirc Pump seal failure by the operating crew will allow for implementation of actions to isolate the leakage source in accordance with Abnormal Operating Procedures. The design of the Recirc Pump seal limits the magnitude of the identified leakage for this event to 60 GPM due to the presence of a breakdown bushing. As a result, RCS inventory will not be siginifcantly effected. The ability to monitor the leak rate is limited to 50 GPM, the upper limit of the Drywell Leak Detection Instrumentation. Drywell Pressure is not expected to reach the High Drywell Pressure Scram setpoint for this event, provided that the isolation was successfully completed within 10 minutes.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SU5
NUMARC Questions and Answers, June 1993, "General" Question #12
NUMARC Questions and Answers, June 1993, "Fission Product Barrier Question #11"
HC.OP-AB.ZZ-0112 (Q), Recirculation Pump Trip
HC.OP-AB.ZZ-0201 (Q), Drywell High Pressure/Loss of Drywell Cooling
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-GP.ZZ-0005 (Q), Drywell Leakage Source Detection
HCGS Technical Specifications, LCO 3.4.3.2

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.1.a

IC Potential Loss of Fuel Clad

EAL

Reactor Water Level <u>REACHES</u> **- 161" (Top of Active Fuel)**, <u>EXCLUDING</u> intentional lowering of Reactor Water Level during an ATWS

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Reactor Water Level reaching -161" (Top of Active Fuel - TAF), excluding intentional lowering of Reactor Water Level during an ATWS, results in an inability to maintain adequate core cooling by core submergence, causing a Potential Loss of the Fuel Clad Barrier. Without core submergence, the integrity of the fuel clad barrier is in jeopardy. Appropriate classification under this EAL is based on <u>reaching</u> Reactor Water Level of -161" (instead of being able to restore and maintain above -161") due to the potentially severe consequences of a loss of core submergence. Reactor Water Level reaching this threshold results from either a LOCA exceeding available makeup capacity or a Total Loss of High Pressure injection capability.

In addition, during an Anticipated Transient Without Scram (ATWS), it is possible that operator actions will be taken to intentionally lower Reactor Water Level to between -161" and -190", for Reactor Power Control purposes. For this event, classification must be made in accordance with EAL Section 5.0

Barrier Analysis

Fuel Clad Barrier has been potentially lost

ESCALATION CRITERIA

Emergency Classification will escalate based upon the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core Submergence is the preferred method of maintaining adequate core cooling. When Reactor Water Level decreases to below TAF, the ability to effectively remove decay heat is being challenged, and as such the Fuel Clad fission product barrier can no longer be considered intact. While the Emergency Operating Procedures provide contingencies to establish adequate core cooling when Reactor Water Level drops below TAF (Steam Cooling with or without injection), these actions are designed to be an alternative method of providing adequate core cooling while actions are taken to reestablish core submergence. Sustained partial or total core uncovery can result in fuel clad damage and a significant release of fission products to the Reactor coolant. Sustained core uncovery can also result in a breach of the Reactor Vessel due to core melt material interaction with the RPV.

A Loss of Core Submergence will occur when the rate of inventory loss is greater than the rate of inventory makeup from High Pressure injection sources. This condition can occur as the result of the following events/sequences (excluding intentional lowering of Reactor Water Level during an ATWS).

- A LOCA will cause Reactor Water Level to reach the Top of Active Fuel when the LOCA is the result of a large break (momentary core uncovery is expected to occur under this condition) or when the LOCA is due to a small or intermediate break in combination with an inability of High Pressure injection sources to keep up with the leakrate.

- A Loss of High Pressure injection sources without the presence of a LOCA will also result in Reactor Water Level decreasing to TAF, due to continued Reactor Steam Flow without makeup.

Either of these events/sequences results in a challenge to the Fuel Clad Barrier when Reactor Water Level reaches TAF due to core uncovery, hence classification at this threshold is appropriate. However, for both these sequences, Low Pressure ECCS are designed to inject to the Reactor as Reactor Pressure decreases below the shutoff head of the pumps. Reactor Depressurization will occur either due to the LOCA or Manual initiation of Emergency Depressurization when Reactor Water Level reaches -161", provided injection systems are available. This will allow for restoration of Reactor Water Level and reestablishment of Core Submergence. Failure of these systems to restore and maintain Reactor Water Level above -200" will require escalation.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, FC2
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0201 (Q)-FC, Alternate Level Control
HC.OP-EO.ZZ-0207 (Q)-FC, Level/Power Control
BWR Owner's Group Emergency Procedure Guidelines, Rev. 4

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.1.b

IC Loss of Fuel Clad

EAL

Reactor Water Level <u>CANNOT BE RESTORED AND MAINTAINED</u> above **-200"** (Minimum Zero Injection RPV Water Level)

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Inability to <u>restore and maintain</u> Reactor Water Level above - 200" (Minimum Zero Injection RPV Water Level), results in a loss of adequate core cooling by all mechanisms, causing a Loss of the Fuel Clad Barrier. Without adequate core cooling, the integrity of the fuel clad barrier can no longer be assured. Appropriate classification under this EAL is based on the failure of injection systems to restore and maintain Reactor Water Level above > -200", following a condition that causes level to decrease below the threshold. For example, a large break LOCA is expected to cause Reactor Water Level to momentarily decrease below -200", due to the response time of Low Pressure ECCS. As these systems initiate and commence injection to the Reactor, water level will begin to increase and should be able to be maintained above -200". In this case, classification under this EAL is not appropriate as plant systems have performed their intended design function and will eventually restore adequate core cooling by core submergence. However, in the event that Low Pressure ECCS and alternate injection system, as defined in the EOPs are in a degraded condition (i.e., Station Blackout, ECCS Suction Strainer plugging, etc.) and Reactor Water Level can not be restored and maintained above -200", then classification under this EAL should occur due to the potential for release of energy to the containment from imminent fuel failure.

Barrier Analysis

Fuel Clad Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based upon the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core submergence is the preferred method for maintaining adequate core cooling. The failure to reestablish Reactor Water Level above -161", the Top of Active Fuel (TAF), for an extended period of time could lead to a significant of fuel damage. With Reactor Water Level below TAF, but above the Minimum Zero Injection RPV Water Level (-200"), adequate core cooling occurs due to the cooling effects of steam generated in the covered portion of the core flowing through the uncovered portion (Steam Cooling). The Minimum Zero Injection RPV Water Level is defined in the Emergency Operating Procedures. This method of cooling precludes any fuel clad temperature in the uncovered portion of the core from exceeding 1800°F. As Reactor Water Level drops below -200" with no injection available, this method of cooling becomes inadequate. Prolonged lack of cooling may result in severe overheating of the fuel clad, additional release of energy from accelerated clad oxidation, and eventual fuel melting. For events starting from full power operation, the failure to promptly reflood could result in some fuel melting. Even under these conditions vessel failure and containment failure with resultant release to the public would not be expected for some time. Reactor Water Level remaining below TAF for an extended amount of time represents an early indicator that significant core damage is in progress while providing sufficient time to initiate public protective actions.

Ample time should be allowed for Low Pressure ECCS and alternate injection systems to restore Reactor Water Level prior to entry into this classification. The time basis for deciding whether or not Reactor Water can be maintained > -200" should be based on the rate of reactor depressurization, the availability of low pressure injection sources, (ECCS and alternate injection systems), and the rate of Reactor coolant inventory loss. Indications such as Reactor Water Level trend, injection flow rates, containment parameter trends, and low pressure injection system operability should also be considered.

In the event, Reactor Water Level can not be restored > -200", containment flooding will be required by the EOPs. This will attempt to flood the containment as a means of flooding the RPV, and use a flooded containment as a heat sink for the nuclear fuel.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, FC2
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0201 (Q)-FC, Alternate Level Control
HC.OP-EO.ZZ-0207 (Q)-FC, Level/Power Control
HC.OP-EO.ZZ-0208 (Q)-FC, Primary Containment Flooding
BWR Owners Group Emergency Procedure Guidelines, Revision 4

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.2

IC Loss of Fuel Clad

EAL

DAPA Radiation Monitor reading ≥ **5000 R/hr**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Drywell Atmosphere Post Accident (DAPA) Radiation monitors indicating 5000 R/hr or greater corresponds to an instantaneous release of Reactor Coolant with a concentration of 300 μCi/gm Dose Equivalent Iodine-131(DEI-131) into the Primary Containment . This value of Reactor Coolant Activity is well above the threshold that could occur as the result of Iodine Spiking, resin/chemical intrusion transients or a HWCI System malfunction. This activity level corresponds to fuel clad damage of approximately 3.8%.

In addition, there are other events that could cause Drywell Atmosphere radiation levels to increase to this threshold, without a LOCA in the Drywell. These events involve shine from the reactor core if it is uncovered. While such events would not necessarily involve the calculated fuel clad damage percentage, they would be classifiable under other EALs at a Site Area Emergency level or higher.

Barrier Analysis

Fuel Clad Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

EAL 3.1.3 provides a core damage analysis showing that a Reactor Coolant activity of 300 µCi/gm Dose Equivalent Iodine-131(DEI) is indicative of 3.8% clad damage. Using Attachment 2 of EPIP 205H, 1% clad damage is indicated by a DAPA reading of 1.4E3 R/hr at 0.1 hrs after shutdown (the most conservative). This is shown on the Attachment as the 0.1% TID line. Extrapolating to the 3.8% clad damage point gives 5.32E3 R/hr. This is rounded to 5.0E3 R/hr. Hence, the Fuel Clad Barrier is lost.

NUMARC EAL RC3 addresses the use of DAPA to assess the status of the RCS Barrier, based on the release of Reactor Coolant into the Drywell. This EAL threshold is calculated assuming the instantaneous release and dispersal of the Reactor Coolant noble gas and iodine inventory associated with normal operating concentrations (within TS limits) into the Drywell Atmosphere. The reading would be lower than the threshold for EAL 3.1.2, thus being indicative of an RCS leak only. <u>However</u>, due to the inability of the DAPA radiation monitors to distinguish between a cloud of released RCS gases and shine from the Reactor Vessel and adajacent piping and components, this EAL is being omitted, as permitted by the NUMARC EALs, and other indications of RCS Leakage are being used. It should be recognized that DAPA exceeding 5000 R/hr would most likely occur due to core uncovery, as Reactor Water Level decreases below the Top of Active Fuel. This condition will result in appropriate escalation to a Site Area Emergency in the Fission Product Barrier Table, and hence use of DAPA exceeding 5000 R/hr is not needed to detect a Loss of the RCS Barrier.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC3
NUMARC NESP-007, RC3
EPIP 205H, TSC - Post Accident Core Damage Assessment
HC.OP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.3

IC Loss of Fuel Clad

EAL

Reactor Coolant Sample Activity ≥ **300 μCi/gm Dose Equivalent I-131**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Reactor Coolant sample analysis with specific activity greater than or equal to 300 μCi/gm Dose Equivalent I-131 (DEI-131) indicates fuel clad damage due to significant clad heating or mechanical stress, causing a Loss of the Fuel Clad Barrier. This threshold is well above the activity level that could occur as the result of Iodine spiking. The use of the term "Valid" as a qualifier for event classification is not required, since Reactor Coolant Activity of this magnitude can only occur as the result of fuel clad damage. This activity level corresponds to approximately 3.8% fuel clad damage.

Barrier Analysis

Fuel Clad Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

The percentage of Fuel Damage that corresponds to an RCS Activity of 300 µCi/gm DEI-131 is calculated as follows (for purposes of this calculation, cc and gm are considered equivalent):

Dose Factors (RG-1.109)

I-131 = 4.39E-3
I-132 = 5.23E-5
I-133 = 1.04E-3
I-134 = 1.37E-5
I-135 = 2.14E-4

Total core inventory (HCGS-UFSAR, table 12.2-135). This table gives 50% inventory, so table values are multiplied by 2.0.

I-131 = 8.64E7 Ci
I-132 = 1.29E8 Ci
I-133 = 1.99E8 Ci
I-134 = 2.32E8 Ci
I-135 = 1.81E8 Ci

Reactor Water Volume = 13000 cubic feet (HCGS-UFSAR, table 12.3-2)

Clad Release Fraction for iodines = 0.02 (Table 4.1, NUREG-1228)

The activity of each isotope in the clad would then be:

I-131 = 8.64E7(0.02) = 1.73E6 Ci
I-132 = 1.29E8(0.02) = 2.58E6 Ci
I-133 = 1.99E8(0.02) = 3.98E6 Ci
I-134 = 2.32E8(0.02) = 4.64E6 Ci
I-135 = 1.81E8(0.02) = 3.62E6 Ci

These activities are equivalent to 2.89E6 Ci DEI-131

$$DEI\text{-}131 = \frac{4.39E\text{-}3(1.73E6)+5.23E\text{-}5(2.58E6)+1.04E\text{-}3(3.98E6)+1.37E\text{-}5(4.64E6)+2.14E\text{-}4(3.62E6)}{4.39E\text{-}3}$$

Calculating the equivalent concentration:

$$Conc = \frac{2.89E6\,Ci(1E6\,\mu Ci/Ci)}{13000\ cf(2.8E4\,cc/cf)} = 7.94E3\,\mu Ci/cc$$

which represents the 100% clad damage concentration.

300 μCi/cc DEI-131 is then equivalent to:

$$\frac{300\,\mu Ci/cc}{7.94E3\,\mu Ci/cc}(100) = 3.78\%$$

This is rounded to 3.8%.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC1
HC.OP-AB.ZZ-0100(Q), High Reactor Coolant Activity
HC.OP-AB.ZZ-0203(Q), Main Steam Line High Radiation
HCGS Technical Specification LCO 3.4.5
NUREG 1228 - Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents, Table 4.1
Reg. Guide 1.109, Table E-9
HCGS-UFSAR, Table 12.2-135 and Table 12.3-2
10 CFR100

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.4

IC Potential Loss or Loss of Fuel Clad

EAL

ANY condition, in the opinion of the EC, that indicates a Potential Loss (3 pts) or Loss (4 pts) of the Fuel Clad Barrier

OPERATIONAL CONDITION - 1, 2, 3

BASIS

This EAL allows the Emergency Coordinator to address any condition that effects the integrity of the Fuel Clad Barrier that is not already covered elsewhere in the Fission Product Barrier Table. A complete loss of the ability to monitor the Fuel Clad Barrier should be considered as a "Potential Loss" of that barrier.

Barrier Analysis

Fuel Clad Barrier has been potentially lost or lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the potential loss or loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

None

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC5

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.1.a

IC Potential Loss of RCS

EAL

Reactor Water Level REACHES **-129"**, EXCLUDING intentional lowering of Reactor Water Level during an ATWS

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Reactor Water Level reaching -129", excluding intentional lowering of Reactor Water Level during an ATWS, indicates that the inventory loss from the RCS exceeds the capacity of available High Pressure injection sources. Below this threshold, a challenge to maintaining Adequate Core Cooling by core submergence exists, based on Reactor Water Level continuing to decrease, thus a Potential Loss of the RCS Barrier exists. Without core submergence, the integrity of the Fuel Clad would be in jeopardy. Appropriate classification under this EAL is based on reaching Reactor Water Level of -129" (instead of being able to restore and maintain above -129"), due to the challenge that exist to core submergence. Reactor Water Level reaching this threshold results from either a LOCA exceeding available makeup capacity or a Total Loss of High Pressure injection capability.

In addition, during an Anticipated Transient Without Scram (ATWS), it is possible that operator action will be taken to intentionally lower Reactor Water Level to below -129". for Reactor Power Control purposes. For this event, classification must be made in accordance with EAL Section 5.0.

Barrier Analysis

RCS Barrier has been potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core Submergence is the preferred method of maintaining adequate core cooling. When Reactor Water Level decreases to -129", a significant challenge to continued core submergence exists. The threshold for this EAL corrresponds to the initiation setpoint l for the low pressure Emergency Core Cooling Systems (ECCS).

Reactor Water Level reaching -129" occurs when the rate of of inventory loss is greater than the rate of inventory from High Pressure injection sources. This condition can occur as the result of the following events/sequences (excluding intentional lowering of Reactor Water level during an ATWS).

- A LOCA will cause Reactor Water Level to reach -129" when the LOCA is the result of a large break (momentary core uncovery is expected to occur under this condition) or when the LOCA is due to a small or intermediate break in combination with an inability of High Pressure injection sources to keep up with the leak rate.

- A Loss of High Pressure injection sources without the presence of a LOCA will also result in Reactor Water Level decreasing to -129" , due to continued Reactor Steam Flow without makeup.

Either of these events/sequences results in a potential challenge to the RCS Barrier when Reactor Water level reaches -129", hence classification at this threshold is appropriate. However, for both these sequences, low Pressure ECCS are designed to inject to the Reactor as Reactor Pressure decreases below the shutoff head of the pumps. Reactor Depressurization will occcur either due to the LOCA or Manual initiation of Emergency Depressurization when Reactor Water Level reaches -161", provided injection systems are available. This will allow for restoration of Reactor Water Level and reestablishment of Core Submergence.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, RC5
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0116 (Q), Containment Isolation and Recovery From An Isolation
HC.OP-AB.ZZ-0200 (Q), Reactor Low Water Level
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP.EO.ZZ-0101 (Q)-FC,Reactor Pressure Vessel (RPV) Control
HCGS Technical Specifications LCO 3/4.3, Instrumentation

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.1.b

IC Loss of RCS

EAL

Reactor Water Level REACHES **-161" (Top of Active Fuel)**, EXCLUDING intentional lowering of Reactor Water Level during an ATWS

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Reactor Water Level reaching -161" (Top of Active Fuel - TAF), excluding intentional lowering of Reactor Water Level during an ATWS, results in an inability to maintain adequate core cooling by core submergence, causing a Loss of the RCS Barrier. Without core submergence, the integrity of the fuel clad barrier is in jeopardy. Appropriate classification under this EAL is based on reaching Reactor Water Level of -161" (instead of being able to restore and maintain above -161") due to the potentially severe consequences of a loss of core submergence. Reactor Water Level reaching this threshold results from either a LOCA exceeding available makeup capacity or a Total Loss of High Pressure injection capability.

In addition, during an Anticipated Transient Without Scram (ATWS), it is possible that operator actions will be taken to intentionally lower Reactor Water Level to between -161" and -190", for Reactor Power Control purposes. For this event, classification must be made in accordance with EAL Section 5.0

Barrier Analysis

RCS Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based upon the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core Submergence is the preferred method of maintaining adequate core cooling. When Reactor Water Level decreases to below TAF, the ability to effectively remove decay heat is being challenged, and as such the Fuel Clad barrier can no longer be considered intact. While the Emergency Operating Procedures provide contingencies to establish adequate core cooling when Reactor Water Level drops below TAF (Steam Cooling with or without injection), these actions are designed to be an alternative method of providing adequate core cooling while actions are taken to reestablish core submergence. Sustained partial or total core uncovery can result in fuel clad damage and a significant release of fission products to the Reactor coolant. Sustained core uncovery can also result in a breach of the Reactor Vessel due to core melt material interaction with the RPV.

A Loss of Core Submergence will occur when the rate of inventory loss is greater than the rate of inventory makeup from High Pressure injection sources. This condition can occur as the result of the following events/sequences (excluding intentional lowering of Reactor Water Level during an ATWS).

- A LOCA will cause Reactor Water Level to reach the Top of Active Fuel when the LOCA is the result of a large break (momentary core uncovery is expected to occur under this condition) or when the LOCA is due to a small or intermediate break in combination with an inability of High Pressure injection sources to keep up with the leak rate.

- A Loss of High Pressure injection sources without the presence of a LOCA will also result in Reactor Water Level decreasing to TAF, due to continued Reactor Steam Flow without makeup.

Either of these events/sequences results in a challenge to the Fuel Clad Barrier when Reactor Water Level reaches TAF due to core uncovery, hence classification at this threshold is appropriate. However, for both these sequences, Low Pressure ECCS are designed to inject to the Reactor as Reactor Pressure decreases below the shutoff head of the pumps. Reactor Depressurization will occur either due to the LOCA or Manual initiation of Emergency Depressurization when Reactor Water Level reaches -161", provided injection systems are available. This will allow for restoration of Reactor Water Level and reestablishment of Core Submergence.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, RC4
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0201 (Q)-FC, Alternate Level Control
HC.OP-EO.ZZ-0207 (Q)-FC, Level/Power Control
BWR Owner's Group Emergency Procedure Guidelines, Rev. 4

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.2.a

IC Potential Loss of RCS

EAL

Unisolable RCS Leak Rate ≥ **50 GPM** INSIDE Primary Containment

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Unisolable RCS Leak Rate exceeding 50 GPM, inside Primary Containment is indicative of a potential loss of the RCS. An unisolable leak rate of this magnitude is significant due to the potential for further break propagation, resulting in a much higher loss of inventory with an inability to isolate the leak source. As such, this threshold is considered a Potential Loss of the RCS. Leakage just above the 50 GPM threshold is well within the capacity of normal and emergency injection systems and is not a significant concern for core uncovery. However, 50 GPM is the minimum leak rate that would be classified under this EAL, with the maximum being equivalent to the leak rate that would result in either Reactor Water Level reaching -129" or Drywell Pressure reaching 1.68 PSIG, since these two conditions are obviously more recognizable to Control Room personnel, than an existing leak rate.

Specifying an unisolable RCS leak as part of the threshold for this EAL, precludes classifying events such as an isolable Reactor Recirculation Pump dual seal failure under this EAL.

Barrier Analysis

RCS Barrier has been potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

It is important to recognize that the unisolable RCS leak rate established in this EAL is inside the Primary Containment. The inability to isolate the leak would eventually lead to a High Drywell Pressure (> 1.68 PSIG) actuation of RPS, ECCS and PCIS. The actuation would lead to an isolation of the Drywell Floor and Equipment Drain sumps, complicating efforts to further identify and quantify any changes in the existing leak rate. In addition, monitoring of the leak rate could be limited by reaching the upper range (50 GPM) of the Drywell Leak Detection channels (9AX313 - Equipment, 9AX314- Floor Drain).

For leakage outside Containment, since quantification of the leak rate is much more difficult due to the physical size of the Reactor Building, receipt of a **Valid** isolation signal has been established as the threshold for classification of this type of leakage.

DEVIATION

None

REFERENCES

NUMARC NESP-007, RC1
NUMARC Questions and Answers, June 1993, "Fission Product Barrier Question #11"
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0116(Q), Containment Isolations and Recovery from an Isolation
HC.OP-AB.ZZ-0201(Q), Drywell High Pressure/Loss of Drywell Cooling
HC.RP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response
HC.OP-EO.ZZ-0100(Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101(Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102(Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103(Q)-FC, Secondary Containment Control
HC.OP-GP.ZZ-0005(Q), Drywell Leakage Source Detection

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.2.b

IC Loss of RCS

EAL

Valid High Drywell Pressure Condition (**≥ 1.68 psig)**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

A **Valid** High Drywell Pressure Condition (≥ 1.68 PSIG) is indicative of the release of high energy Reactor Coolant from the RCS into the Drywell and hence is considered a Loss of the RCS Barrier. **Valid** is defined as the High Drywell Pressure condition specifically due to RCS leakage into the Drywell, ensuring that event classification under this EAL is truly reflective of a degraded RCS Barrier. This precludes unwarranted event declaration as the result of system malfunctions, including a loss of Drywell Cooling or inadvertent Drywell makeup. Indication of an RCS leak should be positively determined by observing Primary Containment parameters, including Drywell Pressure and Temperature trends, Drywell Equipment and Floor Drain sump levels, DAPA Radiation levels, atmospheric pressure, Torus Pressure, and the status of Drywell Cooling systems.

An isolable Reactor Recirculation Pump dual seal failure should not result in Drywell Pressure reaching the threshold for this EAL, hence classification under this EAL should not occur.

Barrier Analysis

RCS Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

RCS Leakage into the Drywell exceeding 50 GPM is substantially greater than the RCS leakage thresholds established in EAL Section 2.1.1, and represents further degradation of the RCS barrier. Inability to isolate the RCS leakage would eventually result in a High Drywell Pressure (>1.68 PSIG) actuation of RPS, ECCS and PCIS. The actuation would lead to an isolation of the Drywell Floor and Equipment Drain sumps, complicating efforts to further identify and quantify any changes in the leak rate. In addition, monitoring of the leak rate could be limited by reaching the upper range (50 GPM) of the Drywell Leak Detection channels (9AX313 - Equipment, 9AX314 - Floor Drain).

There are multiple Control Room indicators and alarms which can be used to determine the presence of a High Drywell Pressure condition. Overhead Annunciators will alarm at 1.5 PSIG and 1.68 PSIG. Plant automatic response to a High Drywell Pressure condition includes: a reactor scram, ECCS initiation, trip of the drywell cooling fans and isolation of the cooling water to the drywell. These actuations may mask the trend in drywell pressure. For example, the scram will result in less heat being added to the containment and the cooling water isolation will result in no heat being removed.

Actions initiated as part of increasing drywell pressure condition include investigation of the source of the increased leakage into the drywell. maximizing drywell cooling and venting the Drywell (if release criteria can be satisfied). These actions are designed to control and relieve increasing drywell pressure.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, RC2
NUMARC Questions and Answers, June 1993, "Fission Product Barrier Question #11"
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0116 (Q), Containment Isolation and Recovery From An Isolation
HC.OP-AB.ZZ-0201 (Q), Drywell High Pressure/Loss of Drywell Cooling
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-GP.ZZ-0005 (Q), Drywell Leak Source Detection
Hope Creek Appendix A based on NEDO-2121, Supplement A to BWR Owners Group Emergency Procedure Guidelines, Revision 4
HCGS Technical Specifications LCO 3/4.3, Instrumentation

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.3.a

IC Potential Loss of RCS

EAL

Main Steam Line Break OUTSIDE Primary Containment , resulting in an AUTOMATIC MSIV Isolation Signal **AND** ALL 4 Main Steam Lines have been successfully isolated based on NO indication of CONTINUING FLOW / LEAKAGE OUTSIDE the Primary Containment AFTER **valve closure** from the Main Control Room has been attempted

OPERATIONAL CONDITION - 1, 2, 3

BASIS

A Main Steam Line Break outside the Primary Containment, resulting in an automatic MSIV Isolation Signal, could result in dose consequences offsite from a "puff" release in excess of 10 millirem, based on design basis accident analysis, even if MSIV closure occurs within design limits. Hence this condition is classified as a Potential Loss of the RCS Barrier. Classification under this EAL is specifically for a Main Steam Line Break outside the Primary Containment, as evidenced by a rapid change in Main Steam Line Flow and Steam Tunnel Temperature, that results in automatic isolation with no indication of continuing leakage. Valve Packing leaks that result in elevated Steam Tunnel temperatures do not require classification under this EAL.

A manual actuation of NSSSS or manual MSIV closure PRIOR to exceeding the setpoints that would result in an automatic isolation of the MSIV should not result in a "puff" release exceeding 10 millirem, and thus should not be classified under this EAL. Verification that continuing leakage does not exist, ensures that any potential release will not significantly exceed the 10 CFR100 limits. This EAL is specific to a break outside the Primary Containment, since a break outside represents a potential challenge to Primary Containment Integrity due to the Containment Bypass condition that would exist until MSIV closure occurred . Failure to completely isolate the effected Main Steam Line(s) as determined by valve position and indication of continuing leakage would result in an additional Loss of the Primary Containment Barrier.

In addition, this EAL ALLOWS for **valve closure** from the Main Control Room to isolate any Main Steam Line that did not completely isolate. **Valve closure** is defined as the closure of ANY valve from the Main Control Room associated with the effected Main Steam Line(s), that did not completely isolate. For example, if the isolation logic fails to cause valve closure, but operator actions implemented in the Main Control Room successfully isolates the effected Main Steam Line(s), then event classification under this EAL is warranted due to the consequences of the event previously discussed. This includes Motor Operated Valves that are not controlled by the isolation logic, but are manually controlled from the Main Control Room. (i.e Main Steam Stop Valves 1ABHV-3631 A/B/C/D). In the event the effected Main Steam Line(s) can not be isolated, escalation of the classification will be required.

Barrier Analysis

RCS Barrier has been potentially lost

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional barriers per EAL section 3.0.

DISCUSSION

The Main Steam System is associated with systems that are part of the RCS boundary and penetrate the Primary Containment. Isolation requirements for these lines are covered in 10CFR50, Appendix A, General Design Criteria 55. These systems form a closed loop outside the Primary Containment and are not open or potentially open to the environment. These systems represent an extension of the RCS Barrier beyond the Primary Containment.

Positive identification of a Main Steam Line Break outside the Primary Containment can be based on receipt of the following Overhead Annunciators:

NSSSS ISLN SIG - STM TNL TEMP HI	(C8-C4)
NSSSS ISLN SIG - MN STM FLOW HI	(C8-B4)
MSIV CLOSURE	(C5-B3)

as well as the following indications:

MSIV TRIP LOGIC TRIPPED
Rapid changes in Main Steam Line Flow and Steam Tunnel Temperatures

DEVIATION

This EAL is being maintained in the Fission Product Barrier Table for ease of use by the operators. It has been categorized as a "Potential loss" since the RCS leak is successfully isolated and an alert classification will still be made as a result of the potential loss of RCS.

REFERENCES

NUMARC NESP-007, RC1
NUMARC Question and Answer, June 1983, "Fission Product Barrier- BWR" Question #4
10 CFR50, App. A, GDC 55
10 CFR 100
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0114(Q), Loss of Primary Containment Integrity
HC.OP-AB.ZZ-0116(Q), Containment Isolations and Recovery from an Isolation
HC.OP-AB.ZZ-0203(Q), Main Steam Line High Radiation
HC.OP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response
HC.OP-AR.ZZ-0011(Q), Annunciator Response Procedures, Window C6
HC.OP-AR.ZZ-0012(Q), Annunciator Response Procedures, Window C8
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103 (Q)-FC, Reactor Building Control
HC.OP-EO.ZZ-0104 (Q)-FC, Radioactive Release Control
HCGS Technical Specifications, LCO 3/4.3
HCGS UFSAR, Section 6.2.4.3.1

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.3.b

IC Loss of RCS

EAL

RCS Line Break OUTSIDE Primary Containment, resulting in a **Valid** Isolation Signal for ANY one of the following systems:

- NSSSS
- HPCI
- RCIC

AND

Indication of CONTINUING FLOW / LEAKAGE OUTSIDE the Primary Containment through the **effected system** AFTER **valve closure** from the Main Control Room has been attempted

OPERATIONAL CONDITION - 1, 2, 3

BASIS

An RCS Line Break outside Primary Containment that results in a **Valid** Isolation Signal for any of the systems listed in the EAL requires closure of the associated Primary Containment Isolation valves to maintain RCS and Primary Containment integrity under abnormal conditions. A failure of these isolation valves to isolate directly allows Reactor Coolant to be released outside the Primary Containment (Containment Bypass), resulting in a Loss of RCS and Loss of Containment. An RCS Line is ANY line that communicates directly with the Reactor. An RCS Line Break with indication of continuing flow is classified under this EAL, due to the continuing discharge of Reactor Coolant outside the Primary Containment along with a potential for further "break propagation". This is the only condition that warrants classification under this EAL.

Valid is defined as the isolation signal specifically being the result of an RCS Line Break, thus ensuring that the RCS discharge is of significant magnitude to pose a threat to the integrity of the

RCS Barrier. This precludes unwarranted Event Classification as the result of condition that result in limited leakage with no potential for "break propagation", including valve packing leaks outside Primary Containment and RWCU Pump Seal Leaks. In addition, isolation signal generated from known failures in other systems, that do not result in Reactor Coolant discharging outside the Primary Containment do not warrant Event Classification under this EAL either. Examples of such failures include a high temperature isolation resulting from a loss of ventilation or cooling water, spurious actuation during I&C surveillance testing or a low Reactor Water Level Condition due to a Loss of High Pressure injection capability.

In addition, this EAL ALLOWS for **valve closure** from the Main Control Room to isolate any systems that did not completely isolate, prior to event classification. **Valve closure** is defined as the closure of ANY valve from the Main Control Room in the system(s) that did not completely isolate. For example, if the isolation logic fails to cause valve closure, but operator actions implemented in the Main Control Room successfully isolates the **effected system**, then classification under this EAL is not warranted. This includes Motor Operated Valves that are not control by the isolation logic, but are manually controlled from the Main Control Room. **Effected system** is defined as the system that is providing the flowpath outside the Primary Containment.

Barrier Analysis

RCS Barrier has been lost

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

NSSSS isolations, as well as HPCI and RCIC steam line isolations, are associated with systems that are part of the RCS boundary and penetrate the Primary Containment. Isolation requirements for these lines are covered in 10CFR50, Appendix A , General Design Criteria 55. These systems form a closed loop outside the Primary Containment, and are not open or potentially open to the environment. They are included in this EAL since they represent an extension of the RCS boundary beyond the Primary Containment, and a potential release path from the RCS to the environment. Without a completed isolation, continuing flow/leakage represents a situation where Reactor Coolant is discharging outside the Primary Containment, including areas in the Reactor Building addressed in the EOPs.

Indication of continuing flow/leakage includes: flow indication through isolated lines, increasing Reactor Building area temperatures, area radiation levels, sump levels, or room levels in spaces associated with affected lines, as well as increases in Plant Vent Effluent levels.

DEVIATION

This EAL is being considered a loss of the reactor coolant boundary since actuation of listed isolation system indicate a leak of significant magnitude, and an isolation failure. The classification for exceeding this EAL remains consistent with NUMARC guide lines.

REFERENCES

NUMARC NESP-007, RC1
10 CFR50, App. A, GDC 55
10 CFR 100
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0114(Q), Loss of Primary Containment Integrity
HC.OP-AB.ZZ-0116(Q), Containment Isolations and Recovery from an Isolation
HC.OP-AB.ZZ-0203(Q), Main Steam Line High Radiation
HC.OP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response
HC.OP-AR.ZZ-0011(Q), Annunciator Response Procedures, Window C6
HC.OP-AR.ZZ-0012(Q), Annunciator Response Procedures, Window C8
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103 (Q)-FC, Reactor Building Control
HC.OP-EO.ZZ-0104 (Q)-FC, Radioactive Release Control
HCGS Technical Specifications LCO 3/4.3, Instrumentation
HCGS UFSAR, Section 6.2.4.3.1

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.4

IC Potential Loss or Loss of RCS

EAL

ANY condition, in the opinion of the EC, that indicates a Potential Loss (3 pts) or Loss (4 pts) of the RCS Barrier

OPERATIONAL CONDITION - 1, 2, 3

BASIS

This EAL allows the Emergency Coordinator to address any condition that effects the integrity of the RCS Barrier that is not already covered elsewhere in the Fission Product Barrier Table. A complete loss of the ability to monitor the RCS barrier should be considered as a "Potential Loss" of that barrier.

Barrier Analysis

RCS Barrier has been potentially lost or lost.

ESCALATION CRITERIA

Emergency Classification will be escalate based on the Potential Loss or Loss of additional barriers per EAL section 3.0.

DISCUSSION

None

DEVIATION

None

REFERENCES

NUMARC NESP-007, RC6

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.1

IC Potential Loss of Containment

EAL

Reactor Water Level CANNOT BE RESTORED AND MAINTAINED above **-200"** (Minimum Zero Injection RPV Water Level)

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Inability to restore and maintain Reactor Water Level above - 200" (Minimum Zero Injection RPV Water Level), results in a loss of adequate core cooling by all mechanisms, causing a Potential Loss of the Fuel Clad Barrier. Without adequate core cooling, the integrity of the Containment is being challenged and can no longer be assured. Appropriate classification under this EAL is based on the failure of injection systems to restore and maintain Reactor Water Level above -200", following a condition that causes level to decrease below the threshold. For example, a large break LOCA is expected to cause Reactor Water Level to momentarily decrease below -200", due to the response time of Low Pressure ECCS. As these systems initiate and commence injection to the Reactor, water level will begin to increase and should be able to be maintained above -200". In this case, classification under this EAL is not appropriate as plant systems have performed their intended design function and will eventually restore adequate core cooling by core submergence. However, in the event that Low Pressure ECCS and alternate injection system, as defined in the EOPs are in a degraded condition (i.e., Station Blackout, ECCS Suction Strainer plugging, etc.) and Reactor Water Level can not be restored and maintained above -200", then classification under this EAL should occur due to the Potential Loss of Containment from the release of energy to the containment from imminent fuel failure.

Barrier Analysis

Primary Containment Barrier has been potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based upon the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core submergence is the preferred method for maintaining adequate core cooling. The failure to reestablish Reactor Water Level above -161", the Top of Active Fuel (TAF), for an extended period of time could lead to significant fuel damage. With Reactor Water Level below TAF, but above the Minimum Zero Injection RPV Water Level (-200"), adequate core cooling occurs due to the cooling effects of steam generated in the covered portion of the core flowing through the uncovered portion (Steam Cooling). The Minimum Zero Injection RPV Water Level is defined in the Emergency Operating Procedures. This method of cooling precludes any fuel clad temperature in the uncovered portion of the core from exceeding 1800°F. As Reactor Water Level drops below -200" with no injection available, this method of cooling becomes inadequate. Prolonged lack of cooling may result in severe overheating of the fuel clad, additional release of energy from accelerated clad oxidation, and eventual fuel melting. For events starting from full power operation, the failure to promptly reflood could result in some fuel melting. Even under these conditions vessel failure and containment failure with resultant release to the public would not be expected for some time. Reactor Water Level remaining below TAF for an extended amount of time represents an early indicator that significant core damage is in progress while providing sufficient time to initiate public protective actions.

Ample time should be provided for Low Pressure ECCS and alternate injection systems restore Reactor Water Level prior to entry into this classification. The time basis for deciding whether or not Reactor Water can be maintained > -200" should be based on the rate of reactor depressurization, the availability of low pressure injection sources, (ECCS and alternate injection systems), and the rate of Reactor coolant inventory loss. Indications such as Reactor Water Level trend, injection flow rates, containment parameter trends, and low pressure injection system operability should also be considered.

In the event, Reactor Water Level can not be restored > -200", containment flooding will be required by the EOPs. This will attempt to flood the containment as a means of flooding the RPV, and use a flooded containment as a heat sink for the nuclear fuel.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, PC4
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP.EO.ZZ-0101 (Q)-FC,Reactor Pressure Vessel (RPV) Control
HC.OP.EO.ZZ-0201 (Q)-FC, Alternate Level Control
HC.OP.EO.ZZ-208 (Q)-FC, Primary Containment Flooding
BWR Owners Group Emergency Procedure Guidelines, Revision 4

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.2.a

IC Potential Loss of Containment

EAL

Containment Venting is Required by the Emergency Operating Procedures (EOPs) <u>EXCLUDING</u> Containment Venting due to an ATWS

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Containment venting required by the EOPs indicates a degrading condition in containment and is implemented in an effort to preclude containment failure. Venting is required before Suppression Chamber pressure reaches 65 PSIG or Hydrogen concentration reaches the Lower Explosive Limit (LEL = 4%) and Oxygen concentration reaches 5%. Exceeding these parameters creates the potential for an unisolable breach of the primary containment, which could result in an uncontrolled, unmonitored, and untreated release of radioactivity to the environment. This EAL represents a Potential Loss of Containment, since containment venting is required due to Containment parameters potentially exceeding their design limits. During an ATWS event, classification should be made in accordance with ECG Section 5, since this will provide a more accurate classification of the condition. Hence, Containment Venting due to an ATWS should not be classified under this EAL. The magnitude of any radiological release is dependant upon events leading to the requirement for emergency venting, including a loss of the RCS and a loss of the Fuel Clad Barriers.

A Downcomer failure, by itself, <u>does not</u> represent a Loss of the Primary Containment Barrier. This failure does, however, render the Primary Containment inoperable per the Technical Specification, as Primary Containment integrity has been compromised. A Downcomer failure combined with a large break LOCA will likely result in a Potential Loss of Primary Containment under this EAL if Containment pressure can not be maintained below 65 PSIG and Containment Venting is required.

Barrier Analysis

Primary Containment Barrier has been potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Venting of the Primary Containment is initiated to preserve containment integrity under accident conditions. Primary Containment venting is required when Suppression Chamber cannot be maintained below 65 psig, which is well above the maximum pressure expected to be present in the Primary Containment during a design basis Loss of Coolant Accident (LOCA). Primary Containment venting is also required based on hydrogen concentrations exceeding 4%.H_2 concentrations in excess of 6.0 % requires Emergency Depressurization and subsequent containment venting. Venting is continued until either H_2 concentration has been reduced to < 6.0% or O_2 levels have been reduced to < 5.0%. Venting with elevated hydrogen concentration conditions ensures that containment failure resulting from a hydrogen detonation or deflagration does not occur.

The elevated hydrogen in the containment may result from excessive zircaloy-water reaction occurring following a LOCA. Additionally, hydrogen and oxygen gas may be introduced into the containment environment from long term disassociation of water in the Suppression Chamber.

EOP procedural guidance in these cases is provided to vent the Primary Containment regardless of off-site dose consequences. Although radiological releases resulting from venting containment may exceed EPA limits, a controlled, monitored, and isolable release is preferred to a potential uncontrolled, unmonitored radiological release that would result from a failure of containment.

DEVIATION

NUMARC PC2 EAL says intentional venting per EOPs is a loss of containment. Per Hope Creek procedures the containment is vented if design pressure or explosive mixture conditions exist. Per NUMARC PC 1 this is considered a potential loss of containment. Since both conditions are essentially the same, PSE&G has decided to call this a potential loss as recommended in NUMARC PC1.

REFERENCES

NUMARC NESP-0007, PC1, PC2
HC.OP-AB.ZZ-0201 (Q), Drywell High Pressure/Loss of Drywell Cooling
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0318 (Q)-FC, Containment Venting
BWR Owners Group Emergency Procedure Guidelines, Revision 4

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.2.b

IC Loss of Containment

EAL

Containment Failure as indicated by a rapid decrease in Drywell pressure following an increase in pressure above **1.68 psig**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Containment failure indicated by a rapid decrease in Drywell pressure following a significant increase in Drywell pressure is indicative of a Loss of the Containment barrier. This EAL specifically represents a Loss of Containment , whereby a unisolable breach of the containment structure has occurred. Conditions that result in a decrease in Drywell pressure following a pressure rise that are not the direct result of a Containment failure do not warrant classification under this EAL. These events include the initiation of Drywell Sprays, the reestablishment of Drywell Cooling, Containment Venting as required by the EOPs, and anticipated Drywell pressure decrease due to ambient losses.

A Downcomer failure, by itself, does not represent a Loss of the Primary Containment Barrier. This failure does, however, render the Primary Containment inoperable per the Technical Specification, as Primary Containment integrity has been compromised. A Downcomer failure combined with a large break LOCA will likely result in a Potential Loss of Primary Containment under EAL 3.3.2.a if Containment pressure can not be maintained below 65 PSIG and Containment Venting is required.

Barrier Analysis

Primary Containment Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Appropriate classification under this EAL occurs as the result of a Containment failure. Drywell pressure reaching 1.68 psig indicates that there is a significant release of reactor coolant to the containment. Unless this source of leakage is isolated or the Reactor is depressurized, Drywell pressure would not be expected to decrease in a rapid manner.

Other indications such as Reactor Building Area Radiation Monitors (ARMs) radiation levels, Reactor Building area temperatures, Reactor Building floor and sump levels, Plant Effluent radiation levels, and containment isolation status should be used to confirm the loss of containment integrity if possible. Reactor Building to Torus vacuum breaker status should be monitored to ensure that this pathway does not result in a loss of containment integrity.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, PC1
HC.OP-AB.ZZ-0114 (Q), Loss of Primary Containment Integrity
HC.OP-AB.ZZ-0116 (Q), Containment Isolations and Recovery from an Isolation
HC.OP-AB.ZZ-0201 (Q), Drywell High Pressure/Loss of Drywell Cooling
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103 (Q)-FC, Reactor Building Control
BWR Owners Group Emergency Procedure Guidelines, Revision 4

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.3

IC Potential Loss of Containment

EAL

DAPA Radiation Monitor reading $\geq$ **28000 R/hr**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Drywell Atmosphere Post Accident (DAPA) monitor reading $\geq$ 28000 R/hr indicates significant fuel damage, well in excess of the level corresponding to the loss of the RCS and Fuel Clad barriers. This threshold corresponds to approximately 20% fuel clad damage. Regardless of whether or not containment is challenged, this amount of activity in containment, if released, could have severe consequences and it is prudent to treat this condition as a Potential Loss of containment.

Barrier Analysis

Primary Containment Barrier is potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

NUREG-1228, "Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents", states that releases of severe magnitude are not possible if plant systems function as designed, and any accident with a release of 20% or greater of the gap region must be considered severe.

Using attachment 2 of EPIP 205H, 10% clad damage is represented by a DAPA reading of 1.4E4 R/hr at 0.1 hrs after shutdown (the most conservative). This is shown on the attachment as the 1% TID line. Extrapolating to 20% clad damage gives a reading of 2.8E4 R/hr.

Exceeding a DAPA reading of 28000 R/hr should meet the criteria for declaration of a General Emergency.

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC3
NUREG-1228 - Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents
EPIP 205H, TSC - Post Accident Core Damage Assessment

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.4.a

IC Potential Loss of Containment

EAL

> **RCS Line Break** OUTSIDE Primary Containment, resulting in a **Valid** Isolation Signal for ANY one of the following systems:
>
> - NSSSS (excluding Main Steam Lines)
> - HPCI
> - RCIC
>
> **AND**
>
> NO indication of CONTINUING FLOW / LEAKAGE OUTSIDE the Primary Containment through the **effected system** AFTER **valve closure** from the Main Control Room has been attempted

OPERATIONAL CONDITION - 1, 2, 3

BASIS

An RCS Line Break outside Primary Containment that results in a **Valid** Isolation Signal for any of the systems listed in the EAL requires closure of the associated Primary Containment Isolation valves to maintain RCS and Primary Containment integrity under abnormal conditions. Successful closure of the required isolation valves that results in NO indication of continuing FLOW / LEAKAGE is classified under this EAL as an Unusual Event, due to the significance of an RCS line break outside the Primary Containment for one of the systems listed in the EAL. An RCS Line is ANY line that communicates directly with the Reactor. A Main Steam Line Break with successful isolation is excluded from this EAL, since it is covered under EAL 3.2.3.a. An RCS Line Break with indication of successful isolation is the only condition that warrants classification under this EAL.

Valid is defined as the isolation signal specifically being the result of an RCS Line Break, thus ensuring that the RCS discharge is of significant magnitude to pose a threat to the integrity of the Primary Containment Barrier. This precludes unwarranted Event Classification as the result of condition that result in limited leakage with no potential for "break propagation", including valve packing leaks outside Primary Containment and RWCU Pump Seal Leaks. In addition, isolation signal generated from known failures in other systems, that do not result in Reactor Coolant discharging outside the Primary Containment do not warrant Event Classification under this EAL either. Examples of such failures include a high temperature isolation resulting from a loss of ventilation or cooling water, spurious actuation during I&C surveillance testing or a low Reactor Water Level Condition due to a Loss of High Pressure injection capability.

In addition, this EAL ALLOWS for **valve closure** from the Main Control Room to isolate any systems that did not completely isolate, prior to event classification. **Valve closure** is defined as the closure of ANY valve from the Main Control Room in the system(s) that did not completely isolate. For example, if the isolation logic fails to cause valve closure, but operator actions implemented in the Main Control Room successfully isolates the **effected system**, then event classification under this EAL is warranted, due to the consequencesof the event previously discussed. This includes Motor Operated Valves that are not control by the isolation logic, but are manually controlled from the Main Control Room. **Effected system** is defined as the system that is providing the flowpath outside the Primary Containment.In the event the effected system(s) can not be isolated, escalation of the classification will be required.

Barrier Analysis

Primary Containment Barrier has been potentially lost

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

NSSSS isolations, as well as HPCI and RCIC steam line isolations, are associated with systems that are part of the RCS boundary and penetrate the Primary Containment. Isolation requirements for these lines are covered in 10CFR50, Appendix A , General Design Criteria 55. These systems form a closed loop outside the Primary Containment, and are not open or potentially open to the environment. They are included in this EAL since they represent an extension of the RCS boundary beyond the Primary Containment, and a potential release path from the RCS to the environment.

Indication of continuing flow/leakage includes: flow indication through isolated lines, increasing Reactor Building area temperatures, area radiation levels, sump levels, or room levels in spaces associated with affected lines, as well as increases in Plant Vent Effluent levels.

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC5
10 CFR50, App. A, GDC 55
10 CFR 100
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0114(Q), Loss of Primary Containment Integrity
HC.OP-AB.ZZ-0116(Q), Containment Isolations and Recovery from an Isolation
HC.OP-AB.ZZ-0203(Q), Main Steam Line High Radiation
HC.OP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response
HC.OP-AR.ZZ-0011(Q), Annunciator Response Procedures, Window C6
HC.OP-AR.ZZ-0012(Q), Annunciator Response Procedures, Window C8
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103 (Q)-FC, Reactor Building Control
HC.OP-EO.ZZ-0104 (Q)-FC, Radioactive Release Control
HCGS Technical Specifications, LCO 3/4.3
HCGS UFSAR, Section 6.2.4.3.1

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.4.b

IC Loss of Containment

EAL

Isolation Signal for <u>ANY</u> one of the following systems:

- NSSSS
- PCIS
- HPCI
- RCIC

<u>**AND**</u>

Indication of <u>CONTINUING FLOW / LEAKAGE OUTSIDE</u> the Primary Containment through the **effected system** <u>AFTER</u> **valve closure** has been attempted from the Main Control Room

OPERATIONAL CONDITION - 1, 2, 3

BASIS

An Isolation Signal for any of the systems listed in the EAL requires closure of the associated Primary Containment Isolation valves to maintain RCS and Primary Containment integrity under abnormal conditions. A failure of these isolation valves to isolate directly allows the transport of Reactor Coolant or containment atmosphere to outside the Primary Containment (Containment Breach or Bypass), resulting in a Loss of Containment. This EAL addresses two condition under which RCS is being transported <u>OUTSIDE</u> the Primary Containment. The first condition is associated with an Isolation signal being generated as the result of an RCS Line Break with a failure of the isolation valves to close. In this condition, a <u>ABNORMAL</u> <u>FLOWPATH</u> exists for RCS to be discharged directly outside the Primary Containment. The second condition is associated with the failure of both Inboard and Outboard Isolation valves to FULLY close following an Isolation signal. In this condition, a flow path from containment atmosphere to areas outside of the Primary Containment exists.

In addition, this EAL ALLOWS for **valve closure** from the Main Control Room to isolate any systems that did not completely isolate, prior to event classification. **Valve closure** is defined as the closure of ANY valve from the Main Control Room in the system(s) that did not completely isolate. For example, if the isolation logic fails to cause valve closure, but operator actions implemented in the Main Control Room successfully isolates the **effected system**, then Unusual Event declaration is not warranted. This includes Motor Operated Valves that are not control by the isolation logic, but are manually controlled from the Main Control Room. **Effected system** is defined as the system that is providing the flowpath outside the Primary Containment.

Barrier Analysis

Primary Containment has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

PCIS Isolations are associated with systems having lines that are either: 1) connect directly to the Primary Containment atmosphere and penetrate the Primary Containment; or 2) penetrate the Primary Containment and are neither part of the RCS boundary and are not connected directly to the Primary Containment atmosphere (e.g. RACS, Chilled Water). Isolation requirements for these lines are covered in 10CFR50, App. A , General Design Criteria 56 and 57 respectively. This event, therefore, may potentially connect the RCS or the Primary Containment atmosphere to the environment. Without a completed isolation, continuing flow/leakage represents a release path from the RCS or Primary containment to the environment.

NSSSS isolations, as well as HPCI and RCIC steam line isolations, are associated with systems that are part of the RCS boundary and penetrate the Primary Containment. Isolation requirements for these lines are covered in 10CFR50, App. A , General Design Criteria 55. These systems form a closed loop outside the Primary Containment, and are not open or potentially open to the environment. They are included in this EAL since they represent an extension of the RCS boundary beyond the Primary Containment, and a potential release path from the RCS to the environment. Without a completed isolation, continuing leakage represents a Primary System discharging outside the Primary Containment (Containment Bypass), including areas in the Reactor Building addressed in the EOPs.

Indication of continuing flow/leakage includes: flow indication through isolated lines, increasing Reactor Building area temperatures, area radiation levels, sump levels, or room levels in spaces associated with affected lines, as well as increases in Plant Vent Effluent levels.

The isolation valve status of all isolation groups is monitored for quick reference on SPDS, to be backed up by operator observation of valve status.

DEVIATION

NUMARC Primary Containment Barrier Example Flowchart (PC2) suggests that for the "Containment Isolation Valve Status after Containment Isolation Signal" EAL, a failure of both valves in any one line to close AND downstream pathway to the environment exists be included as a threshold for classification of an Unusual Event. In order to include the condition where the Inboard Valve fails to close and an RCS Line Break exists between the Primary Containment wall and Outboard Valve, the condition that both valves fail to close is NOT being included in the EAL. Indication of continuing flow / leakage OUTSIDE the Primary Containment will provide an adequate threshold for Event Classification, since both isolation valves must be open for continuing leakage Outside the Primary Containment, except as noted above.

REFERENCES

NUMARC NESP-007, PC2
10CFR50, App. A, GDC 55, 56, 57
10 CFR 100
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0116(Q), Containment Isolations and Recovery from an Isolation
HC.OP-AB.ZZ-0203(Q), Main Steam Line High Radiation
HC.OP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response
HC.OP-AR.ZZ-0011(Q), Annunciator Response Procedures, Window C6
HC.OP-AR.ZZ-0012(Q), Annunciator Response Procedures, Window C8
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103 (Q)-FC, Reactor Building Control
HCGS Technical Specifications LCO 3/4.3, Instrumentation
HCGS UFSAR Sections 6.2.4.3.1, 6.2.4.3.2, 6.2.4.3.3

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.5

IC Potential Loss or Loss of Containment Barrier

EAL

ANY condition, in the opinion of the EC, that indicates a Potential Loss (1 pt) or Loss (2 pts) of the Containment Barrier

OPERATIONAL CONDITION - 1, 2, 3

BASIS

This EAL allows the Emergency Coordinator to address any condition that effects the integrity of the Containment Barrier that is not already covered elsewhere in the Fission Product Barrier Table.
A complete loss of the ability to monitor the Containment Barrier should be considered as a "Potential Loss" of that barrier.

Barrier Analysis

Containment Barrier has been potentially lost or lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional barriers per EAL section 3.0.

DISCUSSION

None

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC6

4.0 EC Discretion

4.1 Emergency Coordinator Discretion

UNUSUAL EVENT - 4.1.1

IC Other Conditions Exist Which In the Judgement of the Emergency Coordinator Warrant Declaration of an Unusual Event

EAL

Events are in progress or have occurred which, in the judgement of the Emergency Coordinator, indicate a **Potential Degradation of Plant Safety**

OPERATIONAL CONDITION - All

BASIS

Emergency Coodrdinator judgement to declare an Unusual Event , based on the determination that the **Potential Degradation of Plant Safety** exists, should be implemented ONLY when conditions are not explicitly addressed elsewhere in the ECG. The phrase **Potential Degradation of Plant Safety** is intended to apply to those conditions that include a likely or actual breakdown of event mitigating actions or that hinder plant personnel from safely operating the plant. The following examples are by no means all inclusive and are not intended to limit the discretion of the SNSS. Examples for consideration include the following:

- inadequate emergency response procedures
- failure or unavailability of emergency systems during an accident/transient condition
- insufficient availability of equipment or support personnel to deal with the ongoing or anticipated events
- aircraft crash on or near site
- explosions near site (within owner controlled area)

Barrier Analysis

Additional guidance on EC judgement for Fission Product Barriers is found on the Fission Product Barrier Table, Section 3.0.

ESCALATION CRITERIA

Emergency Coordinator Judgement

DISCUSSION

Dose consequences from an Unusual Event, if a radiological release is involved, would not require offsite response or field monitoring since any release at this level would be < 10 mRem TEDE. Refer to Section 6 of the ECG if a Radiological release is ongoing.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU1.3, HU5, Section 3.7.

4.0 EC Discretion

4.1 Emergency Coordinator Discretion

ALERT - 4.1.2

IC Other Conditions Exist Which In the Judgement of the Emergency Coordinator Warrant Declaration of an Alert

EAL

Events are in progress or have occurred which, in the judgement of the Emergency Coordinator, indicate plant safety systems **(more than one)** are, or may be degraded **AND** Increased monitoring of plant functions is warranted

OPERATIONAL CONDITION - All

BASIS

Emergency Coordinator judgement to declare an Alert, based on the determination that Plant Systems are, or may be degraded, should be implemented ONLY when conditions are not explicitly addressed elsewhere in the ECG. This includes a determination by the SNSS that hazards exist that have, or may have caused damage to more than one safety system or to a plant vital structure. In addition, if plant conditions degrade to the point where increased monitoring of plant functions is warranted to better determine the plants actual safety status than an Alert classification may be appropriate.

Barrier Analysis

Additional guidance on EC judgement for Fission Product Barriers is found on the Fission Product Barrier Table, Section 3.0.

ESCALATION CRITERIA

Emergency Coordinator Judgement

DISCUSSION

Dose consequences for an Alert, if a radiological release was ongoing, would only be a small fraction of the EPA Protective action Guideline (PAG) plume exposure level, i.e., 10 to 100 mRem TEDE. Refer to ECG Section 6 if a radiological release is ongoing.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA6, HA1.4, Section 3.7.
EPA-400

4.0 EC Discretion

4.1 Emergency Coordinator Discretion

SITE AREA EMERGENCY - 4.1.3

IC Other Conditions Exist Which In the Judgement of the Emergency Coordinator Warrant Declaration of a Site Area Emergency

EAL

Events are in progress or have occurred which, in the judgement of the Emergency Coordinator, indicate EITHER one of the following: • The Potential for an uncontrolled radiological release or the source term available in the Containment atmosphere could result in Site Boundary Dose rates in **excess of 100 mRem/hr** • Criteria for declaration of a Site Area Emergency per the ECG Introduction Section exists

OPERATIONAL CONDITION - All

BASIS

Emergency Coordinator judgement to declare a Site Area Emergency, based on the determination that the potential exists for an uncontrolled radiological release or the source term available in the Containment atmosphere could result in Site Boundary dose rates in excess of 100 mRem/hr, should be implemented ONLY when conditions are not explicitly addressed elsewhere in the ECG. In addition, any criteria that satisifies the definition of a Site Area Emergency in the ECG Introduction Section, also warrants declaration under this EAL. A Site Area Emergency is intended to be anticipatory of potential fission product barrier failure, and allows offsite agencies to commence preparation for emergency response.

Barrier Analysis

Additional guidance on EC judgement for Fission Product Barriers is found on the Fission Product Barrier Table, Section 3.

ESCALATION CRITERIA

Emergency Coordinator Judgement

DISCUSSION

Radiological release rates during a Site Area Emergency declaration are not expected to result in exposure levels which exceed the EPA Protective Action Guideline threshold values except within the Site Boundary. However, plume exposure levels of 100 to < 1000 mRem TEDE may be possible offsite and levels >1000 mRem TEDE could be experienced onsite. Refer to ECG Section 6 if a radiological release is ongoing.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HS3, Section 3.7.
EPA-400

4.0 EC Discretion

4.1 Emergency Coordinator Discretion

GENERAL EMERGENCY - 4.1.4

IC Other Conditions Exist Which In the Judgement of the Emergency Coordinator Warrant Declaration of a General Emergency

EAL

Events are in progress or have occurred which, in the judgement of the Emergency Coordinator, indicateEither one of the following:

- The Potential for an uncontrolled radiological release is expected to **exceed** Protective Action Guideline levels per EAL 6.1.4.a
- Criteria for declaration of a General Emergency per the ECG Introduction Section exists

OPERATIONAL CONDITION - All

BASIS

Emergency Coordinator judgement to declare a General Emergency , based on the determination that the potential for an uncontrolled radionuclide release exists, should be implemented ONLY when conditions are not explicitly addressed elsewhere in the ECG. In addition, any criteria that satisifies the definition of a General Emergency in the ECG Introduction Section, also warrants declaration under this EAL. A General Emergency is intended to be anticipatory of fission product barrier failure, and permits maximum offsite intervention time.

Barrier Analysis

This EAL is intended for EC judgement for declaration at the General Emergency level. Additional guidance on EC judgement for Fission Product Barriers is found on the Fission Product Barrier Table, Section 3.0.

ESCALATION CRITERIA

N/A

DISCUSSION

Radiological release rates during a General Emergency may exceed the EPA Protective Action Guidelines, i.e., >1000mRem TEDE, for more than the immediate site area. ECG Section 6, Radiological Releases/Occurrences should be consulted for releases of this magnitude.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HG2, Section 3.7.
EPA-400

5.0 Failure to Scram

5.1 ATWS

ALERT - 5.1.2.a / 5.1.2.b

IC Failure of the Reactor Protection System (RPS) to Successfully Complete a Reactor Scram (Automatic or Manual)

EAL

<u>EITHER</u> one of the following conditions:

- An Automatic Reactor Scram Condition exists **<u>AND</u>** An Automatic Reactor Scram (RPS) <u>IS NOT</u> **successful**
- <u>ANY</u> Manually Initiated Reactor Scram (RPS) from the Control Room <u>IS NOT</u> **successful**

OPERATIONAL CONDITION - 1, 2

BASIS

Failure of the RPS to **successfully** complete a Reactor Scram (automatic or manual) represents a significant degradation in plant safety, as the primary reactivity control system has failed to perform its design function. The intent of this EAL is to classify events in which <u>either</u> an automatic <u>or</u> manual RPS signal fails to initially complete a **successful** scram when required, even if a subsequent manual or automatic scram is **successful** . The failure of RPS to complete a **successful** scram, is the bases for Alert declaration under this EAL. A **Successful** scram (RPS automatic or RPS manual), as it relates to this EAL, results in a Control Rod configuration by which the Reactor will remain shutdown under all conditions without boron. The three criteria that satisfy this condition are :

1) All Control Rods are inserted to position 02 or beyond (Maximum Subcritical Banked Withdrawal Position)
2) All Control Rods but one being full inserted.
3) Reactor Engineering has determined that the Reactor will remain Shutdown under all conditions without Boron

In addition, for a manual scram to be considered successful, it must be attempted from the Reactor Control console. In the event that ARI completes a **successful** Scram following a failure of automatic or manual RPS, the declaration of an Alert is still warranted, due to the failure of RPS. An inability to physically place the Reactor Mode Switch in the SHUTDOWN position, (i.e. broken key) does not consitute an RPS failure, since the RPS logic has not failed.

Barrier Analysis

This event does not reach the threshold for the loss of Fuel Clad or RCS Barriers, but conditions exist that could lead to a potential loss of those barriers.

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency when a failure of both automatic or manual scram functions occurs, with Reactor power remaining ≥ 4%.

DISCUSSION

The Reactor Protection System (RPS) is designed to function to shut down the reactor (either manually or automatically). The system is "fail safe", that is it deenergizes to function. An Anticipated Transient Without Scram (ATWS) event can be caused either by a failure of RPS (electrical/pneumatic failure) or a failure of the Control Rod Drive system to permit the control rods to insert (hydraulic failure).

The Alternate Rod Insertion (ARI) function of the Redundant Reactivity Control System (RRCS) provides an automatic backup function for an electrical/pneumatic failure of the RPS. A successful scram due to ARI following a failure of the RPS would still be classified under this EAL because of the potentially serious consequences of an RPS failure.

Confirmation indications of an RPS failure to complete a successful scram include control room annunciators, control rod positions, APRM power and downscale indicating lights, IRM/SRM power level, SRM period, and control rod position indication.

A manual scram is defined as any set of actions by the reactor operator(s) at the reactor controls which causes control rods to be rapidly inserted into the core via the RPS in an attempt to place the reactor in a subcritical condition (i.e. mode switch to shutdown, manual scram push buttons). This EAL addresses only those manual scram attempts that are initiated from the Control Room control panels.

A failure of the Reactor Protection System (RPS) to initiate and complete a reactor scram can result in the design limits of the nuclear fuel being compromised. RPS is designed to automatically detect and generate a reactor scram signal when a Technical Specification Limiting Safety System Setting (LSSS) is reached or exceeded. If an LSSS is exceeded without an automatic scram, consideration must be given to the possibility that a Technical Specification Safety Limit may have been exceeded.

DEVIATION

NUMARC EAL SA2 suggests that an Alert classification be based only on a failure of an automatic RPS scram followed by a successful manual RPS scram from the control room, with EAL SS2 escalating to a Site Area Emergency if a manual scram (RPS or ARI) fails to reduce Reactor Power below 4%.

The Alert threshold is set so that unsuccessful manual RPS scrams from the control rooom, as well as unsuccessful automatic RPS scrams via RPS would be classified at the Alert level. This will cover those situations in which a manual RPS scram is attempted in anticipation of a continually degrading plant condition (i.e degrading Main Condenser Vacuum). In addition, this threshold will also address those situations where a manual scram is required by procedure. (i.e. stuck open SRV, Main Steam Line Hi Hi Radiation, Dual Reactor Recirc Pump trip, Power Oscillations) and the manual scram is not successful. In either case, Alert declaration is appropriate when the RPS fails to perform its intended design function.

The SAE threshold is set to include automatic and manual failure (for the reasons stated above), with resulting power ≥ 4% as suggested in NUMARC EAL SS2 bases.

By defining a "Successful" scram as control rod being positioned such that the Reactor will remain Shutdown under all conditions, partial scrams that result in Reactor Power below 4% would be classified as an Alert, whether automatically or manually initiated.

REFERENCES

NUMARC NESP-0007, SA2
NUMARC Questions and Answers, June 1993, "System Malfunctions Question #7"
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0207 (Q)-FC, Level/Power Control
BWR Owners Group Emergency Procedure Guidelines, Revision 4
HCGS Technical Specifications 1.0, Definitions; SL/LSSS 2.1/2.2; LCO 3/4.1, Reactivity Control Systems; LCO 3/4.3, Instrumentation

5.0 Failure to Scram

5.1 ATWS

SITE AREA EMERGENCY - 5.1.3

IC Failure of the Reactor Protection System (RPS) to Successfully Complete a Reactor Scram (Automatic and Manual) and Reactor Power is above than 4%.

EAL

<u>EITHER</u> one of the following conditions:

- An Automatic Reactor Scram Condition exists **<u>AND</u>** An Automatic Reactor Scram (RPS) <u>IS NOT</u> **successful**
- <u>ANY</u> Manually Initiated Reactor Scram (RPS) from the Control Room <u>IS NOT</u> **successful**

<u>AND</u>

<u>ALL</u> Reactor Scram attempts from the Control Room (RPS <u>and</u> ARI) <u>DID NOT REDUCE and MAINTAIN</u> Reactor Power to **$\leq$ 4%**

OPERATIONAL CONDITION - 1, 2

BASIS

Failure of the RPS to **successfully** complete a Reactor Scram (automatic and manual) represents a significant degradation in plant safety, as the primary reactivity control system has failed to perform its design function. In addition, failure of subsequent Reactor Scram attempts (both RPS and ARI) to reduce Reactor Power to less than 4%, represents a potential challenge to the ability to provide continued heat removal from the Reactor. Thus, conditions exist that could lead to an imminent loss or potential loss of both the Fuel Clad and RCS Barriers. The intent of this EAL is to classify events in which <u>both</u> automatic <u>and</u> manual RPS signals fail to complete a **successful** scram when required, and subsequent actions using ARI fails to reduce Reactor Power to less than 4%. The failure of RPS and ARI to complete a **successful** scram with Reactor Power remaining above 4% is the bases for SAE declaration under this EAL. A **Successful** scram (RPS Automatic or Manual), as it relates to this EAL, results in a Control Rod configuration by which the Reactor will remain shutdown under all conditions without boron injection. The three criteria that satisify this condition are :

1) All Control Rods are inserted to position 02 or beyond (Maximum Subcritical Banked Withdrawal Position)
2) All Control Rods but one being full inserted.
3) Reactor Engineering has determined that the Reactor will remain Shutdown under all conditions without Boron

In addition, for a manual scram to be considered successful, it must be attempted from the Reactor control console. In the event that ARI completes a **successful** Scram following a failure of automatic or manual RPS, the declaration of an SAE is not warranted.

Barrier Analysis

This event does not reach the threshold for the loss of Fuel Clad or RCS Barriers, but conditions exist that could lead to an imminent loss or potential loss of those barriers.

ESCALATION CRITERIA

Emergency Classification will escalate to a General Emergency when Reactor Water Level can not be maintained > -190", or Suppression Pool Temperature and Reactor Pressure can not be maintained below the HCTL.

DISCUSSION

The Reactor Protection System (RPS) is designed to function to shut down the reactor (either manually or automatically). The system is "fail safe", that is it deenergizes to function. An Anticipated Transient Without Scram (ATWS) event can be caused either by a failure of RPS (electrical/pneumatic failure) or a failure of the Control Rod Drive system to permit the control rods to insert (hydraulic failure).

The Alternate Rod Insertion (ARI) function of the Redundant Reactivity Control System (RRCS) provides an automatic backup function for an electrical/pneumatic failure of the RPS. A failure of ARI to reduce Reactor Power to ≤ 4% following a failure of the RPS is classified under this EAL because of the potentially serious consequences of a failure of RPS and ARI to reduce Reactor Power.

Confirmation indications of an RPS failure to complete a successful scram include control room annunciators, control rod positions, APRM power and downscale indicating lights, IRM/SRM power level, SRM period, and control rod position indication.

A failure of the Reactor Protection System (RPS) to initiate and complete a reactor scram can result in the design limits of the nuclear fuel being compromised. RPS is designed to automatically detect and generate a reactor scram signal when a Technical Specification Limiting Safety System Setting (LSSS) is reached or exceeded. If an LSSS is exceeded without an automatic scram, consideration must be given to the possibility that a Technical Specification Safety Limit may have been exceeded.

Emergency Operating Procedures (EOPs) establish Reactor Power > 4% coincident with a scram condition as the initiating condition for various actions in response to an ATWS. If the Reactor is isolated (MSIVs closed), the heat generated is transferred to the Primary Containment, thus potentially threatening the integrity of Primary Containment . In an attempt to preclude this condition, EOP guidance includes restoration of the Main Condenser as a heat sink, provided there is no indication of gross fuel failure or a main steam line break. EOP guidance also includes methods of alternate reactivity control, including the use of Standby Liquid Control (SLC), alternate control rod insertion, and intentional lowering of Reactor Water Level to control Reactor Power.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SS2
NUMARC Questions and Answers, June 1993, "System Malfunctions Question #7"
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0207 (Q)-FC, Level/Power Control
BWR Owners Group Emergency Procedure Guidelines, Revision 4
HCGS Technical Specifications 1.0, Definitions; SL/LSSS 2.1/2.2; LCO 3/4.1, Reactivity Control Systems; LCO 3/4.3, Instrumentation

5.0 Failure to Scram

5.1 ATWS

GENERAL EMERGENCY - 5.1.4

IC Failure of the Reactor Protection System (RPS) to Successfully Complete a Reactor Scram (Automatic and Manual) and there is indication of an Extreme Challenge to the Ability to Cool the Core

EAL

EITHER one of the following conditions:

- An Automatic Reactor Scram Condition exists **AND** An Automatic Reactor Scram (RPS) IS NOT **successful**
- ANY Manually Initiated Reactor Scram (RPS) from the Control Room IS NOT **successful**

AND

ALL Reactor Scram attempts from the Control Room (RPS and ARI) DID NOT REDUCE and MAINTAIN Reactor Power to ≤ **4%**

AND

EITHER one of the following:

- Reactor Water Level CANNOT BE MAINTAINED **> -190"**
- The combination of Suppression Pool Temperature and RPV Pressure CANNOT BE MAINTAINED below the HCTL Curve

OPERATIONAL CONDITION - 1, 2

BASIS

Failure of the RPS to **successfully** complete a Reactor Scram (Automatic and Manual) represents a significant degradation in plant safety, as the primary reactivity control system has failed to perform its design function. In addition, failure of subsequent scram attempts (ARI) to reduce Reactor Power to less than 4%, resulting in an inability to MAINTAIN Reactor Water Level above -190" or Suppression Pool Temperature and Reactor Pressure below the Heat Capacity Temperature Limit (HCTL), represents an imminent loss or potential loss of all three fission product barriers. The inabiltiy to MAINTAIN Reactor Water Level above -190" was choosen based on the condition that core cooling is extremely challenged. This threshold corresponds directly to a decision step contained in EOP 207, Level /Power Control (Step LP-18), which requires a determination be made if Reactor Water Level can be MAINTAINED above -190". For cases where Reactor Water Level CAN NOT BE MAINTAINED > -190", a General Emergency declaration is warranted. The intent of this EAL is to classify those ATWS events that result in a challenge to the integrity of these barriers.

A **Successful** scram (RPS Automatic and Manual), as it relates to this EAL, results in a Control Rod configuration by which the Reactor will remain shutdown under all conditions without boron. The three criteria that satisify this condition are :

1) All Control Rods are inserted to position 02 or beyond (Maximum Subcritical Banked Withdrawal Position)
2) All Control Rods but one being full inserted.
3) Reactor Engineering has determined that the Reactor will remain Shutdown under all conditions without Boron

Barrier Analysis

This event reaches the threshold for either a loss or potential loss of all three Fission Product Barriers.

ESCALATION CRITERIA

N/A

DISCUSSION

The Reactor Protection System (RPS) is designed to function to shut down the reactor (either manually or automatically). The system is "fail safe", that is it deenergizes to function. An Anticipated Transient Without Scram (ATWS) event can be caused either by a failure of RPS (electrical/pneumatic failure) or a failure of the Control Rod Drive system to permit the control rods to insert (hydraulic failure). The Alternate Rod Insertion (ARI) function of the Redundant Reactivity Control System (RRCS) provides an automatic backup function for an electrical/pneumatic failure of the RPS.

Confirmation indications of an RPS failure to complete a successful scram include control room annunciators, control rod positions, APRM power and downscale indicating lights, IRM/SRM power level, SRM period, and control rod position indication.

A failure of the Reactor Protection System (RPS) to initiate and complete a reactor scram can result in the design limits of the nuclear fuel being compromised. RPS is designed to automatically detect and generate a reactor scram signal when a Technical Specification Limiting Safety System Setting (LSSS) is reached or exceeded. If an LSSS is exceeded without an automatic scram, consideration must be given to the possibility that a Technical Specification Safety Limit may have been exceeded.

Emergency Operating Procedures (EOPs) establish Reactor Power > 4% coincident with a scram condition as the initiating condition for various actions in response to an ATWS. If the Reactor is isolated (MSIVs closed), the heat generated is transferred to the Primary Containment, thus potentially threatening the integrity of Primary Containment. In an attempt to preclude this condition, EOP guidance includes restoration of the Main Condenser as a heat sink, provided there is no indication of gross fuel failure or a main steam line break. EOP guidance also includes methods of alternate reactivity control, including the use of Standby Liquid Control (SLC), alternate control rod insertion, and intentional lowering of Reactor Water Level to control Reactor Power.

During these actions, adequate core cooling is accomplished by maintaining Reactor Water Level above -190". Although this is below the Top of Active Fuel (Loss of Core Submergence), maintaining Reactor Water Level above -190" will ensure sufficient steam flow from the covered portion of the core to preclude Fuel Clad Temperatures in the uncovered portion of the core from exceeding 1500 Degrees F. This is refered to as the Minimum Steam Cooling RPV Water Level. Inability to maintain this level may result in damage to the fuel.

The EOPs require the initiation of SLC before Suppression Pool Temperature reaches 110 Degrees F. This threshold is refered to as the Boron Injection Initiation Temperature, and is defined as the highest Suppression Pool Temperature at which initiation of boron injection will result in injection of the Hot Shutdown Boron Weight before Suppression Pool Temperature exceeds the Heat Capacity Temperature Limit (HCTL).

Actions required by the EOPs when Reactor Water Level can not be maintained above -190" or the HCTL is exceeded include the initiation of Emergency Depressurization.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SG2
NUMARC Questions and Answers, June 1993, "System Malfunctions Question #7"
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0207 (Q)-FC, Level/Power Control
BWR Owners Group Emergency Procedure Guidelines, Revision 4
HCGS Technical Specifications 1.0, Definitions; SL/LSSS 2.1/2.2; LCO 3/4.1, Reactivity Control Systems; LCO 3/4.3, Instrumentation

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

UNUSUAL EVENT - 6.1.1.a

IC Any **Unplanned** Release of Gaseous Radioactivity to the Environment that Exceeds 2 Times the Radiological Technical Specifications for 60 minutes or longer

EAL

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

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

AND

Release is ongoing for ≥ **60 minutes**

OPERATIONAL CONDITION - All

BASIS

Dose Assessment at or beyond the MEA exceeding the EAL threshold, can result from a Gaseous Radiological Release in excess of 2 times Technical Specifications. This condition results from an uncontrolled release of radioactivity to the environment, resulting in elevated offsite dose rates. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude that was not isolated within 60 minutes. The final integrated dose is very low and is not the primary concern. Classification is based on an ongoing release that does not comply with a license condition. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

Dose Assessment using actual meteorological data provides an accurate indication of release magnitude. The use of dose assessment based EALs is therefore preferred over the use of Release Rate based EALs which utilize calculations which have built-in inaccuracies because ODCM default Meteorological data is used. As long as dose assessment is available, this EAL should be used in place of EAL 6.1.1.d.

It is not intended that the release be averaged over 60 minutes, but exceed 2 times the Technical Specification limit for 60 minutes or longer. In addition, it is intended that the event be declared as soon as it is determined that the release will exceed 2 times the limit for 60 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when the effluent release increases to 200 times the Technical Specification limit.

DISCUSSION

Prorating the 500 mRem/yr criterion for the TEDE 4-day dose: time (8766 hr/yr); the 2 x Tech. Spec. multiplier; and, Artificial Island's Allocation Factor of 0.5 (50% per site), the associated site boundary dose rate would be 0.057 mRem/hr.

$$\textit{TEDE 4-Day MEA Dose Rate} = \left(\frac{500 mRem/yr}{8766 hr/yr}\right)(2)(.5) = 0.057 mRem/hr$$

This is rounded to .05 mRem/hr.

The TEDE 4-day Dose is based on a 4 hour release duration. Therefore .05 mRem/hr*4 hours = 0.2 mRem.

Prorating the 1500 mRem/yr criterion for the Thyroid-CDE Dose: time (8766 hr/yr); the 2 x Tech. Spec. multiplier; and, Artificial Island's Allocation Factor of 0.5 (50% per site), the associated site boundary dose rate would be 0.17 mRem/hr.

$$\textit{Thyroid-CDE MEA Dose Rate} = \left(\frac{1500 \, mRem/yr}{8766 \, hr/yr}\right)(2)(.5) = 0.17 \, mRem/hr$$

The Thyroid-CDE Dose is based on a 4 hour release duration. Therefore 0.17 mRem/hr*4 hours = 0.68 mRem.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AU1.4
Off-Site Dose Calculation Manual, Section 2.0 - Gaseous Effluents
NUMARC Draft White Paper, 7-25-94, 9-10-94.

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

UNUSUAL EVENT - 6.1.1.b

IC Any **Unplanned** Release of Gaseous Radioactivity to the Environment that Exceeds 2 Times the Radiological Technical Specifications for 60 minutes or longer

EAL

Dose Rate measured at the Protected Area Boundary or beyond EXCEEDS **.05 mRem/hr** above normal background **AND** Release is ongoing for $\geq$ **60 minutes**

OPERATIONAL CONDITION - All

BASIS

Measured Dose Rate at or beyond the Protected Area Boundary exceeding the EAL threshold can result from a Gaseous Radiological Release in excess of 2 times Technical Specifications. This condition results from an uncontrolled release of radioactivity to the environment, resulting in elevated offsite dose rates. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude that was not isolated within 60 minutes. The final integrated dose is very low and is not the primary concern. Classification is based on an ongoing release that does not comply with a license condition. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

It is not intended that the release be averaged over 60 minutes, but exceed 2 times Tech. Spec. limits for 60 minutes or longer. Further, it is intended that the event be declared as soon as it is determined that the release will exceed 2 times the limit for 60 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when effluent release increases to 200 times the Technical Specification limit.

DISCUSSION

Prorating the 500 mRem/yr criterion for: time (8766 hr/yr); the 2 x Tech. Spec. multiplier; and, Artificial Island's Allocation Factor of 0.5 (50% per site), the associated site boundary (MEA) dose rate would be 0.057 mRem/hr.

Protected Area Boundary Dose Rate =

$$=\left(\frac{500\ mRem/yr}{8766\ hr/yr}\right)(2)(.5)=0.057\ mRem/hr$$

This is rounded to .05 mRem/hr

DEVIATION

None

REFERENCES

NUMARC NESP-007, AU1.3
Off-Site Dose Calculation Manual, Section 2.0 - Gaseous Effluents
NUMARC Draft White Paper, 7-25-94, 9-10-94.

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

UNUSUAL EVENT - 6.1.1.c

IC Any **Unplanned** Release of Gaseous Radioactivity to the Environment that Exceeds 2 Times the Radiological Technical Specifications for 60 minutes or longer

EAL

Total gaseous effluent release sample analysis for <u>ANY</u> one of the following indicates a concentration of:

- **FRVS:**
  ≥ **5.65E-03 µCi/cc Total Noble Gas**
  ≥ **8.00E-06 µCi/cc I-131**
- **NPV:**
  ≥ **1.21E-03 µCi/cc Total Noble Gas**
  ≥ **1.72E-06 µCi/cc I-131**
- **SPV:**
  ≥ **1.13E-04 µCi/cc Total Noble Gas**
  ≥ **1.61E-07 µCi/cc I-131**

AND

Release is ongoing for ≥ **60 minutes**

OPERATIONAL CONDITION - All

BASIS

Total gaseous effluent release sample analysis exceeding the EAL threshold for any of the plant vents listed (FRVS, NPV, SPV), can result from a Gaseous Radiological Release in excess of 2 times Technical Specifications. This condition results from an uncontrolled release of radioactivity to the environment, resulting in elevated offsite dose rates. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude that was not isolated within 60 minutes. The final integrated dose is very low and is not the primary concern. Classification is based on an ongoing release that does not <u>comply</u> with a license condition. The HTV is not included under this EAL since there are no provisions for collecting a HTV grab sample. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

It is not intended that the release be averaged over 60 minutes, but <u>exceed</u> 2 times the Technical Specification limit for 60 minutes or longer. In addition, it is intended that the event be declared as soon as it is determined that the release will exceed 2 times the limit for 60 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when the effluent release increases to 200 times the Technical Specification limit.

DISCUSSION

Calculation of the threshold sample concentrations are as follows:

FRVS Noble Gas Sample Concentration = $\dfrac{2.40E{+}04\ \mu Ci/sec}{472\ x9000\ cfm} = 5.65E{-}03\ \mu Ci/cc$

FRVS I-131 Sample Concentration = $\dfrac{3.40E{+}01\ \mu Ci/sec}{472\ x9000\ cfm} = 8.00E{-}06\ \mu Ci/cc$

NPV NobleGas Sample Concentration= $\dfrac{2.40E{+}04\ \mu Ci/sec}{472\ x4.19E{+}4\ cfm} = 1.21E{-}03\ \mu Ci/cc$

NPV I-131 Sample Concentration = $\dfrac{3.40E{+}01\ \mu Ci/sec}{472\ x4.19E{+}04\ cfm} = 1.72E{-}06\ \mu Ci/cc$

SPV Noble Gas Sample Concentration = $\dfrac{2.40E{+}04\ \mu Ci/sec}{472\ x4.48E{+}5\ cfm} = 1.13E{-}04\ \mu Ci/cc$

SPV I-131 Sample Concentration = $\dfrac{3.40E{+}01\ \mu Ci/sec}{472\ x4.48E{+}05\ cfm} = 1.61E{-}07\ \mu Ci/cc$

Where: 472 = conversion factor (28,317 cc/ft^3 x 1 min./60 sec.)
9000 cfm = FRVS Vent Flow (maximum)
4.19E+04 cfm = NPV Vent Flow (maximum)
4.48E+05 cfm = SPV Vent Flow (maximum)
The noble gas release rate of 2.40E+04 μCi/sec is obtained by multiplying the Technical Specification release rate of 1.20E+04 μCi/sec times 2.
The iodine release rate of 3.40E+01 μCi/sec is obtained by multiplying the Technical Specification release rate of 1.70E+01 μCi/sec times 2.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AU1.2
Off-Site Dose Calculation Manual, Section 2.0
NUMARC Draft White Paper, 7-25-94, 9-10-94.

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

UNUSUAL EVENT - 6.1.1.d

IC Any **Unplanned** Release of Gaseous Radioactivity to the Environment that Exceeds 2 Times the Radiological Technical Specifications for 60 minutes or longer

EAL

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 1/3; 9RX518)

AND

Total Plant Vent release rate EXCEEDS one of the following limits:

- **2.40E+04 μCi/sec Total Noble Gas**
- **3.40E+01 μCi/sec I-131** (USE FOR NPV & SPV ONLY)

AND

Dose Assessment is NOT available

AND

Release is ongoing for ≥ **60 minutes**

OPERATIONAL CONDITION - All

BASIS

Valid High alarm and effluent release rate values exceeding the EAL threshold, can result from a Gaseous Radiological Release in excess of 2 times Technical Specifications. This condition results from an uncontrolled release of radioactivity to the environment, resulting in elevated offsite dose rates. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude that was not isolated within 60 minutes. The final integrated dose is very low and is not the primary concern. **Valid** is defined as the High alarm actuating specifically due to a Gaseous Release exceeding Technical Specification limits, thus precluding unwarranted event declaration as the result of spurious actuation. Classification is based on an ongoing release that does not comply with a license condition. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit

The EAL value for Total Plant Vent release rate was determined using default X/Q values from the ODCM which provides a less accurate method of evaluation release magnitude then using dose assessment with real time meteorological data. For that reason, this EAL should not be utilized if Dose Assessment is available. Dose Assessment will take in account actual meteorological conditions, plant vent flows and plant vent effluent concentrations to provide a more accurate assessment of a radiological release. If Dose Assessment is available than refer to EAL 6.1.1.a for classification.

It is not intended that the release be averaged over 60 minutes, but exceed 2 times Technical Specification limits for 60 minutes or longer. In addition, it is intended that the event be declared as soon as it is determined that the release will exceed 2 times the limit for 60 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will be escalate to an Alert when the effluent release increases to 200 times the Technical Specification limits.

DISCUSSION

The release rate thresholds for this EAL are obtained by multiplying the Technical Specification release rates of 1.2E+04 µCi/sec and 1.70E+01 µCi/sec, for Noble Gases and Iodine-131 respectively, times 2. Total Noble Gas release rate is the summation of all plant vent release rates.

This EAL includes Iodine Release rates for the NPV and SPV, since these vents have an Iodine monitor. Determination of the Iodine Release Rate from the Iodine monitor is accomplished by multiplying the Iodine reading (in uCi/cc) by the applicable vent flow rate, and 472 (Conversation factor). Iodine Release rates for FRVS and the HTV are excluded since these vents do not include an Iodine detector. The SPDS Total Iodine Offsite Release Rate does not provide useful information because this is based on a default value of 1000 times less than the Total Noble Gas Offsite Release Rate, which could be grossly inaccurate.

Release rates for FRVS and the HTV are not included since these vents to not have an Iodine detector. A gaseous effluent sample is needed to accurately quantify the Iodine Release rate. The SPDS Total Iodine Offsite Release Rate should not be used, as this is based a default value of 1000 times less than the Total Noble Gas Offsite Release Rate. The Technical Specification limits are based on ODCM calculations.

Technical Specification Calculation for Noble Gas

$$\text{uCi/Second} = \frac{\text{500 mrem/year * (Allocation Factor)}}{\text{(ODCM X/Q) * (ODCM DRCF)}}$$

WHERE: **uCi/Second** = Total Noble Gas Release Rate from Salem (Unit 1 & Unit 2) or Hope Creek (all Vents; NPV, SPV, FRVS, and HTV) which would result in a TEDE Dose Rate of 250 mrem/year.

ODCM X/Q = Site Specific (Salem or Hope Creek) dispersion factor at the Site Boundary in sec/m^3.

ODCM DRCF = Site Specific (Salem or Hope Creek) dose rate conversion factor in mrem/year/uCi/m^3.

ODCM X/Q = 2.67E-06
ODCM DRCF = 7.80E+03 mrem/yr/uCi/m^3
Allocation Factor = 5.00E-01

$$\text{1.20E+04 uCi/Second} = \frac{\text{(500 mrem/year) * (5.00E-01)}}{\text{(2.67E-06 sec/m}^3\text{) * (7.80E+03 mrem/yr/uCi/m}^3\text{)}}$$

1.20E+04 uCi/Second is the Hope Creek Technical Specification value.

Technical Specification Calculation for Thyroid Committed Dose

$$\text{uCi/Second} = \frac{\text{1500 mrem/year * (Allocation Factor)}}{\text{(ODCM X/Q) * (ODCM THY DRCF)}}$$

WHERE: **uCi/Second** = Total Iodine 131 release rate from Salem (Unit 1 or 2) or Hope Creek (all Vents; NPV, SPV, FRVS and HTV).

ODCM X/Q = Site Specific (Salem or Hope Creek) dispersion factor at the Site Boundary in sec/m^3.

ODCM DRCF = is the most limiting potential pathway (inhalation, child, thyroid I-131) dose rate conversion factor in mrem/year/uCi/m^3.

ODCM X/Q = 2.67E-06
ODCM DRCF THY = 1.62E+07 mrem/yr/uCi/m^3
Allocation Factor = 5.00E-01

$$\text{1.73E+01 uCi/Second} = \frac{\text{(1500 mrem/year) * (5.00E-01)}}{\text{(2.67E-06 sec/m}^3\text{)* (1.62E+07 mrem/yr/uCi/m}^3\text{)}}$$

1.73E+01 uCi/Second is the Hope Creek Technical Specification value.

DEVIATION

None

REFERENCES

NUMARC NESP-007,AU1.1, AU1.4
HC.OP-AB.ZZ-126(Q), Abnormal Releases of Gaseous Radioactivity
HC.RP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response
Off-Site Dose Calculation Manual, Section 2.0 - Gaseous Effluents
NUMARC Draft White Paper, 7-25-94, 9-10-94.

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

ALERT - 6.1.2.a

IC Any **Unplanned** Release of Gaseous Radioactivity to the Environment that exceeds 200 Times Radiological Technical Specifications for 15 minutes or longer

EAL

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

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

AND

Release is ongoing for ≥ **15 minutes**

OPERATIONAL CONDITION - All

BASIS

Dose Assessment at or beyond the MEA exceeding the EAL threshold , can result from a Gaseous Radiological Release in excess of 200 times Technical Specifications.This condition results from an uncontrolled release of radioactivity to the environment, resulting in significantly elevated offsite dose rates. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude that was not isolated within 15 minutes.. Classification is based on an ongoing release that does not comply with a license condition. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit

Dose Assessment using actual meteorological data provides an accurate indication of release magnitude. The use of dose assessment based EALs is therefore preferred over the use of Release Rate based EALs which utilize calculations which have built-in inaccuracies because ODCM default Meteorological data is used. As long as dose assessment is available, this EAL should be used in place of EAL 6.1.2.d.

It is not intended that the release be averaged over 15 minutes, but exceed 200 times the Technical Specification limit for 15 minutes or longer. In addition, it is intended that the event be declared as soon as it is determined that the release will exceed 200 times the limit for 15 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency when the effluent release increases to a level that would cause a 100 mrem dose at the Protected Area Boundary.

DISCUSSION

Prorating the 500 mRem/yr criterion for the TEDE 4-day dose: time (8766 hr/yr); the 200 x Tech. Spec. multiplier; and, Artificial Island's Allocation Factor of 0.5 (50% per site), the associated site boundary dose rate would be 5.7 mRem/hr.

$$\textit{TEDE 4-Day MEA Dose Rate} = \left(\frac{500 mRem/yr}{8766 hr/yr}\right)(200)(.5) = 5.7 mRem/hr$$

This is rounded to 5.0 mRem/hr.

The TEDE 4-day Dose is based on a default (assumed) 4 hour release duration. Therefore 5.0 mRem/hr*4 hours = 20 mRem.

Prorating the 1500 mRem/yr criterion for the Thyroid-CDE Dose: time (8766 hr/yr); the 200 x Tech. Spec. multiplier; and, Artificial Island's Allocation Factor of 0.5 (50% per site), the associated site boundary dose rate would be 17 mRem/hr.

$$\textit{Thyroid-CDE MEA Dose Rate} = \left(\frac{1500 mRem/yr}{8766 hr/yr}\right)(200)(.5) = 0.17 mRem/hr$$

The Thyroid-CDE Dose is based on a 4 hour release duration. Therefore 17 mRem/hr*4 hours = 68 mRem.

DEVIATION

None

REFERENCES

NUMARC NESP-007,AA1.4
Off-Site Dose Calculation Manual, Section 2.0 - Gaseous Effluents
NUMARC Draft White Paper, 7-25-95, 9-10-94.

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

ALERT - 6.1.2.b

IC Any **Unplanned** Release of Gaseous Radioactivity to the Environment that exceeds 200 Times Radiological Technical Specifications for 15 minutes or longer

EAL

Dose Rate measured at the Protected Area Boundary or beyond EXCEEDS **5 mRem/hr**
AND
Release is ongoing for ≥ **15 minutes**

OPERATIONAL CONDITION - All

BASIS

Measured Dose Rates at or beyond the MEA exceeding the EAL threshold , can result from a Gaseous Radiological Release in excess of 200 times Technical Specifications.This condition results from an uncontrolled release of radioactivity to the environment, resulting in significantly elevated offsite dose rates. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude that was not isolated within 15 minutes.. Classification is based on an ongoing release that does not comply with a license condition. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

It is not intended that the release be averaged over 15 minutes, but exceed 200 times the Technical Specification limit for 15 minutes or longer. In addition, it is intended that the event be declared as soon as it is determined that the release will exceed 200 times the limit for 15 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency when effluents increase to a level that would cause a 100 mrem dose at the Protected Area Boundary.

DISCUSSION

Prorating the 500 mRem/yr criterion for: time (8766 hr/yr); the 200 x Tech. Spec. multiplier; and, Artificial Island's Allocation Factor of 0.5 (50% per site), the associated site boundary dose rate would be 5.7 mRem/hr.

Protected Area Boundary Dose Rate = $$=\left(\frac{500\ mRem/yr}{8766\ hr/yr}\right)(200)(.5)=5.7\ mRem/hr$$

This is rounded to 5.0 mRem/hr

DEVIATION

None

REFERENCES

NUMARC NESP-007,AA1.3
Off-Site Dose Calculation Manual, Section 2.0 - Gaseous Effluents
NUMARC Draft White Paper, 7-25-94, 9-10-94.

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

ALERT - 6.1.2.c

IC Any **Unplanned** Release of Gaseous Radioactivity to the Environment that exceeds 200 Times Radiological Technical Specifications for 15 minutes or longer

EAL

Total gaseous effluent release sample analysis for ANY of the following indicates a concentration of:

- **FRVS:**
  ≥ 5.65E-01 µCi/cc Total Noble Gas
  ≥ 8.00E-04 µCi/cc I-131

- **NPV:**
  ≥ 1.21E-01 µCi/cc Total Noble Gas
  ≥ 1.72E-04 µCi/cc I-131

- **SPV:**
  ≥ 1.13E-02 µCi/cc Total Noble Gas
  ≥ 1.61E-05 µCi/cc I-131

AND

Release is ongoing for ≥ **15 minutes**

OPERATIONAL CONDITION - All

BASIS

Total gaseous effluent release sample analysis exceeding the EAL threshold for any of the plant vents listed (FRVS, NPV, SPV), can result from a Gaseous Radiological Release in excess of 200 times Technical Specifications. This condition results from an uncontrolled release of radioactivity to the environment, resulting in elevated offsite dose rates. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude that was not isolated within 15 minutes. The final integrated dose is very low and is not the primary concern. Classification is based on an ongoing release that does not comply with a license condition. The HTV is not included under this EAL since there are no provisions for collecting a HTV grab sample. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

It is not intended that the release be averaged over 15 minutes, but exceed 200 times the Technical Specification limit for 15 minutes or longer. In addition, it is intended that the event be declared as soon as it is determined that the release will exceed 200 times the limit for 15 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency when effluent release increases to a level that would cause a 100 mRem dose at the Protected Area Boundary.

DISCUSSION

Calculation of the threshold sample concentrations are as follows:

FRVS Noble Gas Sample Concentration = $\frac{2.40E{+}06\ \mu Ci/sec}{472\ x9000\ cfm} = 5.65E\text{-}01\ \mu Ci/cc$

FRVS I-131 Sample Concentration = $\frac{3.40E{+}03\ \mu Ci/sec}{472\ x9000\ cfm} = 8.00E\text{-}04\ \mu Ci/cc$

NPV Noble Gas Sample Concentration = $\frac{2.40E{+}06\ \mu Ci/sec}{472\ x4.19E{+}4\ cfm} = 1.21E\text{-}01\ \mu Ci/cc$

NPV I-131 Sample Concentration = $\frac{3.40E{+}03\ \mu Ci/sec}{472\ x4.19E{+}04\ cfm} = 1.72E\text{-}04\ \mu Ci/cc$

SPV Noble Gas Sample Concentration = $\frac{2.40E{+}06\ \mu Ci/sec}{472\ x4.48E{+}5\ cfm} = 1.13E\text{-}02\ \mu Ci/cc$

SPV I-131 Sample Concentration = $\frac{3.40E{+}03\ \mu Ci/sec}{472\ x4.48E{+}05\ cfm} = 1.61E\text{-}05\ \mu Ci/cc$

Where: 472 = conversion factor (28,317 cc/ft^3 x 1 min./60 sec.)
9000 cfm = FRVS Vent Flow (maximum)
4.19E+04 cfm = NPV Vent Flow (maximum)
4.48E+05 cfm = SPV Vent Flow (maximum)
The noble gas release rate of 2.40E+06 μCi/sec is obtained by multiplying the Technical Specification release rate of 1.20E+04 μCi/sec times 200.
The iodine release rate of 3.40E+03 μCi/sec is obtained by multiplying the Technical Specification release rate of 1.70E+01 μCi/sec times 200.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA1.2
Off-Site Dose Calculation Manual, Section 2.0
NUMARC Draft White Paper, 7-25-94, 9-10-94.

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

ALERT - 6.1.2.d

IC Any **Unplanned** Release of Gaseous Radioactivity to the Environment that exceeds 200 Times Radiological Technical Specifications for 15 minutes or longer

EAL

Valid High Alarm received from ANY of the following 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 1/3; 9RX518)

AND

Total Plant Vent release rate EXCEEDS **2.40E+06 µCi/sec Total Noble Gas**

AND

Dose Assessment is NOT available

AND

Release is ongoing for ≥ **15 minutes**

OPERATIONAL CONDITION - All

BASIS

Valid High alarm and effluent release rate values exceeding the EAL threshold, can result from a Gaseous Radiological Release in excess of 200 times Technical Specifications. This condition results from an uncontrolled release of radioactivity to the environment, resulting in elevated offsite dose rates. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude that was not isolated within

15 minutes. The final integrated dose is very low and is not the primary concern. **Valid** is defined as the High alarm actuating specifically due to a Gaseous Release exceeding Technical Specification limits, thus precluding unwarranted event declaration as the result of spurious actuation. Classification is based on an ongoing release that does not comply with a license condition. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

The EAL value for Total Plant Vent release rate was determined using default X/Q values from the ODCM which provides a less accurate method of evaluation release magnitude then using dose assessment with real time meteorological data. For that reason, this EAL should not be utilized if Dose Assessment is available. Dose Assessment will take in account actual meteorological conditions, plant vent flows and plant vent effluent concentrations to provide a more accurate assessment of a radiological release. If Dose Assessment is available than refer to EAL 6.1.2.a for classification.

It is not intended that the release be averaged over 15 minutes, but exceed 200 times Technical Specification limits for 15 minutes or longer. In addition, it is intended that the event be declared as soon as it is determined that the release will exceed 200 times the limit for 15 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency when effluent release increases to a level that would cause a 100 mRem dose at the Protected Area Boundary

DISCUSSION

The release rate thresholds for this EAL are obtained by multiplying the Technical Specification release rates of 1.2E+04 µCi/sec for Noble Gases, times 200. Total Noble Gas release rate is the summation of all plant vent release rates.

This EAL does not utilize an Iodine Release rate because the corresponding Alert threshold for Iodine is above the upper range of the NPV and SPV Iodine monitoring channels. Iodine Release rates for FRVS and the HTV are excluded since these vents do not include an Iodine detector. A gaseous effluent sample is needed to accurately quantify the Iodine Release rate (Refer to EAL 6.1.2.c). The SPDS Total Iodine Offsite Release Rate does not provide useful information because this is based on a default value of 1000 times less than the Total Noble Gas Offsite Release Rate, which could be grossly inaccurate.

Technical Specification Calculation for Noble Gas

$$\text{uCi/Second} = \frac{\text{500 mrem/year * (Allocation Factor)}}{\text{(ODCM X/Q) * (ODCM DRCF)}}$$

WHERE: **uCi/Second** = Total Noble Gas Release Rate from Salem (Unit 1 & Unit 2) or Hope Creek (all Vents; NPV, SPV, FRVS, and HTV) which would result in a TEDE Dose Rate of 250 mrem/year.

ODCM X/Q = Site Specific (Salem or Hope Creek) dispersion factor at the Site Boundary in sec/m^3.

ODCM DRCF = Site Specific (Salem or Hope Creek) dose rate conversion factor in mrem/year/uCi/m^3.

ODCM X/Q = 2.67E-06
ODCM DRCF = 7.80E+03 mrem/yr/uCi/m^3
Allocation Factor = 5.00E-01

$$\text{1.20E+04 uCi/Second} = \frac{\text{(500 mrem/year) * (5.00E-01)}}{\text{(2.67E-06 sec/m}^3\text{) * (7.80E+03 mrem/yr/uCi/m}^3\text{)}}$$

1.20E+04 uCi/Second is the Hope Creek Technical Specification value.

DEVIATION

None

REFERENCES

NUMARC NESP-007,AA1.1, AA1.4
OP-AB.ZZ-126(Q), Abnormal Releases of Gaseous Radioactivity
Off-Site Dose Calculation Manual, Section 2.0 - Gaseous Effluents
NUMARC Draft White Paper, 7-25-94, 9-10-94.

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

SITE AREA EMERGENCY - 6.1.3.a

IC 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

EAL

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

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

OPERATIONAL CONDITION - All

BASIS

The TEDE 4-Day Dose of 100 mRem corresponds directly to the NUMARC dose of 100 mRem. The Thyroid-CDE Dose of 500 mRem corresponds directly to the NUMARC dose of 500 mRem.

Dose Assessment using actual meteorological data provides an accurate indication of release magnitude. The use of dose assessment based EALs is therefore preferred over the use of Release Rate based EALs which utilize calculations which have built-in inaccuracies because ODCM default Meteorological data is used. **Imminent** is defined as expected to occur within 2 hours.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification escalates to a General Emergency when actual or projected doses exceed EPA Protective Action Guidelines.

DISCUSSION

This value provides a desirable gradient (one order of magnitude) between the Site Area Emergency and General Emergency classifications. No site allocation factor (.5) is used in this calculation due to the assumption that releases of this magnitude will be from one site.

The dose projection code assumes a 4 hour release utilizing current 15 minute average release rate data. For the TEDE 4-Day Dose, 100 mRem/hr*4 hr = 400 mRem. For the Thyroid-CDE Dose, 500 mRem/hr*4 hr = 2000 mRem.

DEVIATION

NUMARC EAL AS1.1 (Classification based on noble gas release rate) is not desirable per the NUMARC Draft White Paper dated 7/25/94 and 9/10/94. The classification could be under-consevative if it were made on the basis of noble gas release rate. Since dose assessment would continue in either case and the classification escalated if necessary, the impact from not having this EAL would be a delay in reaching the appropriate classification. This delay was deemed to be acceptable since in significant release situations, the plant condition EALs should provide the anticipatory classifications necessary for the implementation of offsite protective measures.

REFERENCES

NUMARC NESP-007,AS1.3
EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents
NUMARC Draft White Paper, 7-25-94, 9-10-94

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

SITE AREA EMERGENCY - 6.1.3.b

IC 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

EAL

Dose Rate measured at the Protected Area Boundary or beyond EXCEEDS **100 mr/hr** **AND** Release is ongoing for ≥ **15 minutes**

OPERATIONAL CONDITION - All

BASIS

An actual dose rate of 100 mRem/hr which is expected to continue for ≥ 15 minutes indicates a substantial radiological release which could exceed the 10CFR20 average annual population exposure limit of 100 mRem TEDE, using the assumption of a one hour release duration. **Imminent** is defined as expected to occur within 2 hours..

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a General Emergency when actual or projected doses exceed EPA Protective Action Guidelines.

DISCUSSION

An actual dose of 100 mRem Total Effective Dose Equivalent (TEDE) is based on the 10CFR20 annual average population exposure limit. Unless otherwise indicated, the conversion from whole body dose to TEDE is 1:1. Measured dose rates will be taken at the Protected Area Boundary, and a ≥ 15 minute threshold will be applied to be conservative.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AS1.4
EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents
NUMARC Draft White Paper, 7-25-94, 9-10-94

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

SITE AREA EMERGENCY - 6.1.3.c

IC 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

EAL

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

- **≥ 5.24E+02 CCPM**
- **≥ 4.63E-07 μCi/cc I-131**

OPERATIONAL CONDITION - All

BASIS

The Corrected Counts per Minute (CCPM) value is based on reading(s) obtained using a radiation count rate meter such as a RM-14 or E-140N with an HP260 probe attached. The Iodine-131 field survey sample concentration threshold is based on I-131 dose conversion factors from EPA-400. The thresholds are based on a Thyroid-CDE dose rate of 500 mRem/hr thyroid for I-131. **Imminent** is defined as expected to occur within 2 hours.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a General Emergency when actual or projected doses exceed EPA Protective Action Guidelines.

DISCUSSION

The release sample concentration calculations are as follows.

The sample concentration is calculated using the I-131 Dose Conversion Factor from EPA-400:

Solving the following equation for µCi/cc:

$$mRem/hr = (\mu Ci/cc)(Dose\ Conversion\ Factor)$$

Then;

$$I\text{-}131\ Sample\ Concentration = \left(\frac{500 mRem/hr}{1.08E{+}09 mRem/\mu Ci/cc/hr}\right) = 4.63E\text{-}07 \mu Ci/cc$$

Where 1.08E+09mRem/µCi/cc/hr is the Dose Conversion Factor from EPA-400, Table 5-4 and includes the EPA-400 breathing rate.

The Corrected Counts per Minute reading is calculated using the I-131 Sample concentration, and factors for using an RM-14 or E-140N with an HP260 probe.

Solving the following equation for CCPM:

$$\mu Ci/cc = \frac{CCPM}{(Detector\ Efficiency)(Collection\ Efficiency)(Conversion\ Factor - DPM\ to\ \mu Ci)(Volume - ft^3)(Conversion\ Factor - cc\ to\ ft^3)}$$

Then;

$$CCPM = (4.63E\text{-}07 \mu Ci/cc)(0.9)(2.22E{+}06 DPM/\mu Ci)(2.00E\text{-}03 CCPM/DPM)(10 ft^3)(2.832E{+}04 cc/ft^3)$$

$$= 5.24E{+}02\ CCPM$$

Where:

2.00E-03 =	Detector Efficiency - CCPM/DPM
0.9 (or 90%) =	Collection Efficiency
2.22E+06 =	Conversion factor - DPM/µCi
10 ft^3 =	Volume
2.832E+04 =	Conversion factor - cc to ft^3
CCPM =	Corrected Counts per Minute using an RM-14 or E-140N with an HP260 probe.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AS1.4
EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents
FEMA REP-2, Rev. 1, 7/87, Guidance on Offsite Emergency Radiation Measurement Systems, Phase-1 Airborne Release
SORC Summary 07/10/89
RPCS Thyroid Dose Commitment Factor Paper (NRP-94-0557), 11/22/94

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

GENERAL EMERGENCY - 6.1.4.a

IC 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

EAL

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

- TEDE 4-Day Dose of ≥ **1.0E+03 mRem**
- Thyroid-CDE Dose of ≥ **5.0E+03 mRem**

OPERATIONAL CONDITION - All

BASIS

The TEDE 4-Day Dose of 1000 mRem corresponds directly to the NUMARC dose of 1000 mRem which exceeds EPA Protective Action Guideline criteria for a General Emergency.

The Thyroid-CDE Dose or 5000 mRem corresponds directly to the NUMARC dose of 5000 mRem which exceeds EPA Protective Action Guideline criteria for a General Emergency.

Imminent is defined as expected to occur within 2 hours.

Barrier Analysis

N/A

ESCALATION CRITERIA

N/A

DISCUSSION

No site allocation factor (.5) is used in this calculation due to the assumption that releases of this magnitude will be from one site.

DEVIATION

NUMARC EAL AG1.1 (Classification based on noble gas release rate) is not desirable per the NUMARC Draft White Paper dated 7/25/94 and 9/10/94. The classification could be under-consevative if it were made on the basis of noble gas release rate. Since dose assessment would continue in either case and the classification escalated if necessary, the impact from not having this EAL would be a delay in reaching the appropriate classification. This delay was deemed to be acceptable since in significant release situations, the plant condition EALs should provide the anticipatory classifications necessary for the implementation of offsite protective measures.

REFERENCES

NUMARC NESP-007, AG1.3
EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

GENERAL EMERGENCY - 6.1.4.b

IC 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

EAL

Dose Rate measured at the Protected Area Boundary or beyond <u>EXCEEDS</u> **1000 mRem/hr**
<u>AND</u>
Release is ongoing for ≥ **15 minutes**

OPERATIONAL CONDITION - All

BASIS

An actual dose rate of 1000 mRem/hr indicates the EPA Protective Action Guide may be exceeded for the general public. **Imminent** is defined as expected to occur within 2 hours.

Barrier Analysis

N/A

ESCALATION CRITERIA

N/A

DISCUSSION

An actual projected dose of 1000 mRem Total Effective Dose Equivalent (TEDE) is based on the EPA protective action guidance which indicates that public protective actions are indicated if the dose exceeds 1 Rem whole body. This is consistent with the emergency class description for a General Emergency. A release rate equivalent to 1000 mRem/hr boundary dose rate may also be used if TEDE projections are not available. Unless otherwise indicated, the conversion from whole body dose to TEDE is 1:1.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AG1.4
EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

6.0 Radiological Releases/Occurrences

6.1 Gaseous Effluent Release

GENERAL EMERGENCY - 6.1.4.c

IC 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

EAL

Analysis of field survey samples at the Protected Area Boundary indicates <u>EITHER</u> one of the following: • **≥ 5.24E+03 CCPM** • **≥ 4.63E-06 μCi/cc I-131**

OPERATIONAL CONDITION - All

BASIS

The Corrected Counts per Minute (CCPM) value is based on reading(s) obtained using a radiation count rate meter such as a RM-14 or E-140N with an HP260 probe attached. The Iodine-131 field survey sample concentration threshold is based on I-131 dose factors from EPA-400. The thresholds are based on a dose rate of 5000 mRem/hr Thyroid-CDE for I-131. **Imminent** is defined as expected to occur within 2 hours.

Barrier Analysis

N/A

ESCALATION CRITERIA

N/A

DISCUSSION

No site allocation factor (.5) is used in this calculation due to the assumption that releases of this magnitude will be from one site.

The release sample concentration calculations are as follows.

The sample concentration is calculated using the I-131 Dose Factor from EPA-400:

Solving the following equation for µCi/cc:

$$\text{mRem/hr} = (\mu\text{Ci/cc})(\text{Dose Conversion Factor})$$

Then;

I-131 Sample Concentration = $\left(\dfrac{5000\,mRem/hr}{1.08E{+}09\,mRem/\mu Ci/cc/hr}\right) = 4.63E{-}06\,\mu Ci/cc$

Where 1.08E+09 mRem/µCi/cc/hr is the Dose conversion factor from EPA-400, Table 5-4 and includes the EPA-400 breathing factor.

The Corrected Counts per Minute reading is calculated using the I-131 Sample concentration, and factors for using an RM-14 or E-140N with an HP260 probe.

Solving the following equation for CCPM:

$$\mu\text{Ci/cc} = \frac{\text{CCPM}}{(\text{Detector Efficiency})(\text{Collection Efficiency})(\text{Conversion Factor - DPM to }\mu\text{Ci})(\text{Volume - ft}^3)(\text{Conversion Factor - cc to ft}^3)}$$

Then;

CCPM = $(4.63\,\mu Ci/cc)(0.9)(2.22E{+}06\,DPM/\mu Ci)(2.00E{-}03\,CCPM/DPM)(10\,ft^3)(2.832E{+}04\,cc/ft^3)$

= 5.24E+03 CCPM

Where:

2.00E-03 =	Detector Efficiency - CCPM/DPM
0.9 (or 90%) =	Collection Efficiency
2.22E+06 =	Conversion factor - DPM/µCi
10 ft^3 =	Volume
2.832E+04 =	Conversion factor - cc to ft^3
CCPM =	Corrected Counts per Minute using an RM-14 or E-140N with an HP260 probe.

DEVIATION

None

REFERENCES

NUMARC NESP-007,AG1.4
EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents
FEMA REP-2, Rev. 1/July 1987, Guidance on Offsite Emergency Radiation Measurement Systems, Phase-1 Airborne Release
SORC Summary 07/10/89
RPCS Thyroid Dose Commitment Factor paper NRP-94-0557, 11-22-94

6.0 Radiological Releases/Occurrences

6.2 Liquid Effluent Release

UNUSUAL EVENT - 6.2.1

IC Any **Unplanned** Release of Liquid Radioactivity to the Environment that Exceeds 2 Times the Radiological Technical Specifications for 60 minutes or longer

EAL

Valid Cooling Tower Blowdown Effluent Radiation Monitor High Alarm Condition

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

OPERATIONAL CONDITION - All

BASIS

A **Valid** Cooling Tower Blowdown Effluent Radiation Monitor High alarm condition corresponds to the Technical Specification Liquid Effluent Release Limit . Despite this limit being below the EAL threshold, exceeding this limit with a failure to terminate the discharge may be a precursor to an Unplanned Liquid Radiological Release in excess of 2 times Technical Specifications that continues for greater than 60 minutes. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude, that is not isolated in 60 minutes. The final integrated dose is very low and is not the primary concern. **Valid** is defined as the Cooling Tower Blowdown Effluent Radiation Monitor High Alarm actuating specifically due to a Liquid Release exceeding the Technical Specification limit, thus precluding unwarranted event declaration as the result of spurious actuation. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

It is not intended that the release be averaged over 60 minutes, but exceed 2 times the Technical Specification limit for 60 minutes or longer. In addition, it is intended that the event be declared as soon as it is determined that the release will exceed 2 times the limit for 60 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when the Liquid Effluent Release exceeds 200 times Technical Specification limits.

DISCUSSION

The Cooling Tower Blowdown Effluent Radiation Monitor (9RX506) monitors radioactivity in the cooling tower blowdown before it is discharged into the Delaware River and warns personnel of an excessive amount of radioactivity (greater than Technical Specificatio limits) being released to the environment.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AU1.2
Off-Site Dose Calculation Manual, Section 1.0 - Liquid Effluents
Technical Specifications LCO 3.11.1.1
HC.RP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response

6.0 Radiological Releases/Occurrences

6.2 Liquid Effluent Release

ALERT - 6.2.2

IC Any **Unplanned** Release of Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Technical Specifications for 15 minutes or longer

EAL

Valid Cooling Tower Blowdown Effluent Radiation Monitor High Alarm Condition

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

OPERATIONAL CONDITION - All

BASIS

A **Valid** Cooling Tower Blowdown Effluent Radiation Monitor High alarm condition corresponds to the Technical Specification Liquid Effluent Release Limit. Despite this limit being well below the EAL threshold, exceeding this limit with a failure to terminate the discharge may be a precursor to an Unplanned Liquid Radiological Release in excess of 200 times Technical Specifications that continues for greater than 15 minutes. The threshold for this EAL is NOT based on a specific offsite dose rate, but rather on the loss of plant control implied by a radiological release of this magnitude, that is not isolated in 15 minutes. The release duration was reduced from 60 minutes (UE) to 15 minutes in recognition of the increased severity of a release of this magnitude. **Valid** is defined as the Cooling Tower Blowdown Effluent Radiation Monitor High Alarm actuating specifically due to a Liquid Release exceeding the Technical Specification limit, thus precluding unwarranted event declaration as the result of spurious actuation. **Unplanned** is defined as any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions on the applicable permit.

It is not intended that the release be averaged over 15 minutes, but exce: 200 times the Technical Specification limit for 15 minutes or longer. In addition, it is intended that the event be declared as soon as it is determined that the release will exceed 200 times the limit for 15 minutes or longer.

Barrier Analysis

N/A

ESCALATION CRITERIA

N/A

DISCUSSION

The Cooling Tower Blowdown Effluent Radiation Monitor (9RX506) monitors radioactivity in the cooling tower blowdown before it is discharged into the Delaware River and warns personnel of an excessive amount of radioactivity (greater than Technical Specificatio limits) being released to the environment.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA1.2
Off-Site Dose Calculation Manual, Section 1.0 - Liquid Effluents
HC.RP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

UNUSUAL EVENT - 6.3.1.a

IC Unplanned Increase in Plant Radiation

EAL

Unplanned increase 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

OPERATIONAL CONDITION - All

BASIS

An **Unplanned** increase in radiation levels within the Protected Area by a factor of 1000 times over normal represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant. **Unplanned** is defined as those events or conditions which are not associated with a planned evolution, such that radiation levels are increasing in an uncontrolled manner. This condtion specifically represents an uncontrolled increase in radiation levels within the Protected Area. Planned evolutions which cause elevated radiation levels do not warrant classification under this EAL.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert when radiation levels increase to a level that would impede access to areas required for the safe shutdown of the plant.

DISCUSSION

Normal level is considered as the highest reading in the past 24-hours excluding current peak values. RM-11 computer trends, RMS strip charts, and/or SPDS can be used to confirm these values.

Examples of a plannned evolution that results in increased radiation levels within the Protected Area include, but are not limited to:

- Radiography
- Lifting of the Reactor Vessel Moisture Seperator / Dryer during Refuel Operations
- Performance of a TIP trace
- Relocation of radioactive materials, including radioactive waste

DEVIATION

NUMARC IC AU2 includes unexpected increases in Airborne concentration in addition to plant radiation. The corresponding Hope Creek IC does not address Airborne concentration, since an increase in Airborne concentration is not addressed in the example EALs or the basis for the Unusual Event or Alert. Apparently, the Airborne concentration example EAL was deleted by NUMARC, but the corresponding IC was overlooked.

REFERENCES

NUMARC NESP-007, AU2.4

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

UNUSUAL EVENT - 6.3.1.b

IC Unplanned Increase in Plant Radiation

EAL

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

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

OPERATIONAL CONDITION - 5

BASIS

An **Uncontrolled** lowering of Reactor Cavity Level during Refueling (Operational Condition 5) represents a condition which can result in increased radiation levels, due to the loss of radiation shielding, if the Reactor Cavity level decrease can not be terminated. This event has a long lead time relative to potential for radiological release outside the site boundary, thus the impact to public health and safety is very low. **Uncontrolled** means that the level decrease can not be terminated.

Determination of an **uncontrolled** level decrease is made through either Visual Observation or indication in the Main Control Room. Visual Observation is the preferred method, whenever possible, however it is NOT intended that an individual must be dispatched for classification purposes, if the existing radiation level increase trend prevents personnel from accessing the Refuel Floor, or if cameras are available to remotely verify the condition. In the event visual observation is not available by any means, then Main Control Room indication should be used.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Calssification will escalate to an Alert as a result of uncovery of a fuel assembly and/or indication of high radiation levels on the refueling floor.

DISCUSSION

During Refueling operations, the RPV is flooded and RPV level indication is monitored on the shutdown instrument range. Limitations on evolutions on with a potential for draining the RPV are imposed when refueling is in progress. Lowering of RPV level may result in the loss of Shutdown Cooling if RPV level continues to lower unchecked. This may result in the loss of decay heat removal from the fuel contained in the RPV.

Technical Specifications requires at least 22 feet 2 inches of water be maintained over the top of the reactor pressure vessel flange while in Operating Condition 5 and either fuel assemblies are being handled or the fuel assemblies seated within the reactor vessel are irradiated. The Technical Specification minimum water level in the Reactor Vessel under these conditions is based on the minimum water level required to remove 99% of the assumed 10% iodine gap activity that would be released from the rupture of an irradiated fuel assembly.

DEVIATION

1) NUMARC states that this EAL will be applicable in all modes of operation. In other than Operational Condition 5, the RPV head will be fully tensioned, and lowering of vessel level would be classified by EALs in Section 3.0, Fission Product Barriers, or Section 8.1, Loss of Heat Removal Capability.

2) NUMARC IC AU2 includes unexpected increases in Airborne concentration in addition to plant radiation. The corresponding Hope Creek IC does not address Airborne concentration, since an increase in Airborne concentration is not addressed in the example EALs or the basis for the Unusual Event or Alert. Apparently, the Airborne concentration example EAL was deleted by NUMARC, but the corresponding IC was overlooked.

REFERENCES

NUMARC NESP-0007, AU2.1
HC.OP-AB.ZZ.0142 (Q), Loss of Shutdown Cooiing
HC.OP-AB.ZZ-0144 (Q), Loss of Fuel Pool Inventory/Cooling
HC.OP-AB.ZZ-0101 (Q), Irradiated Fuel Damage
HC.OP-AB.ZZ-126 (Q), Abnormal Release of Gaseous Radioactivity
HCGS Technical Specifications Section 3/4 9.8

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

UNUSUAL EVENT - 6.3.1.c

IC Unplanned Increase in Plant Radiation

EAL

Uncontrolled water level decrease in the Spent Fuel Pool as indicated by EITHER one of the following: • Visual Observation • **Vaild** Fuel Pool Low Level Alarm Condition

OPERATIONAL CONDITION - All

BASIS

An **Uncontrollled** decrease in Spent Fuel Pool Level represents a condition which can result in increased radiation levels, due to the loss of radiation shielding, if the Spent Fuel Pool level decrease can not be terminated. This event has a long lead time relative to potential for radiological release outside the site boundary, thus the impact to public health and safety is very low. **Uncontrolled** means that the level decrease can not be terminated.

Determination of an **uncontrolled** level decrease is made through either Visual Observation or receipt of the Spent Fuel Pool Low Level Alarm in the Main Control Room. Visual Observation is the preferred method, whenever possible, however it is NOT intended that an individual must be dispatched for classification purposes, if the existing radiation level increase trend prevents personnel from accessing the Refuel Floor, or if cameras are available to remotely verify the condition. In the event visual observation is not available by any means, then Main Control Room indication should be used.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert as a result of uncovery of irradiated fuel as indicated by high radiation levels on the refueling.

DISCUSSION

Normal Spent Fuel Pool level is at 40' of water in the pool. This level provides approximately 25' of water above the top fuel stored in pool, and 9' of water above fuel in transit. The low level alarm is set at 39' 9". This is above, but approaching the Technical Specification minimum required water level of 23 feet over the top of irradiated fuel assemblies seated in the spent fuel pool storage racks. The Technical Specification minimum water level in the Spent Fuel Pool is based on the minimum inventory and level required to remove 99% of the assumed 10% iodine gap activity that would be released from the rupture of an irradiated fuel assembly.

To prevent accidental draining of the Spent Fuel Pool, no piping connections are made to the fuel pool below the normal water level. The spent fuel pool cooling water return lines are provided with vacuum breakers to prevent water from being siphoned out of the fuel pool should a break occur in one of these lines. The skimmer surge tanks receive the overflow from the spent fuel pool and serve as the suction source to the fuel pool cooling pumps. Lowering of level in the skimmer surge tank will result in isolation of the pool filter demineralizers. This will result in the loss of the fuel pool cooling pumps. Subsequent heating of the water in the spent fuel pool may occur depending on the heat load present.

DEVIATION

NUMARC IC AU2 includes unexpected increases in Airborne concentration in addition to plant radiation. The corresponding Hope Creek IC does not address Airborne concentration, since an increase in Airborne concentration is not addressed in the example EALs or the basis for the Unusual Event or Alert. Apparently, the Airborne concentration example EAL was deleted by NUMARC, but the corresponding IC was overlooked.

REFERENCES

NUMARC NESP-0007, AU2.2
HC.OP-AR.ZZ-0014(Q), Annuciator Response Procedures, Window D3-A5 (D3834)
HC.OP-AB.ZZ-0144 (Q), Loss of Fuel Pool Inventory/Cooling
HC.OP-AB.ZZ-0101 (Q), Irradiated Fuel Damage
HC.OP-AB.ZZ-126 (Q), Abnormal Release of Gaseous Radioactivity
HCGS Technical Specifications Section 3/4 9.9
HCGS UFSAR, Section 9.2.2.2

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

ALERT - 6.3.2.a

IC 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

EAL

Unplanned increase 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

AND

Unplanned Dose Rates ≥ **2000 mRem/hr** in ANY area of the plant which require ACCESS to maintain plant safety functions (EXCLUDING the Main Control Room and CAS)

OPERATIONAL CONDITION - All

BASIS

An **Unplanned** Dose Rate of 2000 mRem/hr or greater in ANY area of the plant which requires ACCESS to maintain plant safety functions, warrants declaration of an Alert, due to the impaired ability to operate the required plant equipment. **Unplanned** is defined as those events or conditions which are not associated with a planned evolution, such that radiation levels are increasing in an uncontrolled manner. The Dose Rate threshold of **2000 mRem/hr** was chosen based upon NC.NA-AP.ZZ-0024, Radiation Protedction Program Administrative Dose Limits and Extension criteria which requires Senior Radiation Protection Supervisor approval prior to exceeding 2000 mRem/yr. This value is low enough to ensure classification of an Alert before personnel access is severely hampered and high enough to allow any increase in normal radiation level, by a factor of 1000, to be classified as an Unusual Event per EAL 6.3.1.a. Radiation levels could be indicated by ARM or radiological survey.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency when loss of control of radioactive materials causes significant offsite doses.

DISCUSSION

Emergency Coordinator judgement must be used, based on existing plant conditions, to determine areas that contain systems that are required to be operated manually, or require local surveillances to assure reliable support of safe plant operation for the conditions that exist. Areas having equipment that must be operated locally during an accident and areas along associated access routes that require HP coverage and continuous update of changing radiological conditions satisfy the definition of this condition.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA3.2

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

ALERT - 6.3.2.b

IC 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

EAL

Unplanned Dose Rates $\geq$ **15 mRem/hr** in EITHER one of the following:

- Main Control Room
- Security Central Alarm Station (CAS)

OPERATIONAL CONDITION - All

BASIS

An **Unplannned** Dose Rate of greater than or equal to 15 mRem/hr represent a condition which would jeopardize continuous occupancy of the Control Room or Security CAS, and warrants declaration of an Alert. It is the impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. In addition, **unplanned** increases in plant radiation levels represent a degradation in the control of radioactive materials and represent a degradation in the level of safety of the plant. **Unplanned** is defined as those events or conditions which are not associated with a planned evolution, such that radiation levels are increasing in an uncontrolled manner. Radiations levels can be determined by ARM or radiological survey.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency when loss of control of radioactive materials causes significant off-site doses.

DISCUSSION

The Control Room and Security Central Alarm Station general area radiation level threshold is set at 15 mr/hr and was chosen because continuous occupancy is required. This is consistent with General Design Criteria 19, which addresses continuous occupancy of the Control Room for 30 days after an accident.

The Security Secondary Alarm Station (SAS) was excluded because it is fully redundant to the Security CAS. For a radiological event, SAS would be evacuated, with all Security functions performed by the CAS.

Events which require Control Room evacuation will be classified per ECG Section 8.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA3.1
10CFR50

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

ALERT - 6.3.2.c

IC Major Damage to Irradiated Fuel

EAL

Major Damage to Irradiated Fuel has occurred

AND

Valid High Alarm received from ANY one of the following RMS channels:

- Refuel Floor Exhaust Channel A (9RX627)
- Refuel Floor Exhaust Channel B (9RX628)
- Refuel Floor Exhaust Channel C (9RX629)

OPERATIONAL CONDITION - All

BASIS

Major Damage to an irradiated fuel bundle that result in a High Refuel Floor Exhaust Radiation Monitors alarm warrants declaration of an Alert, due to the potential for an offsite release exceeding the Technical Specification limit. The intent of this EAL is to classify those events that result in the actual release of fission products from an irradiated Fuel Bundle, due to physical damage. Events that result in increased radiation levels due to shine, as a result of decreased shielding, but do not involve a release of fission products should not be classified under this EAL, but should be classified EAL 6.3.2.d, when those conditions exist.

Major Damage is defined as physical damage to an Irradiated Fuel Bundle that results from either dropping or physical contact with other components in the Fuel Pool, such that the magnitude of the damage specifically results in actuation of a Refuel Floor Exhaust High Radiation Alarm. **Valid** is defined as the High alarm occurring as a result of the damage to the irradiated fuel bundle.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency when loss of control of radioactive materials causes significant offsite doses.

DISCUSSION

The Refuel Floor Exhaust Rad Monitor are Process Monitor and are designed to detect a release of Fission Product to the atmosphere. Hence, they are included as part of the EAL threshold, to confirm the magnitude of damage to an irradiated fuel bundle. These monitors can also react as Area Radiation Monitors, in the event of increasing radiation levels due to decreased shielding, as would occur during a loss of Fuel Pool inventory event. It is important to distinguish between the cause for increased radiation levels when classifying an event under this EAL.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA2.1
HC.OP-SO.SM-0001(Q), Isolation Systems Operation
HC.OP-AB.ZZ-0116(Q), Containment Isolations and Recovery from an Isolation
HC.RP-AR.SP-0001(Q), Radiation Monitoring System Alarm Reponse, Att. 54, 55, 56
HCGS Technical Specifications, 3.3.2 Table 3.3.2-2
HCGS-UFSAR, Section 11.5.2

6.0 Radiological Releases/Occurrences

6.3 In-Plant Radiation Occurrences

ALERT - 6.3.2.d

IC Events that have or may result in uncovering Irradiated Fuel outside the Reactor Vessel

EAL

Unplanned increase on ANY one of the following Area Rad Monitors or general area rad survey indicates ≥ **2000 mRem/hr**:

- Spent Fuel Storage Pool Area (9RX707)
- New Fuel Criticality Storage Channel A (9RX612)
- New Fuel Criticality Storage Channel B (9RX613)

OPERATIONAL CONDITION - All

BASIS

An **Unplanned** Dose Rate of 2000 mRem/hr as indicated on any of the Refuel Floor Area Radiation Monitors warrants declaration of an Alert, as dose rates of this magnitude could be the result of a loss of shielding of irradiated Fuel Bundles or possible damage to an irradiated Fuel Bundle. Offsite doses during these accidents would be well below the EPA Protective Action Guidelines and the classification as an Alert is therefore appropriate. The intent of this EAL is to classifiy those events that result in increased Dose Rates on the Refuel Floor. Specifically, those events that result in increased radiation levels due to shine, as a result of decreased shielding, but do not involve a release of fission products should be classified under this EAL. Those events that result in physical damage to an irradiated and are accompanied by increasing radiation levels should not be classified under this EAL, but should be classified EAL 6.3.2.c, when those conditions exist.

Unplanned is defined as those events or conditions which are not associated with a planned evolution, such as lifting of the Reactor Vessel Internals, that results in radiation levels are increasing in an uncontrolled manner. The Dose Rate threshold of **2000 mRem/hr** was chosen based upon NC.NA-AP.ZZ-0024, Radiation Protedction Program Administrative Dose Limits and Extension criteria which requires Senior Radiation Protection Supervisor approval prior to exceeding 2000 mRem/yr. This value is low enough to ensure classification of an Alert before personnel access is severely hampered and high enough to allow any increase in normal radiation level, by a factor of 1000, to be classified as an Unusual Event per EAL 6.3.1.a. Radiation levels could be indicated by ARM or radiological survey.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency when loss of control of radioactive materials causes significant offsite doses.

DISCUSSION

The Refuel Floor Area Radiation Monitors are designed to detect an increased radiation level on the Refuel Floor. Hence, they are included as part of the EAL threshold, to determine the magnitude of a loss of shielding to irradiated Fuel Bundles. Actual Damage to an irradiated fuel bundle will also cause an increase in these Area Radiation Monitors, however the Refuel Floor Exhaust Rad Monitors are specifically designed to detect the actual release of fission products to the atmosphere. It is important to distinguish between the cause for increased radiation levels when classifying an event under this EAL.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA2.3, AA2.4
HCGS Technical Specifications, 3.3.7.1, Table 3.3.7.1-1
HC.RP-AR.SP-0001(Q), Radiation Monitoring System Alarm Reponse, Att. 41, 42, 77
NUREG-1229, Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents
EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions For Nuclear Incidents

7.0 Electrical Power

7.1 Loss of AC Power Capabilities

UNUSUAL EVENT - 7.1.1

IC Loss of All Offsite Power to Vital Buses for greater than 15 minutes

EAL

Unplanned Loss of Power from Station Service Transformers 1AX501 **AND** 1BX501 to ALL **4.16 KV Vital Buses** **AND** > **15 minutes** have elapsed

OPERATIONAL CONDITION - All

BASIS

An **Unplanned** Loss of Power from Station Service Transformers 1AX501 AND 1BX501 (Offsite Power Sources) to the **4.16 KV Vital Buses** for greater than **15 minutes**, reduces required plant redundancy and potentially degrades the level of safety by increasing plant vulnerability to a complete loss of all Vital AC power. Reliance on the EDGs to energize the Vital Buses represents a significantly abnormal condition. The intent of the EAL is to classify an Unusual Event when the EDGs are being used to energize their respective Vital Buses, due to a loss of the offsite power sources. In the case where one or more EDGs are unavailable or fail to start for any reason, following the loss of the offsite power sources, an Unusual Event is warranted until only one Onsite or Offsite Power Source remains energized, such that the loss of this energized source would result in a complete loss of all 4.16 KV Vital Power. 15 minutes was chosen to exclude transient or momentary power losses and to allow restoration of available sources. **Unplanned** is defined as the loss not being the result of planned or scheduled maintenance activities.

Although no fission product barriers are directly affected by the loss of the offsite power sources to the Vital 4.16 KV buses, the heat addition to the Primary Containment combined with heat removal capability dependent on Emergency Diesel Generator operation, warrants classification as an Unusual Event, since it is potential precursor to more serious conditions.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert based on a Loss of Offsite Power to Vital 4.16 KV buses coincident with Onsite AC power being reduced to a single Vital 4.16 KV Bus. (Operational Conditions 1, 2, and 3); or having a Loss of all Offsite and Onsite AC power in Operational Conditions 4 or 5.

DISCUSSION

Hope Creek normally has three physically separate , independent 500 KV transmission lines, connecting the Hope Creek 500 KV Switchyard with the Offsite Power Distribution Network (PJM). The three sources are as follows:

- 500 KV Hope Creek - Salem Crosstie line.
- The Keeney Line, referred to as the 5015 line, is 30.1 mile tie to the Keeney Switching Station (located near Newark, Delaware), which feeds the 500 KV Switchyard Bus Section 3.
- The New Freedom Line, referred to as the 5023 line, is a 42.9 mile tie to the New Freedom Switching Station (located northeast of Hope Creek in Camden County), which feeds the 500 KV Switchyard Bus Section 5.

Power is distributed from the 500 KV Switchyard to a 13.8 KV ring bus. Station electrical loads are supplied from the 13.8 KV ring bus through 2 physically independent auxiliary power systems, via Station Service Transformers which supply Vital and Non-Vital Station Loads. Station Service Transformers 1AX501 and 1BX501 normally supply the 4.16 KV Vital Buses. The four 4.16 KV Vital Buses can be supplied by either 1AX501 or 1BX501. Two of the four Vital Buses are normally provided power from 1AX501 with alternate power from 1BX501; the other two are normally supplied power from 1BX501 with alternate power from 1AX501. Loss of the normal power supply to a 4.16 KV Vital Bus initiates a fast transfer (alternate feeder breaker closes) to the alternate source, provided power is available.

Additionally, each 4.16 KV Vital Bus has an Emergency Diesel Generator which will automatically start and provide power to the bus in the event of a sustained loss of power to its associated Vital Bus. Additional automatic EDG starts are initiated on degraded power conditions on both 1AX501 and 1BX501, or under LOCA conditions (EDGs will not automatically provide power to the bus unless the bus has a sustained loss of power).

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SU1
HC.OP-AB.ZZ.0135 (Q),Station Blackout//Loss of Offsite Power//Diesel Generator Malfunction
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP.EO.ZZ-0102 (Q)-FC, Primary Containment Control
HCGS Technical Specifications 3/4.8, Electrical Power Systems

7.0 Electrical Power

7.1 Loss of AC Power Capabilities

ALERT - 7.1.2.a

IC 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

EAL

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) **AND** > **15 minutes** have elapsed

OPERATIONAL CONDITION - 1, 2, 3

BASIS

A degradation of the six **4.16 KV Vital Bus Power Sources,** which consist of the Offsite power sources (1AX501 AND 1BX501) and the Onsite power sources (4 EDGs), available to the **4.16 KV Vital Buses,** such that a loss of any additional single energized source would result in a complete loss of all **4.16 KV Vital Power**, represents a significant challenge to plant safety and are classified under this EAL. These conditions could occur as a result of a Loss of the Offsite power sources with concurrent failure of all but one EDG to supply power to its Vital Bus, or due to a failure of all EDGs concurrent with the Offsite power sources reduced to a single source (even though all 4.16 KV Vital Buses may still be energized). Thiese conditions reduce redundancy and potentially degrade the level of safety by increasing plant vulnerability to a complete Loss of Vital AC power. The intent of this EAL is to classify an Alert in those conditions in which a loss of a single power source to the 4.16 KV Vital Buses would result in the loss of All 4.16 KV Vital power. **Availability** is defined as a power source that can be aligned to provide power to the bus within 15 minutes. This includes the power source, as well as, all required breakers needed to provide power. 15 minutes was chosen to exclude transient or momentary power losses.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classifciation will escalate to a Site Area Emergency based on a Loss of Power to all 4.16 KV Vital Buses for > 15 minutes.

DISCUSSION

Hope Creek normally has three physically separate , independent 500 KV transmission lines, connecting the Hope Creek 500 KV Switchyard with the Offsite Power Distribution Network (PJM). The three sources are as follows:

- 500 KV Hope Creek - Salem Crosstie line.
- The Keeney Line, referred to as the 5015 line, is 30.1 mile tie to the Keeney Switching Station (located near Newark, Delaware), which feeds the 500 KV Switchyard Bus Section 3.
- The New Freedom Line, referred to as the 5023 line, is a 42.9 mile tie to the New Freedom Switching Station (located northeast of Hope Creek in Camden County), which feeds the 500 KV Switchyard Bus Section 5.

Power is distributed from the 500 KV Switchyard to a 13.8 KV ring bus. Station electrical loads are supplied from the 13.8 KV ring bus through 2 physically independent auxiliary power systems, via Station Service Transformers which supply Vital and Non-Vital Station Loads. Station Service Transformers 1AX501 and 1BX501 normally supply the 4.16 KV Vital Buses. The four 4.16 KV Vital Buses can be supplied by either 1AX501 or 1BX501. Two of the four Vital Buses are normally provided power from 1AX501 with alternate power from 1BX501; the other two are normally supplied power from 1BX501 with alternate power from 1AX501. Loss of the normal power supply to a 4.16 KV Vital Bus initiates a fast transfer (alternate feeder breaker closes) to the alternate source, provided power is available.

Additionally, each 4.16 KV Vital Bus has an Emergency Diesel Generator which will automatically start and provide power to the bus in the event of a sustained loss of power to its associated Vital Bus. Additional automatic EDG starts are initiated on degraded power conditions on both 1AX501 and 1BX501, or under LOCA conditions (EDGs will not automatically provide power to the bus unless the bus has a sustained loss of power).

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SA5
HC.OP-AB.ZZ.0135 (Q), Station Blackout / Loss of Offsite Power / Diesel Generator Malfunction
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP.EO.ZZ-0102 (Q)-FC, Primary Containment Control
HCGS Technical Specifications 3/4.8, Electrical Power Systems

7.0 Electrical Power

7.1 Loss of AC Power Capabilities

ALERT - 7.1.2.b

IC Loss of All Offsite Power and All Onsite AC Power to Vital 4.16 KV Buses during either Cold Shutdown or Refueling for greater than 15 minutes

EAL

ALL **4.16 KV Vital Buses** are deenergized **AND** > **15 minutes** have elapsed

OPERATIONAL CONDITION - 4, 5, Defueled

BASIS

A Loss of ALL **4.16 KV Vital Buses** that occurs while the plant is in either Cold Shutdown or Refueling conditions, results in a compromise of plant systems. The intent of this EAL is to classify degraded AC power events that result in a Loss of Offsite power sources (1AX501 AND 1BX501) to the 4.16 KV Vital Buses, along with a Loss of Onsite power sources (EDGs). With the plant in Cold Shutdown or Refueling, the reduced decay heat, and lower Reactor Coolant temperatures and pressures, increases the time available to restore one of the Vital Buses before Fission Product Barriers are threatened relative to classification of this condition in Operational Conditions 1, 2, or 3. Thus this condition is classified as an Alert. 15 minutes was chosen to exclude transient or momentary power losses.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency based on Radiological Release (EAL Section 6.0), or on the long term inability to remove Decay Heat (EAL Section 8.0).

DISCUSSION

Loss of all AC power to the Vital Buses compromises all plant safety systems requiring AC electric power including RHR, ECCS, Spent Fuel Pool Cooling and Service Water. tric power. Depending on the status of power supplies to non-vital buses, some Balance of Plant systems that would assist in maintaining plant conditions (i.e. RWCU, condensate, etc.) may be unavailable. Thus, the ability to remove decay heat and control containment parameters is severely challenged.

During a Loss of all AC power to the Vital Buses, all Class 1E System Instruments remain powered from Class 1E Uninterruptable Power Supplies (UPS), which are powered by DC power via inverters. The 125 VDC Battery Buses will continue to supply DC power from the batteries. Battery power is limited depending on the discharge rate and predischarge condition of the battery. The ability to restore power to AC buses may eventually be threatened as battery power (DC) is depleted due to the lack of DC (control power) for AC power circuit breakers.

Normally, Hope Creek has three physically separate , independent 500 KV transmission lines, connecting the Hope Creek 500 KV Switchyard with the Offsite Power Distribution Network (PJM). The three sources are as follows:

- 500 KV Hope Creek - Salem Crosstie line.
- The Keeney Line, referred to as the 5015 line, is 30.1 mile tie to the Keeney Switching Station (located near Newark, Delaware), which feeds the 500 KV Switchyard Bus Section 3.
- The New Freedom Line, referred to as the 5023 line, is a 42.9 mile tie to the New Freedom Switching Station (located northeast of Hope Creek in Camden County), which feeds the 500 KV Switchyard Bus Section 5.

Power is distributed from the 500 KV Switchyard to a 13.8 KV ring bus. Station electrical loads are supplied from the 13.8 KV ring bus through 2 physically independent auxiliary power systems, via Station Service Transformers which supply Vital and Non-Vital Station Loads. Station Service Transformers 1AX501 and 1BX501 normally supply the 4.16 KV Vital Buses. The four 4.16 KV Vital Buses can be supplied by either 1AX501 or 1BX501. Two of the four Vital Buses are normally provided power from 1AX501 with alternate power from 1BX501; the other two are normally supplied power from 1BX501 with alternate power from 1AX501. Loss of the normal power supply to a 4.16 KV Vital Bus initiates a fast transfer (alternate feeder breaker closes) to the alternate source, provided power is available.

Additionally, each 4.16 KV Vital Bus has an Emergency Diesel Generator which will automatically start and provide power to the bus in the event of a sustained loss of power to its associated Vital Bus. Additional automatic EDG starts are initiated on degraded power conditions on both 1AX501 and 1BX501, or under LOCA conditions (EDGs will not automatically provide power to the bus unless the bus has a sustained loss of power).

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SA1
HC.OP-AB.ZZ.0135(Q), Station Blackout//Loss of Offsite Power//Diesel Generator Malfunction
HC.OP.EO.ZZ-0100(Q)-FC, Reactor Scram
HC.OP.EO.ZZ-0102(Q)-FC, Primary Containment Control
HCGS Technical Specifications 3/4.8, Electrical Power Systems

7.0 Electrical Power

7.1 Loss of AC Power Capabilities

SITE AREA EMERGENCY - 7.1.3

IC 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

EAL

<u>ALL</u> **4.16 KV Vital Buses** are deenergized **<u>AND</u>** > **15 minutes** have elapsed

OPERATIONAL CONDITION - 1, 2, 3

BASIS

A Loss of <u>ALL</u> **4.16 KV Vital Buses** that occurs while the plant is in either Power Operation, Startup or Hot Shutdown warrrants declaration of a Site Area Emergency due to the compromise to all plant safety systems. The intent of this EAL is to classify degraded AC power events that result in a loss of Offsite power source (1AX501 AND 1BX501) to the **4.16 KV Vital Buses**, along with a Loss of Onsite power sources (EDGs). Declaration under this EAL should only occur for a loss of <u>ALL</u> **4.16 KV Vital Buses**. Prolonged loss of Vital AC power may cause Core uncovery and the inability to remove heat from the containment. 15 minutes was chosen to exclude transient or momentary power losses.

Barrier Analysis

Prolonged loss of AC power has the potential to cause a potential loss or loss of the Fission Product Barriers.

ESCALATION CRITERIA

Emergency Classification will be escalate to a General Emergency if the power loss is extended beyond 4 hours, or on loss of Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Loss of all AC power to the Vital Buses compromises all plant safety systems requiring AC electric power including RHR, ECCS, Spent Fuel Pool Cooling and Service Water. Depending on the status of power supplies to non-vital buses, some Balance of Plant systems that would assist in maintaining plant conditions (i.e. RWCU, condensate, etc.) may be unavailable. Thus, the ability to remove decay heat and control containment parameters is severely challenged.

During a Loss of all AC power to the Vital Buses, all Class 1E System Instruments remain powered from Class 1E Uninterruptable Power Supplies (UPS), which are powered by DC power via inverters. The 125 VDC Battery Buses will continue to supply DC power from the batteries. Battery power is limited depending on the discharge rate and predischarge condition of the battery. The ability to restore power to AC buses may eventually be threatened as battery power (DC) is depleted due to the lack of DC (control power) for AC power circuit breakers.

Normally, Hope Creek has three physically separate , independent 500 KV transmission lines, connecting the Hope Creek 500 KV Switchyard with the Offsite Power Distribution Network (PJM). The three sources are as follows:

- 500 KV Hope Creek - Salem Crosstie line.
- The Keeney Line, referred to as the 5015 line, is 30.1 mile tie to the Keeney Switching Station (located near Newark, Delaware), which feeds the 500 KV Switchyard Bus Section 3.
- The New Freedom Line, referred to as the 5023 line, is a 42.9 mile tie to the New Freedom Switching Station (located northeast of Hope Creek in Camden County), which feeds the 500 KV Switchyard Bus Section 5.

Power is distributed from the 500 KV Switchyard to a 13.8 KV ring bus. Station electrical loads are supplied from the 13.8 KV ring bus through 2 physically independent auxiliary power systems, via Station Service Transformers which supply Vital and Non-Vital Station Loads. Station Service Transformers 1AX501 and 1BX501 normally supply the 4.16 KV Vital Buses. The four 4.16 KV Vital Buses can be supplied by either 1AX501 or 1BX501. Two of the four Vital Buses are normally provided power from 1AX501 with alternate power from 1BX501; the other two are normally supplied power from 1BX501 with alternate power from 1AX501. Loss of the normal power supply to a 4.16 KV Vital Bus initiates a fast transfer (alternate feeder breaker closes) to the alternate source.

Additionally, each 4.16 KV Vital Bus has an Emergency Diesel Generator which will automatically start and provide power to the bus in the event of a sustained loss of power to its associated Vital Bus. Additional automatic EDG starts are initiated on degraded power conditions on both 1AX501 and 1BX501, or under LOCA conditions (EDGs will not automatically provide power to the bus unless the bus has a sustained lost of power).

Under a Loss of Vital AC Power condition, operation and control of plant systems is guided by the Station Blackout//Loss of Offsite Power//Diesel Generator Malfunction Abnormal Operating Procedure. Successful coping maintains the following key parameters within given acceptable limits:

1. Reactor water level > (TAF)
2. Suppression pool level low enough to prevent HPCI and/or RCIC steam exhaust line flooding
3. Reactor pressure high enough to maintain HPCI and RCIC operable
4. Containment pressure < design limit
5. Torus temperature < design limits (HPCI/RCIC lube oil temperature concern when suction aligned to suppression pool)
6. Drywell temperature below design limits

RCIC and HPCI operability is dependent on the availability of 125/250 VDC power. The parameters listed above can be maintained as long as battery power remains available. Battery power is limited depending on the discharge rate and predischarge condition of the battery. The HCGS IPE assumes that the batteries will be available for four hours, even though the design battery depletion time is six hours. Additionally, the loss of ventilation to the HPCI and RCIC turbine areas may result in a system isolation due to elevated temperatures.

Other than HPCI and/or RCIC, additional inventory makeup may be possible by using the diesel driven fire pump to inject water (at low pressure), to the RPV via, the RHR/LPCI system. This may require RPV depressurization using the SRVs, which also require 125 VDC power.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SS1
HC.OP-AB.ZZ.0135 (Q), Station Blackout / Loss of Offsite Power / Diesel Generator Malfunction
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP.EO.ZZ-0102 (Q)-FC, Primary Containment Control
HCGS Technical Specifications Section 3/4.8, Electrical Power Systems

7.0 Electrical Power

7.1 Loss of AC Power Capabilities

GENERAL EMERGENCY - 7.1.4.a

IC Prolonged Loss of All Offsite and Onsite AC Power to All Vital AC Buses

EAL

ALL **4.16 KV Vital Buses** are deenergized
AND
Restoration of Power to **at least one** 4.16 KV Vital Bus within **4 hours** is NOT likely

OPERATIONAL CONDITION - 1, 2, and 3

BASIS

A Loss of ALL **4.16 KV Vital Buses** for a prolonged period of time (> 4 Hours) represents a compromise to all plant safety systems. The intent of this EAL is to classify degraded AC power events that result in a Loss of offsite power source (1AX501 AND 1BX501) to the **4.16 KV Vital Buses**, along with a Loss of Onsite power sources (EDGs) for greater than 4 hours. Prolonged loss of Vital AC power may cause Core uncovery and the inability to remove heat from the containment. 4 Hours is based on the assumtions of the Station Blackout Coping Studies for Hope Creek. Beyond the four hour window, Reactor injection capability may no longer be available, and degradation in core cooling will commence. However; a General Emergency should be declared before 4 hours if it can be determined that the power loss cannot be recovered within 4 hours, or if potential loss or loss of fission product barriers is imminent.

Barrier Analysis

Although not directly related to Fission Product Barriers, these events will eventually result in the loss of all three barriers if power cannot be restored. In addition, the extent of the loss of power will result in degraded monitoring capability. It is therefore important is such events to closely monitor the Fission Product Barriers and use judgement related to the IMMINENT Loss or Potential Loss of barriers as directed in

EAL Section 3.0

ESCALATION CRITERIA

N/A

DISCUSSION

10 CFR 50.2 defines a station blackout (SBO) as complete loss of AC power to Vital AND Non-Vital buses. Loss of all AC power to the Vital Buses compromises all plant safety systems requiring AC electric power including RHR, ECCS, Spent Fuel Pool Cooling and Service Water. Depending on the status of power supplies to non-vital buses, some Balance of Plant systems that would assist in maintaining plant conditions (i.e. RWCU, condensate, etc.) may be unavailable. Thus, the ability to remove decay heat and control containment parameters is severely challenged.

During a Loss of all AC power to the Vital Buses, all Class 1E System Instruments remain powered from Class 1E Uninterruptable Power Supplies (UPS), which are powered by DC power via inverters. The 125 VDC Battery Buses will continue to supply DC power from the batteries. Battery power is limited depending on the discharge rate and predischarge condition of the battery. The ability to restore power to AC buses may eventually be threatened as battery power (DC) is depleted due to the lack of DC (control power) for AC power circuit breakers.

Under a Loss of Vital AC Power condition, operation and control of plant systems is guided by the Station Blackout//Loss of Offsite Power//Diesel Generator Malfunction Abnormal Operating Procedure. Successful coping maintains the following key parameters within given acceptable limits:

1. Reactor water level > (TAF)
2. Suppression pool level low enough to prevent HPCI and/or RCIC steam exhaust line flooding
3. Reactor pressure high enough to maintain HPCI and RCIC operable
4. Containment pressure < design limit
5. Torus temperature < design limits (HPCI/RCIC lube oil temperature concern when suction aligned to suppression pool)
6. Drywell temperature below design limits

RCIC and HPCI operability is dependent on the availability of 125/250 VDC power. The parameters listed above can be maintained as long as battery power remains available. Battery power is limited depending on the discharge rate and predischarge condition of the battery. The HCGS IPE assumes (based on the Coping Study) that the batteries will be available for four hours, even though the design battery depletion time is six hours. Additionally, the loss of ventilation to the HPCI and RCIC turbine areas may result in a system isolation due to elevated temperatures.

Other than HPCI and/or RCIC, additional inventory makeup may be possible by using the diesel driven fire pump to inject water (at low pressure), to the RPV via, the RHR/LPCI system. This may require RPV depressurization using the SRVs, which also require 125 VDC power.

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Coordinator a reasonable idea of how quickly he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of fission product barriers is imminent?

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

It is estimated that several hours are required to fully evacuate the 10 mile EPZ. Taking into consideration the above factors, declaring a General Emergency leaves sufficient time for the offsite authorities to implement Protective Actions well before a radioactive release would occur while providing sufficient time for on-site and off-site mitigation activities to restore AC power.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SG1
Station Blackout Coping Studies
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP.EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP.EO.ZZ-0104 (Q)-FC, Radioactive Release Control
HC.OP-AB.ZZ.0135 (Q), Station Blackout / Loss of Offsite Power / Diesel Generator Malfunction
HCGS Technical Specifications Section 3/4.8, Electrical Power Systems
HCGS Individual Plant Evaluation, Section 3.1.1.4.6, 3.1.2.1.6

7.0 Electrical Power

7.1 Loss of AC Power Capabilities

GENERAL EMERGENCY - 7.1.4.b

IC Prolonged Loss of All Offsite and Onsite AC Power to All Vital AC Buses

EAL

<u>ALL</u> **4.16 KV Vital Buses** are deenergized **<u>AND</u>** A Loss of **any 2** Fission Product Barriers with the Potential Loss of the **third** Barrier

OPERATIONAL CONDITION - 1, 2, and 3

BASIS

A Loss of <u>ALL</u> **4.16 KV Vital Buses** that results in a Loss of any 2 Fission Product Barriers with the Potential Loss of the third. The intent of this EAL is to classify degraded AC power events that result in a Loss of offsite power source (1AX501 AND 1BX501) to the **4.16 KV Vital Buses**, along with a Loss of Onsite power sources (EDGs). Prolonged loss of Vital AC power may cause Core uncovery and the inability to remove heat from the containment. Reactor injection capability may no longer be available, and degradation in core cooling will commence; however, a General Emergency should be declared before the loss of the fission product barriers are imminent.

Barrier Analysis

Although not directly related to Fission Product Barriers, these events will eventually result in the loss of all three barriers if power cannot be restored. In addition, the extent of the loss of power will result in degraded monitoring capability. It is therefore important is such events to closely monitor the Fission Product Barriers and use judgement related to the IMMINENT Loss or Potential Loss of barriers as directed in EAL Section 3.0

ESCALATION CRITERIA

N/A

DISCUSSION

10 CFR 50.2 defines a station blackout (SBO) as complete loss of AC power to Vital AND Non-Vital buses. Loss of all AC power to the Vital Buses compromises all plant safety systems requiring AC electric power including RHR, ECCS, Spent Fuel Pool Cooling and Service Water. Depending on the status of power supplies to non-vital buses, some Balance of Plant systems that would assist in maintaining plant conditions (i.e. RWCU, condensate, etc.) may be unavailable. Thus, the ability to remove decay heat and control containment parameters is severely challenged.

During a Loss of all AC power to the Vital Buses, all Class 1E System Instruments remain powered from Class 1E Uninterruptable Power Supplies (UPS), which are powered by DC power via inverters. The 125 VDC Battery Buses will continue to supply DC power from the batteries. Battery power is limited depending on the discharge rate and predischarge condition of the battery. The ability to restore power to AC buses may eventually be threatened as battery power (DC) is depleted due to the lack of DC (control power) for AC power circuit breakers.

Under a Loss of Vital AC Power condition, operation and control of plant systems is guided by the Station Blackout//Loss of Offsite Power//Diesel Generator Malfunction Abnormal Operating Procedure. Successful coping maintains the following key parameters within given acceptable limits:

1. Reactor water level > (TAF)
2. Suppression pool level low enough to prevent HPCI and/or RCIC steam exhaust line flooding
3. Reactor pressure high enough to maintain HPCI and RCIC operable
4. Containment pressure < design limit
5. Torus temperature < design limits (HPCI/RCIC lube oil temperature concern when suction aligned to suppression pool)
6. Drywell temperature below design limits

RCIC and HPCI operability is dependent on the availability of 125/250 VDC power. The parameters listed above can be maintained as long as battery power remains available. Battery power is limited depending on the discharge rate and predischarge condition of the battery. Additionally, the loss of ventilation to the HPCI and RCIC turbine areas may result in a system isolation due to elevated temperatures.

Other than HPCI and/or RCIC, additional inventory makeup may be possible by using the diesel driven fire pump to inject water (at low pressure), to the RPV via, the RHR/LPCI system. This may require RPV depressurization using the SRVs, which also require 125 VDC power.

The likelihood of Potential loss of the **third** Barrier should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, and the potential loss may be mitigated, it is necessary to give the Emergency Coordinator a reasonable idea of how quickly he may need to declare a General Emergency based on these conditions.

It is estimated that several hours are required to fully evacuate the 10 mile EPZ. Taking into consideration the above factors, declaring a General Emergency leaves sufficient time for the offsite authorities to implement Protective Actions well before a radioactive release would occur while providing sufficient time for on-site and off-site mitigation activities to restore AC power.

DEVIATIO.!

None

REFERENCES

NUMARC NESP-0007, SG1
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP.EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP.EO.ZZ-0104 (Q)-FC, Radioactive Release Control
HC.OP-AB.ZZ.0135 (Q), Station Blackout / Loss of Offsite Power / Diesel Generator Malfunction
HCGS Technical Specifications Section 3/4.8, Electrical Power Systems
HCGS Individual Plant Evaluation, Section 3.1.1.4.6, 3.1.2.1.6

7.0 Electrical Power

7.2 Loss of DC Power Capabilities

UNUSUAL EVENT - 7.2.1

IC Unplanned Loss of All Vital 125 VDC Power during either Cold Shutdown or Refueling Mode for greater than 15 minutes

EAL

Unplanned degraded voltage condition for <u>ALL</u> **Vital 125 VDC Buses,** such that voltage is **< 108 VDC** <u>**AND**</u> **> 15 minutes** have elapsed

OPERATIONAL CONDITION - 4, 5

BASIS

An **Unplanned** degraded voltage condition (<108 VDC) for <u>ALL</u> **Vital 125 VDC Buses** for greater than 15 minutes with the unit in Operational Condition 4 or 5 compromises the ability to monitor and control plant functions. The minimum required voltage value is based on the minimum voltage required for Vital 125 VDC bus operability following a battery discharge test per Technical Specification 4.8.2.1.b. Although continued equipment operation may occur with degraded voltage, this value signifies the minimum operable voltage allowed. **Unplanned** is defined as the loss not being the result of planned or scheduled maintenance activities. 15 minutes was chosen to exclude transient or momentary power losses.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate if the condition effects the inability to maintain cold shutdown, based on Loss of Decay Heat Removal Capability EAL section 8.1.

DISCUSSION

Vital 125 VDC provides control power to engineered safety features actuation, diesel generator auxiliaries, plant alarm and indication circuits as well as the control power for the associated loads. If 125 volt DC power is lost for an extended period of time (greater than 15 minutes) critical plant functions such as 4.16 KV Breaker Controls, HPCI, RCIC, CS, and RHR pump controls required to maintain safe plant conditions may not operate, and core uncovery with subsequent reactor coolant system and primary containment failure might occur.

In operating condition 4 or 5, a minimum of two of the four DC power channels are required by Technical Specifications, including either channel A (10D410) or channel B (10D420). The loss of one of the required two 125 VDC distribution systems would require that core alterations be suspended, that handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel be stopped.

The design limits of the 1E battery banks are as follows:

125 VDC Vital Power:

CHANNEL	Switchgear	Battery	CAPACITY
A	10D410	1AD411	1800 AH at 8 hours
B	10D420	1BD411	1800 AH at 8 hours
C	10D430	1CD411	1800 AH at 8 hours
D	10D440	1DD411	1800 AH at 8 hours

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SU7
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-AB.ZZ.0147 (Q), DC System Grounds
HC.OP-AB.ZZ.0150 (Q), 125VDC System Malfunction
HC.OP-AB.ZZ.0151 (Q), + or - 24 Volt DC Malfunction
HC.OP- AB.ZZ-0135 (Q), Station Blackout//Loss of Offsite Power//Diesel Generator Malfunction
HCGS Technical Specifications Section 3.8.2.2; 3.8.3.2
LCR 93-12, HCGS Technical Specifications Section 4.8.2.1 Revision Request

7.0 Electrical Power

7.2 Loss of DC Power Capabilities

SITE AREA EMERGENCY - 7.2.3

IC Unplanned Loss of All Vital 125 VDC Power during either Power Operations, Startup or Hot Shutdown for greater than 15 minutes

EAL

Unplanned degraded voltage condition for ALL **Vital 125 VDC Buses**, such that voltage is **< 108 VDC**

AND

> 15 minutes have elapsed

OPERATIONAL CONDITION - 1, 2, 3

BASIS

An **Unplanned** degraded voltage condition (<108 VDC) for ALL **Vital 125 VDC Buses** for greater than 15 minutes with the unit in Operational Condition 1,2 or 3 compromises the ability to monitor and control plant functions. The minimum required voltage value is based on the minimum voltage required for Vital 125 VDC bus operability following a battery discharge test per Technical Specification 4.8.2.1.b. Although continued equipment operation may occur with degraded voltage, this value signifies the minimum operable voltage allowed. **Unplanned** is defined as the loss not being the result of planned or scheduled maintenance activities. 15 minutes was chosen to exclude transient or momentary power losses.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate based on other EALs indicating Radiological Release (EAL Section 6.0) or loss of Fission Product Barriers (EAL Section 3.0).

DISCUSSION

Vital 125 VDC provides control power to engineered safety features actuation, diesel generator auxiliaries, plant alarm and indication circuits as well as the control power for the associated loads. If 125 volt DC power is lost for an extended period of time (greater than 15 minutes) critical plant functions such as 4.16 KV Breaker Controls, HPCI, RCIC, CS, and RHR pump controls required to maintain safe plant conditions may not operate, and core uncovery with subsequent reactor coolant system and primary containment failure might occur.

Loss of ADS may create a loss of low pressure ECCS availability due to the potential inability to depressurize the reactor. In addition, loss of these buses will eventually lead to MSIV closure and reactor scram due to the loss of the Primary Containment Instrument Gas (PCIG). Subsequent to MSIV closure, much of the equipment noted above will be required for plant stabilization and shutdown.

A sustained loss of 125 VDC power will threaten the ability to remove heat from the reactor core and from the containment. SRVs will remain operable in the relief mode and the heat addition to the containment could result in a loss of the primary containment as a fission product release barrier.

HPCI and RCIC also require 250 VDC vital power for system operability. Loss of vital 250 VDC power will only render the associated system inoperable; it does not affect the operability of the systems listed/discussed above. Loss of all Vital 1E 125 VDC power will also render these systems inoperable for automatic initiation, and from the control room due to loss of control power. The loss Vital 1E 250 VDC system requires that HPCI and/or RCIC be declared inoperable and the respective Technical Specification LCO be entered. Loss of these sources is therefore not included in this EAL.

The design limits of the 1E battery banks are as follows:

125 VDC Vital Power:

Channel	Switchgear	Battery	CAPACITY
A	10D410	1AD411	1800 AH at 8 hours
B	10D420	1BD411	1800 AH at 8 hours
C	10D430	1CD411	1800 AH at 8 hours
D	10D440	1DD411	1800 AH at 8 hours

In operating conditions 1,2, or 3, the loss of any single channel 125 VDC power source would require the channel to be restored within 2 hours or the unit placed in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SS3
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP.EO.ZZ-0202 (Q)-FC, Emergency Depressurization
HC.OP-AB.ZZ.0147 (Q), DC System Grounds
HC.OP-AB.ZZ.0149 (Q), 250VDC System Malfunction
HC.OP-AB.ZZ.0150 (Q), 125VDC System Malfunction
HC.OP-AB.ZZ.0151 (Q), + or - 24 Volt DC Malfunction
HC.OP- AB.ZZ-0135 (Q), Station Blackout//Loss of Offsite Power//Diesel Generator Malfunction
HCGS Technical Specifications Section 3/4.8.2.1, 3/4.8.3.1
LCR 93-12, HCGS Technical Specifications Section 4.8.2.1 Revision Request

8.0 System Malfunctions

8.1 Loss of Heat Removal Capability

ALERT - 8.1.2

IC Inability to Maintain the Plant in Cold Shutdown

EAL

Unplanned, Complete Loss of ALL Technical Specification required systems **available to provide Decay Heat Removal functions**

AND

EITHER one of the following occur:

- RCS Temperature has increased to > **200°F**
  (Excluding a **momentary** increase **>200°F** with **heat removal function** restored)
- An UNCONTROLLED temperature increase is RAPIDLY approaching **200°F**
  (with NO **heat removal functions** restored)

OPERATIONAL CONDITION - 4, 5

BASIS

A loss of decay heat removal capabilities necessary to maintain Cold Shutdown conditions could potentially lead to core damage if corrective actions are not implemented. Declaration of an Alert is warranted when ALL Technical Specification required systems are not available to provide Decay Heat Removal functions and can not be restored to prevent boiling in the core. The specification of a temperature INCREASE, rather than specific equipment failures, recognizes the potential for long heatup times providing adequate time for restoration of some form of alternate cooling. The statement "**Unplanned, Complete Loss** of ALL Technical Specification required systems **available to provide Decay Heat Removal functions**" is intended to represent a complete loss of functions available, or an inadequate ability, to provide core cooling during the Cold Shutdown and Refueling Modes, including alternate decay heat removal methods. This EAL allows for actions taken in accordance with OP-AB.ZZ-0142, Loss of Shutdown Cooling Abnormal Operating Procedure to reestablish RHR in the Shutdown Cooling Mode or provide for an alternate methods of decay heat removal, with the intent of maintaining RCS temperature below 200°F.

For loss of an in-service decay heat removal system with other decay heat removal methods available, actions taken to provide for restoration of a decay heat removal function may require time to implement. If the event results in RCS temperature **momentarily** (not to exceed 15 minutes) rising above 200°F with heat removal capability restored, Emergency Coordinator judgement will be required to determine whether heat removal systems are adequate to prevent boiling in the core and restoration of RCS temperature control. **Momentary (not to exceed 15 minutes) unplanned excursions above 200°F, when alternate decay heat removal capabilities exist, should not be classified under this EAL**. NRC analysis has shown that specific sequences can result in core uncovery within 15 to 20 minutes and severe core damage within an hour after decay heat removal capability has been lost. **Unplanned** is defined as a condition that is not due to scheduled operations or maintenance activities, in which an RHR system is intentionally removed from service.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency based on inability to maintain RPV Water level above the Top of the Active Fuel, or increased Radiological Releases.

DISCUSSION

The Residual Heat Removal (RHR) system provides the normal method for decay heat removal operating in the Shutdown Cooling Mode. With RHR unavailable for shutdown cooling operation, (including the loss of SACS and/or service water which supply cooling water to the RHR heat exchangers), alternate decay heat removal system can be aligned to control decay heat. An unavailability of these systems, can result in a gradual increase in reactor coolant temperature to the values specified in this EAL. The rate of increase in coolant temperature would be dependent on the amount of decay heat present. The threshold for this EAL is the RCS temperature transition value between Operational Condition 4 and Operrational Condtion 3.

Procedural guidance is provided to establish an alternate method of decay heat removal. These alternate methods include: aligning Reactor Water Cleanup system (RWCU), with maximum RACS aligned to the Non-regenerative heat exchanger; aligning condensate transfer via the ECCS injection lines; aligning RPV head spray with RPV Water level established above + 80"; maximizing fuel pool cooling if the RPV head is removed and the reactor cavity flooded; using the "C" RHR pump crosstied to the "A" RHR loop.

If these alternate means are unavailable, or ineffective, decay heat removal must be accomplished by feed-and -bleed using ECCS systems and discharging steam to the Suppression Pool via the SRVs to the Suppression Pool.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SA3
NUMARC Questions and Answers, June 1993, "System Malfunction Question #6b"
HC.OP-AB.ZZ-0142 (Q), Loss of Shutdown Cooling
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
Hope Creek Appendix A based on NEDO-2121, Supplement A to BWR Owners Group Emergency Procedure Guidelines, Revision 4
HCGS Technical Specifications Sections 3/4.3, 3/4.4.9, 3/4.7.1, 3/4.7.2

8.0 System Malfunctions

8.1 Loss of Heat Removal Capability

SITE AREA EMERGENCY - 8.1.3.a

IC Loss of Reactor Water Level that has or will Uncover Fuel in the Reactor Vessel

EAL

Reactor Water Level REACHES **-161" (Top of Active Fuel)**

OPERATIONAL CONDITION - 4, 5

BASIS

Reactor Water Level reaching -161" indicates a loss of core submergence. Without core submergence, the integrity of the fuel clad barrier can no longer be assured, even with the reduced decay heat levels in Cold Shutdown and Refuel. This event is classified based on reaching the Reactor Water level threshold (instead of being able to restore and maintain above the threshold) due to the potentially severe consequences of a loss of core submergence. Since the design of the normal and emergency makeup systems would preclude this condition, an extreme challenge to their ability to provide core cooling by submergence has occurred. Additionally, ECCS availability and Containment Integrity requirements may be relaxed under these Operational Conditions, thus classification at the Site Area Emergency level is warranted.

Barrier Analysis

Fuel Clad Barrier has been potentially lost
RCS Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate to a General Emergency based on abnormal radiological releases.

DISCUSSION

Core Submergence ensures adequate core cooling. When RPV level decreases to below TAF the ability to effectively remove decay heat can no longer be guaranteed, and the fuel cladding fission product barrier can no longer be considered intact. Sustained partial or total core uncovery can result in clad damage and a significant release of fission products to the reactor coolant. Sustained core uncovery can also result in a breach of the reactor vessel, or an unisolated intersystem LOCA with the RHR System.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SS5
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0201 (Q)-FC, Alternate Level Control

8.0 System Malfunctions

8.1 Loss of Heat Removal Capability

SITE AREA EMERGENCY- 8.1.3.b

IC Complete Loss of Functions Needed to Achieve Cold Shutdown Conditions

EAL

Loss of Main Condenser capabilities, as evidenced by an inability to remove Decay Heat from the Reactor

AND

Loss of Torus capabilities as evidenced by EITHER one of the following:

- Entry into an Unsafe region of ANY 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

OPERATIONAL CONDITION - 1, 2, 3

BASIS

A Complete **Loss** of decay heat removal systems required to ACHIEVE Cold Shutdown conditions from a Hot Shudown condition, represents a significant challenge to the plant due to the failure of multiple systems designed for the protection of the public. Hence, declaration of a Site Area Emergency is warranted. This EAL specifically includes a degradation of those plant systems required to ACHIEVE a Cold Shutdown condition. It does NOT include an inability to MAINTAIN a Cold Shutdown condition. The inability to MAINTAIN Cold Shutdown Conditions is specifically addressed by EAL 8.1.2. Hence, a Loss of RHR Shutdown Cooling is not included in this EAL. This EAL includes a Loss of Service Water and/or SACS capabilities, based on the effect a loss of these systems has on the ability to maintain Torus capabilities with the Safe Region of the referenced EOP curves. **Loss** is defined as the systems being unavailable to perform their intended design function. Hence, in the case where the Main Condenser became isolated from the Reactor due to an MSIV Isolation, but the MSIV could be reopened by procedure, then a Loss of the Main Condenser capabilities has not occurred.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a General Emergency based on loss of Fission Product Barriers or Radiological Releases.

DISCUSSION

A loss of both the normal heat sink for the reactor, and an impending severe degradation of alternate heat removal capability to the Torus. Loss of the heat sink for the reactor when in a Hot Shutdown condition will limit the ability to maintain that condition, or to cooldown the reactor if required.

The Main Condenser can be lost for a variety or reasons; loss of Circulating Water, loss of the turbine Control and/or Bypass Valve functions, main steam line isolation, etc. With the Main Condenser not available, heat must be removed from the RCS by the SRVs and absorbed in the Suppression Pool. Loss of the pressure control ability of the SRVs as indicated by the inability to reduce RPV pressure represents a loss of control of a major RCS parameter; which could result in RPV overpressure conditions, or the inability to cooldown if cold shutdown is required.

The Heat Capacity Temperature Limit curve is defined as the highest torus temperature at which initiation of RPV depressurization will not result in exceeding either the suppression pool design temperature or the primary containment pressure limit before the rate of energy transfer from the RPV to the containment is within the capacity of the containment vent. The Heat Capacity Level Limit is defined as the higher of either the elevation of the containment downcomer opening or the lowest torus level at which initiation of RPV depressurization will not result in exceeding the Heat Capacity Temperature Limit. Violation of either curve would require an immediate emergency depressurization, thus ensuring that all the heat immediately present in the reactor has been transferred to the containment while maintaining the containment within design limits. This represents a serious potential threat to the Primary Containment.

DEVIATION

The NUMARC IC asociated with EAL SS4 suggests that the IC should include a Complete Loss of Function needed to achieve or maintain Hot Shutdown. The NUMARC basis includes both reactivity control and decay heat removal. At Hope Creek, as with all other BWRs, the operator action of placing the Reactor Mode Switch in the Shutdown position that results in Control Rod inserting into the core such that the Reactor will remain shutdown under all conditions without boron, places the Reactor in a Hot Shutdown condtion. No additional actions are required to maintain the Reactor in this condition. Systems are required and additional operator actions are required to achieve Cold Shutdown conditions. Based on this, Hope Creek has modified the NUMARC IC for SS4 to apply specifically to a total loss of decay heat removal, since reactivity control concerns are addressed under the ATWS Section. This IC and EAL are consistent with the requirements for declaration ofa SiteArea Emergency.

REFERENCES

NUMARC NESP-0007, SS4
HC.OP-EO.ZZ-0100 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
Hope Creek Appendix A based on NEDO-2121, Supplement A to BWR Owners Group Emergency Procedure Guidelines, Revision 4
HCGS Technical Specifications 3/4.1.3, 3/4.1.5

8.0 System Malfunctions

8.2 Loss of Assessment Capability

UNUSUAL EVENT - 8.2.1.a

IC Unplanned Loss of All Onsite or Offsite Communications Capabilities

EAL

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

- Station Page System (Gaitronics)
- Station Radio System
- Direct Inward Dial System (DID)
- Essex (Centrex) Phone System
- Nuclear Emergency Telephone System (NETS)

OPERATIONAL CONDITION - All

BASIS

An **Unplanned** loss of communication ability significantly degrades the operating crews ability to perform tasks necessary for plant operations and/or the ability to communicate with offsite authorities, warrants declararation of an Unusual Event. The loss of off-site communications capability is more comprehensive than that addressed by 10CFR50.72.b. **Unplanned** is defined as the loss of communication capabilities not being the result of planned maintenance activities, where compensatory measures would be taken.

Barrier Analysis

N/A

ESCALATION CRITERIA

None

DISCUSSION

None

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SU6

8.0 System Malfunctions

8.2 Loss of Assessment Capability

UNUSUAL EVENT - 8.2.1.b

IC Unplanned Loss of All Onsite or Offsite Communications Capabilities

EAL

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)
- Essex (Centrex) Phone System
- NAWAS
- EMRAD
- FTS 2000

OPERATIONAL CONDITION - All

BASIS

An **Unplanned** loss of communication ability significantly degrades the operating crews ability to perform tasks necessary for plant operations and/or the ability to communicate with offsite authorities, warrants declararation of an Unusual Event. The loss of off-site communications capability is more comprehensive than that addressed by 10CFR50.72.b. **Unplanned** is defined as the loss of communication capabilities not being the result of planned maintenance activities, where compensatory measures would be taken.

Barrier Analysis

N/A

ESCALATION CRITERIA

None

ESCALATION CRITERIA

Emergency classification will be escalate to an Alert if a transient is in progress or if CRIDS becomes unavailable.

DISCUSSION

Without Control Room annunciators, there may be difficulty initially recognizing changing plant conditions, as well as, monitoring conditions associated with normal plant operations. SNSS judgement of the severity of the loss should also be based on the need to initiate increased or continuous plant equipment monitoring. Also, specific annunciator loss should be judged against those needed for by the operating staff for operation in abnormal and emergency operating procedures.

Most alarm conditions for the annunciator system have CRIDS digital alarm points as well. By monitoring the CRIDS screens, most alarm conditions can be observed and responded to independent of the overhead annunciators.

A loss of plant inverter 1BD483 will result in a loss of all overhead annunciators, as well as a loss of feedwater level control. If the loss of overhead annunciators was not due to the loss of 1BD483, stable plant operations should continue.

This EAL is not required in modes 4 or 5 due to the limited number of safety systems required for operation.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SU3
HC.OP.AB.ZZ-0143 (Q), Loss of Overhead Annunciators / Loss of Crids
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram

8.0 System Malfunctions

8.2 Loss of Assessment Capability

UNUSUAL EVENT - 8.2.1.c

IC Unplanned Loss of Most or All Safety System Annunciation or Indication in the Control Room for Greater Than 15 Minutes

EAL

Unplanned Loss of > 75% of Main Control Room Overhead Annunciators for ≥ **15 minutes**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

A **Unplanned** Loss of **> 75%** of all Main Control Room Overhead Annunciators without a plant transient in Operational Conditions 1, 2 or 3 for greater than 15 minutes warrants a heightened awareness by Control Room Operators. Qualification of **> 75%** is left to the discretion of the Senior Nuclear Shift Supervisor (SNSS), and is considered approximately 75%. It is not intended that a detailed count be performed, but that a rough approximation be used to determine the severity of the loss. CRIDS is available to provide compensatory indication. 15 minutes is used as a threshold to exclude transient or momentary power losses. The 15 minutes clock starts when the annunciators have been lost, or are determined to have been lost. If upon time of discovery it is determined that the annunciators have been lost for at least 15 minutes prior to discovery, classification must be made under this EAL regardless of time required for restoration. If it is determined that the annunciators had previously been lost for at least 15 minutes but the annunciators were available at the time of discovery, classification is not required under this EAL but a review of the "After The Fact" RAL must be completed. **Unplanned** loss of annunciators excludes scheduled maintenance and testing activities.

Barrier Analysis

N/A

DISCUSSION

None

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SU6

8.0 System Malfunctions

8.2 Loss of Assessment Capability

ALERT - 8.2.2.a

IC Unplanned Loss of Control Room Annunciatiors and a Significant Transient is in Progress or Compensatory Indicators are Unavailable

EAL

Unplanned Loss of > **75%** of Main Control Room Overhead Annunciators for ≥ **15 minutes** **AND** BOTH of the following: • CRIDS • SPDS are NOT AVAILABLE

OPERATIONAL CONDITION - 1, 2, 3

BASIS

An **Unplanned** Loss of > **75%** of Main Control Room Overhead Annunciators with loss of backup control room monitoring significantly hampers operator response. Qualification of > **75%** is left to the discretion of the Senior Nuclear Shift Supervisor (SNSS), and is considered approximately 75%. It is not intended that a detailed count be performed, but that a rough approximation be used to determine the severity of the loss.

15 minutes is used as a threshold to exclude transient or momentary power losses. The 15 minutes clock starts when the annunciators have been lost, or are determined to have been lost. If upon time of discovery it is determined that the annunciators have been lost for at least 15 minutes prior to discovery, classification must be made under this EAL regardless of time required for restoration. If it is determined that the annunciators were lost for at least 15 minutes with the annunciators available at the time of discovery, classification is not required under this EAL but a review of the "After The Fact" RAL must be completed.

Unplanned loss of annunciators excludes scheduled maintenance and testing activities. The fifteen minutes also allows for attempting to restore the CRIDS computer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency based on a loss of control room annunciators with both a failure of CRIDS and a plant transient in progress.

DISCUSSION

The Control Room Integrated Display System (CRIDS) is not essential for the safe shutdown or operation of the plant. However, with the loss of control room annunciators the loss of CRIDS significantly reduces the ability of the operations staff to monitor and evaluate plant conditions. SNSS judgement of the severity of the loss should also be based on the need to initiate increased or continuous plant equipment monitoring. Most alarm conditions for the annunciator system have CRIDS digital alarm points as well. By monitoring the CRIDS screens, most alarm conditions can be observed and responded to, independent of the overhead annunciators.

The safety parameter display system (SPDS) also provides information and indication related to selected plant parameters during a plant transient. Loss of this assessment tool may hamper operators attempt to comply with directions provided in EOPs or may limit the recognition of significant parameter values called out in the EOPs. It is not included in the threshold for this EAL because of the limited scope of the parameters it monitors.

This EAL is not required in modes 4 or 5 due to the limited number of safety systems required for operation.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SA4
HC.OP.AB.ZZ-0143 (Q), Loss of Overhead Annunciators / Loss of Crids
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram

8.0 System Malfunctions

8.2 Loss of Assessment Capability

ALERT - 8.2.2.b

IC Unplanned Loss of Control Room Annunciatiors and a Significant Transient is in Progress or Compensatory Indicators are Unavailable

EAL

Unplanned Loss of > **75%** of Main Control Room Overhead Annunciators for ≥ **15 minutes** **AND** A **significant transient** is in progress

OPERATIONAL CONDITION - 1, 2, 3

BASIS

An **Unplanned** Loss of > **75%** of Main Control Room Overhead Annunciators with a **significant transient** in progress significantly hampers operator response. Qualification of > **75%** is left to the discretion of the Senior Nuclear Shift Supervisor (SNSS), and is considered approximately 75%. It is not intended that a detailed count be performed, but that a rough approximation be used to determine the severity of the loss. **Significant transients** include response to automatic or manually initiated actions such as:

- Reactor Scrams
- Turbine Trips
- Recirc Runbacks reducing reactor power > 25%
- Thermal Power oscillations of 10% or greater
- Stuck open SRVs
- ECCS actuation
- Loss of Feedwater Heating
- Recirc Pump Trip
- Loss of Offsite power source

15 minutes is used as a threshold to exclude transient or momentary power losses. The 15 minutes clock starts when the annunciators have been lost, or are determined to have been lost. If upon time of discovery it is determined that the annunciators have been lost for at least 15 minutes prior

to discovery, classification must be made under this EAL regardless of time required for restoration. If it is determined that the annunciators were lost for at least 15 minutes with the annunciators available at the time of discovery, classification is not required under this EAL but a review of the "After The Fact" RAL must be completed. **Unplanned** loss of annunciators excludes scheduled maintenance and testing activities.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency based on a loss of control room annunciators with both a failure of CRIDS and a plant transient in progress.

DISCUSSION

Without Control Room annunciators, it may be difficult to monitor conditions associated with normal plant operations. During transient event such as those listed in the EAL, the difficulty becomes more acute.

Loss of control room annunciators significantly reduces the ability of the operations staff to monitor and evaluate plant conditions. SNSS judgement of the severity of the loss should also be based on the need to initiate increased or continuous plant equipment monitoring. Most alarm conditions for the annunciator system have CRIDS digital alarm points as well. By monitoring the CRIDS screens, most alarm conditions can be observed and responded to, independent of the overhead annunciators. The safety parameter display system (SPDS) also provides information and indication related to selected plant parameters during a plant transient.

This EAL is not required in modes 4 or 5 due to the limited number of safety systems required for operation.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SA4
HC.OP.AB.ZZ-0143 (Q), Loss of Overhead Annunciators / Loss of Crids
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram

8.0 System Malfunctions

8.2 Loss of Assessment Capability

SITE AREA EMERGENCY - 8.2.3

IC Inability to Monitor a Significant Transient in Progress

EAL

Unplanned Loss of > **75%** of Main Control Room Overhead Annunciators for ≥ **15 minutes**

<u>**AND**</u>

A **significant transient** is in progress

<u>**AND**</u>

Main Control Room Board Indications are <u>NOT AVAILABLE</u> to monitor <u>ANY</u> of the following:

- RCS Status
- Reactivity Control
- ECCS
- Containment Parameters

<u>**AND**</u>

<u>BOTH</u> of the following:
- CRIDS
- SPDS

are <u>NOT AVAILABLE</u>

OPERATIONAL CONDITION - 1, 2, 3

BASIS

An **Unplanned** Loss of > **75%** of Main Control Room Overhead Annunciators with loss of backup control room monitoring, AND while a transient is in progress represents a major loss of ability to properly respond to a transient condition. Qualification of **> 75%** is left to the discretion of the Senior Nuclear Shift Supervisor (SNSS), and is considered approximately 75%. It is not intended that a detailed count be performed, but that a rough approximation be used to determine the severity of the loss. Backup monitoring from CRIDS compounds the ability to monitor the progress of the transient. In addition, a Loss of Main Control Room indications for one of the systems listed in the EAL must also occur. **Significant transients** include response to automatic or manually initiated actions such as:

Reactor Scrams
Turbine Trips
Recirc Runbacks reducing reactor power > 25%
Thermal Power oscillations of 10% or greater
Stuck open SRVs
ECCS actuation
Loss of Feedwater Heating
Recirc Pump Trip
Loss of Offsite power source

15 minutes is used as a threshold to exclude transient or momentary power losses. The 15 minutes clock starts when the annunciators have been lost, or are determined to have been lost. If upon time of discovery it is determined that the annunciators have been lost for at least 15 minutes prior to discovery, classification must be made under this EAL regardless of time required for restoration. If it is determined that the annunciators were lost for at least 15 minutes with the annunciators available at the time of discovery, classification is not required under this EAL but a review of the "After The Fact" RAL must be completed. **Unplanned** loss of annunciators excludes scheduled maintenance and testing activities. The fifteen minutes also allows for attempting to restore the CRIDS computer.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a General Emergency based on either the Loss of Fission Product Barriers; increased plant radiation levels or releases; or EC judgement.

8.0 System Malfunctions

8.3 Loss of Control Room Habitability

ALERT - 8.3.2

IC Main Control Room Evacuation has been Initiated

EAL

Main Control Room Evacuation has been initiated

OPERATIONAL CONDITION - All

BASIS

Main Control Room evacuation represents a serious plant situation since the degree of plant control at the Remote Shutdown Panel is not as complete as from the Main Control Room. The intent of this EAL is to declare an Alert when the determination to evacuate the Main Control Room has been made based on environmental/personnel safety concerns, and the physical process of evacuating the Control Room per OP-AB.ZZ-0130, Control Room Evacuation has commenced.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a Site Area Emergency if control cannot be established within 15 minutes.

DISCUSSION

Control Room evacuation requires establishment of plant control from outside the control room (Remote Shutdown Panels (RSP)). Support from the Technical Support Center (TSC) and/or other Emergency Operations Facility (EOF) is necessary.

Establishing remote system control will bypass many protective trips and interlocks. In addition, most of the instrumentation and assessment tools available in the Main Control Room will not be available. Operator actions upon deciding that the control room should be evacuated include scramming the reactor and closing the MSIVs. With these actions taken all inventory and pressure control can be accomplished at the RSP.

A fire in any one of the following fire zones has the potential to render redundant safe shutdown controls and instrumentation in the Main Control Room inoperable.

Fire Zone	Description
5202	Cable Spreading Room
5302	Control Equipment Room
5403	Control Equipment Room Mezzanie
5510	Main Control Room
5605	Class 1E Panel Room
5620	1E Panel Room HVAC
5704	Diesel Area HVAC

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA5
HC.OP-AB.ZZ-0130 (Q), Control Room Evacuation

DISCUSSION

Without Control Room annunciators, it may be difficult to monitor conditions associated with normal plant operations. During transient event such as those listed in the EAL, the difficulty becomes more acute. Compounding these, a concurrent loss of control room backup monitoring will further hinder operations staff decision making needed to respond to the transient.

The safety parameter display system (SPDS) also provides information and indication related to selected plant parameters during a plant transient. Loss of this assessment tool may hamper operators attempt to comply with directions provided in EOPs or may limit the recognition of significant parameter values called out in the EOPs. It is not included in the threshold for this EAL because of the limited scope of the parameters it monitors.

This EAL is not required in modes 4 or 5 due to the limited number of safety systems required for operation.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SS6
HC.OP.AB.ZZ-0143 (Q), Loss of Overhead Annunciators / Loss of Crids
HC.OP.EO.ZZ-0100 (Q)-FC, Reactor Scram

8.0 System Malfunctions

8.3 Loss of Control Room Habitability

SITE AREA EMERGENCY - 8.3.3

IC Main Control Room Evacuation has been Initiated and Plant Control cannot be established

EAL

Main Control Room Evacuation has been initiated **AND** **Control of the plant** CANNOT be established from outside the Main Control Room within **15 minutes**

OPERATIONAL CONDITION - All

BASIS

Failure to transfer and establish control of safety systems needed to maintain the Reactor in a safe shutdown condition and remove decay heat, could result in damage to the fission product barriers, and the ability to determine plant status may be lost. The **15 minute** time limit for transfer of control is based on a reasonable time period for personnel to leave the control room, arrive at the Remote Shutdown Panel, and reestablish plant control to preclude core uncovery and/or core damage. The term "**control of the plant**" will require SNSS assessment to determinine whether sufficient control has been established to maintain adequate core cooling.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a General Emergency based on loss of fission product barriers, abnormal radiological releases, or Emergency Director judgement.

DISCUSSION

Most of the monitoring capability of the Remote Shutdown Panel is not enabled until control is transferred. During this transitional period the function of monitoring and/or controlling parameters necessary for plant safety may not be occurring and may result in a threat to plant safety. If the transitional period is prolonged, damage to plant systems and safety barriers may occur and worsen without actions being taken to mitigate the consequences.

Control Room evacuation requires establishment of plant control from outside the control room (Remote Shutdown Panels (RSP)). Support from the Technical Support Center (TSC) and/or other Emergency Operations Facility (EOF) is necessary.

A fire in any one of the following fire zones has the potential to render redundant safe shutdown controls and instrumentation in the Main Control Room inoperable.

Fire Zone	Description
5202	Cable Spreading Room
5302	Control Equipment Room
5403	Control Equipment Room Mezzanie
5510	Main Control Room
5605	Class 1E Panel Room
5620	1E Panel Room HVAC
5704	Diesel Area HVAC

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HS2
HC.OP-AB.ZZ-0130 (Q), Control Room Evacuation

8.0 System Malfunctions

8.4 Technical Specifications

UNUSUAL EVENT - 8.4.1

IC Inability to Reach Required Operational Condition within Technical Specification Limits

EAL

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

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Failure to place the unit in an Operational Condition in conpliance with the Technical Specification LCO ACTION Statement warrants declaration of an Unusual Event due to the plant being outside the defined Technical Specification safety envelope. Classification under this EAL is specific to an INABILITY OR FAILURE to comply with the mode change requirements of those Technical Specification LCOs that require the plant placed in a more conservative Operational Condition. Classification should be made under this EAL for a failure to comply with ANY Technical Specification required change in Operational Condition FROM the Operational Conditions in which this EAL applies (Operational Conditions 1,2 and 3). An Unusual Event is declared when the plant FAILS TO COMPLY WITH THE OPERATIONAL CONDITION change stated in the ACTION Statement of an LCO, and NOT as the result of a required ACTION.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate based upon system malfunctions or other conditions covered in various other EAL sections.

DISCUSSION

A shutdown required by the Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when actions are completed within the allowable action statement time in the Technical Specifications. If the times specified within the action statements are not met, the plant may be in an unsafe condition. The declaration is based on exceeding the LCO ACTION STATEMENT time period from the <u>POINT OF RECOGNITION</u> and is not related to how long a plant condition may have existed.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, SU2
HCGS Technical Specifications
10CFR50.72
NUMARC Questions and Answers, June 1993, "General Question #7"
NUMARC Questions and Answers, June 1993, "General Question #8"

9.0 Hazards - Internal/External

9.1 Security Threats

UNUSUAL EVENT - 9.1.1

IC Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant

EAL

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

OPERATIONAL CONDITION - All

BASIS

Security events classified under this EAL represent a potential degradation in the level of safety of the plant. The EAL threshold is satisified if the event is identified as being directed toward the station The intent of this EAL is to classify security events which threaten the Protected Area, but have not been determined to threaten plant vital areas. A security threat exists if physical evidence supporting the threat exists, if information independent from the actual threat exists, or if a specific group claims responsibility for the threat. The SNSS/EC should declare an Unusual Event upon consulting with Security to determine the validity of the entry conditions. Security Contingency

Procedure (SCP) numbers are referenced following each EAL threshold. Since some SCP numbers appear in more than one EAL, the on-duty PSE&G Security Supervisor will provide information concerning the specific event to aid in classification.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert based upon an actual intrusion or malicious acts within the Protected Area.

DISCUSSION

Security events which do not represent a potential degradation in the level of safety of the plant are reported under RAL Section 11.0, One Hour Non-Emergency Safeguards Event (10 CFR 73.71 or 10 CFR 50.72), and will not result in an Unusual Event declaration.

The following is an index of Security Contingency Procedures referenced by this event:

SCP-5 "Security Threat"
SCP-7 "Internal Disturbance"
SCP-8 "Hostage Situation"
SCP-10 "Discovery of Destructive Devices or Evidence of Malicious Acts"
SCP-11 "Civil Disturbance"

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU4.1, HU4.2
Safeguards Contingency Plan

9.0 Hazards - Internal/External

9.1 Security Threats

ALERT - 9.1.2

IC Security Event in a Plant Protected Area

EAL

Confirmed hostile intrusion or malicious acts as evidenced by <u>ANY</u> 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 or destructive device discovered in a Vital Area, resulting in **SCP-10** implementation

OPERATOINAL CONDITION - All

BASIS

Security events classified under this EAL represent an escalated threat to the level of safety of the plant. The EAL threshold is satisified if physical evidence supporting the hostile intrusion or assault exists. The intent of this EAL is to classify security events which represent an actual intrusion into the Plant Protected Area. The SNSS/EC should declare an Alert upon consulting with the Security to determine the validity of the entry conditions. Security Contingency Procedure (SCP) numbers are referenced following each EAL threshold. Since some SCP numbers appear in more than one EAL, the on-duty PSE&G Security Supervisor will provide information concerning the specific event to aid in classification.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will be escalate to a Site Area Emergency based upon a hostile intrusion or act in Plant Vital Areas.

DISCUSSION

The following is an index of Security Contingency Procedures referenced by this event:

SCP-6 "Discovery of Intruders or Attack"
SCP-8 "Hostage Situation"
SCP-10 "Discovery of Destructive Devices or Evidence of Malicious Acts"

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA4.1, HA4.2
Safeguards Contingency Plan

9.0 Hazards - Internal/External

9.1 Security Threats

SITE AREA EMERGENCY - 9.1.3

IC Security Event in a Plant Vital Area

EAL

Confirmed hostile intrusion or malicious acts in Plant Vital Areas as evidenced by :

- Discovery of an intruder(s), armed and violent, within a Vital Area, resulting in **SCP-6** implementation

OPERATIONAL CONDITION - All

BASIS

Security events classified under this EAL represent an escalated threat to plant safety above that contained in an Alert in that a hostile intrusion or assault has progressed from the Protected Area to a Plant Vital Area. These areas contain vital equipment which includes any equipment, system, device or material, the failure, destruction or release of could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems which would be required to function to protect health and safety following such failure, destruction or release are also considered vital. Security Contingency Procedure (SCP) numbers are referenced following each EAL threshold. Since some SCP numbers appear in more than one EAL, the on-duty PSE&G Security Supervisor will provide information concerning the specific event to aid in classification.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to a General Emergency based upon the actual loss of physical control of the Main Control Room or Remote Shutdown Panel.

DISCUSSION

Plant Vital Areas are within the Protected Area and are generally controlled by card key readers. A hostile intrusion into a Plant Vital Area could represent a situation that threatens the safety of plant personnel and the general public.

The following is an index of the Security Contingency Procedure referenced by this event:

SCP-6 "Discovery of Intruders or Attack"

DEVIATION

None

REFERENCES

NUMARC NESP-007, HS1.1, HS1.2
Safeguards Contingency Plan

9.0 Hazards - Internal/External

9.1 Security Threats

GENERAL EMERGENCY - 9.1.4

IC Security Event Resulting in Loss of Ability to Reach and Maintain Cold Shutdown

EAL

Security event resulting in the **actual loss of physical control** of <u>EITHER</u> one of the following: • Main Control Room • Remote Shutdown Panel

OPERATIONAL CONDITION - All

BASIS

Security events classified under this EAL represent conditions under which a hostile force has taken physical control of areas required to reach and maintain cold shutdown. Both the Main Control Room and Remote Shutdown Panel are included, since control of either could hamper the operating crew's ability to perform a safe plant shutdown. **Actual loss of physical control** is defined as the condition where licensed Control Room operators can no longer take required action to operate the plant, including unauthorized transfer of plant control from the Main Control Room.

Barrier Analysis

N/A

ESCALATION CRITERIA

N/A

DISCUSSION

Security threats which meet the threshold for declaration of a General Emergency are an actual loss of physical control of the Main Control Room or the Remote Shutdown Panel. This situation places the plant in a potentially unstable condition with high potential of multiple barrier failures.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HG1.1, HG1.2
Safeguards Contingency Plan

9.0 Hazards - Internal/External

9.2 Fire

UNUSUAL EVENT - 9.2.1

IC Fire within the Protected Area Boundary Not Extinguished within 15 minutes of Detection

EAL

Valid Fire Alarm is received in the Main Control Room **OR**
Report of a **fire** from personnel at the scene

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

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

OPERATIONAL CONDITION - All

BASIS

Fires classified under this EAL include those of a magnitude and extent that may be a potential precursor to damage to **Safety Systems**, and hence has safety significance. This EAL includes plant vital structures and also structures and areas that are contiguous to plant vital structures, due to the potential for a fire to spread from a non-safety related structure to an adjoining safety related structure. A fire alarm received in the Main Control Room is considered to be **Valid** when the alarm is substantiated by the receipt of related independent alarms (fire, temperature, deluge,etc) in the Main Control Room or by visual confirmation if only a single detector is alarming. This EAL EXCLUDES such items as fires in Structures other then those listed in the EAL, waste-basket fires, and other small fires of no safety significance based on the judgement of the SNSS that NO potential to affect a **Safety System** exists. Emergency Coordinator judgement must be exercised to determine if a fire within a plant structure is of any safety significance. The 15 minute clock starts upon receipt of a **Valid** Fire Alarm or report of a fire from personnel at the scene. 15 minutes was determined to be a reasonable time limit for small fires to be extinguished. A **Safety System** is defined as any system or component included within the Technical Specification.

Fire is defined as combustion characterized by the generation heat and smoke. Sources of smoke such as overheated electrical equipment and slipping drive belts, for example, do not constitute fires. Observation of a flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert if the fire damages more than one plant safety system or damages any plant vital structures.

DISCUSSION

The presence of a fire within the specified areas must be evaluted to determine the potential impact on **Safety Systems**, even if initial reports are that the fire is effecting a non-safety related portion of the plant, but has the potential to spread.

Excluded non-vital permanent plant structures include:

Circulating Water Structure
Station Service Transfomer and Switchyard Area
Hope Creek Admin Building
Onsite Warehouses
Onsite Trailers
Main and Aux Guardhouse
Nuclear Services Building
Auxiliary Boiler House

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU2
HCGS Fire & Medical Emergency Response; HC.FP-EO.ZZ-0001(Z)
NUMARC Questions and Answers, June 1993, "Hazards Question #7"

9.0 Hazards - Internal/External

9.2 Fire

ALERT - 9.2.2

IC Fire Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown

EAL

Fire within ANY one of the following Plant Vital Structures:
- Reactor Building
- Control/Aux Building
- Service Water Intake Structure
- Service/Rad Waste Building

AND

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

Damaged Safety System(s) or Plant Vital Structure is required for the present Operating Condition

OPERATIONAL CONDITION - All

BASIS

The primary concern in this EAL is the magnitude of the fire and the effects on **safety systems** required for the present Operating Condition. Specific system degradation is addressed in the System Malfunction EALs. A detailed assessment of system **damage** is not required prior to classification. The term "**Damage**" is defined as evidence that the fire has caused component malfunction (pump trip, breaker trip, etc.) or a report of visible scorching, blistering or other deformation that may have resulted in the equipment/structure being INOPERABLE or otherwise incapable of performing it's design function. A **Safety System** is defined as any system or component included in Technical Specifications. In those cases where it is believed that the fire may have caused damage to **Safety Systems**, then an Alert declaration is warranted, since the full extent of the damage may not be known. For Plant Vital Structure **damage**, classification is required under this EAL if the structure houses or otherwise supports **safety systems** required for the present Operating Condition.

For example, a fire that has been confirmed to be localized to a single piece of equipment, like a 4.16 KV Breaker, with no potential to spread to adjacent equipment, does not warrant classification as an Alert. In the event, however, that the fire has spread or is believed to be spreading to other 4.16 KV Breakers for component(s) required for the present operating condition, then an Alert is warranted.

Fire is defined as combustion characterized by the generation heat and smoke. Sources of smoke such as overheated electrical equipment and slipping drive belts, for example, do not constitute fires. Observation of a flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate based on further damage to plant safety systems, loss of fission product barriers, or abnormal radiological releases.

DISCUSSION

No lengthy and timely assessment of damage is required prior to classification. In this EAL, no attempt is made to quantify the magnitude of the damage to any safety system but instead an attempt is made to identify any damage in order to quantify the magnitude and extent of the fire. In short, if the fire is big enough that it has damaged more than one safety system, or more than one subsystem of a safety system, then the fire is big enough to justify an Alert declaration. Damage to Plant Vital Structures must be to the extent that EC judgement must be used to determine if the structure is still capable of performing its design function. Electrical failures (such as shorts, grounds, arcing, etc.) should be evaluated for the possibility of a fire. Any security aspects of this event should be considered under EAL sections covering Security Events.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA2
HCGS Fire & Medical Emergency Response; HC.FP-EO.ZZ-0001(Z)
HCGS Technical Specifications Section 3/4 7-6, Control Room Emergency Filtration System
NUMARC Questions and Answers, June 1993, "Hazards Question #7"

9.0 Hazards - Internal/External

9.3 Explosion

UNUSUAL EVENT - 9.3.1

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

Confirmed Explosion within the Protected Area **AND** Report of visible damage to Plant equipment or Protected Area Permanent Plant Structures

OPERATIONAL CONDITION - All

BASIS

Occurrence of these event within the Protected Area, that cause visible damage to plant equipment or Protected Area Permanent Plant Structures warrant declaration as an Unusual Event under this EAL. Confirmed Explosions outside the Protected Area should not be classified under this EAL. No attempt should be made to assess the magnitude of the damage. The confirmed occurrence of the explosion with a report of damage (deformation/scorching) is sufficient for declaration. A **confirmed explosion** is defined as visual evidence that a rapid, unconfined combustion, or a catastrophic failure of pressurized equipment that imparts energy of sufficient force to damage or potentially damage permanent plant structures, systems or components.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to Alert if the explosion damages more than one safety systems or damages any plant vital structure.

DISCUSSION

Electrical failures (such as shorts, grounds, arcing, etc.) should not be considered an explosion; however, they should be evaluated for the possibility of a fire. Any security aspects of this event should be considered under EAL sections covering Security Events.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU1.5
HCGS Fire & Medical Emergency Response; HC.FP-EO.ZZ-0001(Z)

9.0 Hazards - Internal/External

9.3 Explosion

ALERT - 9.3.2

IC Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown

EAL

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 Operating Condition

OPERATIONAL CONDITION - All

BASIS

The primary concern in this EAL is the magnitude of the explosion and the effects on **safety systems** required for the present Operating Condition. Specific system degradation is addressed in the System Malfunction EALs. A detailed assessment of system **damage** is not required prior to classification. The term "**Damage**" is defined as evidence that the explosion has caused component malfunction (pump trip, breaker trip, etc.) that may have resulted in the equipment/structure being INOPERABLE or otherwise incapable of performing it's design function. A **Safety System** is defined as any system or component included in Technical Specifications. In those cases where it is believed that the explosion may have caused damage to

Safety Systems, then an Alert declaration is warranted, since the full extent of the damage may not be known. For Plant Vital Structure **damage**, classification is required under this EAL if the structure houses or otherwise supports **safety systems** required for the present Operating Condition.

A **confirmed explosion** is defined as visual evidence that a rapid, unconfined combustion, or a catastrophic failure of pressurized equipment that imparts energy of sufficient force to damage or potentially damage permanent plant structures, systems or components.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate based on further damage to plant safety systems, loss of fission product barriers, or abnormal radiological releases.

DISCUSSION

No lengthy and timely assessment of damage is required prior to classification. In this EAL, no attempt is made to quantify the magnitude of the damage to any safety system but instead an attempt is made to identify any damage in order to quantify the magnitude and extent of the explosion. In short, if the explosion is big enough that it has damaged more than one safety system, or more than one subsystem of a safety system, then the explosion is big enough to justify an Alert declaration. Damage to Plant Vital Structures must be to the extent that EC judgement must be used to determine if the structure is still capable of performing its design function. Electrical failures (such as shorts, grounds, arcing, etc.) should not be considered an explosion; however, they should be evaluated for the possibility of a fire. Any security aspects of this event should be considered under EAL sections covering Security Events.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA2
HCGS Fire & Medical Emergency Response; HC.FP-EO.ZZ-0001(Z)

9.0 Hazards - Internal/External

9.4 Toxic Gases

UNUSUAL EVENT - 9.4.1.a

IC Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant

EAL

Notification by Local, County, or State Officials for the potential need to EVACUATE non-essential personnel due to an Offsite **Toxic Gas** release

AND

SNSS deems evacuation of non-essential personnel is required

OPERATIONAL CONDITION - All

BASIS

Notification by Local, County, or State Officials for the potential need to EVACUATE non-essential personnel due to an Offsite Toxic Gas release, along with SNSS concurrence that such action is appropriate warrants declaration of an Unusual Event, since a release that has occurred offsite, may have an impact on routine plant operations. An offsite event (such as a tanker accident or a barge accident) may place the Protected Area within the evacuation area. The evacuation is determined from the DOT Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials. A **Toxic Gas** is considered to be any substance that is dangerous to life or limb by reason of inhalation or skin contact. A **Toxic Gas** release is considered to be a threat to plant personnel if concentrations are high enough to endanger the health of those personnel.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert if the Toxic Gas enters either a Plant Vital Area or an area contiguous to a Plant Vital Area.

DISCUSSION

None

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU3.1 and HU3.2
HC.OP-AB.ZZ-0129 (Q), High Radiation, Smoke, or Toxic Gases in the Control Room Air Supply
HCGS Technical Specifications Section 3/4 7-6, Control Room Emergency Filtration System

9.0 Hazards - Internal/External

9.4 Toxic Gases

UNUSUAL EVENT - 9.4.1.b

IC Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant

EAL

Uncontrolled Toxic Gas release within the Protected Area in ANY area which does not normally require an atmospheric survey or Respiratory Protection for entry **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

OPERATIONAL CONDITION - All

BASIS

An **uncontrolled Toxic Gas** release within the Protected Area, in high enough concentrations, will adversely affect the health and safety of plant personnel, along with the safe operation of the plant. This EAL specifically addresses those areas within the Protected Area that do not normally require an atmospheric survey or Respiratory Protection for entry, since the atmosphere in an area that does require an atmospheric survey or Respiratory Protection does not meet the intent of this EAL. Releases classified under this EAL include those that originate both onsite and offsite. A **Toxic Gas** is considered to be any substance that is dangerous to life or limb by reason of inhalation or skin contact. **Uncontrolled Toxic Gas** releases are considered to be those releases that can not be isolated / confined to a single compartment or area, or are not as the result of a designed plant safety feature. For example, an **uncontrolled release** of chlorine/ammonia into the Turbine Building warrants declaration of an Unusual Event. A Cardox discharge inside any area that contains this safety feature (i.e. Diesel Bays) does not warrant Unusual Event declaration, unless personnel injuries have occurred as a direct result of the discharge. A **Toxic Gas** release is considered to be IMPEDING normal plant operations if concentrations are high enough to restrict routine operator movements. **Access restrictions** includes t... onditions where access is only possible with appropriate personnel protection equipment, since ...s equipment restricts normal vision and mobility.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert if the Flammable Gas enter either a Plant Vital Area or an area contiguous to a Plant Vital Area.

DISCUSSION

This EAL should not be construed to include confined spaces that must be ventilated prior to entry or situations involving Site Protection personnel who are using respiratory equipment during the performance of their duties unless it also affects personnel not involved with Site Protection activities. These areas include the Drywell (when inerted) and ALL Confined Spaces. In addition, those situations that require personnel to wear respiratory protection equipment as the result of airborne contamination as required by Radiation Protection personnel do not meet the intent of this EAL.

An offsite event (such as a tanker accident or a barge accident) may place the Protected Area within the evacuation area. The evacuation is determined from the DOT Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU3.1 and HU3.2
HC.OP-AB.ZZ-0129 (Q), High Radiation, Smoke, or Toxic Gases in the Control Room Air Supply
HCGS Technical Specifications Section 3/4 7-6, Control Room Emergency Filtration System

9.0 Hazards - Internal/External

9.4 Toxic Gases

UNUSUAL EVENT - 9.4.1.c

IC Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant

EAL

Uncontrolled Flammable Gas release within the Protected Area that RESULTS in Flammable Gas concentrations EXCEEDING **25% of the LEL** **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

OPERATIONAL CONDITION - All

BASIS

An **uncontrolled Flammable Gas** release within the Protected Area, in high enough concentrations, will adversely affect the health and safety of plant personnel, along with the safe operation of the plant. This EAL specifically addresses those conditions where a Flammable Gas concentration EXCEEDING **25% of the LEL** exists anywhere within the Protected Area. Releases classified under this EAL include those that originate both onsite and offsite. A **Flammable Gas** is considered to be any substance that can result in an ignition, sustained burn or detonation. **Uncontrolled Flammable Gas** releases are considered to be those releases that can not be isolated / confined to a single compartment or area. For example, an **uncontrolled release** of hydrogen into the Turbine Building in concentration exceeding 25% of the LEL (Lower Explosive Limit) warrants declaration of an Unusual Event. In comparison, a controlled release of Hydrogen during Generator purging or Hydrogen Tank trailer purging does not warrant event declaration, as these evolutions are controlled. **Flammable Gas** release is considered to be IMPEDING normal plant operations if concentrations are high enough to restrict routine operator movements. **Access restrictions** includes those conditions where access is only possible with appropriate personnel protection equipment, since this equipment restricts normal vision and mobility.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert if the Flammable Gas enter either a Plant Vital Area or an area contiguous to a Plant Vital Area.

DISCUSSION

For Hydrogen Gas, the explosive limit is 4%. Hence, a threshold of 25% of the LEL equates to 1% Hydrogen. This EAL should not be construed to include those controlled evolutions that may discharge a Flammable Gas within the Protected Area, but present no danger to plant safety, since the evolution is planned and controlled.

An offsite event (such as a tanker accident or a barge accident) may place the Protected Area within the evacuation area. The evacuation is determined from the DOT Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU3.1 and HU3.2
HC.OP-AB.ZZ-0129 (Q), High Radiation, Smoke, or Toxic Gases in the Control Room Air Supply
HCGS Technical Specifications Section 3/4 7-6, Control Room Emergency Filtration System

9.0 Hazards - Internal/External

9.4 Toxic Gases

ALERT - 9.4.2.a

IC 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

EAL

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, or dizziness
- The Lower Toxicity Limit being EXCEEDED

AND

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

OPERATIONAL CONDITION - All

BASIS

An **uncontrolled Toxic Gas** release entering any of the plant structures listed in the EAL, that threatens the ability of plant personnel to perform actions required for safe shutdown of the plant, warrants declaration of an Alert. The EAL threshold includes those conditions that present a significant challenge to plant personnel. This EAL specifically addresses only those plant structures that either contain safe shutdown equipment or are contiguous to those areas. Release classified under this EAL include those that originate both onsite and offsite. A **Toxic Gas** is considered to be any substance that is dangerous to life or limb by reason of inhalation or skin contact. **Uncontrolled Toxic Gas** releases are considered to be those releases that can not be isolated / confined to a single compartment or area, or are not as the result of a designed plant safety feature. For example, an **uncontrolled release** of chlorine/ammonia into the Turbine Building that directly effects plant personnel, warrants declaration of an Alert. A Cardox discharge inside any area that contains this safety feature (i.e. Diesel Bays) does not warrant Alert declaration, unless personnel injuries have occurred as a direct result of the discharge.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalated based on further damage to plant safety systems, loss of fission product barriers, or abnormal radiological releases.

DISCUSSION

Access is considered impeded if the Toxic Gas concentrations are life threatening, i.e. require the use of personnel protective equipment. Use of protective equipment also limits the mobility and vision. The cause or magnitude of the gas concentration is not the major concern in this EAL, but rather that access required to an area that may be impeded. An IDLH atmosphere is any atmosphere that is determined to be Immediately Dangerous to Life and Health.

This EAL should not be construed to include confined spaces that must be ventilated prior to entry or situations involving Site Protection personnel who are using respiratory equipment during the performance of their duties unless it also affects personnel not involved with Site Protection activities. These areas include the Drywell (when inerted) and ALL Confined Spaces. In addition, those situations that require personnel to wear respiratory protection equipment as the result of airborne contamination as required by Radiation Protection personnel do not meet the intent of this EAL.

An offsite event (such as a tanker accident or a barge accident) may place the Protected Area within the evacuation area. The evacuation is determined from the DOT Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA3.1 and HA3.2
HC.OP-AB.ZZ-0129 (Q), High Radiation, Smoke, or Toxic Gases in the Control Room Air Supply
HCGS Technical Specifications Section 3/4 7-6, Control Room Emergency Filtration System

9.0 Hazards - Internal/External

9.4 Toxic Gases

ALERT - 9.4.2.b

IC 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

EAL

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

OPERATIONAL CONDITION - All

BASIS

An **uncontrolled Flammable Gas** release entering any of the plant structures listed in the EAL, that threatens the ability of plant personnel to perform actions required for safe shutdown of the plant, warrants declaration of an Alert. The EAL threshold includes those conditions that present a significant challenge to plant personnel. This EAL specifically addresses only those plant structures that either contain safe shutdown equipment or are contiguous to those areas. Release classified under this EAL include those that originate both onsite and offsite. A **Flammable Gas** is considered to be any substance that is capable of being easily ignited or burning quickly. **Uncontrolled Flammable Gas** releases are considered to be those releases that can not be isolated / confined to a single compartment or area, or are not as the result of a designed plant safety feature. For example, an **uncontrolled release** of hydrogen into the Turbine Building in concentration exceeding 50% of the LEL (Lower Explosive Limit) warrants declaration of an Alert. In comparison, a controlled release of Hydrogen during Generator purging does not warrant event declaration, as this evolution is controlled.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalated based on subsequent damage to plant safety systems, loss of fission product barriers, or abnormal radiological releases.

DISCUSSION

For Hydrogen Gas, the explosive limit is 4%. Hence, a threshold of 50% of the LEL equates to 2% Hydrogen. This EAL should not be construed to include those controlled evolutions that may discharge a Flammable Gas within the Protected Area, but present no danger to plant safety, since the evolution is planned and controlled.

An offsite event (such as a tanker accident or a barge accident) may place the Protected Area within the evacuation area. The evacuation is determined from the DOT Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA3.1 and HA3.2
HC.OP-AB.ZZ-0129 (Q), High Radiation, Smoke, or Toxic Gases in the Control Room Air Supply
HCGS Technical Specifications Section 3/4 7-6, Control Room Emergency Filtration System

9.0 Hazards - Internal/External

9.5 Seismic Event

UNUSUAL EVENT - 9.5.1

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

Seismic Event felt by personnel within the Protected Area **AND** **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

OPERATIONAL CONDITION - All

BASIS

A **Valid** Actuation of the Seismic Trigger indicates that a Seismic Event of a magnitude greater than 0.01g has occurred. This threshold warrants declaration of an Unusual Event. **Valid** is defined as the Seismic Trigger actuation being the direct result of a Seismic Event. The condition that the Seismic Event has been felt by personnel within the Protected Area, provides further confirmation that an event has occurred. Classification should be based on a **Valid** actuation of the Seismic Trigger as verified in the Upper Relay Room. Additional information can be obtained by contacting the National Earthquake Center in Denver, Colorado at (303) 273-8500. However, it is important to realize that it will take the Earthquake Center approximately 30 minutes to provide the requested information. The time required to obtain this additional information should not result in a delay of event classification for a **valid** actuation.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert if the a subsequent seismic event occurred in excess of the Operating Basis Earthquake level (0.1g).

DISCUSSION

An earthquake of a magnitude equivalent to 0.01g is not expected to affect the capability of plant safety functions. This threshold value is well below the Operating Basis Earthquake level of 0.1g.

An approximate relationship between acceleration and magnitude is as follows:

An Acceleration of:	is approx. equal to a Richter Scale Magnitude of:
0.02g	4.5
0.1g	5.5
0.2g	6.5

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU1.1
HC.OP-AB.ZZ-0139 (Q), Acts of Nature
HCGS Technical Specification Section 3/4.3.7.2, Seismic Monitoring Instrumentation
HC.OP-SO.SG-0001 (Z), Seismic Instrumentation System Operation
HC.OP-AR.ZZ-0011 (Q), Overhead Annunciator Window Box C6

9.0 Hazards - Internal/External

9.5 Seismic Event

ALERT - 9.5.2

IC Natural and Destructive Phenomena Affecting the Plant Vital Area

EAL

> Seismic Event felt by personnel within the Protected Area
>
> **AND**
>
> **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

OPERATIONAL CONDITION - All

BASIS

A **Valid** Actuation of the Seismic Switch indicates that a Seismic Event of a magnitude greater than 0.1g (Operating Basis Earthquake) has occurred. At this level, plant safety systems are designed to remain functional and within design stress and deformation limits. Thus, an earthquake of this magnitude is not expected to affect the capability of plant safety functions required to shut down the plant and place it in a cold shutdown condition.
This threshold warrrants declaration of an Alert. **Valid** is defined as the Seismic Switch actuation being the direct result of a Seismic Event. The condition that the Seismic Event has been felt by personnel within the Protected Area, along with Seismic Trigger actuation provides

further confirmation that an event has occurred. Classification should be based on a **Valid** actuation of the Seismic Switch as verified in the Upper Relay Room. Additional information can be obtained by contacting the National Earthquake Center in Denver, Colorado at (303) 273-8500. However, it is important to realize that it will take the Earthquake Center approximately 30 minutes to provide the requested information. The time required to obtain this additional information should not result in a delay of event classification for a **valid** actuation.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate if the seismic event caused additional damage to plant safety systems, loss of fission product barriers, or abnormal radiological releases.

DISCUSSION

Seismic Event annunciation on panel 10C673 would alert operators to this event and the active seismic monitoring instrumentation would begin to monitor the event. This threshold value associated with this EAL is well below the Design Basis Earthquake of 0.2g that is the maximum seismic event that is expected to occur based on local geological and seismological factors.

An approximate relationship between acceleration and magnitude is as follows:

Acceleration:	Richter Scale Magnitude (approximate):
0.02g	4.5
0.1g	5.5
0.2g	6.5

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA1.1
HC.OP-AB.ZZ-0139 (Q), Acts of Nature
HCGS Technical Specification Section 3/4.3.7.2, Seismic Monitoring Instrumentation
HC.OP-SO.SG-0001 (Z), Seismic Instrumentation System Operation
HC.OP-AR.ZZ-0011 (Q), Overhead Annunciator Window Box C6

9.0 Hazards - Internal/External

9.6 High Winds

UNUSUAL EVENT - 9.6.1.a

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

Report of a Tornado TOUCHING DOWN within the Protected Area

OPERATIONAL CONDITION - All

BASIS

A tornado touching down within the Protected Area is an observed event with the potential to cause damage to structures containing systems or functions necessary for safe shutdown of the plant. As such, a tornado represents a potential degradation in the level of safety of the plant. Verification of the tornado should be by direct observation and report by plant personnel.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert if the tornado causes damage to Plant Vital Structures or affects the operability of Technical Specification required equipment

DISCUSSION

The National Weather Service can be contacted for further information about existing or projected Adverse Weather Conditions:

Wilmington	(302) 573-6142
Mount Holly	(609) 261-6604
Mount Holly	(609) 261-6602

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU1.2 and HU1.7
HC.OP-AB.ZZ-0139 (Q), Acts of Nature
HCGS Technical Specification Section 3/4, 3.7.3, Meteorological Monitoring Instrumentation
HCGS UFASR Sections 2.3, 3.3.1

9.0 Hazards - Internal/External

9.6 High Winds

UNUSUAL EVENT - 9.6.1.b

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

Sustained wind speeds $\geq$ **75 MPH for 15 minutes,** measured at ANY elevation of the Met Tower

OPERATIONAL CONDITION - All

BASIS

Sustained wind speeds of 75 MPH or greater are of sufficient velocity to have the potential to cause damage to Plant Vital Areas. These conditions are indicative of unstable weather conditions and represent a potential degradation in the level of safety of the plant. The windspeed threshold is well below the structure design basis of 108 mph, and is set at the value used to characterize Hurricane force winds. The EAL threshold is set 5 MPH ABOVE the Salem High Wind Speed threshold (70 MPH) to prevent simultaneous event classification. **Sustained** wind speed means winds in excess of the threshold value for greater than 15 minutes.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert if the high winds cause damage to Plant Vital Structures or affects the operability of Technical Specification required equipment

DISCUSSION

Verification of sustained wind speed will be by observation of meteorological tower data. The Wind Speed indication from the Met Tower instrumentation is full scale at 100 mph.

The National Weather Service can be contacted for further information about existing or projected Adverse Weather Conditions:

Wilmington	(302) 573-6142
Mount Holly	(609) 261-6604
Mount Holly	(609) 261-6602

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU1.2 and HU1.7
HC.OP-AB.ZZ-0139 (Q), Acts of Nature
HCGS Technical Specification Section 3/4, 3.7.3, Meteorological Monitoring Instrumentation
HCGS UFASR Sections 2.3, 3.3.1

9.0 H... rds - Internal/External

9.6 High Winds

ALERT - 9.6.2

IC Natural and Destructive Phenomena Affecting the Plant Vital Area

EAL

EITHER one of the following:
- Report of a Tornado TOUCHING DOWN within the Protected Area
- **Sustained** wind speeds **≥ 75 MPH for 15 minutes,** measured at ANY elevation of the Met Tower

AND
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 Operating Condition

OPERATIONAL CONDITION - All

BASIS

The primary concern in this EAL is the magnitude of the high winds and the effects on **safety systems** required for the present Operating Condition. Specific system degradation is addressed in the System Malfunction EALs. A detailed assessment of system **damage** is not required prior to classification. The term "**Damage**" is defined as evidence that the high winds has caused component malfunction (pump trip, breaker trip, etc.) or a report of visible scorching, blistering or other deformation that may have resulted in the equipment/structure being INOPERABLE or otherwise incapable of performing it's design function. A **Safety System** is defined as any system or component included in Technical Specifications. In those cases where it is believed that the high winds may have caused damage to **Safety Systems**, then an Alert declaration is warranted, since the full extent of the damage may not be known. For Plant Vital Structure **damage**, classification is required under this EAL if the structure houses or otherwise supports **safety systems** required for the present Operating Condition.

It is not intended that a lengthy engineering analysis be performed to determine if damage has affected structural design but EC judgement must determine whether to exclude minor exterior damage which does not affect the structural design capability. The EAL threshold is set 5 MPH ABOVE the Salem High Wind Speed threshold (70 MPH) to prevent simultaneous event classification. **Sustained** wind speed means winds in excess of the threshold value for greater than 15 minutes. A **Safety System** is defined as any system or component included in the Technical Specification.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate based on further damage to plant safety systems, loss of fission product barriers, or abnormal radiological releases.

DISCUSSION

The windspeed threshold is well below the structure design basis of 108 mph, and is set at the value used to characterize Hurricane force winds. The Wind Speed indication from the Met Tower instrumentation is full scale at 100 mph.

The National Weather Service can be contacted for further information about existing or projected Adverse Weather Conditions:

Wilmington	(302) 573-6142
Mount Holly	(609) 261-6604
Mount Holly	(609) 261-6602

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA1.2 and HA1.3
HC.OP-AB.ZZ-0139 (Q), Acts of Nature
HCGS Technical Specification Section 3/4, 3.7.3, Meteorological Monitoring Instrumentation
HCGS UFSAR Sections 2.3, 3.3.1

9.0 Hazards - Internal/External

9.7 Abnormal River Level

UNUSUAL EVENT - 9.7.1.a

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

River Level ≥ **98.0'**

OPERATIONAL CONDITION - All

BASIS

River level greater than 98.0' is indication of impending flood conditions. This EAL threshold is set to correspond to river level conditions that can jeopardize the level of safety of the plant due to potential flooding or loss of Service Water Intake (Ultimate Heat Sink). The high level threshold is based on the historical high river level for the site to provide adequate early notification of impending flood levels, as well as consideration for the high river level used at Salem. Even though the historical high level for both stations is the same, the Hope Creek EAL threshold is set 1' ABOVE the Salem High River Level threshold (97.0') to prevent simultaneous event classification. This is justified because the grade level at the Salem station is lower than that for Hope Creek (Salem = 100', Hope Creek = 102').

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert based on river water level reaching ≥ 101.5'.

DISCUSSION

Prior to Delaware River level reaching 98.0', flood protection measures are required by Technical Specifications and procedure at 95.0'. At this river level precautionary actions are taken, including; filling outside tanks and ensuring that perimeter flood doors are closed. These actions ensure that the facility flood protection features are in place prior to a River level which would necessitate their use. There is a long lead time associated with this level and the river level that would require a plant shutdown (99.5').

The National Weather Service can be contacted for further information about existing or projected Adverse Weather Conditions:

Wilmington	(302) 573-6142
Mount Holly	(609) 261-6604
Mount Holly	(609) 261-6602

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU1.7
HC.OP-AB.ZZ-0139 (Q), Acts of Nature
HCGS Technical Specification Section 3/4, 3.7.3, 3/4.7.1.3, 3/5.7.3
HCGS UFSAR, Section 2.4, Figure 2.4-3
SGS UFSAR, Section 2.4.11.2, Figure 3.4-1

9.0 Hazards - Internal/External

9.7 Abnormal River Level

UNUSUAL EVENT - 9.7.1.b

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

River Level ≤ **80.0'**

OPERATIONAL CONDITION - All

BASIS

River level less than 80.0' is indicative of a potential degradation in the level of plant safety based on the impact upon the service water system.. The Service Water pumps are designed to operate to a low river level of 76.0'. The low level threshold is based on the historical low river level (81.0') for Hope Creek, to provide adequate early notification of impending loss of the Ultimate Heat Sink, as well as consideration for the low river level used at Salem. The Hope Creek EAL threshold is set 1' BELOW the Salem Low River Level threshold (81.0') to prevent simultaneous event classification.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert based on river water level reaching ≤76.0'.

DISCUSSION

The National Weather Service can be contacted for further information about existing or projected Adverse Weather Conditions:

Wilmington	(302) 573-6142
Mount Holly	(609) 261-6604
Mount Holly	(609) 261-6602

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU1.7
HC.OP-AB.ZZ-0139 (Q), Acts of Nature
HCGS Technical Specification Section 3/4, 3.7.3, 3/4.7.1.3, 3/5.7.3
HCGS UFSAR, Section 2.4, Figure 2.4-3
SGS UFSAR, Section 2.4.11.2, Figure 3.4-1

9.0 Hazards - Internal/External

9.7 Abnormal River Level

ALERT - 9.7.2.a

IC Natural and Destructive Phenomena Affecting the Plant Vital Area

EAL

River Level $\geq$ **101.0'**

OPERATIONAL CONDITION - All

BASIS

This EAL indicates river level conditions that can threaten the level of safety at the plant due to flooding. The high level threshold is chose .5' below the grade of Hope Creek to ensure that site access is available when the alert is declared.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate based on damage to plant safety systems, loss of fission product barriers, or abnormal radiological releases in other EAL sections.

DISCUSSION

The Hope Creek EAL threshold is set 2' ABOVE the Salem High River Level threshold (99.0') because the grade level at the Salem station is lower than that for Hope Creek (Salem = 99.5', Hope Creek = 101.5'). Prior to Delaware River level reaching 101.5', flood protection measures are required by Technical Specifications and procedure at 95.0'. At this river level precautionary actions are taken, including; filling outside tanks and ensuring that perimeter flood doors are closed. These actions ensure that the facility flood protection features are in place prior to a River level which would necessitate their use. There is a long lead time associated with this level and the river level that would require a plant shutdown (99.5').

The National Weather Service can be contacted for further information about existing or projected Adverse Weather Conditions:

Wilmington	(302) 573-6142
Mount Holly	(609) 261-6604
Mount Holly	(609) 261-6602

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA1.7
HC.OP-AB.ZZ-0139 (Q), Acts of Nature
HCGS Technical Specification Section 3/4, 3.7.3, 3/4.7.1.3, 3/5.7.3
HCGS UFSAR, Section 2.4, Figure 2.4-3
SGS UFSAR, Section 2.4.11.2, Figure 3.4-1

9.0 Hazards - Internal/External

9.7 Abnormal River Level

ALERT - 9.7.2.b

IC Natural and Destructive Phenomena Affecting the Plant Vital Area

EAL

River Level $\leq$ **76.0'**

OPERATIONAL CONDITION - All

BASIS

River level less than 76.0' is indication that the Ultimate Heat Sink is INOPERABLE. The Service Water pumps are designed to operate to a low river level of 76.0', which corresponds to the EAL threshold. This EAL threshold is set to correspond to river level conditions that jeopardizes the level of safety of the plant due to a loss of the Ultimate Heat Sink. The low level threshold is based on the Minimum River Level required for operability of the Ultimate Heat Sink, as defined in the Technical Specification and UFSAR. The Hope Creek EAL threshold is set 2'-4" BELOW the Salem Low River Level threshold, thus preventing simultaneous event classification. The Alert Low level threshold for Salem station is based on the Salem UFSAR analysis of the minimum stillwater elevation in the vicinity of the Salem station (78.4').

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate based on damage to plant safety systems, loss of fission product barriers, or abnormal radiological releases in other EAL sections.

DISCUSSION

The National Weather Service can be contacted for further information about existing or projected Adverse Weather Conditions:

Wilmington	(302) 573-6142
Mount Holly	(609) 261-6604
Mount Holly	(609) 261-6602

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA1.7
HC.OP-AB.ZZ-0139 (Q), Acts of Nature
HCGS Technical Specification Section 3/4, 3.7.3, 3/4.7.1.3, 3/5.7.3
HCGS UFSAR, Section 2.4, Figure 2.4-3
SGS UFSAR, Section 2.4.11.2, Figure 3.4-1

9.0 Hazards - Internal/External

9.8 Flooding

UNUSUAL EVENT - 9.8.1

IC Internal Flooding in Excess of Sump Handling Capability Affecting Safety Related Areas of the Plant

EAL

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

OPERATIONAL CONDITION - All

BASIS

Visual Observation of **uncontrolled flooding** in the areas listed in the EAL represents the potential to directly impact continued safe operation of the plant. This EAL specifically addresses those areas of the plant where **uncontrolled flooding** presents a challenge **Safety System(s)**. Visual Observation of the flooding should occur prior to classification to validate any alarm conditions. **Uncontrolled flooding** is defined as event or condition that does not result from a controlled evolution. Events classified under this EAL, for example, include the effects of flooding from system malfunctions, component failures, or repair activity failures (such as a failed freeze seal). Those events that result in the flooding of an area as the direct result of a planned evolution, such as system draining in preparation for an equipment outage, do not warrant event classification, unless the draining can not be successfully terminated. **Safety System** is defined as any system or component included in the Technical Specification.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert if the flooding results in damage to equipment required to establish and/or maintain cold shutdown conditions.

DISCUSSION

For the purpose of implementing this EAL, levels in the Reactor Building that would require classification under this EAL are defined as the Maximum Normal Floor Level in the EOPs. Exceeding this level in any of the Reactor Building areas would require running all available sump pumps. If level in these areas cannot be lowered to below the 1" level, then systems discharging into this area are to be isolated, except for systems required to:

- Ensure adequate core cooling
- Shutdown the reactor
- Protect primary containment integrity
- Suppress a fire

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU1.7
HC.OP-EO.ZZ-0103 (Q)-FC, Reactor Building Control
HCGS Technical Specifications Section 3/4 7-3, Flood Protection

9.0 Hazards - Internal/External

9.8 Flooding

ALERT - 9.8.2

IC Internal Flooding Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown

EAL

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 Operating Condition

OPERATIONAL CONDITION - All

BASIS

The primary concern in this EAL is the magnitude of the internal flooding and the effects on **safety systems** required for the present Operating Condition. Specific system degradation is addressed in the System Malfunction EALs. A detailed assessment of system **damage** is not required prior to classification. The term **"Damage"** is defined as evidence that the internal flooding has caused component malfunction (pump trip, breaker trip, etc.) or a report of visible scorching, blistering or other deformation that may have resulted in the equipment/structure being INOPERABLE or otherwise incapable of performing it's design function. **A Safety System** is defined as any system or component included in Technical Specifications. In those cases where it is believed that the internal flooding may have caused damage to **Safety Systems**, then an Alert declaration is warranted, since the full extent of the damage may not be known. For Plant Vital Structure **damage**, classification is required under this EAL if the structure houses or otherwise supports **safety systems** required for the present Operating Condition.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate based on damage to plant systems, loss of fission product barriers, or abnormal radiological releases.

DISCUSSION

Degraded system performance or observation of potential for damage that could degrade system performance is used as the indicator that the safety system operability was actually affected. A report of damage should not be interpreted as mandating a lengthy and timely assessment prior to justification; there is no inference in this EAL that the actual magnitude of damage be qualified or quantified.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA1.7
HCGS Technical Specifications

9.0 Hazards - Internal/External

9.9 Turbine Failure / Vehicle - Missile Impact

UNUSUAL EVENT - 9.9.1.a

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

> 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

OPERATIONAL CONDITION - 1,2,3

BASIS

Main Turbine failure of sufficient magnitude to cause damage to the turbine casing or generator seals increases the potential for leakage of combustible/explosive gases and of combustible liquids to the Turbine Building, warrants declaration of an Unusual Event. The presence of H_2 gas in sufficient quantities may present a flammable/explosive hazard. Oil may also be present which may contribute to the flammability hazard.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate to an Alert based upon damage done by missiles generated by the failure or by any subsequent fire.

DISCUSSION

Turbine rotating component failures may also result in other direct damage to plant systems and components. Damage may rupture the turbine lubricating oil system, which would release flammable liquids to the Turbine Building. Potential rupture of the condenser and condenser tubes may cause flooding in the lower levels of the Turbine Building. This damage should be readily observable.

Escape of hydrogen gas from the generator due to a loss of seal oil pumps or turbine lube oil without a turbine rotating component failure should not be classified under this event.

DEVIATION

Modes 1,2,3 are the only Operational Conditions where Main Steam pressure is high enough to allow for Main Turbine operation.

REFERENCES

NUMARC NESP-007, HU1.6

9.0 Hazards - Internal/External

9.9 Turbine Failure / Vehicle - Missile Impact

UNUSUAL EVENT - 9.9.1.b

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

Vehicle Crash / Missile Impact with or within ANY one of the following Plant Structures: • Reactor Building • Control/Aux Building • Service Water Intake Structure • Service/Radwaste Building • Low Level Radwaste Interim Storage Facility

OPERATIONAL CONDITION - All

BASIS

A **Vehicle Crash / Missile Impact** with or within a listed Plant Structure represents a potential challenge to plant safety. Events classified under this EAL include those of a magnitude and extent that may be a potential precursor to damage to **Safety Systems**, and hence has safety significance. **Vehicle Crash** includes Aircraft, Helicopters, Ships, Barges, or any other vehicle types of sufficient size to potentially damage the structure. **Missile Impact** includes flying objects from offsite, onsite rotating equipment or turbine failure causing turbine casing penetration.

Barrier Analysis

None

ESCALATION CRITERIA

Emergency Classification will escalate to Alert if the vehicle crash or missile impact causes damage to Plant Vital Structures.

DISCUSSION

Any security aspects of this event should be considered under ECG Section 9.1, Security Events.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HU1.4
NUMARC Questions and Answers, June 1993, "Hazards Question #6"

9.0 Hazards - Internal/External

9.9 Turbine Failure / Vehicle - Missile Impact

ALERT - 9.9.2

IC Natural and Destructive Phenomena Affecting the Plant Vital Area

EAL

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 Operating Condition

OPERATIONAL CONDITION - All

BASIS

The primary concern in this EAL is the magnitude of the vehicle crashes / missile impact and the effects on **safety systems** required for the present Operating Condition. Specific system degradation is addressed in the System Malfunction EALs. A detailed assessment of system **damage** is not required prior to classification. The term **"Damage"** is defined as evidence that the vehicle crashes / missile impact has caused component malfunction (pump trip, breaker trip, etc.) or a report of visible scorching, blistering or other deformation that may have resulted in the equipment/structure being INOPERABLE or otherwise incapable of performing it's design function. A **Safety System** is defined as any system or component included in Technical

Specifications. In those cases where it is believed that the vehicle crashes / missile impact may have caused damage to **Safety Systems**, then an Alert declaration is warranted, since the full extent of the damage may not be known. For Plant Vital Structure **damage**, classification is required under this EAL if the structure houses or otherwise supports **safety systems** required for the present Operating Condition.

Barrier Analysis

N/A

ESCALATION CRITERIA

Emergency Classification will escalate based on further damage to plant safety systems, fission product barriers, or abnormal radiation releases in other EAL sections.

DISCUSSION

This EAL is intended to address the threat to safety related equipment imposed by vehicle of missile impacts. No attempt should be made to assess the magnitude of damage to Safety Systems or Plant Vital Structures prior to classification. The evidence of damage is sufficient for declaration.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, HA1.5 and HA1.6
NUMARC Questions and Answers, June 1993, "Hazards Question #6"

11.0 Reportable Action Levels

11.1 Technical Specifications

REPORTABLE ACTION LEVEL - 11.1.1.a

IC INITIATION OF ANY UNIT SHUTDOWN REQUIRED BY THE TECHNICAL SPECIFICATIONS [10CFR50.72(b)(1)(i)(A); 10CFR50.36(c)(1)]

RAL

Unit shutdown is INITIATED to comply with Technical Specifications

OPERATIONAL CONDITION - 1, 2

BASIS

A **Unit Shutdown** initiated to comply with Technical Specification requires a one hour report in accordance with 10CFR50.72(b)(1)(i)(A). This RAL is intended to capture those events for which a Technical Specification required shutdown is initiated Thus, this RAL ensures that the NRC is provided with early warning of safety significant conditions serious enough to warrant a plant shutdown. **Unit Shutdown** is defined as the performance of any action(s) to start reducing reactor power to achieve a Hot Shutdown condition.

A reduction of power for some other purpose, not constituting initiation of a shutdown required by Technical Specifications, is not reportable under this EAL. This includes reducing power only for the purpose of repairing a component.

REFERENCES

10CFR50.72(b)(1)(i)(A)
NUREG 1022, Rev.1, 2nd Draft

11.0 Reportable Action Levels

11.1 Technical Specifications

REPORTABLE ACTION LEVEL - 11.1.1.b

IC EXCEEDING ANY TECHNICAL SPECIFICATION SAFETY LIMIT [10CFR50.72(b)(1)(i)(A); 10CFR50.36(c)(1)]

RAL

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

OPERATIONAL CONDITION - 1, 2, 3, 4, 5

BASIS

This RAL adddresses those conditions requiring a one hour report in accordance with 10CFR50.36(c)(1) which states that exceeding a Technical Specification Safety limit requires a shutdown by Technical Specification. Exceeding a Safety Limit in Technical Specification Section 2.1 in the Operational Condition that the Safety Limit is applicable shall be reported under this RAL.

REFERENCES

10CFR50.36(c)(1)

11.0 Reportable Action Levels

11.1 Technical Specifications

REPORTABLE ACTION LEVEL - 11.1.1.c

IC ANY DEVIATION FROM T/S OR LICENSE CONDITION PURSUANT TO 10CFR50.54(x) [10CFR50.72(b)(1)(i)(B)]

RAL

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 procedures) NOTE: Such action must be approved by at least a licensed SRO

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions that require a one hour report in accordance with 10CFR50.72(b)(1)(i)(B). 10CFR50.54(x) generally permits licensees to take reasonable action in an emergency even though the action departs from license conditions or plant Technical Specifications if 1) the action is immediately needed to protect the public health and safety, and 2) no action consistent with the license conditions and Technical Specifications is immediately apparent that can provide adequate or equivalent protection. Such action requires, at a minimum, prior approval by a licensed Senior Reactor Operator. Refer to NC.NA-AP.ZZ-0005(Q), Station Operating Practices, for more information concerning the use of 10CFR50.54(x).

REFERENCES

10CFR50.54(x)
10CFR50.54(y)
10CFR50.72(b)(1)(i)(B)
NC.NA-AP.ZZ-0005(Q)
NUREG 1022, Rev. 1, 2nd Draft.

11.0 Reportable Action Levels

11.1 Technical Specifications

REPORTABLE ACTION LEVEL - 11.1.2.a

IC VIOLATION OF THE REQUIREMENTS CONTAINED IN THE OPERATING LICENSE [HCGS Operating License, Sections 2.F]

RAL

Violation of the requirements contained in Section 2.C (Items 3 through 13) of the Operating License except as otherwise provided in the Technical Specifications or Environmental Protection Plan

OPERATIONAL CONDITION - All

BASIS

This RAL expresses the conditions for a twenty-four hour report in accordance with Item 2.F of the Hope Creek Operating License.

REFERENCES

HCGS Technical Specification

11.0 Reportable Action Levels

11.1 Technical Specifications

REPORTABLE ACTION LEVEL - 11.1.2.b

IC ANY EVENT REQUIRING AN ENGINEERING EVALUATION BY TECHNICAL SPECIFICATIONS OR COMMITMENT [T/S 3.4.6.1, 3.4.4, 3.7.5]

RAL

Any of the T/S LCOs for RCS heatup or cooldown rates are exceeded (T/S 3.4.6.1)

OPERATIONAL CONDITION - All

BASIS

Conditions reported under this RAL require an engineering evaluation of the effects of the condition on plant materials and future operation. This RAL ensures that timely internal notification is initiated to implement th. . aluations.

REFERENCES

HCGS Technical Specification 3.4.6.1

11.0 Reportable Action Levels

11.1 Technical Specifications

REPORTABLE ACTION LEVEL - 11.1.2.c

IC ANY EVENT REQUIRING AN ENGINEERING EVALUATION BY TECHNICAL SPECIFICATIONS OR COMMITMENT
[T/S 3.4.6.1, 3.4.4, 3.7.5]

RAL

The conductivity, chloride concentration or pH in the RCS is in excess of its specified limits per T/S 3.4.4, thereby requiring an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the RCS

OPERATIONAL CONDITION - All

BASIS

Conditions reported under this RAL require an engineering evaluation of the effects of the condition on plant materials and future operation. This RAL ensures that timely internal notification is initiated to implement the evaluations.

REFERENCES

HCGS Technical Specification 3.4.4

11.0 Reportable Action Levels

11.1 Technical Specifications

REPORTABLE ACTION LEVEL - 11.1.2.d

IC ANY EVENT REQUIRING AN ENGINEERING EVALUATION BY TECHNICAL SPECIFICATIONS OR COMMITMENT
[T/S 3.4.6.1, 3.4.4, 3.7.5]

RAL

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

OPERATIONAL CONDITION - All

BASIS

Conditions reported under this RAL require an engineering evaluation of the effects of the condition on plant materials and future operation. This RAL ensures that timely internal notification is initiated to implement the evaluations.

REFERENCES

HCGS Technical Specification 3.7.5

11.0 Reportable Action Levels

11.2 Design Basis / Unanalyzed Condition

REPORTABLE ACTION LEVEL - 11.2.1

IC ANY EVENT OR CONDITION DURING OPERATION THAT RESULTS IN THE CONDITION OF THE PLANT BEING SERIOUSLY DEGRADED [10CFR50.72(b)(1)(ii)]

RAL

As judged by the SNSS/EDO, an event or condition found during plant operations that results in ANY one of the following: • The condition of the plant, including its principle safety barriers, being seriously degraded. • The plant being in an unanalyzed condition that significantly compromises plant safety. • The plant being in a condition outside the design basis of the plant. • The plant being in a condition not covered by normal/abnormal or emergency operating procedures.

OPERATIONAL CONDITION - 1, 2

BASIS

Reporting at the component, system, and structure level is required per the above condition.

The condition of the plant, including its principle safety barriers, being seriously degraded includes material (e.g., metallurgical or chemical) problems that cause abnormal degradation of the principle safety barriers, (Fuel Clad, RCS, Containment). Examples include:

- Fuel clad failure in reactor or spent fuel pool that exceed expected values, are unique or wide spread, are caused by unexpected factors and involve a release of significant quantities of fission products.
- Cracks and breaks in RCS piping, reactor vessel or major RCS components.
- Significant welding or material defects in the RCS.
- Serious temperature or pressure transients.
- Loss of relief/safety valve functions.
- Loss of containment integrity including excessive containment leakage, loss of containment isolation valve function, loss of containment cooling.

The plant being in an unanalyzed condition that significantly compromises plant safety refers to conditions potentially affecting a system, structure or component which are more than of a minor safety significance. It is not intended that this Action level (RAL) apply to minor variation in Parameters or to problems concerning single pieces of equipment. The NRC understand that PSE&G will use engineering judgement and experience to determine if an unanalyzed condition exist.

If when applying engineering judgement there is doubt as to whether to report or not the NRC recommends that the licensee make the report.

The plant being in a condition that is outside design bases would include errors found in the actual design of structures, systems or components which perform safety functions. It would not include minor infractions such as:

- Cases of technical inoperability where a component is declared inoperable because of surveillance is overdue.
- Case where LCO allowed outage time is slightly exceeded.

Example of conditions that would be reportable under this RAL include:

- Discovery that an ECCS design does not meet single failure criteria
- Discovery that require high energy line break restraints not being installed.
- One train of a safety systems has been incapable of performing its design function for an extended time.

REFERENCES

10CFR50.72(b)(1)(ii)
NUREG 1022, Rev. 1, 2nd Draft.

11.0 Reportable Action Levels

11.2 Design Basis / Unanalyzed Condition

REPORTABLE ACTION LEVEL - 11.2.2.a

IC ANY EVENT FOUND WHILE SHUTDOWN THAT WOULD HAVE SERIOUSLY DEGRADED THE PLANT OR RESULTED IN BEING IN AN UNANALYZED CONDITION [10CFR50.72(b)(2)(i)]

RAL

Any event, found while the reactor is shutdown, that, if had it been found during operation, would have resulted in the plant, including its principle safety barriers being in EITHER one of the following conditions: • Seriously degraded • In an unanalyzed condition that significantly compromises Plant safety

OPERATIONAL CONDITION - 3,4,5, Defueled

BASIS

See RAL 11.2.1 for more information concerning the two plant condition described in the above RAL.

REFERENCES

10CFR50.72(b)(2)(i)
NUREG 1022, Rev.1, 2nd Draft

11.0 Reportable Action Levels

11.2 Design Basis / Unanalyzed Condition

REPORTABLE ACTION LEVEL - 11.2.2.b

IC EVENT/CONDITION THAT ALONE COULD HAVE PREVENTED CERTAIN SAFETY FUNCTIONS [10CFR50.72 (b)(2) (iii)]

RAL

Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to perform ANY one of the following:

- Control the release of radioactive material
- Shutdown the reactor and maintain it in a safe shutdown condition
- Remove residual heat
- Mitigate the consequences of an accident

OPERATIONAL CONDITION - All

BASIS

The intent of this RAL is to require reporting of events or conditions that could have prevented systems from performing their safety functions (actually or potentially) regardless of when the failure was discovered, whether the system was needed at the time, or whether an alternate system or means was available to perform the safety function.

The phrase "alone could have prevented" means the event or condition was, or would be, sufficient by itself to prevent the performance of the safety function(s) of a system or structure (i.e. no additional single failure is assumed or needed to prevent the function).

This RAL covers an event or condition where structures, components or trains of a Safety Systems could have failed to perform their intended functions because of:

- One or more personnel errors including procedure violations or inadequate maintenance.
- Design analysis, fabrication, equipment qualification, construction, or procedural deficiencies.
- Equipment failure if the failure constitutes a condition where there is reasonable doubt that the redundant train or channel is operable. Note: For systems with 3 or

more trains the failure of >2 trains should be reported if the functional capability of overall system is/was jeopardized.

For a single train safety system, loss of the single train would prevent the fulfillment of the safety function of that system and is therefore reportable even though the plant technical specifications may allow such a condition to exist for a limited time.

Individual component failure need not be reported under this RAL if redundant equipment in the same system was operable and available to perform the required safety function.

REFERENCES

10CFR50.72 (b)(2) (iii)
NUREG 1022, Rev. 1, 2nd Draft

11.0 Reportable Action Levels

11.2 Design Basis / Unanalyzed Condition

REPORTABLE ACTION LEVEL - 11.2.2.c

IC PRESENCE OF A LOOSE PART IN THE REACTOR COOLANT SYSTEM [Reg. Guide 1.133]

RAL

Presence of a loose part in the RCS is **Confirmed**

OPERATIONAL CONDITION - ALL

BASIS

This RAL expresses the conditions requiring a prompt notification with written followup report of operating information in accordance with Regulatory Guides 1.133 and 1.16. Presence of a loose part maybe indicated by an overhead alarm and can be monitored both visually and audibly on the on the Loose Parts Monitor (LPM).

The presence of a loose part (i.e., disengaged and drifting) in the primary coolant system can be indication of degraded reactor safety resulting from failure or weakening of a safety restraint component. Loose parts may also come from an item left in the RCS during refueling, or maintenance and can contribute to component damage and material wear by frequently impacting on other parts of the system. In addition, loose parts can pose a serious threat to flow blockage which could lead to localized cladding failure or control rod jamming.

Confirmed indicates that an evaluation of a loose parts alarm has determined that the alarm is due to a loose part and not to detected failure or other plant events.

REFERENCES

Reg. Guide 1.16
Reg. Guide 1.133

11.0 Reportable Action Levels

11.3 Engineered Safety Features

REPORTABLE ACTION LEVEL - 11.3.1

IC Any Event that results or should have resulted in ECCS Discharge into the RCS as the result of a **VALID** signal [10CFR50.72(b)(1)(iv)]

RAL

Valid ECCS Actuation, Manual or Automatic, has or should have occurred <u>AND</u> ECCS Actuation resulted in or should have resulted in, discharge to the vessel

OPERATIONAL CONDITION - All

BASIS

NRC experience has shown that events that involve ECCS discharge to the vessel are generally more serious than ESF actuations without discharge to the vessel and thus warrant a one-hour report. Those events that result in either automatic or manual actuation of ECCS <u>or</u> would have resulted in actuation of the ECCS if some component had not failed or an operator action had not been taken are reportable. For example, if a valid ECCS signal was generated by plant conditions and the operator put all ECCS pumps in pull-to-lock position, although no ECCS discharge to the vessel occurred, the event is reportable. A valid signal refers to an intentional manual actuation or actual plant conditions or parameters satisfying the requirements for ECCS initiation. Excluded from this reporting requirement would be those instances in which instruments drift, spurious signals, human error or other invalid signal causing action (e.g. jarring a cabinet, an error in the use of jumpers or lifted leads, error in actuation of controls or switches, or equipment failures). If the ECCS discharges or should have discharged into the RPV as result of an <u>INVALID</u> signal then a report under this RAL is not required however RAL 11.3.2 (ESF Actuation) should be reviewed for applicability.

REFERENCES

NC.NA-AP-0006(Q)

HCGS UFSAR
10CFR50.72(b)(1)(iv)
10CFR50.73
NUREG 1022, Rev. 1

11.0 Reportable Action Levels

11.3 Engineered Safety Features

REPORTABLE ACTION LEVEL - 11.3.2

IC ACTUATION OF ENGINEERED SAFETY FEATURE (INCLUDING THE REACTOR PROTECTION SYSTEM) EXCEPT PREPLANNED [10CFR50.72(b)(2)(ii)]

RAL

Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), except as part of a preplanned sequence during operation or testing, including the Reactor Protection System (RPS) (Manual or Automatic Scram)

AND

ESF / RPS Actuation is determined to be reportable in accordance with NC.NA-AP-0006(Q)

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions requiring a four hour report in accordance with 10CFR50.72(b)(2)(ii). All ESF actuations, including those of the RPS, are reportable regardless of the plant operating mode or power level, the significance of the structure, system, or component that initiated the event, or whether initiated manually or automatically. The fact that the safety analysis assumes that an ESF system will actuate automatically under certain plant conditions does not preclude the need to report such actuations.

The following exceptions apply:

1 Actuations that result from and are part of the preplanned sequence during testing or reactor operation. This implies that the procedural step indicates the specific ESF RPS actuation that will be generated, and control room personnel are aware of the specific signal generation before its occurrence or indication in the control room.

However, if the ESF actuates during the planned operation or test in such a way that it is not part of the planned procedure, such as at a wrong step, that event is reportable.

2. Invalid actuations that occur when a system has been properly removed from service if all requirements of plant procedures for removing equipment from service have been met. This would include required clearance documentation, equipment and control board tagging, and properly positioned valves and power supply breakers.

NC.NA-AP-0006(Q), Incident Report/Reportable Event Program and Quality/Safety Concern Reporting System, Attachment 4 provides additional guidance on the reportability and reporting requirements for such events.

REFERENCES

NC.NA-AP-0006(Q)
HCGS UFSAR
10CFR50.72(b)(1)(iv)
10CFR50.73
NUREG 1022, Rev. 1, 2nd Draft

11.0 Reportable Action Levels

11.4 Personnel Safety / Overexposure

REPORTABLE ACTION LEVEL - 11.4.1

IC ANY INCIDENT OR EVENT INVOLVING BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL CAUSING ANY OF THE LISTED RESULTS [10CFR20.2202(a)(1); 10CFR20 App. B]

RAL

PERSONNEL OVEREXPOSURE or potential for overexposure as indicated by <u>ANY</u> 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 applies to areas where personnel are not normally stationed during routine operations

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions requiring an immediate report in accordance with 10CFR20.2202(a)(1). Annual Limits on Intake are discussed in Appendix B of 10CFR20.

<u>Terms</u>:

TEDE =	Total Effective Dose Equivalent (integrated dose that consists of the sums of the external dose equivalent, committed effective dose equivalent and 4-day deposition exposure)
LDE =	Lens Dose Equivalent (dose equivalent to the eye)
SDE =	Shallow Dose Equivalent (dose equivalent to the skin or extremities)
CEDE =	Committed Effective Dose Equivalent
ALI =	Annual Limit of Intake

REFERENCES

10CFR20.2202(a)(1)
10CFR20 App. B

11.0 Reportable Action Levels

11.4 Personnel Safety / Overexposure

REPORTABLE ACTION LEVEL - 11.4.2.a

IC ANY INCIDENT OR EVENT INVOLVING BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL CAUSING ANY OF THE LISTED RESULTS [10CFR20.2202(b)(1)]

RAL

PERSONNEL OVEREXPOSURE or potential for overexposure as indicated by <u>ANY</u> 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 applies to areas where personnel are not normally stationed during routine operations.

OPERATIONAL CONDITION - ALL

BASIS

This RAL addresses those conditions requiring a 24 hour report in accordance with 10CFR20.2202(b)(1). Annual Limits on Intake are discussed in Appendix B of 10CFR20. Because events that result in personnel overexposure may result in media interest or notifications to other government agencies, the RAL will result in a 4 Hr. report in accordance with 10CFR50.72(b)(2)(vi).

<u>Terms</u>: (The below listed terms are defined in RAL 11.3.1.f)

TEDE =	Total Effective Dose Equivalent
LDE =	Lens Dose Equivalent
SDE =	Shallow Dose Equivalent
CEDE =	Committed Effective Dose Equivalent
ALI =	Annual Limit of Intake

REFERENCES

10CFR20.2202(b)(1)
10CFR20 App. B
10CFR50.72(b)(2)(vi)

11.0 Reportable Action Levels

11.4 Personnel Safety / Overexposure

REPORTABLE ACTION LEVEL - 11.4.2.b

IC ONSITE FATALITY [10CFR50.72(b)(2)(vi)]

RAL

Any fatality has occurred onsite (within the owner controlled area)

OPERATIONAL CONDITION - All

BASIS

The above condition is reportable because an onsite fatality will most likely involve notification of other government agencies and may involve the media. Other government agencies and the media often rely on the NRC for an independent explanation of the safety implication of events at nuclear power plants; therefore, timely NRC notification is required.

REFERENCES

10CFR50.72(b)(2)(vi)
NUREG 1022, Rev. 1, 2nd Draft

11.0 Reportable Action Levels

11.4 Personnel Safety / Overexposure

REPORTABLE ACTION LEVEL - 11.4.2.c

IC CONTAMINATED INJURED PERSON TRANSPORTED FROM THE SITE TO AN OFFSITE MEDICAL FACILITY [10CFR50.72(b)(2)(v)]

RAL

Transportation of a contaminated or **potentially contaminated individual** from the site to an offsite medical facility

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions requiring a four hour report in accordance with 10CFR50.72(b)(2)(v). Transportation of an injured, contaminated individual to an offsite medical facility has the potential for spreading the contamination to individuals and/or institutions that are not trained or prepared to deal with radioactive materials. The NRC requires notification of any event with the potential to contaminate unrestricted areas in the public domain.

A **potentially contaminated individual** means a person who, due to injuries or first aid treatments cannot be adequately surveyed for contamination prior to transport to an offsite medical facility.

REFERENCES

10CFR50.72(b)(2)(v)
NUREG 1022, Rev. 1

11.0 Reportable Action Levels

11.4 Personnel Safety / Overexposure

REPORTABLE ACTION LEVEL - 11.4.3

IC SIGNIFICANT FITNESS FOR DUTY EVENTS [10CFR26.73(a)]

RAL

Any event that is determined to be reportable by the Medical Review Officer (MRO) or designee I.A.W. PSE&G's Fitness for Duty Program (NC-NA-AP.ZZ-0042(Q)) **AND** The reportable details of the event are made available to the SNSS by the MRO or designee

OPERATIONAL CONDITION - All

BASIS

NC-NA-AP.ZZ-0042(Q) provides the guidance to determine reportability of Fitness for Duty which requires a 24 hour report in accordance with 10CFR26.73. Only the Medical Review Officer or designee may determine reportability of these events for PSE&G, unless the event has safeguards significance, in which case the determination to report is made by security.

REFERENCES

NC-NA-AP.ZZ-0042(Q)
10CFR26.73(a)

11.0 Reportable Action Levels

11.5 Environmental

REPORTABLE ACTION LEVEL - 11.5.2.a

IC SPILL/DISCHARGE OF ANY NON-RADIOACTIVE HAZARDOUS SUBSTANCE [10CFR50.72(b)(2)(vi); N.J.A.C. 7:1E]

RAL

Spill/discharge of an industrial chemical or petroleum product outside of a plant structure within the owner controlled area 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 **1 hour** and no contact with groundwater is suspected

NOTES:
This event may require 15 minute notifications. Do not delay implementation of Attachment 16. Contact licensing per ECG Attachment 9 for guidance concerning reportability as necessary.

OPERATIONAL CONDITION - All

BASIS

This RAL addresses the conditions requiring reports in accordance with PSE&G's DPCC/DCR Plan. The intent of this RAL is to direct implementation of ECG Attachment 16, which will provide direction on reportability based upon the nature of the Discharge/Spill as well as the expertise of licensing personnel concerning requirements.

REFERENCES

10CFR50.72(b)(2)(vi)
N.J.A.C. 7:1E
DPCC/DCR Plan, Part III

11.0 Reportable Action Levels

11.5 Environmental

REPORTABLE ACTION LEVEL - 11.5.2.b

IC SPILL/DISCHARGE OF ANY NON-RADIOACTIVE HAZARDOUS SUBSTANCE INTO OR UPON THE RIVER [10CFR50.72(b)(2) (vi); N.J.A.C.7:1E]

RAL

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 which may have originated from Salem or Hope Creek Station.

NOTES:
This event will require 15 minute notifications. Do not delay implementation of Attachment 16. Contact licensing per ECG Attachment 9 for guidance concerning reportability as necessary.

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions requiring reports in accordance with PSE&G's DPCC/DCR Plan. The intent of this RAL is to direct implementation of ECG Attachment 16, which will provide direction on reportability based upon the nature of the Discharge/Spill as well as the expertise of licensing personnel concerning requirements.

REFERENCES

10CFR50.72(b)(2) (vi)
N.J.A.C.7:1E
DPCC/DCR Plan, Part III

11.0 Reportable Action Levels

11.5 Environmental

REPORTABLE ACTION LEVEL - 11.5.2.c

IC UNUSUAL OR IMPORTANT ENVIRONMENTAL EVENTS [ENVIRONMENTAL PROTECTION PLAN]

RAL

As judged by the SNSS/EDO ANY one of the following events have occurred:

- Unusually large fish kill
- Protected aquatic species impinge on Circulating or Service Water intake screens (ex.; 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

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions requiring reports in accordance with the Environmental Protection Plan. Final determination or reportability will be made by Environmental Licensing as a result of implementing Attachment 15.

REFERENCES

HCGS Technical Specifications, ENVIRONMENTAL PROTECTION PLAN

11.0 Reportable Action Levels

11.5 Environmental

REPORTABLE ACTION LEVEL - 11.5.2.d

IC OVERFLOW ALARM FAILURE ON ABOVE GROUND STORAGE TANKS [10CFR50.72(b)(2)(vi), N.J.A.C7:1E]

RAL

Complete failure of ANY of the below listed storage tank level alarms:

- Circ Water Caustic Storage Tank (0A-T-500) 0DELAH-3552A
- Circ Water Sodium Hypochlorite Storage Tank (0B-T-501) 0DDLAHL-3550B
- Circ Water Sodium Hypochlorite Storage Tank (0C-T-501) 0DDLAHL-3550C
- SW Sodium Hypochlorite Storage Tank (0E-T-501) 0EQLAHL-7800B
- SW Sodium Hypochlorite Storage Tank (0F-T-501) 0EQLAHL-7800C
- Million Gallon Fuel Oil Storage Tank (00-T-516) 0JALAH-3206A
- Aux Boiler Fuel Oil Day Tank (00-T-527) 0JALAHH-3215

NOTES:
This event may require 15 minute notifications. Do not delay implementation of Attachment 16. Contact licensing per ECG Attachment 9 for guidance concerning reportability as necessary

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions requiring reports in accordance with PSE&G's DPCC/DCR Plan. The intent of this RAL is to direct implementation of Attachment 16, which will provide direction on reportability based upon the nature of the Discharge/Spill as well as the expertise of licensing personnel concerning reporting requirements.

REFERENCES

10CFR50.72(b)(2)(vi)
N.J.A.C.7:1E
DPCC/DCR Plan, Part III

11.0 Reportable Action Levels

11.6 After-the-Fact

REPORTABLE ACTION LEVEL - 11.6.1

IC EMERGENCY CONDITIONS DISCOVERED AFTER-THE-FACT

RAL

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

OPERATIONAL CONDITION - All

BASIS

In the event a condition is discovered to have occurred or existed that exceeded an Emergency Action Level threshold, but that no emergency was declared and the basis for the Emergency Classification no longer exists at the time of discovery, then a one hour report is required. This situation might arise due to a condition existing without detection by operating personnel. The NRC does not require actual declaration of the emergency classification to be necessary in these circumstances.

REFERENCES

Hope Creek ECG Introduction Section
NUREG 1022, Rev. 1

11.0 Reportable Action Levels

11.7 Security / Emergency Response Capabilities

REPORTABLE ACTION LEVEL - 11.7.1.a

IC SAFEGUARDS EVENTS THAT ARE DETERMINED TO BE NON-EMERGENCIES BUT ARE REPORTABLE TO THE NRC WITHIN ONE HOUR [10CFR73.71(b)(1)]

RAL

Any Non-Emergency safeguards event that is reportable in accordance with 10CFR73.71 as determined by Security (SCP-15)

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions requiring a one hour report in accordance with 10CFR73.71(b)(1). These non-emergency events are outlined in Security Contingency Procedure #15. The on-duty PSE&G Security Supervisor should provide information concerning the specific event.

REFERENCES

10CFR73.71(b)(1)
SCP-15

11.0 Reportable Action Levels

11.7 Security / Emergency Response Capabilities

REPORTABLE ACTION LEVEL - 11.7.1.b

IC MAJOR LOSS OF EMERGENCY ASSESSMENT CAPABILITY, OFFSITE RESPONSE CAPABILITY, OR COMMUNICATIONS CAPABILITY [10CFR50.72(b)(1)(v))]

RAL

SNSS/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 <u>ANY</u> one of the following:

- Emergency Phone System (NETS) for **> 1 hour**
- ENS for **> 1 Hour** in the Control Room, TSC, or EOF (N/A if reported by the NRC)
- Greater than or equal to **8** Offsite sirens for **> 1 Hour**
- Use of the TSC or EOF for **> 8 Hours**
- All Meteorological data (Hope Creek <u>AND</u> Salem) for **> 8 Hours**
- Site access due to Acts of Nature (snow, flood, etc.)
- SPV & NPV & FRVS plant vent radiation effluent monitors for **> 8 Hours**
- SPDS <u>OR</u> CRIDS for **> 8 Hours**
- All or most (> 75%) OHA's for **< 15 minutes**
- Concurrent multiple accident or emergency condition indicators which in the judgement of the SNSS significantly impairs assessment capabilities

OPERATIONAL CONDITION - All

BASIS

NOTE: If losses are part of a scheduled test or preplanned maintenance activity where compensatory actions have been taken, then no report is required.

1. Loss of the NETS or ENS for > 1 hour directly affects the ability to promptly notify and communicate with the NRC and/or offsite officials. Refer to ECG Section 8.2 if a total loss of communications capabilities has occurred. If notified by the NRC Operations Officer of an inoperable ENS line, no further notification is necessary.

2. Loss of Off-site sirens (>10%) represents a loss of ability to promptly notify a large portion of the population, and warrants an immediate notification. There are 71 offsite sirens in the EPZ; therefore a loss of ≥ 8 represents a 10% loss.

3. Use of the TSC and/or EOF may be vital in responding to an emergency. Loss of use of these facilities, or their supporting equipment, or ability to staff represents a significant loss of emergency response capability.

4. Loss of meteorological data for an extended period of time limits the ability to predict radiological conditions during an emergency situation. An extended loss warrants notification of the loss of this capability.

5. Limited site access may affect the ability to staff the site personnel and/or emergency response facilities, and the ability of off-site agencies to implement emergency plan requirements. If possible, notification should be made when site reaction to anticipated conditions is commenced.

6. Loss of plant vent radiation monitors for an extended period of time limits the ability to predict radiological conditions during an emergency situation. An extended loss warrants notification of the loss of this capability.

7. Loss of SPDS or CRIDS for > 8 hours is considered an event that significantly impairs safety assessments capabilities.

8. A loss of OHAs for a short period of < 15 minutes is considered a loss of emergency assessment capability. If OHAs are lost or were lost for > 15 minutes then Section 8.2 of the ECG should be referred to.

REFERENCES

10CFR50.72(b)(1)(v)
NUREG-1022

11.0 Reportable Action Levels

11.8 Public Interest

REPORTABLE ACTION LEVEL - 11.8.2.a

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

RAL

SNSS/EDO judges that an event or situation has occurred that is related to <u>ANY</u> one of the following:

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

<u>AND</u>

<u>EITHER</u> one of the following:

- A news release is planned
- Notification to a Local, State or Federal agency has been or will be made

OPERATIONAL CONDITION - All

BASIS

Events that require the NRC to respond due to media or public interest, or other government agency involvement are reportable to the NRC. Examples of the events would include, but not be limited to:

- release of contaminated tools or equipment to public areas
- non-routine releases of radioactive effluents
- inadvertent operation of the offsite siren system
- state agency contacted due to fish kill
- toxic material release from the site

PSE&G generally does not have to report media and government interaction or notify the NRC of every press release issued unless they are related to, or are perceived by the public or media to be related to, the radiological health and safety of the public or onsite personnel, or protection of the environment.

REFERENCES

10CFR50.72(b)(2)(vi)
NUREG 1022, Rev. 1

11.0 Reportable Action Levels

11.8 Public Interest

REPORTABLE ACTION LEVEL - 11.8.2.b

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

RAL

As judged by the SNSS/EDO, events which are the responsibility of PSE&G 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.

MODE - All

BASIS

This RAL expresses the conditions for a four hour report in accordance with the Lower Alloways Creek Township Memorandum of Understanding (MOU) with PSE&G.

REFERENCES

LAC -MOU

11.0 Reportable Action Levels

11.9 Accidental Criticality / Special Nuclear Material / Rad Material Shipments

REPORTABLE ACTION LEVEL - 11.9.1.a

IC UNPLANNED / ACCIDENTAL CRITICALITY [10CFR70.52(a)]

RAL

Any unplanned or accidental criticality

OPERATIONAL CONDITION - 1,2, 3, 4, 5

BASIS

Any unplanned or accidental criticality requires a 1 Hour report based on 10CFR 70 52(a).

REFERENCES

10CFR70.52(a)

11.0 Reportable Action Levels

11.9 Accidental Criticality / Special Nuclear Material / Rad Material Shipments

REPORTABLE ACTION LEVEL - 11.9.1.b

IC LOSS AND INVESTIGATION OF THE LOSS OF SPECIAL NUCLEAR MATERIALS/ SPENT FUEL [10CFR73.27(c), 10CFR73.71(a)]

RAL

ANY one of the following events occur involving SNM or Spent Fuel:

- Shipment of Special Nuclear Material (SNM) or Spent Fuel that is lost or unaccounted for after the estimated time of arrival
- A lost or unaccounted for shipment of SNM or Spent Fuel has been recovered or accounted for
- Results of a trace investigation of lost or unaccounted for SNM shipment are received

OPERATIONAL CONDITION - All

BASIS

This RAL address those conditions requiring a one hour report in accordance with 10CFR73.27(c) and 10CFR73.71(a). 10CFR73.71(a)(1) requires a one hour report of a shipment loss, and on recovery of a lost shipment. 10 CFR 73.27(c) requires an immediate trace investigation of lost or unaccounted for shipments and reporting in accordance with 10CFR73.71.

REFERENCES

10CFR73.27(c)
10CFR73.71(a)

11.0 Reportable Action Levels

11.9 Accidental Criticality / Special Nuclear Material / Rad Material Shipments

REPORTABLE ACTION LEVEL - 11.9.1.c

IC THEFT OR LOSS OF LICENSED MATERIAL [10CFR20.2201(a)(1)(i)]

RAL

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

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions requiring an immediate report in accordance with 10CFR20.2201(a)(1)(i).

Definitions:

Licensed material means: source material, special nuclear material, or by-product material received, possessed, used, or transferred under a general or specific license issued by the Commission pursuant to the regulations in 10CFR20.

REFERENCES

10CFR20.2201(a)(1)(i)

11.0 Reportable Action Levels

11.9 Accidental Criticality / Special Nuclear Material / Rad Material Shipments

REPORTABLE ACTION LEVEL - 11.9.1.d

IC RECEIPT OF SNM MATERIAL [10CFR73.27(b)]

RAL

Receipt of shipment of **strategic Special Nuclear Material** (SNM)

OPERATIONAL CONDITION - All

BASIS

This RAL addresses, in part, those conditions requiring an immediate report in accordance with 10CFR73.27(b).

Strategic Special Nuclear Material is Uranium 235 (contained in uranium enriched to 20% or more in the U-235 isotope), U-233 or Plutonium.

REFERENCES

10CFR73.27(b)
10CFR70.4

11.0 Reportable Action Levels

11.9 Accidental Criticality / Special Nuclear Material / Rad Material Shipments

REPORTABLE ACTION LEVEL - 11.9.1.e

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

RAL

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

- **2200 dpm/100 cm^2**
- **200 mR/hr on contact**
- **10 mR/hr at 3 feet**

OPERATIONAL CONDITION - All

BASIS

This RAL addresses those conditions requiring an immediate report in accordance with 10CFR20.1906(d). This requirement refers to values provided in 10CFR71.87(i)(1) for contamination, and to 10CFR71.47 for radiation levels. The RAL contamination level is based on the limit, adjusted for the standard swipe area used at Artificial Island. 10CFR71.87(i)(2) allows contamination levels of 10 times the above limits for Exclusive Use Shipments (Refer to 10CFR71.87(i)(2) and 71.4 for explanation) (see definition). This limit applies only to packages during or after transport, the limits as stated in the EAL apply to packages prior to transport. The radiation levels are the limit values.

DEFINITIONS:

Exclusive Use means: the sole use of a conveyance by a single consignor and for which loading and unloading are carried out with the direction of the consignor or consignee.

REFERENCES

10CFR20.1906(d)
10CFR71.4
10CFR71.47
10CFR71.87(i)(1)/(2)

11.0 Reportable Action Levels

11.9 Accidental Criticality / Special Nuclear Material / Rad Material Shipments

REPORTABLE ACTION LEVEL - 11.9.2.a

IC ACCIDENT OCCURRING DURING TRANSPORTATION OF LICENSED MATERIAL [10CFR71.5(a)(1)(v)]

RAL

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

OPERATIONAL CONDITION - All

BASIS

10CFR71.5(a)(1)(v) refers to 49CFR171.15/16 for transportation of licensed accident reporting. Note: Vehicle breakdowns or delays enroute may also be reported by the driver, but are not reportable to the NRC unless an accident is involved (cargo damage).

Definitions:

Radioactive Material means: Any item, gas, liquid, flowable solid, or material with radioactivity levels in excess of the limits for unconditional release found in Section 5.1.1. of NA-AP.ZZ-029(Q) Radioactive Material Control Program.

REFERENCES

10CFR71.5(A)(1)(V)
49CFR171.15/16

11.0 Reportable Action Levels

11.9 Accidental Criticality / Special Nuclear Material / Rad Material Shipments

REPORTABLE ACTION LEVEL - 11.9.2.b

IC INADVERTANT RELEASE OF RADIOACTIVE CONTAMINATED MATERIAL [10CFR50.72(b)(2)(vi)]

RAL

As judged by the SNSS/EDO, EITHER of the following events has occurred:

- Unusual or abnormal release of radiological effluents
- Release of radiologically contaminated tools or equipment to public areas

AND

EITHER one of the following:

- A news release is planned
- Notification to Local, State or Federal Agencies has been or will be made

OPERATIONAL CONDITION - All

BASIS

The purpose of the RAL is to ensure that the NRC is made aware of issues that will cause heightened public or government concern related to the radiological health and safety of the public or onsite personnel or protection of the environment.

Radiological effluent releases that are >2 times Technical Specifications limits are classified in accordance with ECG Section 6.

REFERENCES

10CFR50.72(b)(2)(vi)
NUREG 1022, Rev.1, 2nd Draft

NUMARC TO HOPE CREEK CROSS REFERENCE

NUMARC EAL No.	HCGS EAL No.	DEVIATIONS
AU1.1	6.1.1.d	None
AU1.2	6.1.1.c 6.2.1	None
AU1.3	6.1.1.b	None
AU1.4	6.1.1a,d	None
AU2.1	6.3.1.b	1) NUMARC states that this EAL will be applicable in all modes of operation. In other than Operational Condition 5, the RPV head will be fully tensioned, and lowering of vessel level would be classified by EALs in Section 3.0, Fission Product Barriers, or Section 8.1, Loss of Heat Removal Capability. 2) NUMARC IC AU2 includes unexpected increases in Airborne concentration in addition to plant radiation. The corresponding Hope Creek IC does not address Airborne concentration, since an increase in Airborne concentration is not addressed in the example EALs or the basis for the Unusual Event or Alert. Apparently, the Airborne concentration example EAL was deleted by NUMARC, but the corresponding IC was overlooked.
AU2.2	6.3.1.c	NUMARC IC AU2 includes unexpected increases in Airborne concentration in addition to plant radiation. The corresponding Hope Creek IC does not address Airborne concentration, since an increase in Airborne concentration is not addressed in the example EALs or the basis for the Unusual Event or Alert. Apparently, the Airborne concentration example EAL was deleted by NUMARC, but the corresponding IC was overlooked.
AU2.3	N/A	HCGS does not have dry spent fuel storage.
AU2.4	6.3.1.a	NUMARC IC AU2 includes unexpected increases in Airborne concentration in addition to plant radiation. The corresponding Hope Creek IC does not address Airborne concentration, since an increase in Airborne concentration is not addressed in the example EALs or the basis for the Unusual Event or Alert. Apparently, the Airborne concentration example EAL was deleted by NUMARC, but the corresponding IC was overlooked.

NUMARC EAL No.	HCGS EAL No.	DEVIATIONS
AA1.1	6.1.2.d	None
AA1.2	6.1.2.c 6.2.2	None
AA1.3	6.1.2.b	None
AA1.4	6.1.2.a,d	None
AA2.1	6.3.2.c	None
AA2.2	N/A	This EAL is not used since if fuel was uncovered it would readout in high radiation alarms, therefore exceeding NUMARC EAL AA21.1
AA2.3	6.3.2.d	None
AA2.4	6.3.2.d	None
AA3.1	6.3.2.b	None
AA3.2	6.3.2.a	None
AS1.1	N/A	NUMARC EAL AS1.1 (Classification based on noble gas release rate) is not desirable per the NUMARC Draft White Paper dated 7/25/94 and 9/10/94. The classification could be under-conservative if it were made on the basis of noble gas release rate. Since dose assessment would continue in either case and the classification escalated if necessary, the impact from not having this EAL would be a delay in reaching the appropriate classification. This delay was deemed to be acceptable since in significant release situations, the plant condition EALs should provide the anticipatory classifications necessary for the implementation of offsite protective measures.
AS1.2	N/A	HCGS does not have telemetered perimeters monitors
AS1.3	6.1.3.a	None - Deviation for AS1.1 is documented in this basis section
AS1.4	6.1.3.b,c	None

NUMARC EAL No.	HCGS EAL No.	DEVIATIONS
AG1.1	N/A	NUMARC EAL AG1.1 (Classification based on noble gas release rate) is not desirable per the NUMARC Draft White Paper dated 7/25/94 and 9/10/94. The classification could be under-conservative if it were made on the basis of noble gas release rate. Since dose assessment would continue in either case and the classification escalated if necessary, the impact from not having this EAL would be a delay in reaching the appropriate classification. This delay was deemed to be acceptable since in significant release situations, the plant condition EALs should provide the anticipatory classifications necessary for the implementation of offsite protective measures.
AG1.2	N/A	HCGS does not have telemetered perimeter monitors
AG1.3	6.1.4.a	None - Deviation for AG1.1 is documented in this basis section
AG1.4	6.1.4.b,c	None
FC1	3.1.3	None
FC2	3.1.1a,b	None
FC3	3.1.2	None
FC4	N/A	HCGS does not have any other site specific indications for this barrier.
FC5	3.1.4	None
RC1	3.2.2.a	None
RC1	3.2.3.a	This EAL is being maintained in the Fission Product Barrier Table for ease of use by the operators. It has been categorized as a "Potential loss" since the RCS leak is successfully isolated and an alert classification will still be made as a result of the potential loss of RCS.
RC1	3.2.3.b	This EAL is being considered a loss of the reactor coolant boundary since actuation of listed isolation system indicate a leak of significant magnitude, and an isolation failure. The classification for exceeding this EAL remains consistent with NUMARC guide lines.
RC2	3.2.2.b	None
RC3	3.1.2	None

NUMARC EAL No.	HCGS EAL No.	DEVIATIONS
RC4	3.2.1.b	None
RC5	3.2.1.a	None
RC6	3.2.4	None
PC1	3.3.2.b	None
PC1	3.3.2.a	NUMARC PC2 EAL says intentional venting per EOPs is a loss of containment. Per Hope Creek procedures the containment is vented if design pressure or explosive mixture conditions exist. Per NUMARC PC 1 this is considered a potential loss of containment. Since both conditions are essentially the same, PSE&G has decided to call this a potential loss as recommended in NUMARC PC1.
PC2	3.3.2.a	NUMARC PC2 EAL says intentional venting per EOPs is a loss of containment. Per Hope Creek procedures the containment is vented if design pressure or explosive mixture conditions exist. Per NUMARC PC 1 this is considered a potential loss of containment. Since both conditions are essentially the same, PSE&G has decided to call this a potential loss as recommended in NUMARC PC1.
PC2	3.3.4.b	NUMARC Primary Containment Barrier Example Flowchart (PC2) suggests that for the "Containment Isolation Valve Status after Containment Isolation Signal" EAL, a failure of both valves in any one line to close AND downstream pathway to the environment exists be included as a threshold for classification of an Unusual Event. in order to include the condition where the Inboard Valve fails to close and an RCS Line Break exists between the Primary Containment wall and Outboard Valve, the condition that both valves fail to close is NOT being included in the EAL. Indication of continuing flow / leakage OUTSIDE the Primary Containment will provide an adequate threshold for Event Classification, since both isolation valves must be open for continuing leakage Outside the Primary Containment, except as noted above.
PC3	3.3.3	None
PC4	3.3.1	None

NUMARC EAL No.	HCGS EAL No.	DEVIATIONS
PC5	3.3.4.a	None
PC6	3.3.5	None
HU1.1	9.5.1	None
HU1.2	9.6.1.a,b	None
HU1.3	4.1.1	None
HU1.4	9.9.1.b	None
HU1.5	9.3.1	None
HU1.6	9.9.1.a	Modes 1,2,3 are the only Operational Conditions where Main Steam pressure is high enough to allow for Main Turbine operation.
HU1.7	9.6.1.a,b 9.7.1.a,b 9.8.1	None
HU2	9.2.1	None
HU3.1	9.4.1.a,b,c	None
HU3.2	9.4.1.a,b,c	None
HU4.1	9.1.1	None
HU4.2	9.1.1	None
HU5	4.1.1	None
HA1.1	9.5.2	None
HA1.2	9.6.2	None
HA1.3	9.6.2	None
HA1.4	4.1.2	None
HA1.5	9.9.2	None
HA1.6	9.9.2	None
HA1.7	9.7.2.a,b 9.8.2	None
HA2	9.2.2 9.3.2	None

NUMARC EAL No.	HCGS EAL No.	DEVIATIONS
HA3.1	9.4.2.a,b	None
HA3.2	9.4.2.a,b	None
HA4.1	9.1.2	None
HA4.2	9.1.2	None
HA5	8.3.2	None
HA6	4.1.2	None
HS1.1	9.1.3	None
HS1.2	9.1.3	None
HS2	8.3.3	None
HS3	4.1.3	None
HG1.1	9.1.4	None
HG1.2	9.1.4	None
HG2	4.1.4	None
SU1	7.1.1	None
SU2	8.4.1	None
SU3	8.2.1.c	None
SU4.1	1.1.1.b,c	NUMARC EAL SU 4.1 suggests that the Operating Mode Applicability for this EAL is ALL. In Operational Condition 5 and Defueled, the MSIVS will be closed, thus rendering the Offgas Pretreatment Radiation Monitors unavailable for detection of increased RCS Activity. Hence, this EAL is applicable in Operational Conditions 1,2,3 and 4.
SU4.2	1.1.1.a	NUMARC EAL SU 4.2 suggests that the Operating Mode Applicability for this EAL is ALL. When the Reactor is defueled, the source term needed to achieve an RCS Activity of 4 uCi/gm Dose Equivalent I-131 is not available. Hence, this EAL is applicable in Operational Conditions 1,2,3,4 and 5.
SU5	2.1.1.a,b,d	None

NUMARC EAL No.	HCGS EAL No.	DEVIATIONS
SU5	2.1.1.c	NUMARC EAL SU5 suggests that exceeding an RCS Identified Leakage limit of 25 gpm warrants the declaration of an Unusual Event because it may be a precursor to a more serious condition. The Hope Creek Technical Specification limit for RCS Identified Leakage is 25 GPM averaged over any 24 hour period. The plant is within the Safety Envelope of the Technical Specification as long as this limit is not exceeded and hence an Unusual Event is not warranted until the limit is exceeded. This philosophy is consistent with that contained in NUMARC EAL SU2, which only requires declaration of an Unusual Event when the plant is outside the Technical Specification Safety Envelope. RCS Pressure Boundary and Unidentified Leakage that exceed the NUMARC EAL threshold will be classified as an Unusual Event, as this leakage exceeds the Technical Specification limit. In addition, NUMARC EAL SU5 appears to apply specifically to those plants that do not allow for averaging of RCS Identified Leakage over a 24 hour period. Furthermore, NUMARC Questions and Answers Document , June 1993, "General" Question #12 addresses those cases where the Technical Specification LCO has been exceeded and the required Action section has been entered (i.e. 4 Hours to identify and reduce the leakage below the limit). The EAL threshold for RCS Identified Leakage does not consider this time for Unusual Event declaration. The Q&A also states that the EAL for RCS Identified Leakage has been significantly raised from 10 to 25 gpm at some plants. Since the Hope Creek Technical Specification limit is already set at 25 gpm averaged over any 24 hour period, the EAL should not be more limiting than the Technical Specifications.
SU6	8.2.1.a,b	None
SU7	7.2.1	None
SA1	7.1.2.b	None

NUMARC EAL No.	HCGS EAL No.	DEVIATIONS
SA2	5.1.2.a,b	NUMARC EAL SA2 suggests that an Alert classification be based only on a failure of an automatic RPS scram followed by a successful manual RPS scram from the control room, with EAL SS2 escalating to a Site Area Emergency if a manual scram (RPS or ARI) fails to reduce Reactor Power below 4%. The Alert threshold is set so that unsuccessful manual RPS scrams from the control room, as well as unsuccessful automatic RPS scrams via RPS would be classified at the Alert level. This will cover those situations in which a manual RPS scram is attempted in anticipation of a continually degrading plant condition (i.e degrading Main Condenser Vacuum). In addition, this threshold will also address those situations where a manual scram is required by procedure. (i.e. stuck open SRV, Main Steam Line Hi Hi Radiation, Dual Reactor Recirc Pump trip, Power Oscillations) and the manual scram is not successful. In either case, Alert declaration is appropriate when the RPS fails to perform its intended design function. The SAE threshold is set to include automatic and manual failure (for the reasons stated above), and resulting power ≥ 4% as suggested in NUMARC EAL SS2 bases. By defining a "Successful" scram as control rod being positioned such that the Reactor will remain Shutdown under all conditions, partial scrams that result in Reactor Power below 4% would be classified as an Alert, whether automatically or manually initiated.
SA3	8.1.2	None
SA4	8.2.2.a,b	None
SA5	7.1.2.a	None
SS1	7.1.3	None
SS2	5.1.3	None
SS3	7.2.3	None

NUMARC EAL No.	HCGS EAL No.	DEVIATIONS
SS4	8.1.3.b	The NUMARC IC associated with EAL SS4 suggests that the IC should include a Complete Loss of Function needed to achieve or maintain Hot Shutdown. The NUMARC basis includes both reactivity control and decay heat removal. At Hope Creek, as with all other BWRs, the operator action of placing the Reactor Mode Switch in the Shutdown position that results in Control Rod inserting into the core such that the Reactor will remain shutdown under all conditions without boron, places the Reactor in a Hot Shutdown condition. No additional actions are required to maintain the Reactor in this condition. Systems are required and additional operator actions are required to achieve Cold Shutdown conditions. Based on this, Hope Creek has modified the NUMARC IC for SS4 to apply specifically to a total loss of decay heat removal, since reactivity control concerns are addressed under the ATWS Section. This IC and EAL are consistent with the requirements for declaration of a Site Area Emergency.
SS5	8.1.3.a	None
SS6	8.2.3	None
SG1	7.1.4.a,b	None
SG2	5.1.4	None
FB Q#7	1.1.2	None

file: C:\AIEALS\HCGSDATA\hccross.ref