Nebraska Public Power District

"Always there when you need us"

50.54(q)

NLS2009008
February 26, 2009

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Emergency Action Level Changes
Cooper Nuclear Station; Docket No. 50-298, DPR-46

Reference: NEI 99-01, Revision 5, Methodology for Development of Emergency Action Levels, February 2008

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District to request Nuclear Regulatory Commission (NRC) approval for adopting revised Emergency Action Level (EAL) documents based on Nuclear Energy Institute (NEI) 99-01, Revision 5 (Reference 1), in accordance with 10 CFR 50.54(q), for use in the Cooper Nuclear Station (CNS) Emergency Plan. Attachment 1 provides the CNS EAL Technical Bases which is an explanation and rational for each proposed EAL. Attachment 2 provides the EAL Comparison Matrix which provides a line-by-line comparison of the EAL contained in NEI 99-01, Revision 5 to the current CNS EALs. Attachment 3 provides the EAL charts for Modes 1 through 5 and Defuel Mode.

The CNS submittal does not contain any major deviations from the industry guidance provided in NEI 99-01 Revision 5. The two minor deviations, included in this submittal, are further described in Table 3 of the EAL Comparison Matrix (Attachment 2).

NRC approval is requested by May 21, 2009, with a 180-day implementation period. These proposed EAL changes have been reviewed by the necessary safety review committees (Station Operations Review Committee and Safety Review and Audit Board). This request is submitted under oath pursuant to 10 CFR 50.30(b).

By copy of this letter and its attachments, the appropriate State of Nebraska official is notified in accordance with 10 CFR 50.91(b)(1). Copies to the NRC Region IV office and the CNS Resident Inspector are also being provided in accordance with 10 CFR 50.4(b)(1).

Should you have any questions concerning this matter, please contact David Van Der Kamp, Licensing Manager, at (402) 825-2904.

COOPER NUCLEAR STATION
P.O. Box 98 / Brownville, NE 68321-0098
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AX45
NRR

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 2/26/09
(Date)

Sincerely,

Stewart B. Minahan
Vice President – Nuclear and
Chief Nuclear Officer

Wilma M Werner 2-26-09

WILMA M. WERNER
MY COMMISSION EXPIRES
October 26, 2010

STATE OF NEBRASKA GENERAL NOTARY

/dm

Attachments

cc: Regional Administrator w/attachments
USNRC - Region IV

Cooper Project Manager w/attachments
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/attachments
USNRC - CNS

Nebraska Health and Human Services w/attachments
Department of Regulation and Licensure

NPG Distribution w/o attachments

CNS Records w/attachments

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©[4]

Correspondence Number: NLS2009008

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

| COMMITMENT | COMMITMENT NUMBER | COMMITTED DATE OR OUTAGE |
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| None | | |
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NLS2009008
Attachment 1
324 Pages Total

# Attachment 1

## Emergency Action Level Technical Bases

## Cooper Nuclear Station, Docket No. 50-298, DPR-46

# Cooper Nuclear Station

# Emergency Action Level Technical Bases

Revision 0

Prepared by: ______________________ ______________
Date

Approved by: ______________________ ______________
Manager, Emergency Preparedness
Date

Table of Contents

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Table of Contents

Table of Contents

## 1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Cooper Nuclear Station (CNS). It should be used to facilitate review of the CNS EALs, provide historical documentation for future reference and serve as a resource for training. Decision-makers responsible for implementation of Emergency Plan Implementing Procedure (EPIP) Procedure 5.7.1, Emergency Classification, may use this document as a technical reference in support of EAL interpretation.

The expectation is that emergency classifications are to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes or less in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

## 2.0 DISCUSSION

### 2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the CNS Emergency Plan.

In 1992, the Nuclear Regulatory Commission (NRC) endorsed NUMARC (Nuclear Management and Resources Council )/ National Environmental Studies Project (NESP)-007 "Methodology for Development of Emergency Action Levels" as an alternative to NUREG-0654 EAL guidance.

Nuclear Engergy Institute (NEI 99-01) (NUMARC/NESP-007) Revision 5 represents the most recently accepted methodology. Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.

- Initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations (ISFSIs).
- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Using NEI 99-01 Rev. 5, CNS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

## 2.2 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon loss or potential loss of one or more of the three fission product barriers. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials; "potential loss" infers an increased probability of barrier loss and decreased certainty of maintaining the barrier.

The primary fission product barriers are:

A. Fuel Clad (FC): The Fuel Clad barrier consists of the zircalloy fuel bundle tubes that contain the fuel pellets.

B. Reactor Coolant System (RCS): The RCS barrier is the reactor coolant system pressure boundary and includes the reactor vessel and all reactor coolant system piping up to the isolation valves.

C. Containment (PC): The Primary Containment barrier includes the drywell, the wetwell (torus), their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves.

## 2.3 Emergency Classification Based on Fission Product Barrier Degradation

The following criteria are the bases for event classification related to fission product barrier loss or potential loss:

*Unusual Event:*

*Any loss or any potential loss of Primary Containment*

*Alert:*

*Any loss or any potential loss of either Fuel Clad or RCS*

*Site Area Emergency:*

*Loss or potential loss of any two barriers*

*General Emergency:*

*Loss of any two barriers and loss or potential loss of third barrier*

## 2.4 EAL Relationship to EOPs

Where possible, the EALs have been made consistent with and utilize the conditions defined in the CNS Emergency Operating Procedures (EOPs). While the symptoms that drive operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. When these symptoms are clearly representative of one of the NEI Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the EOPs, classification of emergencies using these EALs is not dependent upon EOP entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

## 2.5 Symptom-Based vs. Event-Based Approach

To the extent possible, the EALs are symptom-based. That is, the action level threshold is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. However, a purely symptom-based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

## 2.6 EAL Organization

The CNS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - EALs applicable only under Modes 1, 2 or 3 – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Startup or Power Operation mode.
  - EALs applicable only under Modes 4, 5 or Defueled – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

  The purpose of the groups is to avoid review of EALs that cannot be applicable in the current operating mode of the plant. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL-user for a given plant condition, reduces EAL-user reading burden and, thereby, speeds identification of the EAL that applies to the emergency.

- Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The proposed CNS EAL categories/subcategories and their relationship to NEI Recognition Categories are listed below.

## EAL Groups, Categories and Subcategories

| EAL Group/Category | EAL Subcategory |
|---|---|
| Any Operating Mode: | |
| A – Abnormal Rad Release / Rad Effluent | 1 – Offsite Rad Conditions<br>2 – Onsite Rad Conditions & Spent Fuel Pool Events |
| H – Hazards | 1 – Natural or Destructive Phenomena<br>2 – Fire or Explosion<br>3 – Hazardous Gas<br>4 – Security<br>5 – Control Room Evacuation<br>6 - Judgment |
| E – ISFSI | None |
| Modes 1, 2 or 3: | |
| S – System Malfunction | 1 – Loss of Power<br>2 – ATWS / Criticality<br>3 – Inability to Reach Shutdown Conditions<br>4 – Instrumentation / Communications<br>5 – Fuel Clad Degradation<br>6 – RCS Leakage |
| F – Fission Product Barrier Degradation | None |
| Modes 4, 5 or DEF: | |
| C – Cold Shutdown / Refuel System Malfunction | 1 – Loss of Power<br>2 – RPV Level<br>3 – RCS Temperature<br>4 – Communications<br>5 –Inadvertent Criticality |

The primary tool for determining the emergency classification level is the EAL classification matrix. The user of the EAL classification matrix may (but is not required to) consult the EAL Technical Bases in order to obtain additional information concerning the EALs under classification consideration. The user should consult Sections 2.7 and 2.8, and Attachments 1 and 2 of this document for such information.

## 2.7 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group, EAL category (A, C, H, S, E and F) and EAL subcategory. A summary explanation of each category and subcategory is given at the beginning of the technical bases discussions of the EALs included in the category. For each EAL, the following information is provided:

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

<u>Initiating Condition (IC)</u>

Site-specific description of the generic IC given in NEI 99-01.

<u>EAL Identifier (enclosed in rectangle)</u>

Each EAL is assigned a unique identifier to support accurate communication of the emergency classification to onsite and offsite personnel. Four characters define each EAL identifier:

1. First character (letter): Corresponds to the EAL category as described above (A, C, H, S, E or F)
2. Second character (letter): The emergency classification (G, S, A or U)
3. Third character (number): Subcategory number within the given category. Subcategories are sequentially numbered beginning with the number one (1). If a category does not have a subcategory, this character is assigned the number one (1).
4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

Classification (enclosed in rectangle):

Unusual Event (U), Alert (A), Site Area Emergency (S) or General Emergency (G)

EAL (enclosed in rectangle)

Exact wording of the EAL as it appears in the EAL classification matrix

Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operation, 2 – Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling, D - Defueled (DEF), All or N/A - Not Applicable (See Section 2.8 for operating mode definitions.).

NEI 99-01 Basis:

Provides a description of the rationale for the EAL as provided in NEI 99-01

CNS Basis:

Provides CNS-relevant information concerning the EAL

CNS Basis Reference(s):

Site-specific source documentation from which the EAL is derived

## 2.8 Operating Mode Applicability

1 Power Operation

Reactor mode switch is in RUN.

2 Startup

The mode switch is in either REFUEL (with all reactor vessel head closure bolts fully tensioned) or STARTUP/HOT STANDBY.

3 Hot Shutdown

The mode switch is in SHUTDOWN with all reactor vessel head closure bolts fully tensioned and reactor coolant temperature is > 212°F.

4 Cold Shutdown

The mode switch is in SHUTDOWN with all reactor vessel head closure bolts fully tensioned and reactor coolant temperature is ≤ 212°F.

5 Refueling

The mode switch is in either REFUEL or SHUTDOWN with one or more reactor vessel head closure bolts less than fully tensioned.

D Defueled

RPV contains no irradiated fuel.

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action is initiated in response to the condition) should be compared to the mode applicability of the EALs. If a lower or higher plant operating mode is reached before the emergency classification is made, the declaration shall be based on the mode that existed at the time the event occurred.

## 2.9 Validation of Indications, Reports and Conditions

All classifications are to be based upon valid indications, reports or conditions. Indications, reports or conditions are considered valid when they are verified by (1) an instrument channel check, or (2) indications on related or redundant indications, or (3) by direct observation by plant personnel, such that doubt related to the indication's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

## 2.10 Planned vs. Unplanned Events

Planned evolutions involve preplanning to address the limitations imposed by the condition, the performance of required surveillance testing, and the implementation of specific controls prior to knowingly entering the condition in accordance with the specific requirements of the CNS Technical Specifications. Activities which cause operation beyond that allowed by Technical Specifications, planned or unplanned, may result in an EAL threshold being met or exceeded. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the operating license. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

## 2.11 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses (e.g., coolant radiochemistry sampling) may be necessary. Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

Existing guidance for classifying transient events addresses the period of time of event recognition and classification (15 minutes). However, in cases when EAL declaration criteria may be met momentarily during the normal expected response of the plant, declaration requirements should not be considered to be met when the conditions are a part of the designed plant response, or result from appropriate operator actions.

There may be cases in which a plant condition that exceeded an EAL was not recognized at the time of occurrence but is identified well after the condition has occurred (e.g., as a result of routine log or record review), and the condition no longer exists. In these cases, an emergency should not be declared.

Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, should be applied.

## 2.12 Imminent EAL Thresholds

Although the majority of the EALs provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded. While this is particularly prudent at the higher emergency classification levels (the early classification may permit more effective implementation of protective measures), it is nonetheless applicable to all emergency classification levels.

## 3.0 REFERENCES

3.1 Developmental

3.1.1 NEI 99-01 Revision 5, Methodology for Development of Emergency Action Levels, February 2008

3.2 Implementing

3.2.1 EPIP Procedure 5.7.1 Emergency Classification

3.2.2 EAL Comparison Matrix

3.3 Commitments

None

## 4.0 DEFINITIONS & ACRONYMS

### Definitions

**Affecting Safe Shutdown**

Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable hot or cold shutdown condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

> Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in hot shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is not "affecting safe shutdown."

> Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in cold shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is "affecting safe shutdown."

**Bomb**

Refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

**Civil Disturbance**

A group of people violently protesting station operations or activities at the site.

**Confinement Boundary**

Is the barrier(s) between areas containing radioactive substances and the environment.

**Containment Closure**

Is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

**Explosion**

Is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

**Extortion**

Is an attempt to cause an action at the station by threat of force.

**Fire**

Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

**Flooding**

Flooding, as used within the EALs, describes a condition where water is entering a room faster than installed equipment is capable of removal, resulting in a rise of water level within the room.

**Hostage**

Person(s) held as leverage against the station to ensure that demands will be met by the station.

**Hostile Action**

An act toward CNS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.

Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

**Hostile Force**

One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

**Imminent:**

Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where IMMINENT timeframes are specified, they shall apply.

**Inoperable**

Not able to perform its intended function

**Intruder**

Person(s) present in a specified area without authorization.

**Intrusion**

The act of entering without authorization. Discovery of a bomb in a specified area is indication of intrusion into that area by a hostile force.

**Independent Spent Fuel Storage Installation (ISFSI)**

A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

**Normal Plant Operations**

Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

**Projectile**

An object directed toward CNS that could cause concern for its continued operability, reliability, or personnel safety.

**Protected Area**

An area which normally encompasses all controlled areas within the security protected area fence as depicted in Technical Specifications Figure 4.1-1, Site and Exclusion Area Boundaries and Low Population Zone.

**Sabotage**

Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of sabotage until this determination is made by security supervision.

**Security Condition**

Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

**Significant Transient**

An unplanned event involving any of the following:

- Runback > 25% thermal power
- Electrical load rejection > 25% full electrical load
- Reactor scram

- ECCS injection
- Thermal power oscillations > 10%

**Strike Action**

Work stoppage within the Protected Area by a body of workers to enforce compliance with demands made on CNS. The strike action must threaten to interrupt Normal Plant Operations.

**Unisolable**

A breach or leak that cannot be promptly isolated.

**Unplanned**

A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

**Valid**

An indication, report, or condition, is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

**Visible Damage**

Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

**Vital Area**

Any area, normally within the Protected Area, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

## <u>Acronyms</u>

AC ....................Alternating Current

ADS..................Automatic Depressurization System

APRM................Average Power Range Meter

ATWS................Anticipated Transient Without Scram

BIIT ...................Boron Injection Initiation Temp

BWR..................Boiling Water Reactor

CCW .................Component Cooling Water

CDE...................Committed Dose Equivalent

CFR...................Code of Federal Regulations

cps.....................Counts per Second

CRD ..................Control Rod Drive

CS .....................Core Spray

CST ...................Condensate Storage Tank

CTMT/CNMT.....Containment

DBA...................Design Basis Accident

DC .....................Direct Current

Demin................Demineralizer

DHRP ................Decay Heat Removal Pressure

DOT...................Department of Transportation

DW ....................Drywell

DWSIL..............Drywell Spray Initiation Limit

EAL ...................Emergency Action Level

ECCS ................Emergency Core Cooling System

ECL ...................Emergency Classification Level

ED .....................Emergency Director

El. ......................Elevation

EOF...................Emergency Operations Facility

EOP...................Emergency Operating Procedure

EPA ...................Environmental Protection Agency

EPG...................Emergency Procedure Guideline

EPIP ..................Emergency Plan Implementing Procedure

EPRI..................Electric Power Research Institute

ERD...................Emergency RPV Depressurization

ESF ...................Engineered Safety Feature

ESW ..................Emergency Service Water

FAA ....................Federal Aviation Administration

FBI......................Federal Bureau of Investigation

FEMA ................Federal Emergency Management Agency

FSAR.................Final Safety Analysis Report

ft .........................Feet

gal ......................Gallon(s)

GE ......................General Emergency

GPM ..................Gallons per Minute

HCTL.................Heat Capacity Temperature Limit

HCU ..................Hydraulic Control Unit

HOO..................Headquarters (NRC) Operations Officer

HPCI..................High Pressure Coolant Injection

$H_2$ .......................Hydrogen

hr ........................Hour

HX ......................Heat Exchanger

IC........................Initiating Condition

in. .......................Inch(es)

IPEEE................Individual Plant Examination of External Events (Generic Letter 88-20)

ISFSI .................Independent Spent Fuel Storage Installation

Keff.....................Effective Neutron Multiplication Factor

lb ........................Pound(s)

LCO...................Limiting Condition of Operation

LER ...................Licensee Event Report

LOCA ................Loss of Coolant Accident

LPCI ..................Low Pressure Coolant Injection

LWR..................Light Water Reactor

MDRIR ..............Minimum Debris Retention Injection Rate

MDSL ................Minimum Debris Submergence Level

MELLL...............Maximum Extended Load Line Limit

min ....................Minimum, minute

mR.....................milliRoentgen

mRem................milliRem

MSCP................Minimum Steam Cooling Pressure

MSIV .................Main Steam Isolation Valve

MSL...................Main Steam Line

MW....................Megawatt

N/A ....................Not applicable

NEI ....................Nuclear Energy Institute

NESP ................National Environmental Studies Project

NORAD ............North American Aerospace Defense Command

NPP...................Nuclear Power Plant

NR.....................Narrow Range

NRC ..................Nuclear Regulatory Commission

NSSS ................Nuclear Steam Supply System

NUMARC ..........Nuclear Management and Resources Council

$O_2$......................Oxygen

OBE...................Operating Basis Earthquake

OCA ..................Owner Controlled Area

ODCM/ODAM ...Off-site Dose Calculation (Assessment) Manual

ORO..................Off-site Response Organization

PA .....................Protected Area

PAG...................Protective Action Guideline

PC .....................Primary Containment

PCPL.................Primary Containment Pressure Limit

PMIS .................Plant Management Information System

POAH................Point of Adding Heat

PRA/PSA...........Probabilistic Risk Assessment / Probabilistic Safety Assessment

PRM ..................Process Radiation Monitor

psig....................Pounds per square inch (gauge)

PSP...................Pressure Suppression Pressure

PSTG ................Plant Specific Technical Guidelines

R........................Roentgen
RB.....................Reactor Building
RCC..................Reactor Control Console
RCIC.................Reactor Core Isolation Cooling
RCS...................Reactor Coolant System
rem....................Roentgen Equivalent Man
RETS.................Radiological Effluent Technical Specifications
RHR..................Residual Heat Removal
RPS...................Reactor Protection System
RPV...................Reactor Pressure Vessel
RWCU...............Reactor Water Cleanup
SAG...................Severe Accident Guideline
SFP...................Spent Fuel Pool
SGT...................Stand-By Gas Treatment
SBO...................Station Blackout
SLC...................Standby Liquid Control
SPDS................Safety Parameter Display System
SRO..................Senior Reactor Operator
SRV...................Safety Relief Valve
SSE...................Safe Shutdown Earthquake
TAF...................Top of Active Fuel
TEDE.................Total Effective Dose Equivalent
TSC...................Technical Support Center
UE.....................Notification Of Unusual Event
USAR................Updated Final Safety Analysis Report
WR....................Wide Range
'..........................Feet
"..........................Inches
%.......................Percent
&........................Ampersand ("and")

°F ......................Degrees Fahrenheit

> .........................Greater Than

< .........................Less Than

≥ .........................Greater Than or Equal To

≤ .........................Less Than or Equal To

## 5.0 CNS-TO-NEI 99-01 EAL CROSSREFERENCE

This cross-reference is provided to facilitate association and location of a CNS EAL within the NEI 99-01 IC/EAL identification scheme. Further information regarding the development of the CNS EALs based on the NEI guidance can be found in the EAL Comparison Matrix.

| **CNS** | **NEI 99-01** | |
|---|---|---|
| **EAL** | **IC** | **Example EAL** |
| AU1.1 | AU1 | 1 |
| AU1.2 | AU1 | 2 |
| AU1.3 | AU1 | 3 |
| AU2.1 | AU2 | 1 |
| AU2.2 | AU2 | 2 |
| AA1.1 | AA1 | 1 |
| AA1.2 | AA1 | 2 |
| AA1.3 | AA1 | 4 |
| AA2.1 | AA2 | 2 |
| AA2.2 | AA2 | 1 |
| AA2.3 | AA3 | 1 |
| AS1.1 | AS1 | 1 |
| AS1.2 | AS1 | 2 |
| AS1.3 | AS1 | 3 |
| AG1.1 | AG1 | 1 |
| AG1.2 | AG1 | 2 |
| AG1.3 | AG1 | 4 |
| CU1.1 | CU3 | 1 |
| CU1.2 | CU7 | 1 |

| CNS | NEI 99-01 | |
|---|---|---|
| **EAL** | **IC** | **Example EAL** |
| CU2.1 | CU1 | 1 |
| CU2.2 | CU2 | 1 |
| CU2.3 | CU2 | 2 |
| CU3.1 | CU4 | 1 |
| CU3.2 | CU4 | 2 |
| CU4.1 | CU6 | 1, 2 |
| CU5.1 | CU8 | 1 |
| CA1.1 | CA3 | 1 |
| CA2.1 | CA1 | 1, 2 |
| CA3.1 | CA4 | 1, 2 |
| CS2.1 | CS1 | 1 |
| CS2.2 | CS1 | 2 |
| CS2.3 | CS1 | 3 |
| CG2.1 | CG1 | 1 |
| CG2.2 | CG1 | 2 |
| FU1.1 | FU1 | 1 |
| FA1.1 | FA1 | 1 |
| FS1.1 | FS1 | 1 |
| FG1.1 | FG1 | 1 |
| HU1.1 | HU1 | 1 |
| HU1.2 | HU1 | 2 |
| HU1.3 | HU1 | 4 |
| HU1.4 | HU1 | 3 |
| HU1.5 | HU1 | 5 |

| CNS | NEI 99-01 | |
|---|---|---|
| EAL | IC | Example EAL |
| HU2.1 | HU2 | 1 |
| HU2.2 | HU2 | 2 |
| HU3.1 | HU3 | 1 |
| HU3.2 | HU3 | 2 |
| HU4.1 | HU4 | 1, 2, 3 |
| HU6.1 | HU5 | 1 |
| HA1.1 | HA1 | 1 |
| HA1.2 | HA1 | 2 |
| HA1.3 | HA1 | 4 |
| HA1.4 | HA1 | 3 |
| HA1.5 | HA1 | 6 |
| HA1.6 | HA1 | 5 |
| HA2.1 | HA2 | 1 |
| HA3.1 | HA3 | 1 |
| HA4.1 | HA4 | 1, 2 |
| HA5.1 | HA5 | 1 |
| HA6.1 | HA6 | 1 |
| HS4.1 | HS4 | 1 |
| HS5.1 | HS2 | 1 |
| HS6.1 | HS3 | 1 |
| HG4.1 | HG1 | 1, 2 |
| HG6.1 | HG2 | 1 |
| SU1.1 | SU1 | 1 |
| SU2.1 | SU8 | 1 |

| CNS | NEI 99-01 | |
|---|---|---|
| **EAL** | **IC** | **Example EAL** |
| SU3.1 | SU2 | 1 |
| SU4.1 | SU3 | 1 |
| SU4.2 | SU6 | 1, 2 |
| SU5.1 | SU4 | 1 |
| SU5.2 | SU4 | 2 |
| SU6.1 | SU5 | 1, 2 |
| SA1.1 | SA5 | 1 |
| SA2.1 | SA2 | 1 |
| SA4.1 | SA4 | 1 |
| SS1.1 | SS1 | 1 |
| SS1.2 | SS3 | 1 |
| SS2.1 | SS2 | 1 |
| SS4.1 | SS6 | 1 |
| SG1.1 | SG1 | 1 |
| SG2.1 | SG2 | 1 |
| EU1.1 | E-HU1 | 1 |

## 6.0 ATTACHMENTS

6.1 Attachment 1, EAL Bases

6.2 Attachment 2, Fission Product Barrier Loss / Potential Loss Matrix and Basis

## Category A – Abnormal Rad Release / Rad Effluent

EAL Group: ANY (EALs in this category are applicable to any plant condition)

Many EALs are based on actual or potential degradation of fission product barriers because of the elevated potential for offsite radioactivity release. Degradation of fission product barriers though is not always apparent via non-radiological symptoms. Therefore, direct indication of elevated radiological effluents or area radiation levels are appropriate symptoms for emergency classification.

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in plant may also be indicative of the failure of containment systems or preclude access to plant vital equipment necessary to ensure plant safety.

Events of this category pertain to the following subcategories:

1. Offsite Rad Conditions

   Direct indication of effluent radiation monitoring systems provides a rapid assessment mechanism to determine releases in excess of classifiable limits. Projected offsite doses, actual offsite field measurements or measured release rates via sampling indicate doses or dose rates above classifiable limits.

2. Onsite Rad Conditions & Spent Fuel Pool Events

   Sustained general area radiation levels in excess of those indicating loss of control of radioactive materials or those levels which may preclude access to vital plant areas also warrant emergency classification.

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **Any** release of gaseous or liquid radioactivity to the environment greater than two times the ODAM limits for 60 minutes or longer

**EAL:**

| | |
|---|---|
| **AU1.1** | **Unusual Event** |
| **Any** gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2) | |

**Note 2:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Table A-1 Effluent Monitor Classification Thresholds**

| | Monitor | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
|---|---|---|---|---|---|
| GASEOUS | ERP | 3.50E+08 μCi/sec | 3.50E+07 μCi/sec | 2.80E+06 μCi/sec | 2.24E+05 μCi/sec |
| GASEOUS | Rx Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.45E+05 μCi/sec | 8.48E+04 μCi/sec |
| GASEOUS | Turb Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.62E+05 μCi/sec | 9.02E+04 μCi/sec |
| GASEOUS | RW / ARW Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.64E+05 μCi/sec | 9.08E+04 μCi/sec |
| LIQUID | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| LIQUID | Service Water Effluent | ----- | ----- | 4.80E-04 μCi/cc | 4.80E-06 μCi/cc |

* with effluent discharge **not** isolated

**Mode Applicability:**

All

**NEI 99-01 Basis:**

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODAM multiples are specified in AU1.1 and AA1.1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the EAL.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

**CNS Basis:**

Releases in excess of two times the site ODAM (ref. 1) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

**CNS Basis Reference(s):**

1. Offsite Dose Assessment Manual -ODAM- For Assessment of Gaseous and Liquid Effluents at COOPER NUCLEAR STATION
2. IOP 4.15 Elevated Release Point and Building Radiation Monitoring Systems
3. COR001-18-01 Radiation Monitoring

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **Any** release of gaseous or liquid radioactivity to the environment greater than two times the ODAM limits for 60 minutes or longer

**EAL:**

**AU1.2 Unusual Event**

**Any** liquid effluent monitor reading > Table A-1 column “UE” for ≥ 60 min. (Note 2)

**AND**

Effluent discharge is **not** isolated

**Note 2:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Table A-1 Effluent Monitor Classification Thresholds**

| | Monitor | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
|---|---|---|---|---|---|
| GASEOUS | ERP | 3.50E+08 μCi/sec | 3.50E+07 μCi/sec | 2.80E+06 μCi/sec | 2.24E+05 μCi/sec |
| | Rx Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.45E+05 μCi/sec | 8.48E+04 μCi/sec |
| | Turb Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.62E+05 μCi/sec | 9.02E+04 μCi/sec |
| | RW / ARW Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.64E+05 μCi/sec | 9.08E+04 μCi/sec |
| LIQUID | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| | Service Water Effluent | ----- | ----- | 4.80E-04 μCi/cc | 4.80E-06 μCi/cc |

* with effluent discharge **not** isolated

**Mode Applicability:**

All

**NEI 99-01 Basis:**

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODAM multiples are specified in AU1.2 and AA1.2 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

. This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the EAL.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

**CNS Basis:**

Liquid releases in excess of two times the site ODAM (ref. 1) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

**CNS Basis Reference(s):**

1. Offsite Dose Assessment Manual -ODAM- For Assessment of Gaseous and Liquid Effluents at COOPER NUCLEAR STATION
2. IOP 4.15 Elevated Release Point and Building Radiation Monitoring Systems
3. COR001-18-01 Radiation Monitoring

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **Any** release of gaseous or liquid radioactivity to the environment greater than two times the ODAM limits for 60 minutes or longer

**EAL:**

| | |
|---|---|
| **AU1.3** | **Unusual Event** |
| Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODAM limits for ≥ 60 min. (Note 2) | |

**Note 2:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Mode Applicability:**

All

**NEI 99-01 Basis:**

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODAM multiples are specified in AU1.3 and AA1.3 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

. This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.

**CNS Basis**

Releases in excess of two times the site Offsite Dose Assessment Manual (ODAM) (ref. 1) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

**CNS Basis Reference(s):**

1. Offsite Dose Assessment Manual -ODAM- For Assessment of Gaseous and Liquid Effluents at COOPER NUCLEAR STATION

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODAM limits for 15 minutes or longer

**EAL:**

**AA1.1 Alert**

**Any** gaseous monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

**Note 2:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Table A-1 Effluent Monitor Classification Thresholds**

| | Monitor | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
|---|---|---|---|---|---|
| GASEOUS | ERP | 3.50E+08 μCi/sec | 3.50E+07 μCi/sec | 2.80E+06 μCi/sec | 2.24E+05 μCi/sec |
| GASEOUS | Rx Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.45E+05 μCi/sec | 8.48E+04 μCi/sec |
| GASEOUS | Turb Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.62E+05 μCi/sec | 9.02E+04 μCi/sec |
| GASEOUS | RW / ARW Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.64E+05 μCi/sec | 9.08E+04 μCi/sec |
| LIQUID | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| LIQUID | Service Water Effluent | ----- | ----- | 4.80E-04 μCi/cc | 4.80E-06 μCi/cc |

* with effluent discharge **not** isolated

**Mode Applicability:**

All

**NEI 99-01 Basis:**

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODAM multiples are specified in AU1.1 and AA1.1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

**CNS Basis**:

If the Alert thresholds were set at two-hundred times the ODAM limit, the resultant value would exceed the SAE threshold, which is based on 10% of the PAG limits. For this reason, the Alert gaseous thresholds have been set at the log-average of the UE threshold (two times ODAM limit) and the SAE threshold (CNS-DOSE dose assessment calculation) (ref. 4). This provides a reasonable escalation in classification from the UE to the Alert and SAE thresholds.

**CNS Basis Reference(s):**

1. Offsite Dose Assessment Manual -ODAM- For Assessment of Gaseous and Liquid Effluents at COOPER NUCLEAR STATION
2. IOP 4.15 Elevated Release Point and Building Radiation Monitoring Systems
3. COR001-18-01 Radiation Monitoring
4. EPIP Procedure 5.7.17 Dose Assessment, 4. Computer Dose Projection (CNS-DOSE)

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODAM limits for 15 minutes or longer

**EAL:**

**AA1.2 Alert**

**Any** liquid effluent monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

**AND**

Effluent discharge is **not** isolated

**Note 2:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Table A-1 Effluent Monitor Classification Thresholds**

| | Monitor | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
|---|---|---|---|---|---|
| GASEOUS | ERP | 3.50E+08 µCi/sec | 3.50E+07 µCi/sec | 2.80E+06 µCi/sec | 2.24E+05 µCi/sec |
| GASEOUS | Rx Bldg Vent | 3.50E+07 µCi/sec | 3.50E+06 µCi/sec | 5.45E+05 µCi/sec | 8.48E+04 µCi/sec |
| GASEOUS | Turb Bldg Vent | 3.50E+07 µCi/sec | 3.50E+06 µCi/sec | 5.62E+05 µCi/sec | 9.02E+04 µCi/sec |
| GASEOUS | RW / ARW Bldg Vent | 3.50E+07 µCi/sec | 3.50E+06 µCi/sec | 5.64E+05 µCi/sec | 9.08E+04 µCi/sec |
| LIQUID | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| LIQUID | Service Water Effluent | ----- | ----- | 4.80E-04 µCi/cc | 4.80E-06 µCi/cc |

* with effluent discharge **not** isolated

**Mode Applicability:**

All

**NEI 99-01 Basis:**

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODAM multiples are specified in AU1.2 and AA1.2 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

**CNS Basis:**

This event escalates from the Unusual Event by escalating the magnitude of the release by a factor of 100.

**CNS Basis Reference(s):**

1. Offsite Dose Assessment Manual -ODAM- For Assessment of Gaseous and Liquid Effluents at COOPER NUCLEAR STATION
2. IOP 4.15 Elevated Release Point and Building Radiation Monitoring Systems
3. COR001-18-01 Radiation Monitoring

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODAM limits for 15 minutes or longer

**EAL:**

| | |
|---|---|
| **AA1.3** | **Alert** |
| Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM limits for ≥ 15 min. (Note 2) | |

**Note 2:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

**Mode Applicability:**

All

**NEI 99-01 Basis:**

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODAM multiples are specified in AU1.3 and AA1.3 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.

**CNS Basis**:

None

**CNS Basis Reference(s):**

1. Offsite Dose Assessment Manual -ODAM- For Assessment of Gaseous and Liquid Effluents at COOPER NUCLEAR STATION

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 0.1 Rem TEDE or 0.5 Rem thyroid CDE for the actual or projected duration of the release

**EAL:**

**AS1.1 Site Area Emergency**

**Any** gaseous monitor reading > Table A-1 column "SAE" for ≥ 15 min. (Note 1)

**Note 1:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (See EAL AS1.2). Do **not** delay declaration awaiting dose assessment results.

**Table A-1 Effluent Monitor Classification Thresholds**

| | Monitor | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
|---|---|---|---|---|---|
| GASEOUS | ERP | 3.50E+08 μCi/sec | 3.50E+07 μCi/sec | 2.80E+06 μCi/sec | 2.24E+05 μCi/sec |
| GASEOUS | Rx Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.45E+05 μCi/sec | 8.48E+04 μCi/sec |
| GASEOUS | Turb Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.62E+05 μCi/sec | 9.02E+04 μCi/sec |
| GASEOUS | RW / ARW Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.64E+05 μCi/sec | 9.08E+04 μCi/sec |
| LIQUID | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| LIQUID | Service Water Effluent | ----- | ----- | 4.80E-04 μCi/cc | 4.80E-06 μCi/cc |

* with effluent discharge **not** isolated

## Mode Applicability:

All

## NEI 99-01 Basis:

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

**CNS Basis:**

The Table A-1 Site Are Emergency thresholds have been determined using CNS-DOSE dose projection calculations (ref. 1). The Site Area Emergency effluent monitor readings are one decade less than the General Emergency values.

For the purposes of this EAL, the Site Boundary for CNS is a one mile radius around the plant (ref. 2).

**CNS Basis Reference(s):**

1. EPIP Procedure 5.7.17 Dose Assessment, 4. Computer Dose Projection (CNS-DOSE)
2. CNS Drawing DWG.2.2 (P3-A-45)

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 0.1 Rem TEDE or 0.5 Rem thyroid CDE for the actual or projected duration of the release

**EAL:**

| | |
|---|---|
| **AS1.2** | **Site Area Emergency** |
| Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem thyroid CDE at or beyond the site boundary | |

**Mode Applicability:**

All

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

**CNS Basis**:

The dose rate EALs are based on a Site Boundary dose rate of 0.1 Rem/hr TEDE or 0.5 Rem/hr CDE thyroid, whichever is more limiting. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

For the purposes of this EAL the Site Boundary for CNS is a one mile radius around the plant.

**CNS Basis Reference(s):**

1. EPIP Procedure 5.7.17 Dose Assessment
2. CNS-DOSE
3. CNS Drawing DWG.2.2 (P3-A-45)

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 0.1 Rem TEDE or 0.5 Rem thyroid CDE for the actual or projected duration of the release

**EAL:**

**AS1.3 Site Area Emergency**

Field survey indicates closed window dose rates > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary

**OR**

Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

**CNS Basis**:

The 0.5 Rem integrated CDE thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. In establishing the field survey emergency action levels, a duration of one hour is assumed. Therefore, the dose rate EALs are based on a Site Boundary dose rate of 0.1 Rem/hr TEDE or 0.5 Rem for 1 hour of inhalation CDE thyroid, whichever is more limiting.

For the purposes of this EAL, the Site Boundary for CNS is a one mile radius around the plant.

**CNS Basis Reference(s):**

1. CNS Drawing DWG.2.2 (P3-A-45)

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1 Rem TEDE or 5 Rem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

| | |
|---|---|
| **AG1.1** | **General Emergency** |
| **Any** gaseous monitor reading > Table A-1 column "GE" for ≥ 15 min. (Note 1) | |

**Note 1:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (See EAL AG1.2). Do **not** delay declaration awaiting dose assessment results.

**Table A-1 Effluent Monitor Classification Thresholds**

| | Monitor | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
|---|---|---|---|---|---|
| GASEOUS | ERP | 3.50E+08 µCi/sec | 3.50E+07 µCi/sec | 2.80E+06 µCi/sec | 2.24E+05 µCi/sec |
| GASEOUS | Rx Bldg Vent | 3.50E+07 µCi/sec | 3.50E+06 µCi/sec | 5.45E+05 µCi/sec | 8.48E+04 µCi/sec |
| GASEOUS | Turb Bldg Vent | 3.50E+07 µCi/sec | 3.50E+06 µCi/sec | 5.62E+05 µCi/sec | 9.02E+04 µCi/sec |
| GASEOUS | RW / ARW Bldg Vent | 3.50E+07 µCi/sec | 3.50E+06 µCi/sec | 5.64E+05 µCi/sec | 9.08E+04 µCi/sec |
| LIQUID | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| LIQUID | Service Water Effluent | ----- | ----- | 4.80E-04 µCi/cc | 4.80E-06 µCi/cc |

* with effluent discharge **not** isolated

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

**CNS Basis:**

The Table A-1 General Emergency thresholds have been determined using CNS-DOSE dose projection calculations (ref. 1). The General Emergency effluent monitor readings are one decade greater than the Site Area Emergency values.

For the purposes of this EAL, the Site Boundary for CNS is a one mile radius around the plant (ref. 2).

**CNS Basis Reference(s):**

1. EPIP Procedure 5.7.17 Dose Assessment, 4. Computer Dose Projection (CNS-DOSE)
2. CNS Drawing DWG.2.2 (P3-A-45)

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1 Rem TEDE or 5 Rem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

| **AG1.2** | **General Emergency** |
|---|---|
| Dose assessment using actual meteorology indicates doses > 1 Rem TEDE or > 5 Rem thyroid CDE at or beyond the site boundary | |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

**CNS Basis**:

The General Emergency values are based on the boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1 Rem TEDE or 5 Rem CDE thyroid for the actual or projected duration of the release. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

For the purposes of this EAL the Site Boundary for CNS is a one mile radius around the plant.

**CNS Basis Reference(s):**

1. EPIP Procedure 5.7.17 Dose Assessment
2. CNS-DOSE
3. CNS Drawing DWG.2.2 (P3-A-45)

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 1 – Offsite Rad Conditions

**Initiating Condition:** Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1 Rem TEDE or 5 Rem thyroid CDE for the actual or projected duration of the release using actual meteorology

**EAL:**

**AG1.3 General Emergency**

Field survey results indicate closed window dose rates > 1 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary

**OR**

Analyses of field survey samples indicate thyroid CDE > 5 Rem for 1 hr of inhalation at or beyond the site boundary

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

**CNS Basis**:

The 5 Rem integrated CDE thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. In establishing the dose rate emergency action levels, a duration of one hour is assumed. Therefore, the dose rate EALs are based on a Site Boundary dose rate of 1 Rem/hr TEDE or 5 Rem for 1 hour of inhalation CDE thyroid, whichever is more limiting.

For the purposes of this EAL the Site Boundary for CNS is a one mile radius around the plant.

**CNS Reference(s):**

1. CNS Drawing DWG.2.2 (P3-A-45)

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Pool Events

**Initiating Condition:** Unplanned rise in plant radiation levels

**EAL:**

**AU2.1 Unusual Event**

Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by **any** of the following:

- LI-86 (calibrated to 1001' elev.)
- Spent fuel pool low level alarm
- Visual observation

**AND**

Area radiation monitor reading rise on RMA-RA-1 or RMA-RA-2 (or by survey)

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

The refueling pathway is the combination of refueling cavity, and spent fuel pool. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

For example, a refueling bridge ARM reading may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Generally, increased radiation monitor indications will need to combined with another indicator (or personnel report) of water loss.

For refueling events where the water level drops below the RPV flange classification would be via CU2.1. This event escalates to an Alert per AA2.1 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1-3.

**CNS Basis**:

Loss of inventory from the reactor cavity or Spent Fuel Pool (SFP) may reduce water shielding above spent fuel and cause unexpected increases in plant radiation. Classification as an Unusual Event is warranted as a precursor to a more serious event.

The major item of concern on loss of inventory from the reactor cavity and SFP is to maintain adequate water level for personnel shielding and cooling of the irradiated fuel. Normal SFP water level is 37'6 1/2" above the bottom. A low SFP level alarm can be determined by annunciator 9-4-2/A-3, “Fuel Pool Cooling Trouble,” alarming due to Annunciator Panel 25-15, “Fuel Pool Low Level at 4” below normal.” Decreases in SFP water level can also be detected through visual observation. The Skimmer Surge Tank low level alarm (annunciator 9-4-2/C-3 at 100 $ft^3$ in the skimmer surge tank, elevation 981' 3") alone may not be conclusive evidence of an uncontrolled loss of inventory from the SFP. SFP weir wall design should prevent inadvertent draining of the SFP through Fuel Pool Cooling and Demineralizer System connections. A Skimmer Surge Tank low level alarm needs to be confirmed by visual observation to determine the extent of inventory loss from the SFP (ref. 1, 2, 3).

During refueling when the RPV head is removed, Shutdown Range RPV water level instrument NBI-LI-86 is recalibrated to read vessel cavity level up to the 1001' elevation (Refuel Floor). With the reactor cavity in communication with the Spent Fuel Pool via the fuel transfer canal, uncontrolled inventory loss can be remotely monitored via this indicator. NBI-LI-86 can be used only if it has been set up to read to 1001' elev. as specified in Procedure 4.6.1, Reactor Vessel Water Level Indication (ref. 4).

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. Technical Specification Section LCO 3.7.6 (ref. 5) requires spent fuel storage pool water level be maintained at least 21 ft 6 in. over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. Technical Specification LCO 3.9.6 (ref. 6) requires RPV water level to be maintained at least 21 ft above the top of the RPV flange. During refueling, this maintains sufficient water level in the refueling cavity and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

Area radiation monitors that may indicate a loss of shielding of spent fuel in the SFP or refueling cavity include (ref. 7, 8):

- RMA-RA-1 (1448) RX BLDG FUEL POOL (HR) AREA
- RMA-RA-2 (1449) RX BLDG FUEL POOL (LR) AREA

Portable radiation monitors are routinely employed to conduct radiation surveys in the Reactor Building. This source of information should not be excluded when considering emergency classification under this EAL.

**CNS Basis Reference(s):**

1. Procedure 2.2.32 Fuel Pool Cooling and Demineralizer System
2. Procedure 2.4FPC Fuel Pool Cooling Trouble
3. Procedure 2.3, 9-4-2, C-3
4. Procedure 4.6.1 Reactor Vessel Water Level Indication
5. Technical Specification LCO 3.7.6
6. Technical Specification LCO 3.9.6
7. Procedure 2.3, 9-3-1
8. Procedure 5.1 RAD Building Radiation Trouble

**Category:** A – Radioactivity Release / Area Radiation

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Pool Events

**Initiating Condition:** Unplanned rise in plant radiation levels

**EAL:**

**AU2.2 Unusual Event**

Unplanned area radiation monitor reading or survey results rise by a factor of 1,000 over normal levels*

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in UNPLANNED increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

This EAL addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive materials. A specific list of ARMs is not required as it would restrict the applicability of the Threshold. The intent is to identify loss of control of radioactive material in any monitored area.

**CNS Basis:**

The ARMs monitor the gamma radiation levels in units of mR/hr at selected areas throughout the station. If radiation levels exceed a preset limit in any channel, the Control Room annunciator and local alarms will be energized to warn of abnormal or significantly changing radiological conditions. The alarm limit is normally set at approximately 10 times normal background for each channel (ref. 1, 2).

Routine and work specific surveys are conducted throughout the station at frequencies specified by RP management. Routine surveys are scheduled per the RP Department surveillance schedule. Work specific surveys are conducted in accordance with the Radiation Work Permit (RWP) (ref. 3).

**CNS Basis Reference(s):**

1. Procedure 2.3, 9-3-1
2. Procedure 5.1 RAD Building Radiation Trouble
3. Procedure 9.ALARA.4 Radiation Work Permits

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Pool Events

**Initiating Condition:** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the RPV

**EAL:**

**AA2.1 Alert**

Damage to irradiated fuel **OR** loss of water level (uncovering irradiated fuel outside the RPV) that causes **EITHER** of the following:

RMA-RA-1 Fuel Pool Area Rad reading > 50 R/hr

**OR**

RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant.

This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.

Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

Escalation of this emergency classification level, if appropriate, would be based on EALs in Subcategory A1.

**CNS Basis**:

When considering classification, information may come from:

- Radiation monitor readings
- Sampling and surveys
- Dose projections/calculations
- Reports from the scene regarding the extent of damage (e.g., refueling crew, radiation protection technicians)

This EAL is defined by the specific areas where irradiated fuel is located, such as the refueling cavity or Spent Fuel Pool (SFP).

The bases for the ventilation radiation Hi-Hi alarm is a spent fuel handling accident (ref. 1). Fuel Pool area radiation > 50 R/hr represents 100 times the high alarm setpoint (HR) and is unambiguously indicative of spent fuel damage or uncovery (ref. 2).

**CNS Basis Reference(s):**

1. Procedure 2.3 9-4-1, E-4
2. Procedure 2.3 9-3-1, A-10

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Pool Events

**Initiating Condition:** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the RPV

**EAL:**

| **AA2.2** | **Alert** |
|---|---|
| A water level drop in the reactor refueling cavity or spent fuel pool that will result in irradiated fuel becoming uncovered | |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant.

Escalation of this emergency classification level, if appropriate, would be based on EALs in Subcategory A1.

**CNS Basis:**

When considering classification, information may come from:

- Radiation monitor readings
- Sampling and surveys
- Dose projections/calculations
- Reports from the scene regarding the extent of damage (e.g., refueling crew, radiation protection technicians)

The major item of concern on loss of inventory from the Spent Fuel Pool and Refueling Cavity is to maintain adequate water level for personnel shielding and cooling of the irradiated fuel . Normal Spent Fuel Pool water level is 37'6 1/2" above the bottom. A low pool level alarm occurs at 4" below the normal water level. Decreases in Spent Fuel Pool water level can also be detected only through visual observation and the existence of the Skimmer Surge Tank low level alarm (9-4-2/C-3) at 100 $ft^3$ in the skimmer surge tank which is at elevation 981' 3" (ref. 1, 2, 3).

During refueling when the RPV head is removed, Shutdown Range RPV water level instrument NBI-LI-86 is recalibrated to read vessel cavity level up to the 1001' elevation (Refuel Floor). With the reactor cavity in communication with the Spent Fuel Pool via the fuel transfer canal, uncontrolled inventory loss can be remotely monitored via this indicator. NBI-LI-86 can be used only if it has been set up to read to 1001' elev. as specified in Procedure 4.6.1, Reactor Vessel Water Level Indication (ref. 4).

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. Technical Specification Section LCO 3.7.6 (ref. 5) requires spent fuel storage pool water level be maintained at least 21 ft 6 in. over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. Technical Specification LCO 3.9.6 (ref. 6) requires RPV water level to be maintained at least 21 ft above the top of the RPV flange. During refueling, this maintains sufficient water level in the refueling cavity and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

**CNS Basis Reference(s):**

1. Procedure 2.2.32 Fuel Pool Cooling and Demineralizer System
2. Procedure 2.4FPC Fuel Pool Cooling Trouble
3. Procedure 2.3, 9-4-2, C-3
4. Procedure 4.6.1 Reactor Vessel Water Level Indication
5. Technical Specification LCO 3.7.6
6. Technical Specification LCO 3.9.6

**Category:** A – Abnormal Rad Release / Rad Effluent

**Subcategory:** 2 – Onsite Rad Conditions & Spent Fuel Pool Events

**Initiating Condition:** Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

**EAL:**

**AA2.3 Alert**

Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:

Main Control Room (RM-RA-20)

**OR**

CAS

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses increased radiation levels that: impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown.

The cause and/or magnitude of the increase in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved.

. The value of 15mRem/hr is derived from the GDC 19 value of 5 Rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements", provides that the 15 mRem/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.

Areas requiring continuous occupancy include the Main Control Room and the Central Alarm Station (CAS).

**CNS Basis**:

Areas that meet this threshold include the Main Control Room and the Central Alarm Station. The Central Alarm Station is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations.

There are no permanently installed CAS area radiation monitors that may be used to assess this EAL threshold. Therefore, this portion of the EAL threshold must be assessed using local radiation survey (ref. 1, 2).

**CNS Basis Reference(s):**

1. Procedure 2.3 9.3.1, B-10
2. Procedure 5.1 RAD Building Radiation Trouble

## Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Modes 4, 5, DEF

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, Containment Closure, and fuel clad integrity for the applicable operating modes (4 - Cold Shutdown, 5 - Refueling, D – Defueled).

The events of this category pertain to the following subcategories:

1. Loss of Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 4160 V emergency buses and loss of vital 125-Volt DC power sources.

2. RPV Level

RPV water level is a measure of inventory available to ensure adequate core cooling and, therefore, maintain fuel clad integrity. The RPV provides a volume for the coolant that covers the reactor core. The RPV and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

3. RCS Temperature

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of safety functions.

4. Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

5. Inadvertent Criticality

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 1 – Loss of Power

**Initiating Condition:** AC power capability to critical buses reduced to a single power source for 15 minutes or longer such that **any** additional single failure would result in loss of **all** AC power to critical buses

**EAL:**

**CU1.1 Unusual Event**

AC power capability to critical 4160V buses 1F and 1G reduced to a single power source (Table C-4) for ≥ 15 min. such that **any** additional single failure would result in loss of **all** AC power to critical buses (Note 3)

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| **Table C-4 AC Power Sources** |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a loss of all ac power to critical buses. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency bus. The subsequent loss of this single power source would escalate the event to an Alert in accordance with CA1.1.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

. **CNS Basis**:

The 4160V critical buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following offsite sources of power: Figure C-1 illustrates the 4160V AC distribution system (ref. 1, 2).

- Startup Transformer – The Startup Transformer provides a source of offsite AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of offsite AC power to the station whenever the main generator is off-line. The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the Station Normal Transformer fails or that the main generator trips off-line.

- Emergency Transformer -The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.

- Backfeeding power from the 345 kv line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power.

The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed its power to the safety-related buses may be considered an offsite power source.

Onsite power sources are the emergency diesel generators (DG-1 and DG-2).

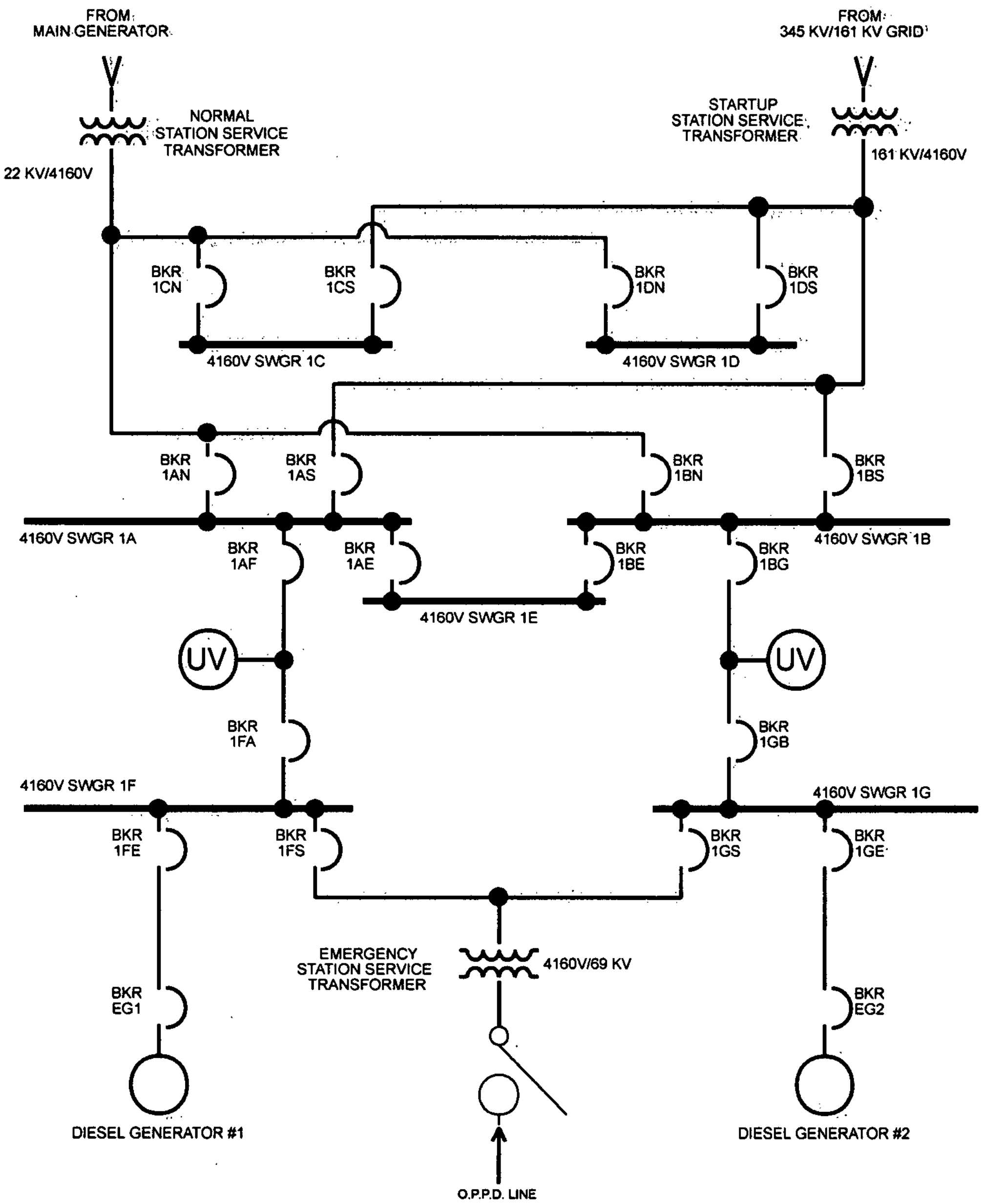
If critical bus AC power is reduced to a single source for greater than 15 minutes, an Unusual Event is declared under this EAL.

This cold condition EAL is equivalent to the hot condition loss of AC power EAL SA1.1.

**CNS Basis Reference(s):**

1. BR 3001 One Line Diagram
2. BR 3002 sh1
3. Procedure 2.1.15 Startup Transformer
4. Procedure 2.1.16 Normal Station Service Transformer
5. Procedure 2.1.17 Emergency Station Service Transformer
6. Procedure 2.2.18 4160V Auxiliary Power Distribution System
7. Procedure 2.2.20 Standby AC Power System (Diesel Generator)
8. Procedure 5.3SBO Station Blackout
9. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Rev. 2

## Figure C-1: 4160V AC Distribution System



**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 1 – Loss of Power

**Initiating Condition:** Loss of required DC power for 15 minutes or longer

**EAL:**

| | |
|---|---|
| **CU1.2** | **Unusual Event** |
| < 105 VDC bus voltage indications on **all** Technical Specification required 125 VDC buses for ≥ 15 min. (Note 3) | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations.

This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

The plant will routinely perform maintenance on a train related basis during shutdown periods The required buses are the minimum allowed by Technical Specifications for the mode of operation. It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per CA3.1.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

**CNS Basis**:

105 VDC is the minimum design bus voltage (ref. 2).

The 125 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. 125 VDC Distribution Panels supply control and instrument power for annunciators control logic power, and protective relaying. Figure C-2 illustrates the 125 VDC power system (ref. 1).

If 125 VDC Distribution Panel A is lost, the following major equipment is affected : RRMG A speed and breaker control, 4160V Bus 1A, 1 E, and 1 F breaker control and undervoltage logics, 480V Bus 1A and 1 F breaker control, the right light in all Control Room annunciators, annunciator panels for Water Treatment, SW A Gland Water , RHR A Gland Water, Auxiliary Steam Boiler C, DG-1 starting and breaker control logics, CS A, RCIC, and RHR A control logics, TIP valve control monitors, main generator voltage regulation, RFPT A trip logic, and ARI solenoid valve power .

If 125 VDC Distribution Panel B is lost, the following major equipment is affected; RRMG B speed and breaker control, 4160V Bus 1B and 1G breaker control and undervoltage logics, 480V Bus 1B and 1G breaker control, the left light in all Control Room annunciators, annunciator panels for ALRW, SW B Gland Water, RHR B Gland Water, Auxiliary Steam Boiler D, DG-2 starting and breaker control logics, CS B , HPCI, and RHR B control logics, main generator trip logic, main generator and transformer protective relaying, bypass valves fail to control pressure after turbine trip and RFPT B trip logic .

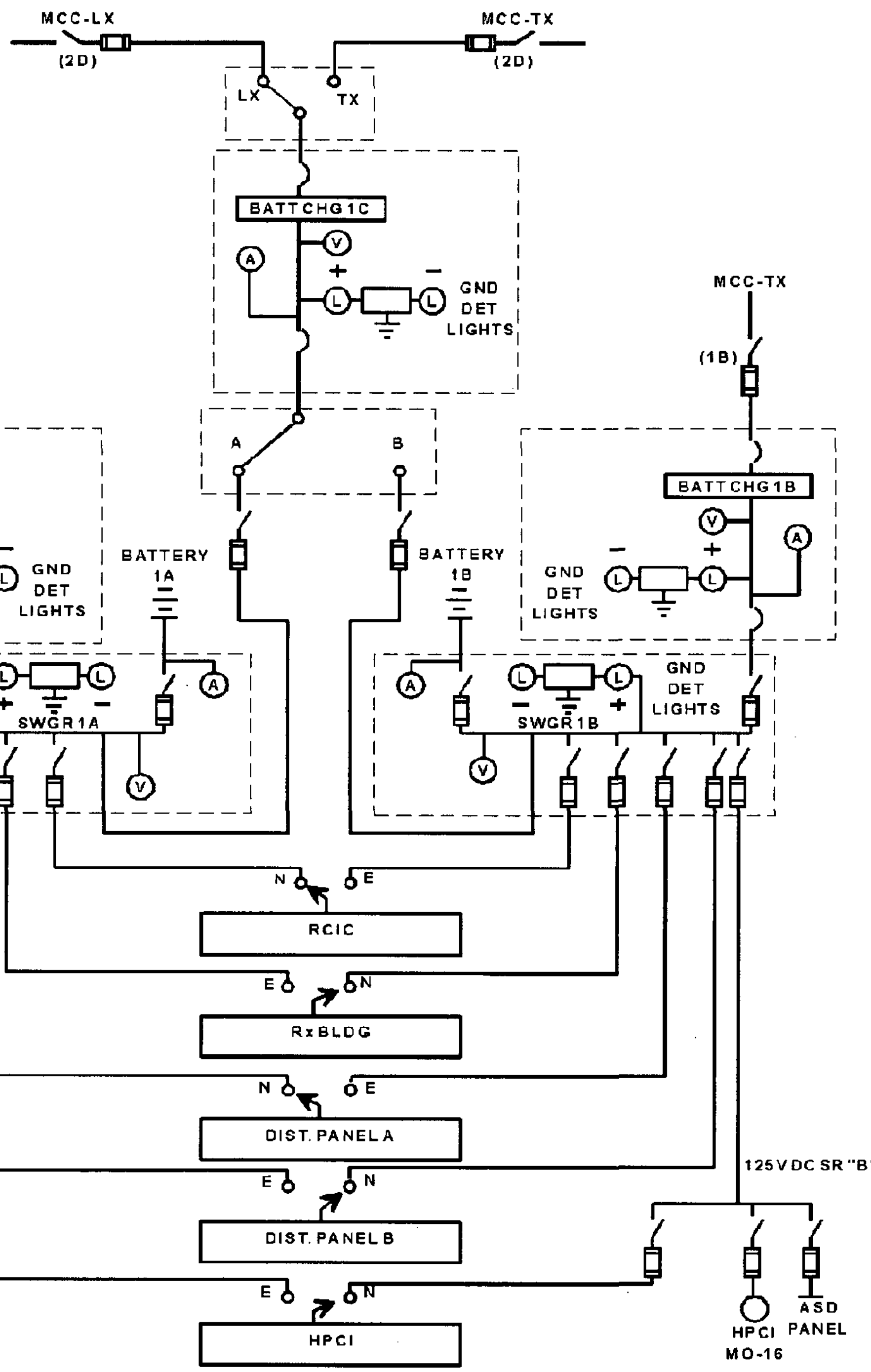
Battery chargers receive their power from 460V critical motor control centers. Each 125 VDC bus receives power from either a 125 VDC battery or a 125 VDC battery charger. The battery chargers receive their power from 460V critical motor control centers. The 250 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. Although 250 VDC Buses 1A and 1B provide vital DC emergency power, 250 VDC Safety System loads (such as motor operated valves) also require 125 VDC control power. Loss of 125 VDC buses alone, therefore, would render most Safeguard System loads inoperable (ref. 2, 3, 4).

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS1.2.

**CNS Basis Reference(s):**

1. BR 3058 DC One Line Diagram
2. Technical Specifications B 3.8.4
3. USAR Section VIII-6.2
4. USAR Section VIII-6.3
5. 5.3DC125 LOSS OF 125 VDC
6. 6.EE.607 125V STATION BATTERY PERFORMANCE DISCHARGE TEST

**Figure C-2: 125 VDC Power System**



**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 1 – Loss of Power

**Initiating Condition:** Loss of **all** offsite and **all** onsite AC power to critical buses for 15 minutes or longer

**EAL:**

| | |
|---|---|
| **CA1.1** | **Alert** |
| Loss of **all** offsite and **all** onsite AC power (Table C-4) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3) | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| Table C-4 AC Power Sources |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling, D - Defueled

**NEI 99-01 Basis:**

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, containment heat removal, spent fuel heat removal and the ultimate heat sink.

The event can be classified as an Alert when in cold shutdown, refueling, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the critical busses, relative to that specified for the Site Area Emergency EAL.

Escalating to Site Area Emergency, if appropriate, is by Abnormal Rad Levels / Radiological Effluent EALs.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

**CNS Basis**:

The 4160V critical buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following offsite sources of power: Figure C-1 illustrates the 4160V AC distribution system (ref. 1, 2).

- Startup Transformer – The Startup Transformer provides a source of offsite AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of offsite AC power to the station whenever the main generator is off-line (<160 Mwe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the Station Normal Transformer fails or that the main generator trips off-line.

- Emergency Transformer -The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.

- Backfeeding power from the 345 kv line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed its power to the safety-related buses may be considered an offsite power source.

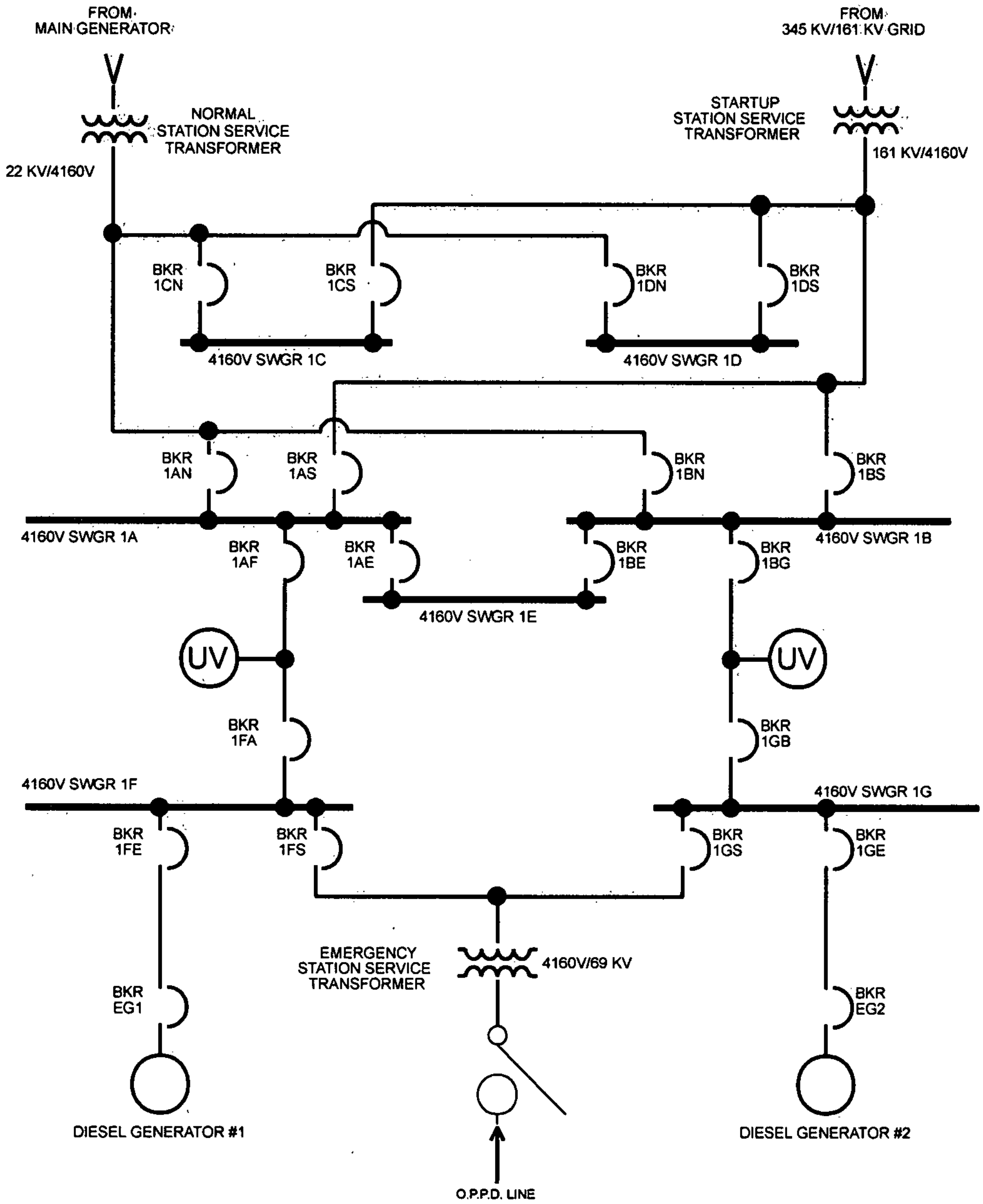
Onsite power sources are the emergency diesel generators (DG-1 and DG-2).

This EAL is the cold condition equivalent of the hot condition loss of all AC power EAL SS1.1. When in Cold Shutdown, Refueling, or Defueled mode, the event can be classified as an Alert because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency buses, relative to that existing when in hot conditions.

**CNS Basis Reference(s):**

1. BR 3001 One Line Diagram
2. BR 3002 sh1
3. Procedure 2.1.15 Startup Transformer
4. Procedure 2.1.16 Normal Station Service Transformer
5. Procedure 2.1.17 Emergency Station Service Transformer
6. Procedure 2.2.18 4160V Auxiliary Power Distribution System
7. Procedure 2.2.20 Standby AC Power System (Diesel Generator)
8. Procedure 5.3SBO Station Blackout
9. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Rev. 2

## Figure C-1: 4160V AC Distribution System



**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – RPV Level

**Initiating Condition:** RCS leakage

**EAL:**

| | |
|---|---|
| **CU2.1** | **Unusual Event** |
| RPV level **cannot** be restored and maintained > +3 in. for ≥ 15 min. (Note 3) due to RCS leakage | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

**Mode Applicability:**

4 - Cold Shutdown

**NEI 99-01 Basis:**

This EAL is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level is indicative of loss of RCS inventory.

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated.

Prolonged loss of RCS inventory may result in escalation to the Alert emergency classification level via either CA2.1 or CA3.1.

The difference between CU2.1 and CU2.2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In Cold Shutdown the RPV will normally be intact and RPV level is typically controlled below the elevation of the RPV flange and above the low-end of the normal control band. In the Refueling mode the RPV is not intact and any planned evolutions to lower RPV level below the elevation of the RPV flange must be carefully controlled.

**CNS Basis:**

The condition of this EAL may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. When RPV level drops to +3 in. (low level scram setpoint), level is well below the normal control band and automatic RPS and PCIS actuations are required (ref. 1, 2).

RPV level is normally monitored using the following instruments (ref. 3, 4):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.)
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.)
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.)
- Narrow Range NBI-LI-94A, B & C (0 to 60 in.)
- Shutdown Range NBI-LI-86 (0 to 400 in.)

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

This Cold Shutdown EAL represents the hot condition EAL SU6.1, in which RCS leakage is associated with Technical Specification limits. In Cold Shutdown, these limits are not applicable; hence, the use of RPV level as the parameter of concern in this EAL (ref. 5).

**CNS Basis Reference(s):**

1. EOP-1A RPV Control
2. Technical Specification Table 3.3.1.1-1
3. 2.4RXLVL RPV Water Level Control Trouble
4. Procedure 4.6.1 Reactor Vessel Water Level Indication
5. NEI/NRC EAL FAQ #2006-014

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – RPV Level

**Initiating Condition:** Unplanned loss of RPV inventory

**EAL:**

| | |
|---|---|
| **CU2.2** | **Unusual Event** |

Unplanned RPV level drop for ≥ 15 min (Note 3) below **EITHER**:

RPV flange (LI-86: 206 in. normal calibration, 113.75 in. elevated calibration)

**OR**

RPV level band when the RPV level band is established below the RPV flange

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

**Mode Applicability:**

5 - Refueling

**NEI 99-01 Basis:**

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RPV water level below the RPV flange are carefully planned and procedurally controlled. An unplanned event that results in water level decreasing below the RPV flange, or below the planned RPV water level for the given evolution (if the planned RPV water level is already below the RPV flange), warrants declaration of an Unusual Event due to the reduced RCS inventory that is available to keep the core covered.

The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists.

Continued loss of RCS inventory will result in escalation to the Alert emergency classification level via either CA2.1 or CA3.1.

The difference between CU2.1 and CU2.2 deals with the RPV conditions that exist between cold shutdown and refuel modes. In cold shutdown the RCS will normally be intact and standard RPV inventory and level monitoring means are available. In the refuel mode the RCS is not intact and RPV level and inventory may be monitored by different means.

This EAL involves a decrease in RPV level below the top of the RPV flange or a decrease below the RPV level band (when the RPV level band is established below the RPV flange) that continues for 15 minutes due to an unplanned event. This EAL is not applicable to decreases in flooded reactor cavity level, which is addressed by AU2.1, until such time as the level decreases to the level of the vessel flange.

If RPV level continues to decrease and reaches the Low-Low ECCS actuation setpoint then escalation to CA2.1 would be appropriate.

**CNS Basis**:

The RPV flange is 722.75 in. above the RPV bottom head. RPV water level at this elevation is normally indicated by the Shutdown Range instrument (LI-86 Shutdown, 0 - 400 in.). When calibrated for normal plant operations, the Shutdown Range instrument reads 206 in. at the RPV flange. With the RPV head removed, the instrument is calibrated to indicate reactor cavity water levels as high as the refuel floor. When calibrated for elevated indication, the Shutdown Range instrument reads 113.75 in. at the RPV flange. Visual observation of water level in the reactor cavity and RPV is also used during refuel operations.

**CNS Basis Reference(s):**

1. Procedure 2.1.20.3 RPV Refueling Preparation, Attachment 1
2. Procedure 14.15.3 Reactor Vessel Open Head Monitor System
3. Procedure 4.6.1 Reactor Vessel Water Level Indication

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – RPV Level

**Initiating Condition:** Unplanned loss of RPV inventory

**EAL:**

| **CU2.3** | **Unusual Event** |
|---|---|
| RPV level **cannot** be monitored with **any** unexplained RPV leakage indication, Table C-1 | |

| **Table C-1 RPV Leakage Indications** |
|---|
| • Drywell equipment drain sump level rise<br>• Drywell floor drain sump level rise<br>• Reactor Building equipment drain sump level rise<br>• Reactor Building floor drain sump level rise<br>• Torus water level rise<br>• RPV make-up rate rise<br>• Observation of unisolable RCS leakage |

**Mode Applicability:**

5 - Refueling

**NEI 99-01 Basis:**

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either CA2.1 or CA3.1.

This EAL addresses conditions in the refueling mode when normal means of core temperature indication and RPV level indication may not be available. Redundant means of RPV level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RPV inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump level changes listed in Table C-1. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Escalation to the Alert emergency classification level would be via either CA2.1 or CA3.1.

**CNS Basis**:

RPV level is normally monitored using the following instruments (ref. 1, 2):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.)
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.)
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.)
- Narrow Range NBI-LI-94A, B & C (0 to 60 in.)
- Shutdown Range NBI-LI-86 (0 to 400 in.)

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

In this EAL, all water level indication is unavailable and the RPV inventory loss should be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 3). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage (ref. 4). If the make-up rate to the

RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

**CNS Basis Reference(s):**

1. 2.4RXLVL RPV Water Level Control Trouble
2. Procedure 4.6.1 Reactor Vessel Water Level Indication
3. Procedure 2.2.27 Equipment, Floor and Chemical Drain System
4. Procedure 2.2.69 Residual Heat Removal

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – RPV Level

**Initiating Condition:** Loss of RPV inventory

**EAL:**

**CA2.1 Alert**

RPV level < -42 in.

**OR**

RPV level **cannot** be monitored for ≥ 15 min. (Note 3) with **any** unexplained RPV leakage indication, Table C-1

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| Table C-1 RPV Leakage Indications |
|---|
| • Drywell equipment drain sump level rise<br>• Drywell floor drain sump level rise<br>• Reactor Building equipment drain sump level rise<br>• Reactor Building floor drain sump level rise<br>• Suppression pool water level rise<br>• RPV make-up rate rise<br>• Observation of unisolable RCS leakage |

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

This EAL serves as precursor to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level decrease and potential core uncovery. This condition will result in a minimum classification of Alert. The low-low ECCS Actuation setpoint was chosen because it is a recognized setpoint.. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. In the Refueling mode, the RCS is not intact and RPV level may be monitored by different means, including the ability to monitor level visually.

In the Refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RPV inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump level changes listed in Table C-1. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS2.1 Site Area Emergency EAL duration. The 15-minute duration allows CA2.1 to be an effective precursor to CS2.1. Significant fuel damage is not expected to occur until the core has been uncovered for greater than one hour. Therefore this EAL meets the definition for an Alert.

If RPV level continues to decrease then escalation to Site Area Emergency will be via EAL CS2.1.

**CNS Basis:**

The threshold RPV level of -42 in. is the low-low ECCS actuation setpoint (ref. 1).

RPV level is normally monitored using the following instruments (ref. 2, 3):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.)
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.)
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.)
- Narrow Range NBI-LI-94A, B & C (0 to 60 in.)
- Shutdown Range NBI-LI-86 (0 to 400 in.)

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 4). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage (ref. 5). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

**CNS Basis Reference(s):**

1. Technical Specification Table 3.3.5.1-1
2. 2.4RXLVL RPV Water Level Control Trouble
3. Procedure 4.6.1 Reactor Vessel Water Level Indication
4. Procedure 2.2.27 Equipment, Floor and Chemical Drain System
5. Procedure 2.2.69 Residual Heat Removal

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – RPV Level

**Initiating Condition:** Loss of RPV inventory affecting core decay heat removal capability

**EAL:**

| | |
|---|---|
| **CS2.1** | **Site Area Emergency** |
| With Containment Closure **not** established (Note 4), RPV level < -48 in. | |

**Note 4:** Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

Under the conditions specified by this EAL, continued decrease in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via CG2.1 or AG1.1.

**CNS Basis:**

When RPV level decreases to -48 in., water level is six inches below the low-low ECCS actuation setpoint: -42 in. – 6 in. = -48 in. (ref. 1).

RPV level is normally monitored using the following instruments (ref. 2, 3):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.)
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.)
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.)
- Narrow Range NBI-LI-94A, B & C (0 to 60 in.)

- Shutdown Range NBI-LI-86 (0 to 400 in.)

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

Containment Closure is the action taken to secure Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety (ref. 4).

**CNS Basis Reference(s):**

1. Technical Specification Table 3.3.5.1-1
2. 2.4RXLVL RPV Water Level Control Trouble
3. Procedure 4.6.1 Reactor Vessel Water Level Indication
4. Administrative Procedure 0.50.5 Outage Shutdown Safety

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – RPV Level

**Initiating Condition:** Loss of RPV inventory affecting core decay heat removal capability

**EAL:**

| | |
|---|---|
| **CS2.2** | **Site Area Emergency** |
| With Containment closure established (Note 4), RPV level < -158 in. | |

**Note 4:** Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

Under the conditions specified by this EAL, continued decrease in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via CG2.1or AG1.1.

**CNS Basis**:

When RPV level drops to -158 in., core uncovery is about to occur (ref. 1).

RPV level is normally monitored using the following instruments (ref. 2, 3):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.)
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.)
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.)
- Narrow Range NBI-LI-94A, B & C (0 to 60 in.)
- Shutdown Range NBI-LI-86 (0 to 400 in.)

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

Containment Closure is the action taken to secure Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety (ref. 4).

**CNS Basis Reference(s):**

1. NEDC 97-089
2. 2.4RXLVL RPV Water Level Control Trouble
3. Procedure 4.6.1 Reactor Vessel Water Level Indication
4. Administrative Procedure 0.50.5 Outage Shutdown Safety

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – RPV Level

**Initiating Condition:** Loss of RPV inventory affecting core decay heat removal capability

**EAL:**

**CS2.3 Site Area Emergency**

RPV level **cannot** be monitored for ≥ 30 min. (Note 3) with a loss of inventory as indicated by **EITHER**:

Unexplained RPV leakage indication, Table C-1

**OR**

Erratic Source Range Monitor indication

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| Table C-1 RPV Leakage Indications |
|---|
| • Drywell equipment drain sump level rise<br>• Drywell floor drain sump level rise<br>• Reactor Building equipment drain sump level rise<br>• Reactor Building floor drain sump level rise<br>• Suppression pool water level rise<br>• RPV make-up rate rise<br>• Observation of unisolable RCS leakage |

**Mode Applicability:**

4 – Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

Under the conditions specified by this EAL, continued decrease in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via CG2.2 or AG1.1.

In the cold shutdown mode, normal RPV level instrumentation systems will usually be available. In the refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RPV inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump level changes listed in Table C-1. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

The 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

**CNS Basis:**

RPV level is normally monitored using the instruments in Figure C-3 (ref. 1, 2).

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.)
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.)
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.)
- Narrow Range NBI-LI-94A, B & C (0 to 60 in.)
- Shutdown Range NBI-LI-86 (0 to 400 in.)

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1 or erratic Source Range Monitor (SRM) indication:

- Table C-1 level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 3). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage (ref. 4). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

- Source Range Monitor (SRM) indication is provided in the main Control Room by NMS-I-43A-D, SRM A-D LOG COUNT RATE, NM-NR-45, SRM 2 PEN RECORDER, and SPDS (ref. 5).

**CNS Basis Reference(s):**

1. 2.4RXLVL RPV Water Level Control Trouble
2. Procedure 4.6.1 Reactor Vessel Water Level Indication
3. Procedure 2.2.27 Equipment, Floor and Chemical Drain System
4. Procedure 2.2.69 Residual Heat Removal
5. Procedure 4.1.1 Source Range Monitoring System

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – RPV Level

**Initiating Condition:** Loss of RPV inventory affecting fuel clad integrity with containment challenged

**EAL:**

| | |
|---|---|
| **CG2.1** | **General Emergency** |
| RPV level < -158 in. for ≥ 30 min. (Note 3) | |
| **AND** | |
| **Any** Containment Challenge indication, Table C-5 | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| **Table C-5 Containment Challenge Indications** |
|---|
| • Containment Closure **not** established (Note 4)<br>• Deflagration concentrations exist inside PC<br>≥ 6% $H_2$ in drywell or torus<br>**AND**<br>≥ 5% $O_2$ in drywell or torus<br>• Unplanned rise in PC pressure<br>• Secondary Containment area radiation > 1000 mR/hr (EOP-5A Table 10) |

**Note 4:** Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

**Mode Applicability:**

4 - Cold Shutdown, 5 – Refueling

**NEI 99-01 Basis:**

This EAL represents the inability to restore and maintain RPV level to above the top of active fuel. Fuel damage is probable if RPV level cannot be restored, as available decay heat will cause boiling, further reducing the RPV level.

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

The General Emergency is declared on the occurrence of the loss or imminent loss of function of <u>all three</u> barriers. Based on the above discussion, RCS barrier failure resulting in core uncovery for 30 minutes or more may cause fuel clad failure. With the Primary Containment and Secondary Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a General Emergency.

Containment Closure is the action taken to secure containment (Primary or Secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure should not be confused with refueling containment integrity as defined in technical specifications. Site shutdown contingency plans typically provide for re-establishing Containment Closure following a loss of heat removal or RPV inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the RCS barrier and Fuel Clad barrier EALs, escalation to General Emergency would not occur.

The use of secondary containment radiation monitors should provide indication of increased release that may be indicative of a challenge to secondary containment. The radiation monitor values are based on the EOP "maximum safe values" because these values are easily recognizable and have an emergency basis.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.

**CNS Basis**:

When RPV level drops to -158 in., core uncovery is about to occur (ref. 1).

RPV level is normally monitored using the following instruments (ref. 2, 3):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.)
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.)
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.)
- Narrow Range NBI-LI-94A, B & C (0 to 60 in.)
- Shutdown Range NBI-LI-86 (0 to 400 in.)

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

Four conditions are associated with a challenge to Primary Containment (PC) integrity:

1. Containment Closure is the action taken to secure Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety (ref. 4)

2. Deflagration (explosive) mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAMGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be

indicative, therefore, of a potential threat to primary containment integrity. The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen) (ref. 5).

3. Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned Primary Containment pressure increases indicate containment closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

4. 1,000 mR/hr is the Secondary Containment Maximum Safe Operating radiation value. Exceeding this value is indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-5A, Secondary Containment Control, Table 10. As indicated by Note 5 in EOP-5A Table 10, RP surveys and ARM teledosimetry system may be used for these indications (ref. 7).

## EOP-5A Table 10 – Secondary Containment Radiation Levels

| 10 | SECONDARY CONTAINMENT RADIATION LEVELS (5) SPDS 15 | | | | |
|---|---|---|---|---|---|
| Maximum Normal Operating Value | | | Maximum Safe Operating Value | | |
| Area | Any ARM Alarmed | Range (mR/hr) | Area | Value (mR/hr) | Actual Value |
| FUEL POOL AREA | RMA-RA-1 | 100 - $10^8$ | 1001' EL | 1000 | |
| FUEL POOL AREA | RMA-RA-2 | .01 - 100 | 1001' EL | | |
| RWCU PRECOAT AREA | RMA-RA-4 | 0.1 - 1000 | 958' EL | | |
| RWCU SLUDGE AND DECANT PUMP AREA | RMA-RA-5 | 0.1 - 1000 | 931' EL | 1000 | |
| CRD HYDRAULIC EQUIP AREA (SOUTH) | RMA-RA-8 | .01 - 100 | 903' EL | | |
| CRD HYDRAULIC EQUIP AREA (NORTH) | RMA-RA-9 | .01 - 100 | | | |
| HPCI PUMP ROOM | RMA-RA-10 | .01 - 100 | HPCI Room | | |
| RHR PUMP ROOM, (SOUTHWEST) | RMA-RA-11 | .01 - 100 | SW Quad | 1000 | |
| TORUS HPV AREA (SOUTHWEST) | RMA-RA-27 | 1.0 - 10000 | SW Torus | | |
| RHR PUMP ROOM, (NORTHWEST) | RMA-RA-12 | .01 - 100 | NW Quad | 1000 | |
| RCIC/CORE SPRAY PUMP ROOM, (NORTHEAST) | RMA-RA-13 | .01 - 100 | NE Quad | 1000 | |
| CORE SPRAY PUMP ROOM, (SOUTHEAST) | RMA-RA-14 | .01 - 100 | SE Quad | 1000 | |

(5)
Area radiation levels can be monitored by RP surveys or ARM teledosimetry system

### CNS Basis Reference(s):

1. NEDC 97-089
2. 2.4RXLVL RPV Water Level Control Trouble
3. Procedure 4.6.1 Reactor Vessel Water Level Indication
4. Administrative Procedure 0.50.5 Outage Shutdown Safety
5. AMP-TBD00, Step PC/H
6. Procedure 2.2.60.1 Containment $H_2/O_2$ Monitoring System
7. EOP-5A Secondary Containment Control, Table 10

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – RPV Level

**Initiating Condition:** Loss of RPV inventory affecting fuel clad integrity with containment challenged

**EAL:**

**CG2.2 General Emergency**

RPV level **cannot** be monitored for ≥ 30 min. (Note 3) with core uncovery indicated by **EITHER**:

Unexplained RPV leakage indication, Table C-1

**OR**

Erratic Source Range Monitor indication

**AND**

**Any** Containment Challenge indication, Table C-5

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| **Table C-1 RPV Leakage Indications** |
|---|
| • Drywell equipment drain sump level rise<br>• Drywell floor drain sump level rise<br>• Reactor Building equipment drain sump level rise<br>• Reactor Building floor drain sump level rise<br>• Suppression pool water level rise<br>• RPV make-up rate rise<br>• Observation of unisolable RCS leakage |

| Table C-5 Containment Challenge Indications |
|---|
| • Containment Closure **not** established (Note 4)<br>• Deflagration concentrations exist inside PC<br>≥ 6% $H_2$ in drywell or torus<br>**AND**<br>≥ 5% $O_2$ in drywell or torus<br>• Unplanned rise in PC pressure<br>• Secondary Containment area radiation > 1000 mR/hr (EOP-5A Table 10) |

**Note 4:** Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

In the Cold Shutdown mode, normal RPV level and RPV level instrumentation systems will normally be available. However, if all level indication were to be lost during a loss of RPV inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump level changes. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Primary Containment to ensure they are indicative of RCS leakage.

In the Refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RPV inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump level changes listed in Table C-1.

For both Cold Shutdown and Refueling modes sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Primary Containment to ensure they are indicative of RCS leakage.

A number of variables, such as initial RPV level, or shutdown heat removal system design, can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that significant core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

The General Emergency is declared on the occurrence of the loss or imminent loss of function of all three barriers. Based on the above discussion, RCS barrier failure resulting in core uncovery for 30 minutes or more may cause fuel clad failure. With the Primary Containment and Secondary Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a General Emergency.

Containment Closure is the action taken to secure either Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure should not be confused with refueling containment integrity as defined in technical specifications. Site shutdown contingency plans provide for re-establishing containment closure following a loss of heat removal or RPV inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the RCS Barrier and Fuel Clad Barrier EALs, escalation to General Emergency would not occur.

The use of Secondary Containment radiation monitors should provide indication of increased release that may be indicative of a challenge to Secondary Containment. The radiation monitor values are based on the EOP "maximum safe values" because these values are easily recognizable and have an emergency basis.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.

**CNS Basis:**

RPV level is normally monitored using the following instruments (ref. 1, 2):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.)
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.)
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.)
- Narrow Range NBI-LI-94A, B & C (0 to 60 in.)
- Shutdown Range NBI-LI-86 (0 to 400 in.)

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1 or erratic Source Range Monitor (SRM) indication:

- Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 3). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage (ref. 4). If the make-up rate to the

RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

- Source Range Monitor (SRM) indication is provided in the main Control Room by NMS-I-43A-D, SRM A-D LOG COUNT RATE, NM-NR-45, SRM 2 PEN RECORDER, and SPDS (ref. 5).

Four conditions are associated with a challenge to Primary Containment (PC) integrity:

1. Containment Closure is the action taken to secure Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety (ref. 6).

2. Deflagration (explosive) mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAMGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to primary containment integrity. The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen) (ref. 7).

3. Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned Primary

Containment pressure increases indicate containment closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

4. 1,000 mR/hr is the Secondary Containment Maximum Safe Operating radiation value. Exceeding this value is indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-5A, Secondary Containment Control, Table 10. As indicated by Note 5 in EOP-5A Table 10, RP surveys and ARM teledosimetry system may be used for these indications (ref. 9).

## EOP-5A Table 10 – Secondary Containment Radiation Levels

| 10 | SECONDARY CONTAINMENT RADIATION LEVELS SPDS 15 ⑤ | | | | |
|---|---|---|---|---|---|
| Maximum Normal Operating Value | | | Maximum Safe Operating Value | | |
| Area | Any ARM Alarmed | Range (mR/hr) | Area | Value (mR/hr) | Actual Value |
| FUEL POOL AREA | RMA-RA-1 | $100 - 10^6$ | 1001' EL | 1000 | |
| FUEL POOL AREA | RMA-RA-2 | .01 - 100 | 1001' EL | | |
| RWCU PRECOAT AREA | RMA-RA-4 | 0.1 - 1000 | 958' EL | | |
| RWCU SLUDGE AND DECANT PUMP AREA | RMA-RA-5 | 0.1 - 1000 | 931' EL | 1000 | |
| CRD HYDRAULIC EQUIP AREA (SOUTH) | RMA-RA-8 | .01 - 100 | 903' EL | | |
| CRD HYDRAULIC EQUIP AREA (NORTH) | RMA-RA-9 | .01 - 100 | | | |
| HPCI PUMP ROOM | RMA-RA-10 | .01 - 100 | HPCI Room | | |
| RHR PUMP ROOM, (SOUTHWEST) | RMA-RA-11 | .01 - 100 | SW Quad | 1000 | |
| TORUS HPV AREA (SOUTHWEST) | RMA-RA-27 | 1.0 - 10000 | SW Torus | | |
| RHR PUMP ROOM, (NORTHWEST) | RMA-RA-12 | .01 - 100 | NW Quad | 1000 | |
| RCIC/CORE SPRAY PUMP ROOM, (NORTHEAST) | RMA-RA-13 | .01 - 100 | NE Quad | 1000 | |
| CORE SPRAY PUMP ROOM, (SOUTHEAST) | RMA-RA-14 | .01 - 100 | SE Quad | 1000 | |

⑤
Area radiation levels can be monitored by RP surveys or ARM teledosimetry system

## CNS Basis Reference(s):

1. 2.4RXLVL RPV Water Level Control Trouble
2. Procedure 4.6.1 Reactor Vessel Water Level Indication
3. Procedure 2.2.27 Equipment, Floor and Chemical Drain System
4. Procedure 2.2.69 Residual Heat Removal
5. Procedure 4.1.1 Source Range Monitoring System
6. Administrative Procedure 0.50.5 Outage Shutdown Safety
7. AMP-TBD00, Step PC/H

8. Procedure 2.2.60.1 Containment $H_2/O_2$ Monitoring System
9. EOP-5A Secondary Containment Control, Table 10

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Temperature

**Initiating Condition:** Unplanned loss of decay heat removal capability with irradiated fuel in the RPV

**EAL:**

| | |
|---|---|
| **CU3.1** | **Unusual Event** |
| **Any** unplanned event results in RCS temperature > 212°F due to loss of decay heat removal capability | |

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RPV inventory. Since the RCS usually remains intact in the Cold Shutdown mode a large inventory of water is available to keep the core covered. In Cold Shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the Refueling mode with irradiated fuel in the RPV (note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling). In addition, the operators should be able to monitor RCS temperature and RPV level so that escalation to the alert level via CA2.1 or CA3.1 will occur if required.

During refueling operations, the level in the RPV will normally be maintained above the vessel flange. Refueling evolutions that decrease water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS temperatures depending on the time since shutdown. Escalation to the Alert level via CA3.1 may therefore be required should an unplanned event result in RCS temperature exceeding the Technical Specification Cold Shutdown temperature limit for an extended period of time. The allowed time varies and is dependent on the status of the Primary Containment and Secondary Containment barriers and the integrity of the RCS barrier.

The Emergency Director must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

**CNS Basis:**

Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212°F) (ref. 1). These include (ref. 2, 3):

- NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21)
- Vessel Drain, PMIS Point M180, or NBI-TR-89 - Point 06 if M180 is not available
- Vessel Bottom Head, PMIS Point M184, or NBI-TR-89 - Point 10 if M184 is not available
- Bottom Head Adjacent to Support Skirt, PMIS Point M183 or NBI-TR-89 - Point 09
- RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4)

PMIS Points M174 through M185 can be used to monitor RPV temperatures. Thermocouples associated with computer Points M180, M183, and M185 do not respond as quickly nor register as high a temperature as other thermocouples due to their locations.

Inservice leak testing, hydrostatic testing and control rod scram time testing in which RCS temperature is intentionally raised above 212°F per Technical Specification LCO 3.10.1 are not applicable to this EAL (ref. 4).

**CNS Basis Reference(s):**

1. Technical Specifications Table 1.1-1
2. Procedure 2.1.1 Startup Procedure
3. Procedure 2.2.69.2 RHR System Shutdown Operations
4. Technical Specifications LCO 3.10.1

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Temperature

**Initiating Condition:** Loss of decay heat removal capability with irradiated fuel in the RPV

**EAL:**

| **CU3.2** | **Unusual Event** |
|---|---|
| Loss of **all** RCS temperature and RPV level indication for ≥ 15 min. (Note 3) | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

This EAL is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RPV inventory. Since the RCS usually remains intact in the Cold Shutdown mode a large inventory of water is available to keep the core covered. In Cold Shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the Refueling mode with irradiated fuel in the RPV. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling outage. In addition, the operators should be able to monitor RCS temperature and RPV level so that escalation to the alert level via CA2.1 or CA3.1 will occur if required.

During refueling operations, the level in the RPV will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS temperatures depending on the time since shutdown. Escalation to the Alert level via CA3.1 may therefore be required should an unplanned event result in RCS temperature exceeding the Technical Specification Cold Shutdown temperature limit for an extended period of time. The allowed time varies and is dependent on the status of the Primary Containment and Secondary Containment barriers and the integrity of the RCS barrier.

Unlike the Cold Shutdown mode, normal means of core temperature indication and RCS level indication may not be available in the Refueling mode. Redundant means of RPV level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling modes, this EAL would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be under CA2.1 based on an inventory loss or CA3.1 based on exceeding its temperature criteria (212°F, ref. 1).

The Emergency Director must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

**CNS Basis**:

RPV level is normally monitored using the following instruments (ref. 1, 2).

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.)
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.)
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.)
- Narrow Range NBI-LI-94A, B & C (0 to 60 in.)
- Shutdown Range NBI-LI-86 (0 to 400 in.)

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212°F). These include (ref. 3, 4):

- NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21)
- Vessel Drain, PMIS Point M180, or NBI-TR-89 - Point 06 if M180 is not available
- Vessel Bottom Head, PMIS Point M184, or NBI-TR-89 - Point 10 if M184 is not available
- Bottom Head Adjacent to Support Skirt, PMIS Point M183 or NBI-TR-89 - Point 09
- RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4)

PMIS Points M174 through M185 can be used to monitor RPV temperatures. Thermocouples associated with computer Points M180, M183, and M185 do not respond as quickly nor register as high a temperature as other thermocouples due to their locations.

**CNS Basis Reference(s):**

1. 2.4RXLVL RPV Water Level Control Trouble
2. Procedure 4.6.1 Reactor Vessel Water Level Indication
3. Procedure 2.1.1 Startup Procedure
4. Procedure 2.2.69.2 RHR System Shutdown Operations

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Temperature

**Initiating Condition:** Inability to maintain plant in cold shutdown

**EAL:**

**CA3.1 Alert**

**Any** unplanned event results in **EITHER**:

RCS temperature > 212°F for > Table C-3 duration (Note 4)

**OR**

RPV pressure increase > 10 psig due to a loss of RCS cooling

**Note 4:** Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

| Table C-3 RCS Reheat Duration Thresholds | |
|---|---|
| * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable | |
| 1. RCS intact (Containment Closure N/A) | 60 min.* |
| 2. Containment Closure established **AND** RCS **not** intact | 20 min.* |
| 3. Containment Closure **not** established **AND** RCS **not** intact | 0 min. |

**Mode Applicability:**

4 - Cold Shutdown, 5 – Refueling

**NEI 99-01 Basis:**

The first condition of this EAL addresses events in which RCS temperature exceeds the CU3.1 EAL threshold of 212°F (ref. 1) for the durations identified in Table C-3.

Table C-3 duration #3 addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when neither containment closure nor RCS integrity are established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation. No delay time is allowed for duration #3 because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

Table C-3 duration #2 addresses the complete loss of functions required for core cooling for > 20 minutes during Refueling and Cold Shutdown modes when containment closure is established but RCS integrity is not established. RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation. The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" and is believed to be conservative given that a low pressure containment barrier to fission product release is established. The table note indicates that this duration is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the 20 minute time frame.

Table C-3 duration #1 addresses complete loss of functions required for core cooling for greater than 60 minutes during Refueling and Cold Shutdown modes when RCS integrity is established. As in duration #2 and #3, RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation. The status of containment closure in this EAL is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

The 10 psig pressure increase covers situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. The table note indicates that duration #1 is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the 60 minute time frame assuming that the RCS pressure increase has remained less than 10 psig.

Escalation to Site Area would be via CS1.1 should boiling result in significant RPV level loss leading to core uncovery.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary unplanned excursion above 212°F when the heat removal function is available and either the RCS is intact or Containment Closure is established.

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

**CNS Basis**:

A 10 psig RPV pressure increase can be read on (ref. 2, 3):

- RFC-PI-90A (Panel 9-5, 0 – 1200 psig)
- RFC-PI-90B (Panel 9-5, 0 – 1200 psig)
- RFC-PI-90C (Panel 9-5, 0 – 1200 psig)
- Reactor pressure on PID B025

Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212°F). These include (ref. 4, 5):

- NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21)
- Vessel Drain, PMIS Point M180, or NBI-TR-89 - Point 06 if M180 is not available

- Vessel Bottom Head, PMIS Point M184, or NBI-TR-89 - Point 10 if M184 is not available
- Bottom Head Adjacent to Support Skirt, PMIS Point M183 or NBI-TR-89 - Point 09
- RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4)

PMIS Points M174 through M185 can be used to monitor RPV temperatures. Thermocouples associated with computer Points M180, M183, and M185 do not respond as quickly nor register as high a temperature as other thermocouples due to their locations.

Inservice leak testing, hydrostatic testing and control rod scram time testing in which RCS temperature is intentionally raised above 212°F per Technical Specification LCO 3.10.1 are not applicable to this EAL (ref. 6).

**CNS Basis Reference(s):**

1. Technical Specifications Table 1.1-1
2. Procedure 4.6.2 Reactor Vessel Pressure Indication
3. Procedure 5.9SAMG, Att. 2 (CPA TSG)
4. Procedure 2.1.1 Startup Procedure
5. Procedure 2.2.69.2 RHR System Shutdown Operations
6. Technical Specifications LCO 3.10.1

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 4 – Communications

**Initiating Condition:** Loss of **all** onsite or offsite communications capabilities

**EAL:**

**CU4.1 Unusual Event**

Loss of **all** Table C-2 onsite (internal) communication methods affecting the ability to perform routine operations

**OR**

Loss of **all** Table C-2 offsite (external) communication methods affecting the ability to perform offsite notifications

**Table C-2 Communications Systems**

| System | Onsite (internal) | Offsite (external) |
|---|---|---|
| Station Intercom System "Gaitronics" | X | |
| Site UHF Radio Paging System | X | |
| Alternate Intercom | X | |
| CNS On-Site Cell Phone System | X | X |
| Telephone system (PBX) | X | X |
| Federal Telecommunications System (FTS 2001) | | X |
| Microwave Telephone Network | | X |
| Local Telephones (C.O. Lines) | | X |
| CNS State Notification Telephones | | X |

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling, 6 - Defueled

**NEI 99-01 Basis:**

The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

The Table C-2 list for onsite communications loss encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system and radios / walkie talkies).

The Table C-2 list for offsite communications loss encompasses the loss of all means of communications with offsite authorities. This should include the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

**CNS Basis**:

Onsite/offsite communications include one or more of the systems listed in Table C-2 (ref. 1).

- Station Intercom System “Gaitronics”: Permits communication between the different parts of the plant and it also incorporates a public address system for plant wide announcements.

- Site UHF Radio Paging System: The site 450 MHz (UHF) radio system uses two repeaters, Base 1 and Base 2. These repeaters operate on different frequencies. All remote control, portable, and mobile units are capable of selecting either repeater.

- Alternate Intercom: Provides an alternate in-plant communications network utilizing the back-up tone commander PBX system. This system is located in the ERP shack and has battery back-up.

- CNS On-Site Cell Phone System

- Telephone system (PBX): Provides voice communication between virtually all buildings, offices, and operation facilities within the station. The telephone system also provides communications between the plant and offsite facilities via the telephone switchboard network. The system allows operating crews to alert plant personnel in emergencies. The telephone company provides the normal and leased line services.

- Federal Telecommunications System (FTS 2001): The Health Physics Network (HPN) and Emergency Notification System (ENS ) provides communications between NRC and CNS during an emergency.

- Microwave Telephone Network

- Local Telephones (C.O. Lines)

- CNS State Notification Telephones: The CNS State Notification Telephone System is the primary means for the plant to make emergency notifications to state and local authorities. This system provides direct communication with the Nebraska State Patrol, the Missouri State Patrol, the Atchison County Sheriffs Department, and the Nemaha and Richardson County Sheriffs Departments.

This EAL is the cold condition equivalent of the hot condition EAL SU4.2.

**CNS Basis Reference(s):**

1. Procedure 5.7COMMUN, Attachment 1

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 5 – Inadvertent Criticality

**Initiating Condition:** Inadvertent criticality

**EAL:**

| **CU5.1** | **Unusual Event** |
|---|---|
| An unplanned sustained positive period observed on nuclear instrumentation | |

**Mode Applicability:**

4 - Cold Shutdown, 5 - Refueling

**NEI 99-01 Basis:**

This EAL addresses criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated) which are addressed in the companion EAL SU2.1.

The terms "sustained" is used in order to allow exclusion of expected short term positive periods from planned fuel bundle or control rod movements during core alteration. These short term positive periods are the result of the increase in neutron population due to subcritical multiplication.

Escalation to higher classification levels would be by the judgment EALs in Category H (EAL HA6.1, HS6.1 or HG6.1).

**CNS Basis**:

SRM A-D period meters NMS-I-44A-D on Panel 9-5 identify this condition as well as Panel 9-5 Panel 9-5 amber light and SRM Period (> 50 sec.) annunciator 9-5-1/F-8 (ref. 1, 2). However, an SRM period alarm caused by SRM channel noise does not result in entry into this EAL (ref. 2).

**CNS Basis Reference(s):**

1. Procedure 4.1.1 Source Range Monitoring System
2. Procedure 2.3, 9-5-1, F-8, SRM Period

## Category H – Hazards

EAL Group: ANY (EALs in this category are applicable to any plant condition)

Hazards are non-plant, system-related events that can directly or indirectly affect plant operation, reactor plant safety or personnel safety.

The events of this category pertain to the following subcategories:

1. Natural or Destructive Phenomena

Natural events include earthquakes, tornados, high winds, and high/low river levels that have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety. Non-naturally occurring events that can cause damage to plant facilities and include vehicle crashes, missile impacts, internal flooding, etc.

2. Fire or Explosion

Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are fires within the site Protected Area or which may affect operability of vital equipment.

3. Hazardous Gas

Non-naturally occurring events that can cause damage to plant facilities and include toxic, corrosive, asphyxiant or flammable gas leaks.

4. Security

Unauthorized entry attempts into the Protected Area, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

5. Control Room Evacuation

Events indicative of loss of Control Room habitability. If the Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

6. Judgment

The EALs defined in other categories specify the predetermined symptoms or events that are indicative of emergency or potential emergency conditions and thus warrant classification. While these EALs have been developed to address the full spectrum of possible emergency conditions which may warrant classification and subsequent implementation of the Emergency Plan, a provision for classification of emergencies based on operator/management experience and judgment is still necessary. The EALs of this category provide the Emergency Director the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Director judgment.

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

**HU1.1 Unusual Event**

Seismic event identified by **any** two of the following:

- SMA-3 Strong Motion Accelograph actuated or Alarm B-3/B-1 SEISMIC EVENT
- Earthquake felt in plant
- National Earthquake Information Center

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.

As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.

The National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant.

**CNS Basis**:

The method of detection with respect to emergency classification relies on the agreement of the shift operators on-duty in the Control Room that the suspected ground motion is a "felt earthquake" as well as the actuation of the CNS seismic instrumentation. Consensus of the Control Room operators with respect to ground motion helps avoid unnecessary classification if the seismic switches inadvertently trip or detect vibrations not related to an earthquake.

CNS seismic instrumentation actuates at 0.01 g. The TS-2 Starter (MI-STR-ACS1), located in the metal enclosure north of the Intake Structure, consists of a vertical starter and a horizontal (bubble type) starter. When the TS-2 Starter senses ground motion greater than 0.01 g, the SMA-3 accelograph actuates causing the following to occur:

- The event indicator changes from black to white.
- The yellow event alarm light turns on.
- Annunciator B-3/B-1, SEISMIC EVENT, alarms.
- The three accelerometers located in the metal enclosure north of the Intake Structure, the Reactor Building NW Quad 859' and the Reactor Building 1001' level are recorded.

The National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant. Refer to the National Earthquake Information Center website:

**http://earthquake.usgs.gov/eqcenter/**

This event escalates to an Alert under EAL HA1.1 if the earthquake exceeds Operating Basis Earthquake (OBE) levels.

**CNS Basis Reference(s):**

1. Procedure 5.1Quake
2. Procedure 4.12 Seismic Instrumentation
3. Procedure 2.3_B-3/B-1
4. USAR Section II-5.2.4 and Table II-5-1

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

| | |
|---|---|
| **HU1.2** | **Unusual Event** |

Tornado striking within Protected Area boundary

**OR**

Sustained high winds > 100 mph

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL is based on a tornado striking (touching down) or high winds within the Protected Area.

Escalation of this emergency classification level, if appropriate, would be based on visible damage, or by other in plant conditions, via HA1.2.

**CNS Basis:**

A tornado striking (touching down) within the Protected Area warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

The design wind pressure for the station and structures is 30 $lb/ft^2$ which is equivalent of sustained winds up to 100 mph (ref. 1).

The Protected Area refers to the designated security area around the process buildings and is depicted in Technical Specifications Figure 4.1-1 (ref. 2).

Sustained winds are of a prolonged duration and, therefore, do not include gusts. Sustained winds are not intermittent or of a transitory nature. Since the inauguration of the Automatic Surface Observation System (ASOS), the National Weather Service has adopted a two minute average standard for its sustained wind definition (ref. 3)

**CNS Basis Reference(s):**


1. USAR Section II-3.2.2
2. Technical Specifications Figure 4.1-1 Site and Exclusion Area Boundaries and Low Population Zone
3. National Weather Service webpage "http://www.aoml.noaa.gov/hrd/tcfaq/D4.html"

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

| **HU1.3** | **Unusual Event** |
|---|---|
| Main turbine failure resulting in casing penetration or damage to turbine or generator seals | |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.

Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual fires and flammable gas build up are appropriately classified via HU2.1 and HU3.1.

This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Escalation of this emergency classification level, if appropriate, would be to HA1.3 based on damage done by projectiles generated by the failure or by any radiological releases.

**CNS Basis:**

The main turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external missiles will be released. These ejected missiles may impact various plant structures, including those housing safety related equipment.

In the event of missile ejection, the probability of a strike on a plant region is a function of the energy and direction of an ejected missile and of the orientation of the turbine with respect to the plant region.

**CNS Basis Reference(s):**

None

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

| **HU1.4 Unusual Event** |
|---|
| Flooding in **any** Table H-1 area that has the potential to affect safety-related equipment needed for the current operating mode |

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps.

Escalation of this emergency classification level, if appropriate, would be based visible damage via HA1.4, or by other plant conditions.

**CNS Basis:**

The internal flooding areas of concern are listed in Table H-1 (ref. 1-6).

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room.

Flooding in these areas could have the potential to cause a reactor trip and could result in consequential failures to important systems. The potential for flooding in these areas was determined by an examination of piping systems in the area and also considered propagation of water from one area to another.

The accumulation of water resulting in a rising water level in the area constitutes flooding.

**CNS Basis Reference(s):**

1. USAR Section XII 2.1.2.1 Principal Class I Structures Required for Safe Shutdown
2. Site Services Procedure 1.1 Station Security
3. Drawing CNS-EE-187 CNS Safe Shutdown Component Locations & Emergency Route Lighting - Site Plan
4. Fire Hazard Analysis for Fire Protection Review to Appendix A Branch Technical Position APCSB 9.5-1
5. CNS-FP-60 Fire Area Boundary Drawing Index
6. USAR Section X.18 Appendix R Safe Shutdown

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting the Protected Area

**EAL:**

**HU1.5 Unusual Event**

High river/forebay water level > 899' MSL

**OR**

Low river level/forebay < 870' MSL

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses other site specific phenomena (such as flood) that can also be precursors of more serious events.

**CNS Basis:**

This EAL covers high river/forebay water level conditions that could be a precursor of more serious events as well as low river/forebay water level conditions which may threaten operability of plant cooling systems

Procedure 5.1FLOOD entry is based on any of the following:

- River water level $\geq$ 895' MSL
- Upstream dam failure
- Projected river water level $\geq$ 902' MSL within next 36 hours

899' MSL is the Probable Maximum Flood level (ref.1).

870' MSL is the Minimum Probable river level (ref. 3). A further level drop may threaten availability of cooling systems and heat sink.

The forebay refers to the area between the east Intake Structure wall and the guide wall (ref. 4).

**CNS Basis Reference(s):**

1. USAR Section II-4.2.2.2
2. Procedure 5.1FLOOD
3. USAR Section II-4.2.3.1
4. USAR Section II-2.7.2

**Category:** H – Hazards

**Subcategory:** 1 - Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.1 Alert**

Seismic event > 0.1g as indicated by SMA-3 Strong Motion Accelograph or Alarm B-3/A-1 EMERGENCY SEISMIC HIGH LEVEL

**AND**

Earthquake confirmed by **any** of the following:

- Earthquake felt in plant
- National Earthquake Information Center
- Control Room indication of degraded performance of systems required for the safe shutdown of the plant

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL escalates from HU1.1 in that the occurrence of the event has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other EALs (e.g., System Malfunction).

Seismic events of this magnitude can result in a vital area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

The National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant.

**CNS Basis**:

CNS seismic instrumentation actuates at 0.01 g. The TS-2 Starter (MI-STR-ACS1), located in the metal enclosure north of the Intake Structure, consists of a vertical starter and a horizontal (bubble type) starter. When the TS-2 Starter senses ground motion greater than 0.01 g, the SMA-3 accelograph actuates causing the following to occur:

- The event indicator changes from black to white.
- The yellow event alarm light turns on.
- Annunciator B-3/B-1, SEISMIC EVENT, alarms.
- The three accelerometers located in the metal enclosure north of the Intake Structure, the Reactor Building NW Quad 859' and the Reactor Building 1001' level are recorded.
- Alarm B-3/A-1, EMERGENCY SEISMIC HIGH LEVEL, is received if the seismic activity exceeds 0.1g.

Seismic event > 0.1g as indicated by SMA-3 Strong Motion Accelograph requires engineering to evaluate the recordings.

The National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant. Refer to the National Earthquake Information Center website:

**http://earthquake.usgs.gov/eqcenter/**

**CNS Basis Reference(s):**

1. Procedure 5.1Quake
2. Procedure 4.12 Seismic Instrumentation
3. Procedure 2.3_B-3/B-1
4. USAR Section II-5.2.4 and Table II-5-1

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.2 Alert**

Tornado striking or high winds > 100 mph resulting in **EITHER**:

Visible damage to **any** Table H-1 area structure containing safety systems or components

**OR**

Control Room indication of degraded performance of safety systems

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL escalates from HU1.2 in that the occurrence of the event has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occurs on the basis of other EALs (e.g., System Malfunction).

This EAL is based on a tornado striking (touching down) or high winds that have caused visible damage to structures containing functions or systems required for safe shutdown of the plant.

**CNS Basis**:

This threshold addresses events that may have resulted in Safe Shutdown Areas being subjected to forces (tornado or high winds > 100 mph) (ref. 1) beyond design limits. Table H-1 safe shutdown areas house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained shutdown (ref. 2-7).

A tornado striking (touching down) within the Protected Area resulting in visible damage warrants declaration of an Alert regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

The design wind pressure for the station and structures is 30 $lb/ft^2$ which is the equivalent of sustained winds up to 100 mph (ref. 1).

The Protected Area refers to the designated security area around the process buildings and is depicted in Technical Specifications Figure 4.1-1 (ref. 8).

**CNS Basis Reference(s):**

1. USAR Section II-3.2.2
2. USAR Section XII 2.1.2.1 Principal Class I Structures Required for Safe Shutdown
3. Site Services Procedure 1.1 Station Security
4. Drawing CNS-EE-187 CNS Safe Shutdown Component Locations & Emergency Route Lighting - Site Plan
5. Fire Hazard Analysis for Fire Protection Review to Appendix A Branch Technical Position APCSB 9.5-1
6. CNS-FP-60 Fire Area Boundary Drawing Index
7. USAR Section X.18 Appendix R Safe Shutdown

8. Technical Specifications Figure 4.1-1 Site and Exclusion Area Boundaries and Low Population Zone

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.3 Alert**

Main turbine failure-generated projectiles result in **EITHER**:

Visible damage to or penetration of **any** Table H-1 area structure containing safety systems or components

**OR**

Control Room indication of degraded performance of safety systems

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

The EAL escalates from HU1.3 in that the occurrence of the event has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occurs on the basis of other EALs (e.g., System Malfunction).

**CNS Basis**:

The main turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external missiles will be released. These ejected missiles may impact various plant structures, including those housing safety related equipment.

In the event of missile ejection, the probability of a strike on a plant region is a function of the energy and direction of an ejected missile and of the orientation of the turbine with respect to the plant region.

The list of Table H-1 areas includes all areas containing safety-related equipment, their controls, and their power supplies (ref. 1-6).

**CNS Basis Reference(s):**

1. USAR Section XII 2.1.2.1 Principal Class I Structures Required for Safe Shutdown
2. Site Services Procedure 1.1 Station Security
3. Drawing CNS-EE-187 CNS Safe Shutdown Component Locations & Emergency Route Lighting - Site Plan
4. Fire Hazard Analysis for Fire Protection Review to Appendix A Branch Technical Position APCSB 9.5-1
5. CNS-FP-60 Fire Area Boundary Drawing Index
6. USAR Section X.18 Appendix R Safe Shutdown

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.4 Alert**

Flooding in **any** Table H-1 area resulting in **EITHER**:

An electrical shock hazard that precludes access to operate or monitor safety equipment

**OR**

Control Room indication of degraded performance of safety systems

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

The EAL escalates from HU1.4 in that the occurrence of the event has resulted in an electrical shock hazard precluding access to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occurs on the basis of other EALs (e.g., System Malfunction).

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

**CNS Basis**:

The internal flooding areas of concern are listed in Table H-1 (ref. 1-6).

Flooding in these areas could have the potential to cause a reactor trip and could result in consequential failures to important systems. The potential for flooding in this area was determined by an examination of piping systems in the area and also considered propagation of water from one area to another.

The accumulation of water resulting in a rising water level in the area constitutes flooding.

**CNS Basis Reference(s):**

1. USAR Section XII 2.1.2.1 Principal Class I Structures Required for Safe Shutdown
2. Site Services Procedure 1.1 Station Security
3. Drawing CNS-EE-187 CNS Safe Shutdown Component Locations & Emergency Route Lighting - Site Plan
4. Fire Hazard Analysis for Fire Protection Review to Appendix A Branch Technical Position APCSB 9.5-1
5. CNS-FP-60 Fire Area Boundary Drawing Index

6. USAR Section X.18 Appendix R Safe Shutdown

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

| | |
|---|---|
| **HA1.5** | **Alert** |

High river/forebay water level > 902' MSL

**OR**

Low river/forebay level < 865' MSL

**Mode Applicability:**

All

**NEI 99-01 Basis:**

Escalation to higher classifications occurs on the basis of other EALs (e.g., System Malfunction).

This EAL addresses other site specific phenomena that result in visible damage to vital areas or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant (such as flood) that can also be precursors of more serious events.

**CNS Basis:**

HU1.5 covers high river/forebay water level conditions that could pose a significant threat to plant safety as well as low river/forebay water level conditions which may threaten operability of vital emergency plant cooling systems.

A river level of 902 ft requires reactor shutdown and represents the maximum possible (10,000 year) flood stage (ref. 5).

A river level of 865 ft MSL corresponds to the Safe Shutdown low river level and threatens availability of cooling systems and heat sink (ref. 1, 2, 3).

The forebay refers to the area between the east Intake Structure wall and the guide wall (ref. 4).

**CNS Basis Reference(s):**

1. USAR Section II-4.2.2.2
2. Procedure 5.1FLOOD
3. USAR Section II-4.2.3.2
4. USAR Section II-2.7.2
5. USAR Section II-4.2.2.1

**Category:** H – Hazards

**Subcategory:** 1 – Natural or Destructive Phenomena

**Initiating Condition:** Natural or destructive phenomena affecting Vital Areas

**EAL:**

**HA1.6 Alert**

Vehicle crash resulting in **EITHER**:

Visible damage to **any** Table H-1 area structure containing safety systems or components

**OR**

Control Room indication of degraded performance of safety systems

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL is based on the occurrence of a vehicle crash that has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other EALs (e.g., System Malfunction).

This EAL addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.

**CNS Basis:**

Table H-1 Safe Shutdown Areas house equipment the operation of which may be needed to ensure the reactor reaches and is maintained in shutdown (ref. 1-6).

The Protected Area refers to the designated security area around the process buildings and is depicted in Technical Specifications Figure 4.1-1 (ref. 7).

If the vehicle crash is determined to be hostile in nature, the event is classified under security based EALs.

**CNS Basis Reference(s):**

1. USAR Section XII 2.1.2.1 Principal Class I Structures Required for Safe Shutdown
2. Site Services Procedure 1.1 Station Security
3. Drawing CNS-EE-187 CNS Safe Shutdown Component Locations & Emergency Route Lighting - Site Plan
4. Fire Hazard Analysis for Fire Protection Review to Appendix A Branch Technical Position APCSB 9.5-1
5. CNS-FP-60 Fire Area Boundary Drawing Index
6. USAR Section X.18 Appendix R Safe Shutdown
7. Technical Specifications Figure 4.1-1 Site and Exclusion Area Boundaries and Low Population Zone

**Category:** H – Hazards

**Subcategory:** 2 – Fire or Explosion

**Initiating Condition:** Fire within the Protected Area not extinguished within 15 minutes of detection or explosion within the Protected Area

**EAL:**

| **HU2.1** **Unusual Event** |
|---|
| Fire in **any** Table H-1 area **not** extinguished within 15 min. of Control Room notification or receipt of a valid Control Room alarm due to fire (Note 3) |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses the magnitude and extent of fires that may be potentially significant precursors of damage to safety systems. It addresses the fire, and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and report by plant personnel or sensor alarm indication.

The 15 minute time period begins with a credible notification that a fire is occurring, or indication of a fire detection system alarm/actuation. Verification of a fire detection system alarm/actuation includes actions that can be taken within the control room or other nearby site specific location to ensure that it is not spurious. An alarm is assumed to be an indication of a fire unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.

The intent of this 15 minute duration is to size the fire and to discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket).

**CNS Basis**:

Fire, as used in this EAL, means combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

**CNS Basis Reference(s):**

1. USAR Section XII 2.1.2.1 Principal Class I Structures Required for Safe Shutdown
2. Site Services Procedure 1.1 Station Security
3. Drawing CNS-EE-187 CNS Safe Shutdown Component Locations & Emergency Route Lighting - Site Plan
4. Fire Hazard Analysis for Fire Protection Review to Appendix A Branch Technical Position APCSB 9.5-1
5. CNS-FP-60 Fire Area Boundary Drawing Index
6. USAR Section X.18 Appendix R Safe Shutdown

**Category:** H – Hazards

**Subcategory:** 2 – Fire or Explosion

**Initiating Condition:** Fire within the Protected Area not extinguished within 15 minutes of detection or explosion within the Protected Area

**EAL:**

| | |
|---|---|
| **HU2.2** | **Unusual Event** |
| Explosion within the Protected Area | |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses the magnitude and extent of explosions that may be potentially significant precursors of damage to safety systems. It addresses the explosion, and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and report by plant personnel or sensor alarm indication.

This EAL addresses only those explosions of sufficient force capable of causing damage to permanent structures or equipment within the Protected Area.

No attempt is made to assess the actual magnitude of any damage. The occurrence of the explosion is sufficient for declaration.

The Emergency director also needs to consider any security aspects of the explosion, if applicable.

Escalation of this emergency classification level, if appropriate, would be based on HA2.1.

**CNS Basis**:

The Protected Area refers to the designated security area around the process buildings and is depicted in Technical Specifications Figure 4.1-1 (ref. 2).

As used here, an explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized equipment that potentially imparts significant energy to nearby structures and materials.

A steam line break or steam explosion that damages surrounding permanent structures or equipment would be classified under this EAL. This does not mean the emergency is classified simply because the steam line break occurred. The method of damage is not as important as the degradation of plant structures or equipment. The need to classify the steam line break itself is considered in fission product barrier degradation monitoring (EAL Category F).

If the explosion is determined to be hostile in nature, the event is classified under security based EALs.

**CNS Basis Reference(s):**

1. Technical Specifications Figure 4.1-1 Site and Exclusion Area Boundaries and Low Population Zone

**Category:** H – Hazards

**Subcategory:** 2 – Fire or Explosion

**Initiating Condition:** Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown

**EAL:**

| | |
|---|---|
| **HA2.1** | **Alert** |

Fire or explosion resulting in **EITHER**:

Visible damage to **any** Table H-1 area containing safety systems or components

**OR**

Control Room indication of degraded performance of safety systems

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

Visible damage is used to identify the magnitude of the fire or explosion and to discriminate against minor fires and explosions.

The reference to structures containing safety systems or components is included to discriminate against fires or explosions in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the fire or explosion was large enough to cause damage to these systems.

The use of visible damage should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform detailed damage assessments.

The Emergency Director also needs to consider any security aspects of the explosion.

Escalation of this emergency classification level, if appropriate, will be based on System Malfunctions, Fission Product Barrier Degradation or Abnormal Rad Levels / Radiological Effluent EALs.

**CNS Basis**:

Fire, as used in this EAL, means combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

An explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized equipment that potentially imparts significant energy to nearby structures and materials.

A steam line break or steam explosion that damages permanent structures or equipment would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment. The need to classify the steam line break itself is considered in fission product barrier degradation monitoring (EAL Category F).

**CNS Basis Reference(s):**

1. USAR Section XII 2.1.2.1 Principal Class I Structures Required for Safe Shutdown
2. Site Services Procedure 1.1 Station Security
3. Drawing CNS-EE-187 CNS Safe Shutdown Component Locations & Emergency Route Lighting - Site Plan
4. Fire Hazard Analysis for Fire Protection Review to Appendix A Branch Technical Position APCSB 9.5-1
5. CNS-FP-60 Fire Area Boundary Drawing Index
6. USAR Section X.18 Appendix R Safe Shutdown

**Category:** H – Hazards

**Subcategory:** 3 – Hazardous Gas

**Initiating Condition:** Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations

**EAL:**

| | |
|---|---|
| **HU3.1** | **Unusual Event** |
| Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could affect normal plant operations | |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations.

The fact that SCBA may be worn does not eliminate the need to declare the event.

This EAL is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

Escalation of this emergency classification level, if appropriate, would be based on HA3.1.

**CNS Basis**:

As used in this EAL, affecting normal plant operations means that activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures have been impacted. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from normal plant operations and thus would be considered to have been affected. Administrative Procedure 0.36.6, MONITORING FOR INDUSTRIAL GASES, may be used for help in assessing this EAL. Such review, however, does not constitute a departure from normal operations.

The release may have originated within the Site Boundary, or it may have originated offsite and subsequently drifted onto the Site Boundary. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on normal plant operations.

At CNS there are various potential sources of atmospheric contamination. Some of these sources are:

- Inert gas used for oxygen exclusion (nitrogen).
- Combustion products.
- Carbon dioxide from fire extinguishing.
- Welding gases (enclosed areas).
- Vapors from painting (enclosed areas).
- Vapors from petroleum products.
- Hydrogen (OWC hydrogen gas generation system, generator cooling system, batteries, and disassociation of water in the reactor).
- Asphyxiants and irritants, found most often in confined areas (water and oil storage tanks, open manholes).
- Methane from bacterial action (tanks and pits).

Some of the gases which could affect normal plant operations under this EAL are:

- Carbon Monoxide - One of the most common asphyxiants encountered in industry. It is formed by the incomplete combustion of fuel containing carbon. It may be found in the vicinity of a fire or a leak in an exhaust system (flue gas or internal combustion engines).
- Oxygen - Oxygen has two fundamentally important properties: it supports combustion and it supports life. Since oxygen is necessary for life it must be present in sufficient quantity. Oxygen deficiency occurs in confined spaces where the level of oxygen has been reduced below the limit to support life. Oxygen content in the air can become fatally low in a brief period of time. Some of the more common causes of this problem are oxidation of metals, bacterial action, combustion, and displacement by other gases. An enriched oxygen atmosphere will accelerate combustion.
- Hydrogen - Used in generator cooling. Hydrogen gas is also produced by the OWC Gas Generation System, located in the OWC Building, and subsequently injected into the condensate system just upstream of the condensate booster pumps. Hydrogen is also produced by the disassociation of water from radiation in the reactor, which is seen in the off-gas. The presence of hydrogen will be especially significant in the Off-Gas and Augmented Off-Gas Systems. Hydrogen is also a by-product of battery charging. It is lighter than air so it will be found in pockets at the ceiling of enclosures.
- Argon - Commonly used during the welding of certain metals. It is denser than air so it will settle in pockets below the welding area.
- Carbon Dioxide - Used to fight fire. Being heavier than air, carbon dioxide will settle in pockets and displace oxygen.
- Nitrogen - Used primarily to purge primary containment. Since it is approximately the same density as air, it can be dispersed by proper ventilation. Areas of poor ventilation may contain greater than expected concentrations of nitrogen and consequently may be deficient in oxygen.

- Combustible Gases and Vapors - Includes naturally occurring gases (such as methane and hydrogen gas) and the vapors of a large group of liquids which are used as fuels and solvents. Monitoring shall be required in fuel tanks and other areas where explosive mixtures may be present.

- Hydrogen Sulfide - Classified as an irritant in low concentrations, but is even more toxic than carbon monoxide, because it inflames the mucus membranes and results in the lungs filling with fluid. This colorless gas has a characteristic rotten egg odor, which renders the sense of smell ineffective. Hydrogen sulfide may be found in sewage treatment or wherever organic matter containing sulfur decomposes and shall be monitored constantly during work.

- Methane - The chief constituent of natural gas and is extremely explosive. It is non-toxic, but may reduce the oxygen content of an atmosphere, causing asphyxiation. Methane is often found in the vicinity of sanitary landfills and has been detected in tanks where bacterial action is taking place (i.e., reactor water cleanup and condensate phase separator tanks). It is lighter than air and tends to accumulate in high spots or pockets. This can present a dangerous situation in storage tanks or sewers where access is normally gained at the top of the confined area.

- Ethyl Benzene - Used primarily as an additive to diesel fuel. Acute exposure results in a local irritant effect on the skin and mucous membranes. Chronic exposure can lead to nervous system disorders and upper respiratory tract inflammation. Monitoring is required when entering a diesel fuel tank.

- Chlorine - Used in chemical treatment of Circulate Water and Service Water Systems. Chlorine gas can be recognized by its pungent, irritating odor, which is like the odor of bleach. Chlorine is not flammable but can react explosively with other chemicals such as turpentine or ammonia. Chlorine gas stays close to the ground and spreads rapidly. When chlorine gas comes in contact with moist human tissues, such as the eyes throat and lungs, an acid is produced that can damage these tissues.

- Chlorine Dioxide - This is a yellow to reddish-yellow manufactured gas which does not occur naturally in the environment. When added to water, chlorine dioxide forms chlorite ion, which is also a very reactive chemical. High levels of chlorine dioxide can be irritating to the nose, eyes, throat, and lungs.

- Hydrogen Chloride - This is a colorless to slightly yellowish gas with a pungent odor. On exposure to air, the gas forms dense white vapors due to condensation with atmospheric moisture. The vapor is corrosive and air concentrations above 5 ppm can cause irritation. When mixed with water or atmospheric moisture, a highly corrosive atmosphere is formed. The most common source of Hydrogen Chloride gas is from Muriatic (Hydrochloric) Acid.

Should the release affect access to plant Safe Shutdown Areas, escalation to an Alert would be based on EAL HA3.1. Should an explosion or fire occur due to flammable gas within an affected plant area, an Alert may be appropriate based on EAL HA2.1.

**CNS Basis Reference(s):**

1. Procedure 0.36.6 MONITORING FOR INDUSTRIAL GASES

**Category:** H – Hazards

**Subcategory:** 3 – Toxic, Corrosive, Asphyxiant & Flammable Gas

**Initiating Condition:** Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations

**EAL:**

| | |
|---|---|
| **HU3.2** | **Unusual Event** |
| Recommendation by local, county or state officials to evacuate or shelter site personnel based on an offsite event | |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations.

The fact that SCBA may be worn does not eliminate the need to declare the event.

This EAL is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

Escalation of this emergency classification level, if appropriate, would be based on HA3.1.

**CNS Basis**:

This EAL is based on the existence of an uncontrolled release originating offsite and local, county or state officials have reported the need for evacuation or sheltering of site personnel. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) are considered in this EAL because they may adversely affect normal plant operations.

State officials may determine the evacuation area for offsite spills by using the Department of Transportation (DOT) Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials. If the evacuation area extends to any portion of the Owner Controlled Area, the EAL threshold is met.

Should the release affect plant Safe Shutdown Areas, escalation to an Alert would be based on EAL HA3.1. Should an explosion or fire occur due to flammable gas within an affected plant area, an Alert may be appropriate based on EAL HA2.1.

**CNS Basis Reference(s):**

1. Procedure 0.36.6 MONITORING FOR INDUSTRIAL GASES

**Category:** H – Hazards

**Subcategory:** 3 – Toxic, Corrosive, Asphyxiant & Flammable Gas

**Initiating Condition:** Access to a vital area is prohibited due to release of toxic, corrosive, asphyxiant or flammable gases which jeopardizes operation of operable equipment required to maintain safe operations or safely shutdown the reactor

**EAL:**

| | |
|---|---|
| **HA3.1** | **Alert** |
| Access to **any** Table H-1 area is prohibited due toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor (Note 7) | |

**Note 7:** If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should **not** be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

Gases in a Safe Shutdown Area can affect the ability to safely operate or safely shutdown the reactor.

The fact that SCBA may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gasses which can ignite/support combustion.

Escalation of this emergency classification level, if appropriate, will be based on System Malfunctions, Fission Product Barrier Degradation or Abnormal Rad Levels / Radioactive Effluent EALs.

**CNS Basis**:

This EAL is based on gases that have entered a plant structure in concentrations that could be unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Table H-1 safe shutdown areas contain systems that are operated to establish or maintain safe shutdown (ref. 1-6).

At CNS there are various potential sources of atmospheric contamination. Some of these sources are:

- Inert gas used for oxygen exclusion (nitrogen).
- Combustion products.
- Carbon dioxide from fire extinguishing.
- Welding gases (enclosed areas).
- Vapors from painting (enclosed areas).
- Vapors from petroleum products.
- Hydrogen (OWC hydrogen gas generation system, generator cooling system, batteries, and disassociation of water in the reactor).
- Asphyxiants and irritants, found most often in confined areas (water and oil storage tanks, open manholes).
- Methane from bacterial action (tanks and pits).

Some of the gases which could affect normal plant operations under this EAL are:

- Carbon Monoxide - One of the most common asphyxiants encountered in industry. It is formed by the incomplete combustion of fuel containing carbon. It may be found in the vicinity of a fire or a leak in an exhaust system (flue gas or internal combustion engines).

- Oxygen - Oxygen has two fundamentally important properties: it supports combustion and it supports life. Since oxygen is necessary for life it must be present in sufficient quantity. Oxygen deficiency occurs in confined spaces where the level of oxygen has been reduced below the limit to support life. Oxygen content in the air can become fatally low in a brief period of time. Some of the more common causes of this problem are oxidation of metals, bacterial action, combustion, and displacement by other gases. An enriched oxygen atmosphere will accelerate combustion.

- Hydrogen - Used in generator cooling. Hydrogen gas is also produced by the OWC Gas Generation System, located in the OWC Building, and subsequently injected into the condensate system just upstream of the condensate booster pumps. Hydrogen is also produced by the disassociation of water from radiation in the reactor, which is seen in the off-gas. The presence of hydrogen will be especially significant in the Off-Gas and Augmented Off-Gas Systems. Hydrogen is also a by-product of battery charging. It is lighter than air so it will be found in pockets at the ceiling of enclosures.

- Argon - Commonly used during the welding of certain metals. It is denser than air so it will settle in pockets below the welding area.

- Carbon Dioxide - Used to fight fire. Being heavier than air, carbon dioxide will settle in pockets and displace oxygen.

- Nitrogen - Used primarily to purge primary containment. Since it is approximately the same density as air, it can be dispersed by proper ventilation. Areas of poor ventilation may contain greater than expected concentrations of nitrogen and consequently may be deficient in oxygen.

- Combustible Gases and Vapors - Includes naturally occurring gases (such as methane and hydrogen gas) and the vapors of a large group of liquids which are used as fuels and solvents. Monitoring shall be required in fuel tanks and other areas where explosive mixtures may be present.

- Hydrogen Sulfide - Classified as an irritant in low concentrations, but is even more toxic than carbon monoxide, because it inflames the mucus membranes and results in the lungs filling with fluid. This colorless gas has a characteristic rotten egg odor, which renders the sense of smell ineffective. Hydrogen sulfide may be found in sewage treatment or wherever organic matter containing sulfur decomposes and shall be monitored constantly during work.

- Methane - The chief constituent of natural gas and is extremely explosive. It is non-toxic, but may reduce the oxygen content of an atmosphere, causing asphyxiation.

Methane is often found in the vicinity of sanitary landfills and has been detected in tanks where bacterial action is taking place (i.e., reactor water cleanup and condensate phase separator tanks). It is lighter than air and tends to accumulate in high spots or pockets. This can present a dangerous situation in storage tanks or sewers where access is normally gained at the top of the confined area.

- Ethyl Benzene - Used primarily as an additive to diesel fuel. Acute exposure results in a local irritant effect on the skin and mucous membranes. Chronic exposure can lead to nervous system disorders and upper respiratory tract inflammation. Monitoring is required when entering a diesel fuel tank.

- Chlorine - Used in chemical treatment of Circulate Water and Service Water Systems. Chlorine gas can be recognized by its pungent, irritating odor, which is like the odor of bleach. Chlorine is not flammable but can react explosively with other chemicals such as turpentine or ammonia. Chlorine gas stays close to the ground and spreads rapidly. When chlorine gas comes in contact with moist human tissues, such as the eyes throat and lungs, an acid is produced that can damage these tissues.

- Chlorine Dioxide - This is a yellow to reddish-yellow manufactured gas which does not occur naturally in the environment. When added to water, chlorine dioxide forms chlorite ion, which is also a very reactive chemical. High levels of chlorine dioxide can be irritating to the nose, eyes, throat, and lungs.

- Hydrogen Chloride - This is a colorless to slightly yellowish gas with a pungent odor. On exposure to air, the gas forms dense white vapors due to condensation with atmospheric moisture. The vapor is corrosive and air concentrations above 5 ppm can cause irritation. When mixed with water or atmospheric moisture, a highly corrosive atmosphere is formed. The most common source of Hydrogen Chloride gas is from Muriatic (Hydrochloric) Acid.

This EAL does not apply to routine inerting of the primary containment.

**CNS Basis Reference(s):**

1. USAR Section XII 2.1.2.1 Principal Class I Structures Required for Safe Shutdown
2. Site Services Procedure 1.1 Station Security
3. Drawing CNS-EE-187 CNS Safe Shutdown Component Locations & Emergency Route Lighting - Site Plan
4. Fire Hazard Analysis for Fire Protection Review to Appendix A Branch Technical Position APCSB 9.5-1
5. CNS-FP-60 Fire Area Boundary Drawing Index
6. USAR Section X.18 Appendix R Safe Shutdown
7. Procedure 0.36.6 MONITORING FOR INDUSTRIAL GASES

**Category:** H – Hazards

**Subcategory:** 4 – Security

**Initiating Condition:** Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant

**EAL:**

| | |
|---|---|
| **HU4.1** | **Unusual Event** |

A security condition that does **not** involve a hostile action as reported by the Security Shift Supervisor

**OR**

A credible site-specific security threat notification

**OR**

A validated notification from NRC providing information of an aircraft threat

**Mode Applicability:**

All

**NEI 99-01 Basis:**

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as hostile actions are classifiable under HA4.1, HS4.1 and HG4.1.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. Consideration should be given to upgrading the emergency response status and emergency classification level in accordance with the Physical Security Plan and Emergency Plan.

## 1st Threshold

Reference is made to the Security Shift Supervisor because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan.

This threshold is based on the CNS Security and Safeguards Contingency Plan. Security Plans are based on guidance provided by NEI 03-12.

## 2nd Threshold

The second threshold is to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Unusual Event.

The determination of "credible" is made through use of information found in the Physical Security Plan.

## 3rd Threshold

The intent of this EAL threshold is to ensure that notifications for the aircraft threat are made in a timely manner and that off-site response organizations and plant personnel are at a state of heightened awareness regarding the credible threat.

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level via HA4.1 would be appropriate if the threat involves an airliner less than 30 minutes away from the plant.

**CNS Basis:**

**Hostile Action**: An act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile Action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

**CNS Basis Reference(s):**

1. CNS Security and Safeguards Contingency Plan

**Category:** H – Hazards

**Subcategory:** 4 – Security

**Initiating Condition:** Hostile action within the owner controlled area or airborne attack threat

**EAL:**

| | |
|---|---|
| **HA4.1** | **Alert** |

A hostile action is occurring or has occurred within the **Owner Controlled Area** as reported by the Security Shift Supervisor

**OR**

A validated notification from NRC of an airliner attack threat within 30 min. of the site

**Mode Applicability:**

All

**NEI 99-01 Basis:**

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

These EAL thresholds address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

**<u>1st Threshold</u>**

This EAL threshold addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Owner Controlled Area. Those events are adequately addressed by other EALs.

Although nuclear plant security officers are well trained and prepared to protect against hostile action, it is appropriate for off-site response organizations to be notified and encouraged to begin activation (if they do not normally) to be better prepared should it be necessary to consider further actions.

**2nd Threshold**

This EAL threshold addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this EAL threshold is to ensure that notifications for the airliner attack threat are made in a timely manner and that off-site response organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This EAL threshold is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner. The status and size of the plane may be provided by NORAD through the NRC.

If not previously notified by the NRC that the airborne hostile action was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

**CNS Basis:**

Reference is made to the Security Shift Supervisor because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred.

**Hostile Action**: An act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile Action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

**CNS Basis Reference(s):**

1. CNS Security and Safeguards Contingency Plan

**Category:** H – Hazards

**Subcategory:** 4 – Security

**Initiating Condition:** Hostile action within the Protected Area

**EAL:**

**HS4.1 Site Area Emergency**

A hostile action is occurring or has occurred within the **Protected Area** as reported by the Security Shift Supervisor

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This condition represents an escalated threat to plant safety above that contained in the Alert in that a hostile force has progressed from the Owner Controlled Area to the Protected Area.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires off-site response organizations readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Protected Area.

Although nuclear plant security officers are well trained and prepared to protect against hostile action, it is appropriate for off-site response organizations to be notified and encouraged to begin preparations for public protective actions (if they do not normally) to be better prepared should it be necessary to consider further actions.

If not previously notified by NRC that the airborne hostile action was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

**CNS Basis**:

Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs. Reference is made to the Security Shift Supervisor because this individual is the designated on-site person qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the CNS Security and Safeguards Contingency Plan (ref. 1).

**Hostile Action**: An act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile Action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

**CNS Basis Reference(s):**

1. CNS Security and Safeguards Contingency Plan

**Category:** H – Hazards

**Subcategory:** 4 – Security

**Initiating Condition:** Hostile action resulting in loss of physical control of the facility

**EAL:**

**HG4.1 General Emergency**

A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions

**OR**

A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool

**Mode Applicability:**

All

**NEI 99-01 Basis:**

**<u>1st Threshold</u>**

This EAL threshold encompasses conditions under which a hostile action has resulted in a loss of physical control of Vital Areas (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.

These safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink). .

Loss of physical control of the control room or remote shutdown capability alone may not prevent the ability to maintain safety functions per se. Design of the remote shutdown capability and the location of the transfer switches should be taken into account. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

**2nd Threshold**

This EAL threshold addresses failure of spent fuel cooling systems as a result of hostile action if imminent fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool.

**Basis**:

A freshly off-loaded reactor core in pool consists of recently discharged fuel that has been out of the reactor for less than one year (ref. 1).

**CNS Basis Reference(s):**

1. Procedure 10.6 (restricted access for B.5.b)
2. CNS Security and Safeguards Contingency Plan

**Category:** H – Hazards

**Subcategory:** 5 – Control Room Evacuation

**Initiating Condition:** Control Room evacuation has been initiated

**EAL:**

| **HA5.1** **Alert** |
|---|
| Control Room evacuation has been initiated |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facilities may be necessary.

Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

**CNS Basis**:

Procedures 5.1ASD, Alternate Shutdown (ref. 1) and 5.4FIRE-S/D, Fire Induced Shutdown From Outside The Control Room (ref. 2), provide the instructions for scramming the unit, and maintaining RCS inventory from outside the Control Room. The Shift Manager (SM) determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions.

**CNS Basis Reference(s):**

1. Procedure 5.1ASD Alternate Shutdown
2. Procedure 5.4FIRE-S/D Fire Induced Shutdown From Outside The Control Room

**Category:** H – Hazards

**Subcategory:** 5 – Control Room Evacuation

**Initiating Condition:** Control Room evacuation has been initiated and plant control **cannot** be established

**EAL:**

| | |
|---|---|
| **HS5.1** | **Site Area Emergency** |

Control Room evacuation has been initiated

**AND**

Control of the plant **cannot** be established within 15 min.

**Mode Applicability:**

All

**NEI 99-01 Basis:**

The intent of this EAL is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated). The fifteen minute time for transfer starts when the Control Room begins to be evacuated (not when Procedure 5.1ASD, Alternate Shutdown, is entered). The time interval is based on how quickly control must be reestablished without core uncovery and/or core damage.

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink).

The determination of whether or not control is established from outside the Control Room is based on Emergency Director (ED) judgment. The Emergency Director is expected to make a reasonable, informed judgment that control of the plant from the Alternate Shutdown Panels cannot be established within the fifteen minute interval.

. Escalation of this emergency classification level, if appropriate, would be by Fission Product Barrier Degradation or Abnormal Rad Levels/Radiological Effluent EALs.

**CNS Basis**:

Procedures 5.1ASD, Alternate Shutdown (ref. 1) and 5.4FIRE-S/D, Fire Induced Shutdown From Outside The Control Room (ref. 2), provide the instructions for scramming the unit, and maintaining RCS inventory from outside the Control Room. The Shift Manager determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions.

The 15 minute criterion applies from the time that the Control Room begins to be evacuated.

**CNS Basis Reference(s):**

1. Procedure 5.1ASD Alternate Shutdown
2. Procedure 5.4FIRE-S/D Fire Induced Shutdown From Outside The Control Room

**Category:** H – Hazards

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE

**EAL:**

**HU6.1 Unusual Event**

Other conditions exist which in the judgment of the Emergency Director indicate that **EITHER**:

Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant

**OR**

A security threat to facility protection has been initiated

**No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event emergency class.

**CNS Basis:**

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the CNS Emergency Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual

usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

**CNS Basis Reference(s):**

1. Emergency Plan for Cooper Nuclear Station, Section 5.0

**Category:** H – Hazards

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert

**EAL:**

| **HA6.1** | **Alert** |
|---|---|
| Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:<br>An actual or potential substantial degradation of the level of safety of the plant<br>**OR**<br>A security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action<br>**Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels beyond the site boundary. | |

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

**CNS Basis:**

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the CNS Emergency Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual

usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref.1).

For the purposes of this EAL, the Site Boundary for CNS is a one mile radius around the plant.

**CNS Basis Reference(s):**

1. Emergency Plan for Cooper Nuclear Station, Section 5.0
2. CNS Drawing DWG.2.2 (P3-A-45)

**Category:** H – Hazards

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions exist which in the judgment of the Emergency Director warrant declaration of Site Area Emergency

**EAL:**

**HS6.1 Site Area Emergency**

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:

An actual or likely major failures of plant functions needed for protection of the public

**OR**

Hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public

**Any** releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary.

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

**CNS Basis**:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the CNS Emergency Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

For the purposes of this EAL, the Site Boundary for CNS is a one mile radius around the plant.

**CNS Basis Reference(s):**

1. Emergency Plan for Cooper Nuclear Station, Section 5.0
2. CNS Drawing DWG.2.2 (P3-A-45)

**Category:** H – Hazards

**Subcategory:** 6 – Judgment

**Initiating Condition:** Other conditions exist which in the judgment of the Emergency Director warrant declaration of General Emergency

**EAL:**

**HG6.1 General Emergency**

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:

Actual or imminent substantial core degradation or melting with potential for loss of containment integrity

**OR**

Hostile action that results in an actual loss of physical control of the facility

Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary.

**Mode Applicability:**

All

**NEI 99-01 Basis:**

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

**CNS Basis**:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the CNS Emergency Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual

usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the Site Boundary.

For the purposes of this EAL, the Site Boundary for CNS is a one mile radius around the plant.

**CNS Basis Reference(s):**

1. Emergency Plan for Cooper Nuclear Station, Section 5.0
2. CNS Drawing DWG.2.2 (P3-A-45)

## Category S – System Malfunction

EAL Group: Modes 1, 2 or 3

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category involves total losses of vital plant 125 VDC power sources.

2. ATWS / Criticality

Events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor scrams. In the plant licensing basis, postulated failures of the RPS to complete a reactor scram comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification however, ATWS is intended to mean any scram failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to Fuel Clad, RCS and Containment integrity. Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

3. Inability to Reach Shutdown Conditions

One EAL falls into this subcategory. It is related to the failure of the plant to be brought to the required plant operating condition required by technical specifications if a limiting condition for operation (LCO) is not met.

4. Instrumentation / Communications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Loss of annunciators or indicators is in this subcategory.

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

5. Fuel Clad Degradation

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (1% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

6. RCS Leakage

The Reactor Vessel provides a volume for the coolant that covers the reactor core. The Reactor Vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

Excessive RCS leakage greater than Technical Specification limits are utilized to indicate potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Containment integrity.

**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of Power

**Initiating Condition:** Loss of **all** off-site AC power to critical buses for 15 minutes or longer

**EAL:**

| **SU1.1** | **Unusual Event** |
|---|---|
| Loss of **all** offsite AC power (Table S-3) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3) | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

| **Table S-3 AC Power Sources** |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

**CNS Basis**:

The 4160V critical buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following offsite sources of power: Figure S-1 illustrates the 4160V AC distribution system (ref. 1, 2).

- Startup Transformer – The Startup Transformer provides a source of offsite AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of offsite AC power to the station whenever the main generator is off-line (<160 Mwe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the Station Normal Transformer fails or that the main generator trips off-line.

- Emergency Transformer -The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.

- Backfeeding power from the 345 kv line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed its power to the safety-related buses may be considered an offsite power source.

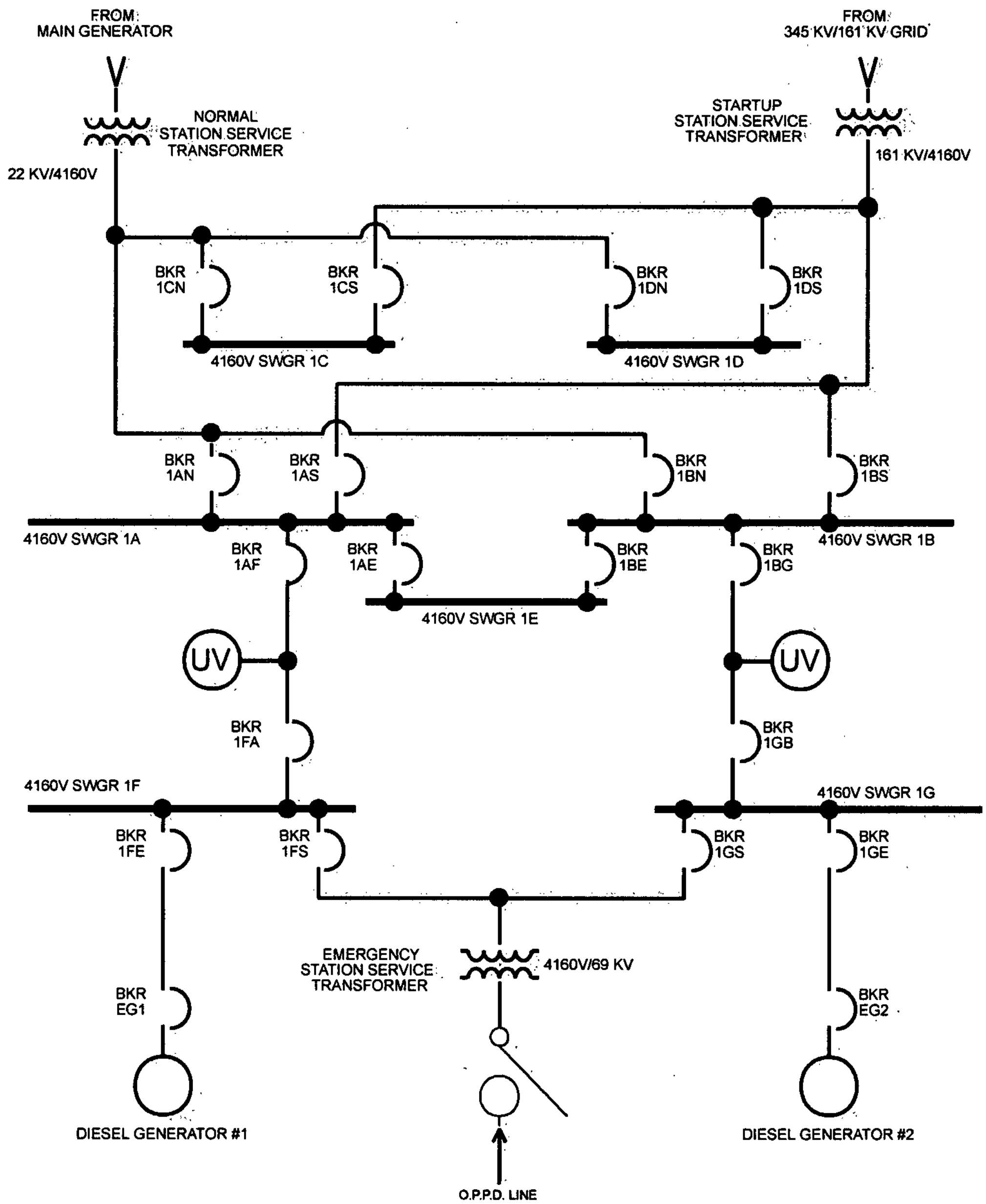
Onsite power sources are the emergency diesel generators (DG-1 and DG-2).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. If neither emergency bus is energized by an offsite source within 15 minutes, an Unusual Event is declared under this EAL.

**CNS Basis Reference(s):**

1. BR 3001 One Line Diagram
2. BR 3002 sh1
3. Procedure 2.1.15 Startup Transformer
4. Procedure 2.1.16 Normal Station Service Transformer
5. Procedure 2.1.17 Emergency Station Service Transformer
6. Procedure 2.2.18 4160V Auxiliary Power Distribution System
7. Procedure 2.2.20 Standby AC Power System (Diesel Generator)
8. Procedure 5.3SBO Station Blackout
9. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Rev. 2

## Figure S-1: 4160V AC Distribution System



**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of Power

**Initiating Condition:** AC power capability to critical buses reduced to a single power source for 15 minutes or longer such that **any** additional single failure would result in loss of **all** AC power to critical buses

**EAL:**

| **SA1.1** | **Alert** |
|---|---|
| AC power capability to critical 4160V buses 1F and 1G reduced to a single power source (Table S-3) for ≥ 15 min. (Note 3) such that **any** additional single failure would result in loss of **all** AC power to emergency buses | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| **Table S-3 AC Power Sources** |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

This EAL is intended to provide an escalation from IC SU1, "Loss of all off-site AC power to critical buses for 15 minutes or longer."

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a loss of all AC power to the critical buses. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its critical bus. Another related condition could be the loss of all off-site power and loss of on-site emergency generators with only one train of critical buses being backfed from the unit main generator, or the loss of on-site emergency generators with only one train of critical buses being backfed from off-site power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with SS1.1.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

. **CNS Basis**:

The 4160V critical buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following offsite sources of power: Figure S-1 illustrates the 4160V AC distribution system (ref. 1, 2).

- Startup Transformer – The Startup Transformer provides a source of offsite AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of offsite AC power to the station whenever the main generator is off-line (<160 Mwe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the Station Normal Transformer fails or that the main generator trips off-line.

- Emergency Transformer -The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.

- Backfeeding power from the 345 kv line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed its power to the safety-related buses may be considered an offsite power source.

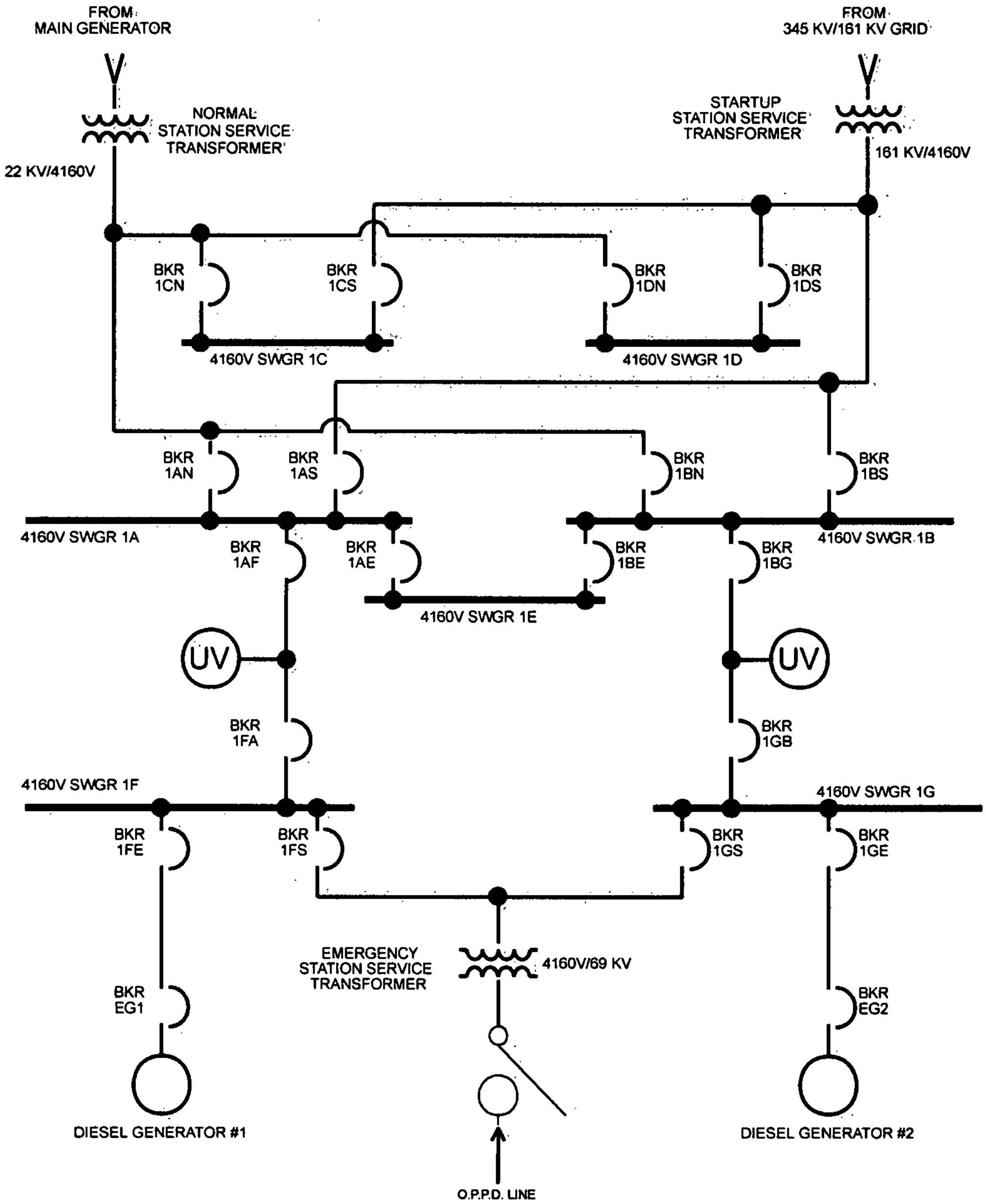
Onsite power sources are the emergency diesel generators (DG-1 and DG-2).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL.

**CNS Basis Reference(s):**

1. BR 3001 One Line Diagram
2. BR 3002 sh1
3. Procedure 2.1.15 Startup Transformer
4. Procedure 2.1.16 Normal Station Service Transformer
5. Procedure 2.1.17 Emergency Station Service Transformer
6. Procedure 2.2.18 4160V Auxiliary Power Distribution System
7. Procedure 2.2.20 Standby AC Power System (Diesel Generator)
8. Procedure 5.3SBO Station Blackout
9. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Rev. 2

# Figure S-1: 4160V AC Distribution System



**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of Power

**Initiating Condition:** Loss of **all** off-site and **all** on-site AC power to critical buses for 15 minutes or longer

**EAL:**

| **SS1.1** | **Site Area Emergency** |
|---|---|
| Loss of **all** offsite and **all** onsite AC power (Table S-3) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3) | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| **Table S-3 AC Power Sources** |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

Loss of all AC power to critical buses compromises all plant safety systems requiring electric power including RHR, ECCS, containment heat removal and the ultimate heat sink. Prolonged loss of all AC power to critical buses will lead to loss of Fuel Clad, RCS, and Primary Containment, thus this event can escalate to a General Emergency.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of off-site power.

Escalation to General Emergency is via Fission Product Barrier Degradation or IC SG1, "Prolonged loss of all off-site power and prolonged loss of all on-site AC power."

**CNS Basis**:

The 4160V critical buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following offsite sources of power: Figure S-1 illustrates the 4160V AC distribution system (ref. 1, 2).

- Startup Transformer – The Startup Transformer provides a source of offsite AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of offsite AC power to the station whenever the main generator is off-line (<160 Mwe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the Station Normal Transformer fails or that the main generator trips off-line.

- Emergency Transformer -The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.

- Backfeeding power from the 345 kv line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed its power to the safety-related buses may be considered an offsite power source.

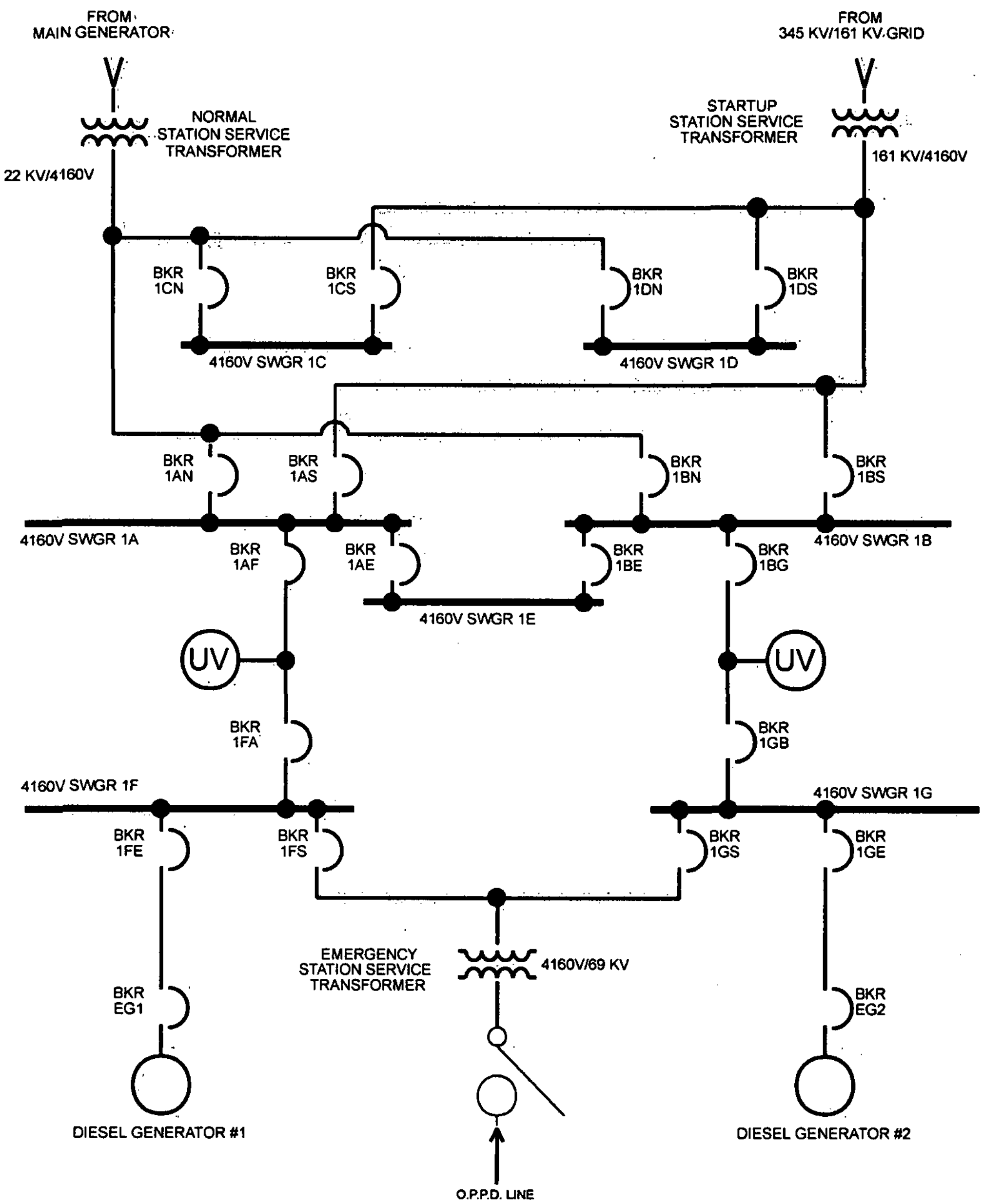
Onsite power sources are the emergency diesel generators (DG-1 and DG-2).

This EAL is the hot condition equivalent of the cold condition loss of all AC power EAL CA1.1. When in Cold Shutdown, Refueling, or Defueled mode, the event can be classified as an Alert because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the critical buses, relative to that existing when in hot conditions.

**CNS Basis Reference(s):**

1. BR 3001 One Line Diagram
2. BR 3002 sh1
3. Procedure 2.1.15 Startup Transformer
4. Procedure 2.1.16 Normal Station Service Transformer
5. Procedure 2.1.17 Emergency Station Service Transformer
6. Procedure 2.2.18 4160V Auxiliary Power Distribution System
7. Procedure 2.2.20 Standby AC Power System (Diesel Generator)
8. Procedure 5.3SBO Station Blackout
9. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Rev. 2

**Figure S-1: 4160V AC Distribution System**



**Category:** S – System Malfunction

**Subcategory:** 1 – Loss of Power

**Initiating Condition:** Loss of **all** vital DC power for 15 minutes or longer

**EAL:**

| | |
|---|---|
| **SS1.2** | **Site Area Emergency** |
| < 105 VDC bus voltage indications on **all** vital 125 VDC buses (1A and 1B) for ≥ 15 min. (Note 3) | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation.

**CNS Basis**:

105 VDC is the minimum design bus voltage (ref. 2).

The 125 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. 125 VDC Distribution Panels supply control and instrument power for annunciators control logic power, and protective relaying. Figure S-2 illustrates the 125 VDC power system (ref. 1).

If 125 VDC Distribution Panel A is lost, the following major equipment is affected : RRMG A speed and breaker control, 4160V Bus 1A, 1 E, and 1 F breaker control and undervoltage logics, 480V Bus 1A and 1 F breaker control, the right light in all Control Room annunciators, annunciator panels for Water Treatment, SW A Gland Water , RHR A Gland Water, Auxiliary Steam Boiler C, DG-1 starting and breaker control logics, CS A, RCIC, and RHR A control logics, TIP valve control monitors, main generator voltage regulation, RFPT A trip logic, and ARI solenoid valve power .

If 125 VDC Distribution Panel B is lost, the following major equipment is affected; RRMG B speed and breaker control, 4160V Bus 1B and 1G breaker control and undervoltage logics, 480V Bus 1B and 1G breaker control, the left light in all Control Room annunciators, annunciator panels for ALRW, SW B Gland Water, RHR B Gland Water, Auxiliary Steam Boiler D, DG-2 starting and breaker control logics, CS B , HPCI, and RHR B control logics, main generator trip logic, main generator and transformer protective relaying, bypass valves fail to control pressure after turbine trip and RFPT B trip logic .

Battery chargers receive their power from 460V critical motor control centers. Each 125 VDC bus receives power from either a 125 VDC battery or a 125 VDC battery charger. The battery chargers receive their power from 460V critical motor control centers. The 250 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. Although 250 VDC Buses 1A and 1B provide vital DC emergency power, 250 VDC Safety System loads (such as motor operated valves) also require 125 VDC control power. Loss of 125 VDC buses alone, therefore, would render most Safeguard System loads inoperable (ref. 2, 3, 4).

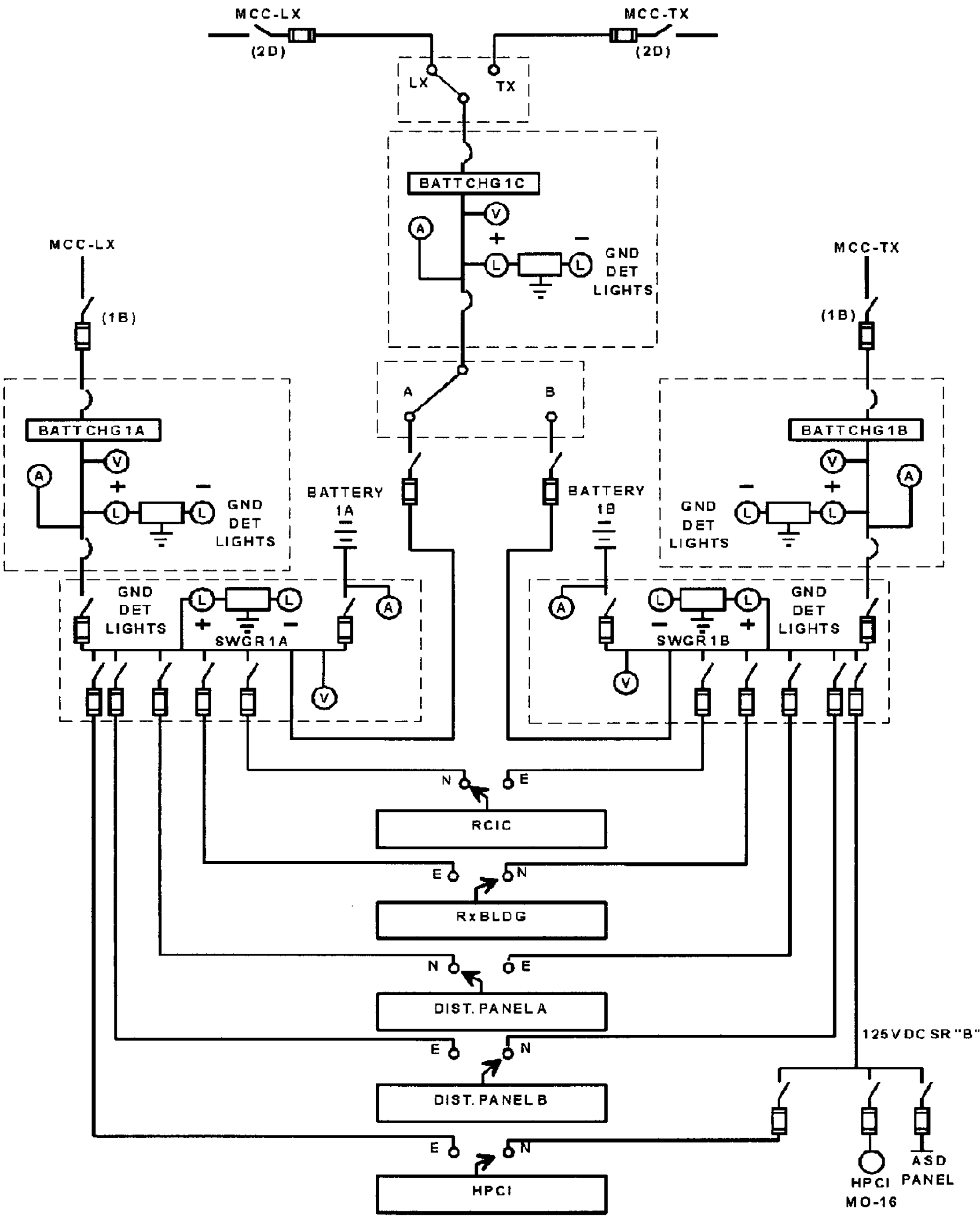
This EAL is the hot condition equivalent of the cold condition loss of DC power EAL CU1.2.

**CNS Basis Reference(s):**

1. BR 3058 DC One Line Diagram
2. Technical Specifications B 3.8.4
3. USAR Section VIII-6.2
4. USAR Section VIII-6.3
5. 5.3DC125 LOSS OF 125 VDC

6. 6.EE.607 125V STATION BATTERY PERFORMANCE DISCHARGE TEST

## Figure S-2: 125 VDC Power System



**Category:** S –System Malfunction

**Subcategory:** 1 – Loss of Power

**Initiating Condition:** Prolonged loss of **all** off-site and **all** on-site AC power to critical emergency busses

**EAL:**

**SG1.1 General Emergency**

Loss of **all** offsite and **all** onsite AC power (Table S-3) to critical 4160V buses 1F and 1G

**AND EITHER:**

Restoration of at least one emergency bus in < 4 hours is **not** likely

**OR**

RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined

| **Table S-3 AC Power Sources** |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

Loss of all AC power to critical buses compromises all plant safety systems requiring electric power including RHR, ECCS, containment heat removal and the ultimate heat sink. Prolonged loss of all AC power to critical buses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a General Emergency.

This EAL is specified to assure that in the unlikely event of a prolonged loss of all critical bus AC power, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one critical 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 Director a reasonable idea of how quickly (s)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?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

**CNS Basis**:

The 4160V critical buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following offsite sources of power: Figure S-1 illustrates the 4160V AC distribution system (ref. 1, 2).

- Startup Transformer – The Startup Transformer provides a source of offsite AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of offsite AC power to the station whenever the main generator is off-line (<160 Mwe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the Station Normal Transformer fails or that the main generator trips off-line.

- Emergency Transformer -The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.

- Backfeeding power from the 345 kv line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. Note that the time required to effect the backfeed is likely longer than the fifteen-minute interval. If off-normal plant conditions have already established the backfeed, however, its power to the safety-related buses may be considered an offsite power source.

Onsite power sources are the emergency diesel generators (DG-1 and DG-2).

4 hours is the CNS Station Blackout Coping Analysis time (ref. 9, 10).

Indication of continuing core cooling degradation is manifested by an RPV level instrument reading of < -158 in. (RPV level is below the top of active fuel). When RPV level is at or above the top of active fuel, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below the top of active fuel, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to the top of active fuel, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

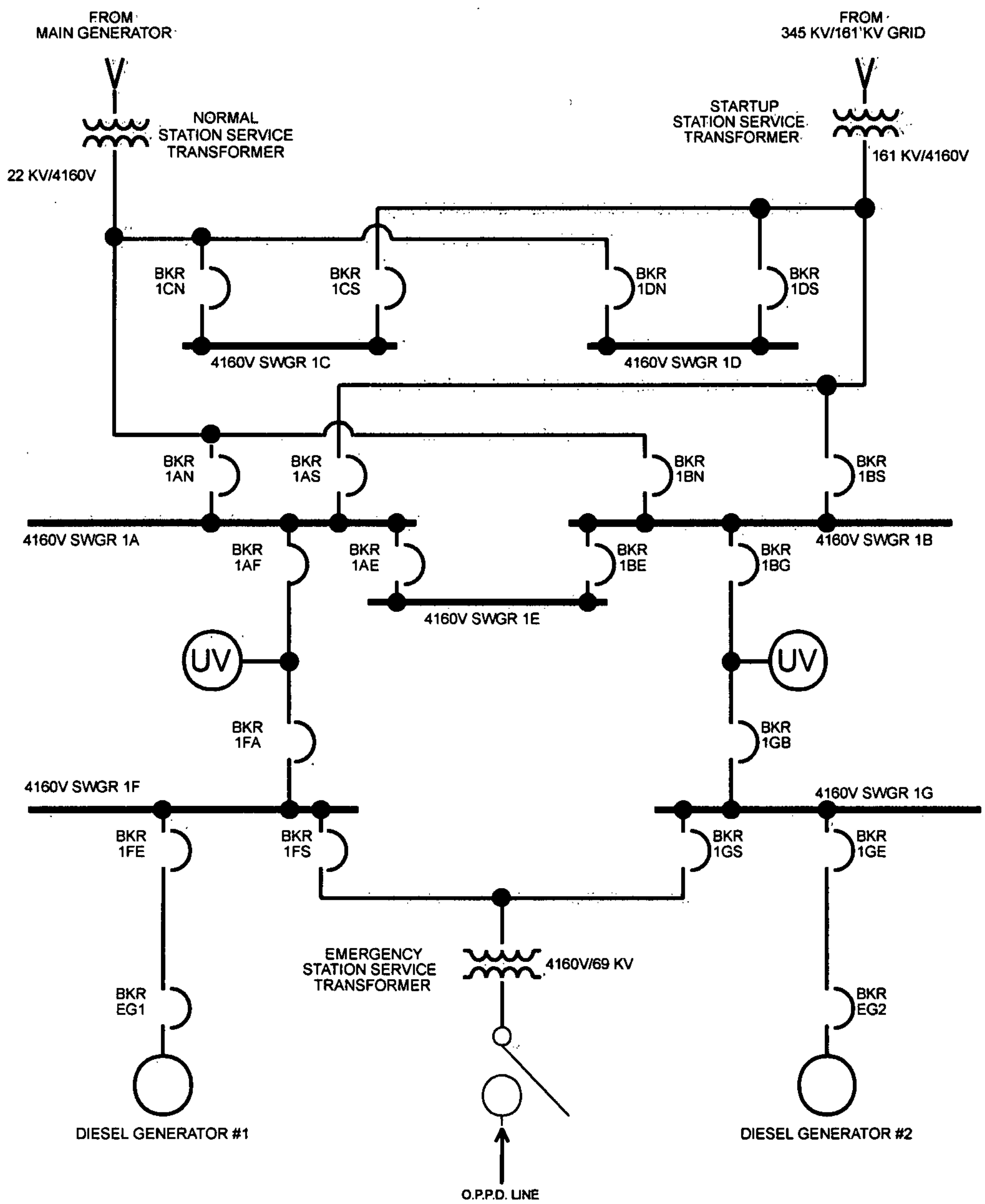
When RPV level cannot be determined, EOPs require entry to EOP-2B, RPV Flooding or EOP-7B, RPV Flooding (Failure-to-Scram). RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-2B/7B specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold the Minimum Steam Cooling Pressures (in scram-failure events). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

**CNS Basis Reference(s):**

1. BR 3001 One Line Diagram
2. BR 3002 sh1
3. Procedure 2.1.15 Startup Transformer
4. Procedure 2.1.16 Normal Station Service Transformer
5. Procedure 2.1.17 Emergency Station Service Transformer
6. Procedure 2.2.18 4160V Auxiliary Power Distribution System

7. Procedure 2.2.20 Standby AC Power System (Diesel Generator)
8. Procedure 5.3SBO Station Blackout
9. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Rev. 2
10. USAR Section VIII-6.2.7
11. NEDC 97-089
12. EOP-2B RPV Flooding
13. EOP-7B RPV Flooding (Failure-to-Scram)

## Figure S-1: 4160V AC Distribution System



**Category:** S – System Malfunction

**Subcategory:** 2 – ATWS / Criticality

**Initiating Condition:** Inadvertent criticality

**EAL:**

| | |
|---|---|
| **SU2.1** | **Unusual Event** |
| An unplanned sustained positive period observed on nuclear instrumentation | |

**Mode Applicability:**

3 - Hot Shutdown

**NEI 99-01 Basis:**

This EAL addresses inadvertent criticality events. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This EAL **EXCLUDES** inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

Escalation would be by the Fission Product Barrier Table, as appropriate to the operating mode at the time of the event.

**CNS Basis:**

SRM A-D period meters NMS-I-44A-D on Panel 9-5 identify this condition as well as Panel 9-5 Panel 9-5 amber light and SRM Period (> 50 sec.) annunciator 9-5-1/F-8 (ref. 1, 2). However, an SRM period alarm caused by SRM channel noise does not result in entry into this EAL (ref. 2).

**CNS Basis Reference(s):**

1. Procedure 4.1.1 Source Range Monitoring System
2. Procedure 2.3, 9-5-1, F-8, SRM Period

**Category:** S – System Malfunction

**Subcategory:** 2 – ATWS / Criticality

**Initiating Condition:** Automatic scram fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

**EAL:**

| **SA2.1** | **Alert** |
|---|---|
| An automatic scram failed to shut down the reactor<br>**AND**<br>Manual actions taken at the reactor control console successfully shut down the reactor as indicated by reactor power < 3% | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup

**NEI 99-01 Basis:**

Manual scram (trip) actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

This condition indicates failure of the automatic protection system to scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient. Thus the plant safety has been compromised because design limits of the fuel may have been exceeded. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad or RCS and because of the failure of the Reactor Protection System to automatically shutdown the plant.

If manual actions taken at the reactor control console fail to shutdown the reactor, the event would escalate to a Site Area Emergency.

**CNS Basis:**

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. A reactor scram may be the result of manual or automatic action in response to any of the following parameters (ref. 1):

- APRM Fixed Neutron Flux – High
- APRM Fixed Neutron Flux – High (Setdown)
- APRM Flow Biased – High
- IRM – High
- Reactor Steam Dome Pressure – High
- Reactor Vessel Water Level Low – Level-3
- Turbine Stop Valve Closure
- Turbine Control Valve Fast Closure
- MSIV Closure
- Scram Discharge Volume (SDV) Level – High
- Drywell Pressure – High

Following a successful reactor scram, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative period. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-scram response from an automatic reactor scram signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful scram has therefore occurred when there is sufficient rod insertion to bring the reactor power below the APRM downscale setpoint of 3% (ref. 2, 3).

The significance of the second condition of this EAL is that a potential degradation of a safety system exists because a front line automatic protection system did not function in response to a plant transient. Thus, plant safety has been compromised.

Following any automatic RPS scram signal, Procedure 2.1.5 prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved (ref. 3). Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Alert.

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

In the event that the operator identifies a reactor scram is imminent and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. Methods of inserting a manual scram are limited to those that can be taken rapidly at the reactor control console (Panel 9-5) and include:

- Both manual Reactor Scram push buttons
- Reactor Mode switch in SHUTDOWN
- Manual actuation of ARI

The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. If manual reactor scram actions fail to reduce reactor power below 3% (ref. 2, 3), the event escalates to the Site Area Emergency under EAL SS2.1.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

**CNS Basis Reference(s):**

1. USAR Table VII-2-2
2. AMP-TBD00 Appendix B, Step RC/Q-4
3. Procedure 2.1.5 Reactor Scram

**Category:** S – System Malfunction

**Subcategory:** 2 – ATWS / Criticality

**Initiating Condition:** Automatic scram fails to shutdown the reactor and manual actions taken from the reactor control console are **not** successful in shutting down the reactor

**EAL:**

| | |
|---|---|
| **SS2.1** | **Site Area Emergency** |

An automatic scram failed to shut down the reactor

**AND**

Manual actions taken at the reactor control console (Note 5) do **not** shut down the reactor as indicated by reactor power $\geq$ 3%

Note 5: Manual scram methods for EAL SS2.1 are the following:

- Reactor Scram push buttons
- Reactor Mode switch in SHUTDOWN
- Manual or auto actuation of ARI

**Mode Applicability:**

1 - Power Operation, 2 - Startup

**NEI 99-01 Basis:**

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS.

Manual scram actions taken at the reactor control console are any set of actions by the reactor operator(s) at which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Manual scram actions are not considered successful if action away from the reactor control console is required to scram the reactor. This EAL is still applicable even if actions taken away from the reactor control console are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shutdown the plant.

Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core-cooling or heat removal.

**CNS Basis**:

This EAL addresses any automatic reactor scram signal followed by a manual scram that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed. Methods of inserting a manual scram are limited to those that can be taken rapidly at the reactor control console (Panel 9-5) and include (ref. 1):

- Both manual Reactor Scram push buttons.
- Reactor Mode switch in SHUTDOWN
- Manual or auto actuation of ARI

Auto actuation of ARI is included in the list of methods because the operator, by procedure, always ensures actuation of ARI has occurred if the ARI actuation setpoints are exceeded. This means action to depress the ARI pushbuttons is taken if the automatic ARI actuation setpoints are exceeded but failed to actuate. If ARI properly actuates automatically, the ARI pushbuttons are not depressed. Reactor shutdown achieved by use of the alternate rod insertion methods listed in ESP 5.8.3 do not constitute a successful manual scram (ref. 2).

The APRM downscale trip setpoint (3%) is a minimum reading on the power range scale that indicates power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 3% power (ref. 1, 3).

Escalation of this event to a General Emergency would be under EAL SG2.1 or Emergency Director judgment.

**CNS Basis Reference(s):**

1. Procedure 2.1.5 Reactor Scram
2. ESP 5.8.3 Alternate Rod Insertion Methods
3. AMP-TBD00 Appendix B, Step RC/Q-4

**Category:** S – System Malfunction

**Subcategory:** 2 – ATWS / Criticality

**Initiating Condition:** Automatic scram and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists

**EAL:**

| | |
|---|---|
| **SG2.1** | **General Emergency** |

Automatic and **all** manual scrams were **not** successful

**AND**

Reactor power ≥ 3%

**AND EITHER** of the following exist or have occurred due to continued power generation:

RPV level **cannot** be restored and maintained > -183 in. or **cannot** be determined

**OR**

Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7)

**Mode Applicability:**

1 - Power Operation, 2 - Startup

**NEI 99-01 Basis:**

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful.

The extreme challenge to the ability to cool the core is intended to mean that the reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level as described in the EOP bases.

Considerations include inability to remove heat via the main condenser, or via the suppression pool or torus (e.g., due to high pool water temperature).

In the event either of these challenges exists at a time that the reactor has not been brought below the power associated with the safety system design a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier table declaration to permit maximum off-site intervention time.

**CNS Basis**:

This EAL addresses the following:

- Any automatic reactor scram signal followed by failure of the automatic scram and all subsequent manual scrams to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SS2.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of the alternate rod insertion methods listed in ESP 5.8.3 are credited as a successful manual scram provided reactor power can be reduced below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1).

The APRM downscale trip setpoint is a minimum reading on the power range scale that indicates power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 3% power (ref. 2, 3)..

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above -183 in. (or cannot be determined). -183 in. is the Minimum Steam Cooling RPV Water Level (MSCRWL). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F (ref. 4). This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence. When RPV level cannot be determined, EOPs require entry to EOP-7B, RPV Flooding (Failure-to-Scram). RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-7B specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold the Minimum Steam Cooling Pressures.

The Heat Capacity Temperature Limit (HCTL) is the highest torus temperature from which Emergency RPV Depressurization will not raise torus pressure above the Primary Containment Pressure Limit (PCPL), while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent. The HCTL is a function of RPV pressure and torus level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. This threshold is met when EOP-3A, Primary Containment Control, Step SP/T-5 is reached (ref. 5). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

**CNS Basis Reference(s):**

1. ESP 5.8.3 Alternate Rod Insertion Methods
2. Procedure 2.1.5 Reactor Scram
3. AMP-TBD00 Appendix B, Step RC/Q-4
4. NEDC 97-090J
5. EOP-3A Primary Containment Control

**Category:** S – System Malfunction

**Subcategory:** 3 – Inability to Reach Shutdown Conditions

**Initiating Condition:** Inability to reach required shutdown within Technical Specification limits

**EAL:**

| **SU3.1** | **Unusual Event** |
|---|---|
| Plant is **not** brought to required operating mode within Technical Specifications LCO action statement time | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a four hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate Unusual Event is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of an Unusual Event is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other EALs.

**CNS Basis**:

None

**CNS Basis Reference(s):**

1. Technical Specifications

**Category:** S – System Malfunction

**Subcategory:** 4 – Instrumentation / Communications

**Initiating Condition:** Unplanned loss of safety system annunciation or indication in the Control Room for 15 minutes or longer

**EAL:**

| **SU4.1** | **Unusual Event** |
|---|---|
| Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3) | |

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical

Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU3.1 "Inability to Reach Required Shutdown Within Technical Specification Limits."

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

This UE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a SIGNIFICANT TRANSIENT is in progress during the loss of annunciation or indication.

**CNS Basis**:

The availability of computer-based monitoring capability (i.e., PMIS, SPDS) is not a factor at the Unusual Event emergency classification level. Safety system annunciation and indication considered in this EAL is found on Control Room Panels 9-3, 9-4, 9-5, and C. The other annunciators and indicators are important to plant operation but are not important to safety (ref. 1-14).

**CNS Basis Reference(s):**

1. Procedure 2.3 9-3-1
2. Procedure 2.3 9-3-2
3. Procedure 2.3 9-3-3
4. Procedure 2.3 9-4-1
5. Procedure 2.3 9-4-2
6. Procedure 2.3 9-4-3
7. Procedure 2.3 9-5-1
8. Procedure 2.3 9-5-2
9. Procedure 2.3 C-1
10. Procedure 2.3 C-2
11. Procedure 2.3 C-3
12. Procedure 2.3 C-4
13. Procedure 2.2.64, Ronan Annunciator System
14. Procedure 2.4ANN, Annunciator Failure

**Category:** S – System Malfunction

**Subcategory:** 4 – Instrumentation / Communications

**Initiating Condition:** Loss of **all** onsite or offsite communications capabilities

**EAL:**

**SU4.2 Unusual Event**

Loss of **all** Table S-2 onsite (internal) communications capability affecting the ability to perform routine operations

**OR**

Loss of **all** Table S-2 offsite (external) communications methods affecting the ability to perform offsite notifications

| Table S-2 Communications Systems | | |
|---|---|---|
| **System** | **Onsite** (internal) | **Offsite** (external) |
| Station Intercom System "Gaitronics" | X | |
| Site UHF Radio Paging System | X | |
| Alternate Intercom | X | |
| CNS On-Site Cell Phone System | X | X |
| Telephone system (PBX) | X | X |
| Federal Telecommunications System (FTS 2001) | | X |
| Microwave Telephone Network | | X |
| Local Telephones (C.O. Lines) | | X |
| CNS State Notification Telephones | | X |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

This EAL addresses loss of communications capability that either prevents the plant operations staff ability to perform routine tasks necessary for plant operations or inhibits the ability to communicate problems externally to offsite authorities from the Control Room. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

. **CNS Basis**:

Onsite/offsite communications include one or more of the systems listed in Table S-2 (ref. 1).

- Station Intercom System "Gaitronics": Permits communication between the different parts of the plant and it also incorporates a public address system for plant wide announcements.
- Site UHF Radio Paging System: The site 450 MHz (UHF) radio system uses two repeaters, Base 1 and Base 2. These repeaters operate on different frequencies. All remote control, portable, and mobile units are capable of selecting either repeater.
- Alternate Intercom: Provides an alternate in-plant communications network utilizing the back-up tone commander PBX system. This system is located in the ERP shack and has battery back-up.
- CNS On-Site Cell Phone System
- Telephone system (PBX): Provides voice communication between virtually all buildings, offices, and operation facilities within the station. The telephone system also provides communications between the plant and offsite facilities via the telephone switchboard

network. The system allows operating crews to alert plant personnel in emergencies. The telephone company provides the normal and leased line services.

- Federal Telecommunications System (FTS 2001): The Health Physics Network (HPN) and Emergency Notification System (ENS ) provides communications between NRC and CNS during an emergency.
- Microwave Telephone Network
- Local Telephones (C.O. Lines)
- CNS State Notification Telephones: The CNS State Notification Telephone System is the primary means for the plant to make emergency notifications to state and local authorities. This system provides direct communication with the Nebraska State Patrol, the Missouri State Patrol, the Atchison County Sheriffs Department, and the Nemaha and Richardson County Sheriffs Departments.

This EAL is the hot condition equivalent of the cold condition EAL CU4.1.

**CNS Basis Reference(s):**

1. Procedure 5.7COMMUN, Attachment 1

**Category:** S – System Malfunction

**Subcategory:** 4 – Instrumentation / Communications

**Initiating Condition:** Unplanned loss of safety system annunciation or indication in the Control Room with **EITHER** (1) a significant transient in progress, or (2) compensatory indicators unavailable

**EAL:**

**SA4.1 Alert**

Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C (Note 3) for ≥ 15 min. (Note 3)

**AND EITHER:**

**Any** significant transient is in progress, Table S-1

**OR**

Compensatory indications are unavailable

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| Table S-1 Significant Transients |
|---|
| Reactor scram |
| Runback > 25% thermal power |
| Electrical load rejection > 25% full electrical load |
| ECCS injection |
| Thermal power oscillations > 10% |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a SIGNIFICANT TRANSIENT.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL:SU3.1 "Inability to Reach Required Shutdown Within Technical Specification Limits."

"Compensatory indications" in this context includes computer based information such as PMIS/SPDS. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a SIGNIFICANT TRANSIENT in progress during the loss of annunciation or indication.

**CNS Basis**:

PMIS and SPDS serve as redundant compensatory indicators which may be utilized in lieu of normal control room indicators. Safety system annunciation and indication considered in this EAL is found on Control Room Panels 9-3, 9-4, 9-5, and C. The other annunciators and indicators are important to plant operation but are not important to safety (ref. 1-14).

Significant transients are listed in Table S-1 and include response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load, ECCS injections, or thermal power oscillations of 10% or greater.

**CNS Basis Reference(s):**


1. Procedure 2.3 9-3-1
2. Procedure 2.3 9-3-2
3. Procedure 2.3 9-3-3
4. Procedure 2.3 9-4-1
5. Procedure 2.3 9-4-2
6. Procedure 2.3 9-4-3
7. Procedure 2.3 9-5-1
8. Procedure 2.3 9-5-2
9. Procedure 2.3 C-1
10. Procedure 2.3 C-2
11. Procedure 2.3 C-3
12. Procedure 2.3 C-4
13. Procedure 2.2.64, Ronan Annunciator System
14. Procedure 2.4ANN, Annunciator Failure

**Category:** S – System Malfunction

**Subcategory:** 4 – Instrumentation / Communications

**Initiating Condition:** Inability to monitor a significant transient in progress

**EAL:**

**SS4.1 Site Area Emergency**

Loss of > approximately 75% of the annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)

**AND**

**Any** significant transient is in progress, Table S-1

**AND**

Compensatory indications are unavailable

**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

| Table S-1 Significant Transients |
|---|
| Reactor scram |
| Runback > 25% thermal power |
| Electrical load rejection > 25% full electrical load |
| ECCS injection |
| Thermal power oscillations > 10% |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

This EAL is intended to recognize the threat to plant safety associated with the complete loss of capability of the control room staff to monitor plant response to a SIGNIFICANT TRANSIENT.

"Planned" and "UNPLANNED" actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU3.1 "Inability to Reach Required Shutdown Within Technical Specification Limits."

A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress.

Site specific indications needed to monitor safety functions necessary for protection of the public must include control room indications, computer generated indications and dedicated annunciation capability.

"Compensatory indications" in this context includes computer based information such as PMIS/SPDS. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

**CNS Basis**:

The availability of computer-based monitoring capability (i.e., PMIS, SPDS) is a factor at the Site Area Emergency classification level because they are compensatory non-alarming indication. Safety system annunciation and indication considered in this EAL is found on Control Room Panels 9-3, 9-4, 9-5, and C. The other annunciators and indicators are important to plant operation but are not important to safety (ref. 1-14).

Significant transients are listed in Table S-1 and include response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load, ECCS injections, or thermal power oscillations of 10% or greater.

Due to the limited number of safety systems in operation during Cold Shutdown, Refueling and Defueled modes, this EAL is not applicable during these modes of operation.

**CNS Basis Reference(s):**

1. Procedure 2.3 9-3-1
2. Procedure 2.3 9-3-2
3. Procedure 2.3 9-3-3
4. Procedure 2.3 9-4-1
5. Procedure 2.3 9-4-2
6. Procedure 2.3 9-4-3
7. Procedure 2.3 9-5-1
8. Procedure 2.3 9-5-2
9. Procedure 2.3 C-1
10. Procedure 2.3 C-2
11. Procedure 2.3 C-3
12. Procedure 2.3 C-4
13. Procedure 2.2.64, Ronan Annunciator System
14. Procedure 2.4ANN, Annunciator Failure

**Category:** S – System Malfunction

**Subcategory:** 5 – Fuel Clad Degradation

**Initiating Condition:** Fuel clad degradation

**EAL:**

| **SU5.1** | **Unusual Event** |
|---|---|
| SJAE monitor > 1.58E+3 mR/hr | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the Fission Product Barriers.

This threshold addresses site-specific radiation monitor readings that provide indication of a degradation of fuel clad integrity.

**CNS Basis:**

Steam Jet Air Ejectors (SJAEs) remove all non-condensable gases from the condensers including air in-leakage and disassociated products originating in the reactor and exhausts them to the offgas holdup volume. A rise in offgas activity could therefore indicate damage to the fuel cladding, a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The Technical Specification allowable limit is ≤ 1 Ci/sec. The SJAE monitor Hi-Hi radiation setpoint is set at 50% of the instantaneous release limit and represents approximately 0.1% fuel cladding damage. The SJAE monitor Hi-Hi radiation setpoint has been selected because it is operationally significant and is readily recognizable by the Control Room operating staff (ref. 1-6). The Offgas system isolates after a 15 minute time delay (ref. 1, 2).

In the Hot modes, a steam source is available from which non-condensibles can be separated for processing by the offgas system. The Cold Shutdown, Refueling and Defueled modes do not afford a transfer mechanism from which the offgas radiation monitors can draw a valid sample. The radiation monitors lose a valid sample source when the air ejectors are not in service (ref. 4, 5, 7).

**CNS Basis Reference(s):**

1. Procedure 2.3_9-4-1, C-4, OFFGAS TIMER INITIATED
2. Procedure 2.3_9-4-1, C-5, OFFGAS HIGH RAD
3. Technical Specification LCO 3.7.5, Air Ejector Off-Gas
4. Procedure 2.4OG, Off-Gas Abnormals
5. Procedure 5.2FUEL, Fuel Failure
6. NEDC 02-004, Estimation of the Steam Jet Air Ejector Radiation Monitor, RMP-RM-150A(B), Readings Following a 1% Fuel Clad release (Degraded Core) in the Reactor Coolant System
7. Procedure 2.2.55, Main Condenser Gas Removal System

**Category:** S – System Malfunction

**Subcategory:** 5 – Fuel Clad Degradation

**Initiating Condition:** Fuel clad degradation

**EAL:**

| **SU5.2** | **Unusual Event** |
|---|---|
| Coolant activity ≥ 4.0 μCi/gm dose equivalent I-131 | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the Fission Product Barriers.

This threshold addresses coolant samples exceeding coolant technical specifications for transient iodine spiking limits.

**CNS Basis**:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding Technical Specification LCO 3.4.6, which is applicable in Hot operating modes (ref. 1).

**CNS Basis Reference(s):**

1. Technical Specification LCO 3.4.6

**Category:** S – System Malfunction

**Subcategory:** 6 – RCS Leakage

**Initiating Condition:** RCS leakage

**EAL:**

| | |
|---|---|
| **SU6.1** | **Unusual Event** |
| Unidentified or pressure boundary leakage > 10 gpm<br>**OR**<br>Identified leakage > 30 gpm (Note 6) | |

Note 6: See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

This EAL is included as an Unusual Event because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated.

The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this EAL to the Alert level is via Fission Product Barrier Degradation EALs.

**CNS Basis**:

Leakage is monitored by utilizing the following techniques (ref. 1):

- Sensing excess flow in piping systems
- Sensing pressure and temperature changes in the primary containment
- Monitoring for high flow and temperature through selected drains,
- Sampling airborne particulate and gaseous radioactivity.
- Drywell floor and equipment drain sump leak rate alarm system

The 10 gpm value for the unidentified drywell leakage was selected because it is observable with normal Control Room measurement of sump pumpout rates (e.g., Drywell Sump Pump Flow RW-FR-528, red/blue pen, etc.). Drywell equipment Sump G and drywell floor drain Sump F each have a FILL UP RATE HIGH annunciator on Panel 9-4-2. If either sump fills from the low-level switch reset point to the high-level pump start point before a preset timer has timed out, the annunciator will alarm indicating the sumps are filling at an excessive rate. Sumps F and G will overflow to each other through a trench system. Drywell equipment and floor drain sump pump isolation valves isolate on RPV low water level (≥ 3 in.) or high drywell pressure (≤1.84 psig).

**CNS Basis Reference(s):**

1. Procedure 2.2.27, Equipment, Floor and Chemical Drain System
2. Procedure 6.LOG.601, Daily Surveillance Log
3. Technical Specification LCO 3.4.4, RCS Operational Leakage
4. Technical Specification LCO 3.4.5, RCS Leakage Detection Instrumentation
5. Procedure 9-4-2/B-2, DRYWELL FLOOR DRN SUMP F HI FILL-UP RATE
6. Procedure 9-4-2/B-1, DRYWELL EQUIP SUMP G HIGH FILL-UP RATE
7. USAR Section X-14.0, Equipment and Floor Drainage Systems

**Category E – ISFSI**

EAL Group: ANY (The EAL in this category is applicable to any plant condition.)

An Independent Spent Fuel Storage Installation (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a cask must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel. Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

An Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated. This includes classification based on a loaded fuel storage cask confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

A security event that leads to a potential loss of level of safety of the ISFSI is a classifiable event under Security category EAL HU4.1.

Minor surface damage that does not affect storage cask boundary is excluded from the scope of these EALs.

**Category:** E – ISFSI

**Subcategory:** None

**Initiating Condition:** Damage to a loaded cask confinement boundary

**EAL:**

| | |
|---|---|
| **EU1.1** | **Unusual Event** |
| Damage to a loaded cask confinement boundary | |

**Mode Applicability:**

N/A

**NEI 99-01 Basis:**

An Unusual Event in this EAL is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

**CNS Basis**:

Minor surface damage that does not affect storage cask boundary is excluded from the scope of this EAL.

**CNS Basis Reference(s):**

1. Certificate of Compliance No. 1004 Amendment 9, April 17, 2007

## Category F – Fission Product Barrier Degradation

EAL Group: Mode 1, 2 or 3

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

A. Fuel Clad (FC): The Fuel Clad barrier consists of the zircalloy fuel bundle tubes that contain the fuel pellets.

B. Reactor Coolant System (RCS): The RCS barrier is the reactor coolant system pressure boundary and includes the reactor vessel and all reactor coolant system piping up to the isolation valves.

C. Primary Containment (PC): The Primary Containment barrier includes the drywell, the wetwell (torus), their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

*Unusual Event:*
*Any loss or any potential loss of Primary Containment*

*Alert:*
*Any loss or any potential loss of either Fuel Clad or RCS*

*Site Area Emergency:*
*Loss or potential loss of any two barriers*

*General Emergency:*
*Loss of any two barriers and loss or potential loss of third barrier*

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Primary Containment barrier. UE EALs associated with RCS and Fuel Clad barriers are addressed under System Malfunction EALs.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS barrier "loss" EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was **no** immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.
- The Primary Containment barrier should **not** be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Primary Containment barrier. When **no** event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Primary Containment barrier status is addressed by Technical Specifications.

Determine which combination of the three barriers are lost or have a potential loss and use FU1.1, FA1.1, FS1.1 and FG1.1 to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent. In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

**Initiating Condition:** **Any** loss or **any** potential loss of Primary Containment

**EAL:**

| **FU1.1** | **Unusual Event** |
|---|---|
| **Any** loss or **any** potential loss of Primary Containment (Table F-1) | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

None

**CNS Basis:**

Fuel Clad, RCS and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

Fuel Clad and RCS barriers are weighted more heavily than the Primary Containment barrier. Unlike the Fuel Clad and RCS barriers, the loss of either of which results in an Alert (EAL FA1.1), loss of the Primary Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or potential loss of the Primary Containment barrier in combination with the loss or potential loss of either the Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

**CNS Basis Reference(s):**

None

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

**Initiating Condition:** **Any** loss or **any** potential loss of either Fuel Clad or RCS

**EAL:**

| | |
|---|---|
| **FA1.1** | **Alert** |
| **Any** loss or **any** potential loss of either Fuel Clad or RCS (Table F-1) | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

None

**CNS Basis:**

Fuel Clad, RCS and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Primary Containment barrier. Unlike the Primary Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Primary Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.

**CNS Basis Reference(s):**

None

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

**Initiating Condition:** Loss or potential loss of **any** two barriers

**EAL:**

| | |
|---|---|
| **FS1.1** | **Site Area Emergency** |
| Loss or potential loss of **any** two barriers (Table F-1) | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

None

**CNS Basis:**

Fuel Clad, RCS and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Primary Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency

Director would have greater assurance that escalation to a General Emergency is less imminent.

**CNS Basis Reference(s):**

None

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

**Initiating Condition:** Loss of **any** two barriers and loss or potential loss of third barrier

**EAL:**

| | |
|---|---|
| **FG1.1** | **General Emergency** |
| Loss of **any** two barriers<br>**AND**<br>Loss or potential loss of third barrier (Table F-1) | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

**NEI 99-01 Basis:**

None

**CNS Basis:**

Fuel Clad, RCS and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Primary Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Primary Containment barrier
- Loss of RCS and Primary Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Primary Containment barriers with potential loss of RCS barrier

**CNS Basis Reference(s):**

None

## Introduction

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Primary Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RPV Level

B. PC Pressure / Temperature

C. Isolation

D. Rad

E. Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers so that they can be easily identified.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Primary Containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1 and FU1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Primary Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,…,E.

## Table F-1 Fission Product Barrier Matrix

| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Primary Containment Barrier | |
|---|---|---|---|---|---|---|
| | **Loss** | **Potential Loss** | **Loss** | **Potential Loss** | **Loss** | **Potential Loss** |
| **A. RPV Level** | 1. PC flooding is required due to **any** of the following:<br>• RPV water level **cannot** be restored and maintained > -183 in.<br>• RPV water level **cannot** be restored and maintained ≥ -209 in. and no core spray subsystem flow can be restored and maintained ≥ 4,750 gpm<br>• RPV water level **cannot** be determined and core damage is occurring | 5. RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined | 7. RPV level cannot be restored and maintained > -158 in. or **cannot** be determined | None | None | 22. PC Flooding required |
| **B. PC Pressure / Temperature** | None | None | 8. PC pressure > 1.84 psig due to RCS leakage | None | 16. PC pressure rise followed by a rapid unexplained drop in PC pressure<br>17. PC pressure response **not** consistent with LOCA conditions | 23. PC pressure > 56 psig and rising<br>24. Deflagration concentrations exist inside PC ≥ 6% $H_2$ in drywell or torus (or **cannot** be determined) **AND** ≥ 5% $O_2$ in drywell or torus (or **cannot** be determined)<br>25. Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7) |
| **C. Isolation** | None | None | 9. Release pathway exists outside primary containment resulting from isolation failure in **any** of the following (excluding normal process system flowpaths from an unisolable system):<br>• Main steam line<br>• HPCI steam line<br>• RCIC steam line<br>• RWCU<br>• Feedwater | 13. RCS leakage > 50 gpm inside the drywell<br>14. Unisolable primary system discharge outside primary containment as indicated by exceeding **any** secondary containment Maximum Normal Operating temperature or radiation value (EOP-5A Tables 9 and 10) | 18. Failure of **all** valves in **any** one line to close **AND** Direct downstream pathway to the environment exists after PC isolation signal<br>19. Intentional PC venting per EOPs<br>20. Unisolable primary system discharge outside PC as indicated by exceeding **any** secondary containment Maximum Safe Operating temperature or radiation value (EOP-5A Tables 9 and 10) | None |
| **D. ERD** | None | None | 10. Emergency RPV depressurization is required | None | None | None |
| **E. Rad** | 2. Drywell radiation monitor (RMA-RM-40A/B) > 2.50E+03 Rem/hr<br>3. Primary coolant activity > 300 μCi/gm dose equivalent I-131 | None | 11. Drywell radiation monitor (RMA-RM-40A/B) > 2.40E+02 Rem/hr | None | None | 26. Drywell radiation monitor (RMA-RM-40A/B) > 5.00E+04 Rem/hr |
| **F. Judgment** | 4. **Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier | 6. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier | 12. **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier | 15. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier | 21. **Any** condition in the opinion of the Emergency Director that indicates loss of the PC barrier | 27. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier |

**Barrier:** Fuel Clad

**Category:** A. RPV Level

**Degradation Threat:** Loss

**Threshold:**

1. Primary Containment flooding is required due to **any** of the following:
   - RPV water level **cannot** be restored and maintained > -183 in.
   - RPV water level **cannot** be restored and maintained ≥ -209 in. and **no** core spray subsystem flow can be restored and maintained ≥ 4,750 gpm
   - RPV water level **cannot** be determined and core damage is occurring

**NEI 99-01 Basis:**

The "Loss" threshold value corresponds to the level used in EOPs to indicate challenge of core cooling. This is the minimum value to assure core cooling without further degradation of the clad.

**CNS Basis**:

EOP-1A, EOP-2B, EOP-7A and EOP-7B specify entry to the SAGs when core cooling is severely challenged and Primary Containment flooding is required. SAG entry signifies the need to flood the primary containment. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. Primary Containment flooding and SAG entry is required when any of the following conditions exist (ref. 1):

- RPV water level cannot be restored and maintained above -183 in. (MSCRWL, EOP-1A/7A) (ref. 2, 4, 5).
- RPV water level cannot be restored and maintained at or above -209 in. (elevation of the jet pump suction) and no core spray subsystem flow can be restored and

maintained equal to or greater than 4,750 gpm (design core spray flow, EOP-1A) (ref. 3, 4)

- RPV water level cannot be determined and core damage is occurring (EOP-2B/7B) (ref. 6, 7)

The above EOP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Potential Loss of the Primary Containment barrier (PC P-Loss 22). Since the EOP requirement for Primary Containment flooding is reached after core uncovery has occurred a Loss of the RCS barrier exists (RCS Loss 7). Primary Containment flooding and SAG entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

**CNS Basis Reference(s):**

1. AMP-TBD00 PSTG/SATG Technical Bases, Contingency #1, #4, #5
2. NEDC 97-090J
3. NEDC 97-089
4. EOP-1A RPV Control
5. EOP-7A RPV Level (Failure-to-Scram)
6. EOP-2B RPV Flooding
7. EOP-7B RPV Flooding (Failure-to-Scram)

**Barrier:** Fuel Clad

**Category:** A. RPV Level

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 5. RPV level cannot be restored and maintained > -158 in. or **cannot** be determined |

**NEI 99-01 Basis:**

This threshold is the same as the RCS barrier "Loss" threshold A.1 and corresponds to the water level at the top of the active fuel. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

**CNS Basis**:

An RPV level instrument reading of -158 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops below TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

When RPV level cannot be determined, EOPs require entry to EOP-2B, RPV Flooding, or EOP-7B, RPV Flooding (Failure-to-Scram). RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-2B/7B specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold the Minimum Steam Cooling Pressures (in scram-failure events). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists (ref. 2, 3).

Note that EOP-7A, RPV Level (Failure-to-Scram), may require intentionally lowering RPV water level to TAF and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and TAF (ref. 4). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - ATWS Criticality EALs.

**CNS Basis Reference(s):**

1. NEDC 97-089
2. EOP-2B RPV Flooding
3. EOP-7B RPV Flooding (Failure-to-Scram)
4. EOP-7A RPV Level (Failure-to-Scram)

**Barrier:** Fuel Clad

**Category:** B. PC Pressure / Temperature

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** B. PC Pressure / Temperature

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** C. Isolation

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** C. Isolation

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** D. ERD

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** D. ERD

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** E. Rad

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 2. Drywell radiation monitor (RMA-RM-40A/B) > 2.50E+03 Rem/hr |

**NEI 99-01 Basis:**

2.50E+03 Rem/hr is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell.

The reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of approximately 300 μCi/gm dose equivalent I-131 (~ 1% clad damage) into the drywell atmosphere.

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage.

This value is higher than that specified for RCS barrier Loss threshold 11. Thus, this threshold indicates a loss of both Fuel Clad barrier and RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

**CNS Basis**:

Procedure 5.7.17 Attachment 7 provides a method of calculating percent fuel clad damage and fuel melt based on drywell radiation. Under LOCA conditions, a reading of 2.44E+6 Rem/hr corresponds to 100% core melt on drywell radiation monitors RMA-RM-40A/B. A value of 2.44E+3 Rem/hr (rounded to 2.50E+03 Rem/hr) yields 1% fuel clad damage using this method.

In order to reach this Fuel Clad barrier Potential Loss threshold, a loss of the RCS barrier has already occurred (see RCS Loss 11). This threshold, therefore, represents at least a Site Area Emergency classification.

**CNS Basis Reference(s):**

1. Procedure 5.7.17 Attachment 7

**Barrier:** Fuel Clad

**Category:** E. Rad

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 3. Primary coolant activity > 300 µCi/gm dose equivalent I-131 |

**NEI 99-01 Basis:**

. Coolant activity of 300 µCi/gm dose equivalent I-131 is well above that expected for iodine spikes and corresponds to about 1% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

**CNS Basis:**

None

**CNS Basis Reference(s):**

1. EPIP Procedure 5.7.17 Dose Assessment, Section 4.6
2. NEI 99-01 Revision 5

**Barrier:** Fuel Clad

**Category:** E. Rad

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** F. Judgment

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 4. **Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier |

**NEI 99-01 Basis:**

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost. In addition, the inability to monitor the barrier should also be considered as a factor in Emergency Director judgment that the barrier may be considered lost.

**CNS Basis:**

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC

power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

**CNS Basis Reference(s):**

None

**Barrier:** Fuel Clad

**Category:** F. Judgment

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 6. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier |

**NEI 99-01 Basis:**

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is potentially lost. In addition, the inability to monitor the barrier should also be considered as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

**CNS Basis**:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC

power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

**CNS Basis Reference(s):**

None

**Barrier:** Reactor Coolant System

**Category:** A. RPV Level

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 7. RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined |

**NEI 99-01 Basis:**

The Loss threshold for RPV water level corresponds to the level that is used in EOPs to indicate challenge of core cooling.

This threshold is the same as Fuel Clad Barrier Potential Loss threshold #5 and corresponds to a challenge to core cooling. Thus, this threshold indicates a Loss of RCS barrier and Potential Loss of Fuel Clad barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with this item.

**CNS Basis**:

An RPV level instrument reading of -158 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment barriers, and initiation of all ECCS. If RPV level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a Loss of the RCS barrier.

When RPV level cannot be determined, EOPs require entry to EOP-2B, RPV Flooding, or EOP-7B, RPV Flooding (Failure-to-Scram). The instructions in EOP-2B/7B specify emergency depressurization of the RPV, which is defined to be a Loss of the RCS barrier (RCS Loss 10) (ref. 2, 3).

Note that EOP-7A, RPV Level (Failure-to-Scram), may require intentionally lowering RPV water level to TAF and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and TAF (ref. 4). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - ATWS Criticality EALs.

**CNS Basis Reference(s):**

1. NEDC 97-089
2. EOP-2B RPV Flooding
3. EOP-7B RPV Flooding (Failure-to-Scram)
4. EOP-7A RPV Level (Failure-to-Scram)

**Barrier:** Reactor Coolant System

**Category:** A. RPV Level

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** B. PC Pressure / Temperature

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 8. PC pressure > 1.84 psig due to RCS leakage |

**NEI 99-01 Basis:**

The threshold pressure value is the primary containment high pressure scram setpoint and is indicative of a LOCA event that requires ECCS response.

There is no Potential Loss threshold associated with this item.

**CNS Basis**:

The primary containment (PC) high pressure scram setpoint is an entry condition to EOP-1A, RPV Control, and EOP-3A, Primary Containment Control (ref. 1, 2). Normal primary containment pressure control functions (e.g., operation of drywell cooling, SBGT, etc.) are specified in EOP-3A in advance of less desirable but more effective functions (e.g., operation of drywell or torus sprays, etc.).

In the CNS design basis, primary containment pressures above the high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the increasing pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control primary containment vent/purge (ref. 3).

The threshold phrase "...due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect primary containment pressure. PC pressure greater than 1.84 psig with corollary indications (drywell temperature, humidity, etc.) should therefore be considered a Loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 1.84 psig should not be considered an RCS barrier Loss.

**CNS Basis Reference(s):**

1. EOP-1A RPV Control
2. EOP-3A Primary Containment Control
3. USAR Section XIV-6.3

**Barrier:** Reactor Coolant System

**Category:** B. PC Pressure / Temperature

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** C. Isolation

**Degradation Threat:** Loss

**Threshold:**

9. Release pathway exists outside primary containment resulting from isolation failure in **any** of the following (excluding normal process system flowpaths from an unisolable system):
    - Main steam line
    - HPCI steam line
    - RCIC steam line
    - RWCU
    - Feedwater

**NEI 99-01 Basis:**

An unisolable RCS break outside Primary Containment is a breach of the RCS barrier. Thus, this threshold is included for consistency with the Alert emergency classification level.

Large high-energy line breaks such as HPCI, Feedwater, RWCU, or RCIC that are unisolable represent a significant loss of the RCS barrier and should be considered as MSL breaks for purposes of classification.

**CNS Basis**:

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside primary containment exists when flow is not prevented by downstream isolations. In the case of a failure of both isolation valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required. Similarly, if the emergency response requires the normal process flow of a system outside primary containment (e.g., EOP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is not met. The combination of these threshold conditions represent the loss of both the RCS and Primary Containment (see PC Loss 20) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an unisolable break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS.

**CNS Basis Reference(s):**

1. Procedure 2.2.56 Main Steam System
2. BR 2041 Reactor Building Main Steam System
3. Procedure 2.2.33 High Pressure Coolant Injection System
4. BR 2044 HPCI System
5. Procedure 2.2.67 Reactor Core Isolation Cooling System
6. BR 2043
7. Procedure 2.2.66 Reactor Water Cleanup
8. BR 2042

**Barrier:** Reactor Coolant System

**Category:** C. Isolation

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 13. RCS leakage > 50 gpm inside the drywell |

**NEI 99-01 Basis:**

This threshold is based on leakage set at a level indicative of a small breach of the RCS but which is well within the makeup capability of normal and emergency high pressure systems. Core uncovery is not a significant concern for a 50 gpm leak, however, break propagation leading to significantly larger loss of inventory is possible.

If primary system leak rate information is unavailable, other indicators of RCS leakage should be used.

**CNS Basis:**

RCS leakage inside the drywell is normally determined by monitoring drywell equipment and floor drain sump pumpout rates. This method of monitoring leakage may be isolated as part of the drywell isolation, and thus may be unavailable. If primary system leak rate information is unavailable, other indicators of RCS leakage should be used (ref. 1-7). Inventory loss events, such as a stuck open SRV, should not be considered when referring to "RCS leakage" because they are not indications of a break, which could propagate.

**CNS Basis Reference(s):**

1. Procedure 2.2.27 Equipment, Floor and Chemical Drain System
2. Procedure 6.LOG.601 Daily Surveillance Log
3. Technical Specifications LCO 3.4.4 RCS Operational Leakage
4. Technical Specifications LCO 3.4.5 RCS Leakage Detection Instrumentation
5. Procedure 9-4-2/B-2 DRYWELL FLOOR DRN SUMP F HI FILL-UP RATE
6. Procedure 9-4-2/B-1 DRYWELL EQUIP SUMP G HIGH FILL-UP RATE

7. USAR Section X-14.0, Equipment and Floor Drainage Systems

**Barrier:** Reactor Coolant System

**Category:** C. Isolation

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 14. Unisolable primary system discharge outside primary containment as indicated by exceeding **any** secondary containment Maximum Normal Operating temperature or radiation value (EOP-5A Tables 9 and 10) |

**NEI 99-01 Basis:**

Potential loss of RCS based on primary system leakage outside the Primary Containment is determined from temperature or area radiation Max Normal Operating Limits values (EOP-5A, Tables 9 and 10) in the areas of the main steam line tunnel, main turbine generator, RCIC, HPCI, etc., which indicate a direct path from the RCS to areas outside Primary Containment.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage warrant an Alert classification. An unisolable leak which is indicated by a high alarm setpoint escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold 20 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

**CNS Basis**:

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. The Maximum Normal Operating values define this RCS threshold because they signify the onset of abnormal system operation. When parameters reach this level, equipment failure or misoperation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-5A, Secondary Containment Control, Tables 9 and 10 (ref. 1) (see below).

In general, multiple indications should be used to determine if a primary system is discharging outside primary containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment. As indicated by Note 5 in EOP-5A Table 10, RP surveys and ARM teledosimetry system may be used for these indications.

**EOP-5A Table 9 – Secondary Containment Temperatures**

| 9 | SECONDARY CONTAINMENT TEMPERATURES<br>SPDS 16 | | | |
|---|---|---|---|---|
| Maximum Normal Operating Value | | Maximum Safe Operating Value | | |
| Area | Any Temp. Switch Alarmed | Area | Value (°F) | Actual Value |
| NE Quad | RCIC-TS-77A<br>RCIC-TS-77C | NE Quad | 195 | |
| SE Quad | RWCU-TS-117F | SE Quad | 195 | |
| NW Quad | RHR-TS-99C | NW Quad | 195 | |
| SW Quad<br>and<br>HPCI Room | RHR-TS-99G<br>HPCI-TS-105B<br>HPCI-TS-105D | SW Quad<br>and<br>HPCI Room | 195 | |
| 1001' El<br>976' El<br>958' El | RWCU-TS-117B | 1001' El<br>976' El<br>958' El | 195 | |
| 903' El<br>and<br>931' El | RHR-TS-99A<br>RHR-TS-99E<br>MS-TS-126A<br>MS-TS-126C<br>RWCU-TS-117E<br>RWCU-TS-117A<br>HPCI-TS-105A | 903' El<br>and<br>931' El | 195 | |

## EOP-5A Table 10 – Secondary Containment Radiation Levels

**10** SECONDARY CONTAINMENT RADIATION LEVELS
SPDS 15 ⑤

| Maximum Normal Operating Value | | | Maximum Safe Operating Value | | |
|---|---|---|---|---|---|
| Area | Any ARM Alarmed | Range (mR/hr) | Area | Value (mR/hr) | Actual Value |
| FUEL POOL AREA | RMA-RA-1 | 100 - $10^8$ | 1001' EL | 1000 | |
| FUEL POOL AREA | RMA-RA-2 | .01 - 100 | 1001' EL | | |
| RWCU PRECOAT AREA | RMA-RA-4 | 0.1 - 1000 | 958' EL | | |
| RWCU SLUDGE AND DECANT PUMP AREA | RMA-RA-5 | 0.1 - 1000 | 931' EL | 1000 | |
| CRD HYDRAULIC EQUIP AREA (SOUTH) | RMA-RA-8 | .01 - 100 | 903' EL | | |
| CRD HYDRAULIC EQUIP AREA (NORTH) | RMA-RA-9 | .01 - 100 | | | |
| HPCI PUMP ROOM | RMA-RA-10 | .01 - 100 | HPCI Room | | |
| RHR PUMP ROOM, (SOUTHWEST) | RMA-RA-11 | .01 - 100 | SW Quad | 1000 | |
| TORUS HPV AREA (SOUTHWEST) | RMA-RA-27 | 1.0 - 10000 | SW Torus | | |
| RHR PUMP ROOM, (NORTHWEST) | RMA-RA-12 | .01 - 100 | NW Quad | 1000 | |
| RCIC/CORE SPRAY PUMP ROOM, (NORTHEAST) | RMA-RA-13 | .01 - 100 | NE Quad | 1000 | |
| CORE SPRAY PUMP ROOM, (SOUTHEAST) | RMA-RA-14 | .01 - 100 | SE Quad | 1000 | |

⑤
Area radiation levels can be monitored by RP surveys or ARM teledosimetry system

### CNS Basis Reference(s):

1. EOP-5A Secondary Containment Control

**Barrier:** Reactor Coolant System

**Category:** D. ERD

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 10. Emergency RPV Depressurization is required |

**NEI 99-01 Basis:**

Plant symptoms requiring Emergency RPV Depressurization are specified in the EOPs (ref. 1, 2, 3, 4, 5) and are indicative of a loss of the RCS barrier. If Emergency RPV depressurization is required, the plant operators are directed to open safety relief valves (SRVs) and keep them open regardless of any subsequent radiological release rate (ref. 1, 5). Even though the RCS is being vented into the suppression pool, a loss of the RCS should be considered to exist due to the diminished effectiveness of the RCS pressure barrier to a release of fission products beyond its boundary.

**CNS Basis:**

None

**CNS Basis Reference(s):**

1. EOP-1A RPV Control
2. EOP-2A Steam Cooling
3. EOP-3A Primary Containment Control
4. EOP-5A Secondary Containment Control, Radioactivity Release Control
5. EOP-7A RPV Control (Failure-to-Scram)

**Barrier:** Reactor Coolant System

**Category:** E. Rad

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 11. Drywell radiation monitor (RMA-RM-40A/B) > 2.40E+02 Rem/hr |

**NEI 99-01 Basis:**

The 2.40E+02 Rem/hr value indicates the release of reactor coolant to the Primary Containment.

The reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the drywell atmosphere.

This reading is less than that specified for Fuel Clad barrier Loss threshold 2. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that value specified by Fuel Clad Barrier threshold, fuel damage would also be indicated.

There is no Potential Loss threshold associated with this item.

**CNS Basis**:

Procedure 5.7.17 Attachment 7 provides a method of calculating percent fuel clad damage and fuel melt based on drywell radiation. A reading of 2.44E+6 Rem/hr corresponds to 100% core melt on RMA-RM-40A/B. A value of 2.44E+2 Rem/hr (rounded to 2.40E+02 Rem/hr) yields 0.1% fuel clad damage using this method. This amount of clad damage is approximately the equivalent of Technical Specification coolant activity discharged uniformly throughout the primary containment (ref. 1).

**CNS Basis Reference(s):**

1. Procedure 5.7.17 Attachment 7

**Barrier:** Reactor Coolant System

**Category:** E. Rad

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** F. Judgment

**Degradation Threat:** Loss

**Threshold:**

12. **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier

**NEI 99-01 Basis:**

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost. In addition, the inability to monitor the barrier should also be considered in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost. )

**CNS Basis:**

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC

power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

**CNS Basis Reference(s):**

None

**Barrier:** Reactor Coolant System

**Category:** F. Judgment

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 15. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier |

**NEI 99-01 Basis:**

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is potentially lost. In addition, the inability to monitor the barrier should also be considered in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

**CNS Basis**:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC

power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

**CNS Basis Reference(s):**

None

**Barrier:** Primary Containment

**Category:** A. RPV Level

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Primary Containment

**Category:** A. RPV Level

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 22. PC Flooding required |

**NEI 99-01 Basis:**

The potential loss requirement for Primary Containment Flooding indicates adequate core cooling cannot be established and maintained and that core melt is possible. Entry into the SAGs (Primary Containment Flooding procedures) is a logical escalation in response to the inability to maintain adequate core cooling.

SAGs direct the operators to perform Containment Flooding when Reactor Vessel Level cannot be restored and maintained greater than specified values or RPV level cannot be determined with indication that core damage is occurring.

The condition in this potential loss threshold represents a potential core melt sequence which, if not corrected, could lead to vessel failure and increased potential for containment failure. In conjunction with Reactor Vessel water level "Loss" thresholds in the Fuel Clad and RCS barrier columns, this threshold will result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third.

**CNS Basis**:

EOP-1A, EOP-2B, EOP-7A and EOP-7B specify entry to the SAGs when core cooling is severely challenged. SAG entry signifies the need to flood the primary containment. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. SAG entry is required when (ref. 1):

- RPV water level cannot be restored and maintained above -183 in. (MSCRWL) (ref. 2).

- RPV water level cannot be restored and maintained at or above -209 in. (elevation of the jet pump suction) and no core spray subsystem flow can be restored and maintained equal to or greater than 4,750 gpm (design core spray flow) (ref. 3).
- RPV water level cannot be determined and core damage is occurring (ref. 4, 5).

The above EOP conditions, if not restored and maintained, represent a potential core melt sequence which could lead to RPV failure and increased potential for containment failure.

This threshold is also a Loss of the Fuel Clad barrier (FC Loss 1). Since SAG entry occurs after core uncovery has occurred, a Loss of the RCS barrier exists (RCS Loss 7). SAG entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

**CNS Basis Reference(s):**

1. AMP-TBD00 PSTG/SATG Technical Bases, Contingency #1, #4, #5
2. NEDC 97-090J
3. NEDC 97-089
4. EOP-2B RPV Flooding
5. EOP-7B RPV Flooding (Failure-to-Scram)

**Barrier:** Primary Containment

**Category:** B. PC Pressure / Temperature

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 16. PC pressure rise followed by a rapid unexplained drop in PC pressure |

**NEI 99-01 Basis:**

Rapid unexplained loss of pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase from a high energy line break indicates a loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

**CNS Basis:**

None

**CNS Basis Reference(s):**

None

**Barrier:** Containment

**Category:** B. PC Pressure / Temperature

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 17. PC pressure response **not** consistent with LOCA conditions |

**NEI 99-01 Basis:**

Primary Containment pressure should increase initially as a result of mass and energy release into containment from a LOCA. Thus, Primary Containment pressure not initially increasing under these conditions indicates a loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

**CNS Basis:**

Analysis of the primary containment response to a postulated DBA LOCA event gives a peak drywell pressure of 54.4 psig and a peak drywell temperature of 301.4°F. These peak values were obtained for the power/flow point of 102%P/75%F (MELLL point). Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. For example, blowdown mass flowrate may be only 60-80% of the analyzed rate. The unexpected response is important because it is the indicator for a containment bypass condition (ref. 1).

**CNS Basis Reference(s):**

1. USAR Section XIV-6.3.7

**Barrier:** Primary Containment

**Category:** B. PC Pressure / Temperature

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 23. PC pressure > 56 psig and rising |

**NEI 99-01 Basis:**

The 56 psig for Potential Loss of containment is based on the Primary Containment design pressure.

**CNS Basis:**

The primary containment internal design pressure is 56 psig (ref. 1). If this threshold is exceeded, a challenge to the primary containment structure has occurred because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a Potential Loss of the Primary Containment barrier even if a containment breach has not occurred.

**CNS Basis Reference(s):**

1. USAR Table V-2-1

**Barrier:** Primary Containment

**Category:** B. PC Pressure / Temperature

**Degradation Threat:** Potential Loss

**Threshold:**

24. Deflagration concentrations exist inside PC

 ≥ 6% $H_2$ in drywell or torus
 (or **cannot** be determined)

 **AND**

 ≥ 5% $O_2$ in drywell or torus
 (or **cannot** be determined)

**NEI 99-01 Basis:**

BWRs specifically define the limits associated with explosive (deflagration) mixtures in terms of deflagration concentrations of hydrogen and oxygen. For Mk I/II containments the deflagration limits are "6% hydrogen and 5% oxygen in the drywell or suppression chamber".

**CNS Basis**:

Deflagration (explosive) mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAMGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to primary containment integrity.

Except for brief periods during plant startup and shutdown, oxygen concentration in the primary containment is maintained at insignificant levels by nitrogen inertion. The specified values for this Potential Loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen) and readily recognizable because 6% hydrogen is well above the EOP-3, Primary Containment Control, entry condition (ref. 1, 2). Since the EOPs/SAGs require deflagration concentration actions to be performed when hydrogen and oxygen concentrations cannot be determined, the phrase has been added to the meaning of explosive mixtures. The minimum global deflagration hydrogen/oxygen concentrations (6% and 5%, respectively) require intentional primary containment venting, which is defined to be a Loss of Containment (PC Loss 20).

Drywell and suppression chamber atmosphere is monitored for $H_2$ and $O_2$ by a divisionally separated $H_2/O_2$ Monitoring System. The system consists of two $H_2/O_2$ analyzers (PC-AN-H2/O2I and PC-AN-H2/O2II), two remote process panels (PC-CS-H2/O2I and PC-CS-H2/O2II), two $H_2$ recorders (PC-R-H2I and PC-R-H2II), two $O_2$ recorders (PC-R-O2I and PC-R-O2II), an $O_2$ digital indicator (PC-I-1), associated control switches and sample stream indicating lights. $H_2/O_2$ analyzers are located in the Reactor Building at 976', remote process panels are located in the Cable Spreading Room, recorders are located on VBD-P1 and VBD-P2, the $O_2$ digital indicator and sample stream lights are located on VBD-H. Div 2 is normally in service providing $O_2$ concentration on VBD-H and $H_2$ and $O_2$ concentrations on PMIS (ref. 3).

**CNS Basis Reference(s):**

1. BWROG EPG/SAG Revision 2, Sections PC/G
2. EOP-3A Primary Containment Control
3. Procedure 2.2.60.1 Containment $H_2/O_2$ Monitoring System

**Barrier:** Primary Containment

**Category:** B. PC Pressure / Temperature

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 25. Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7) |

**NEI 99-01 Basis:**

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized, or
- Suppression chamber pressure above Primary Containment Pressure Limit A, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

**CNS Basis**:

This threshold is met when EOP-3, Primary Containment Control, Step SP/T-5 is reached (ref. 1).

**CNS Basis Reference(s):**

1. EOP-3A Primary Containment Control

**Barrier:** Primary Containment

**Category:** C. Isolation

**Degradation Threat:** Loss

**Threshold:**

18. Failure of **all** valves in **any** one line to close

    **AND**

    Direct downstream pathway to the environment exists after PC isolation signal

**NEI 99-01 Basis:**

These thresholds address incomplete containment isolation that allows direct release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

**CNS Basis**:

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the unisolable open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of primary containment integrity.

Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include unisolable Main steam line, HPCI steam line or RCIC steam line breaks, unisolable RWCU system breaks, and unisloable containment atmosphere vent paths. If the main condenser is available with an unisolable main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using the Category A, Abnormal Rad Release / Rad Effluent, EALs.

The threshold is met if the breach is not isolable from the Control Room or an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation from the Control Room should be made prior to the emergency classification. If operator actions from the Control Room are successful, this threshold is not applicable. Credit is not given for operator actions taken in-plant (outside the Control Room) to isolate the breach.

EOP-3A, Primary Containment Control, Step PC/P-6 may specify primary containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). Under these conditions with a valid containment isolation signal, the Containment barrier should be considered lost.

**CNS Basis Reference(s):**

1. EOP-3A Primary Containment Control

**Barrier:** Primary Containment

**Category:** C. Isolation

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 19. Intentional PC venting per EOPs |

**NEI 99-01 Basis:**

The EOPs may direct containment isolation valve logic(s) to be intentionally bypassed, regardless of radioactivity release rates. Under these conditions with a valid containment isolation signal, the containment should also be considered lost if containment venting is actually performed.

Intentional venting of Primary Containment for Primary Containment pressure or combustible gas control per EOPs to the secondary containment and/or the environment is considered a loss of containment. Containment venting for pressure when not in an accident situation should not be considered.

**CNS Basis**:

EOP-3A, Primary Containment Control, Step PC/P-6 may specify primary containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). The threshold is met when the operator begins venting the primary containment in accordance with EOP-3A, not when actions are taken to bypass interlocks prior to opening the vent valves. Purge and vent actions specified in EOP-3A Step PC/P-1 to control primary containment pressure below the primary containment high pressure scram setpoint does not meet this threshold because such action is only permitted if offsite radioactivity release rates will remain below ODAM limits.

**CNS Basis Reference(s):**

1. EOP-3A Primary Containment Control

**Barrier:** Primary Containment

**Category:** C. Isolation

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| 20. Unisolable primary system discharge outside PC as indicated by exceeding **any** secondary containment Maximum Safe Operating temperature or radiation value (EOP-5A Tables 9 and 10) |

**NEI 99-01 Basis:**

The presence of area radiation levels or area temperatures above any Maximum Safe Operating value indicates unisolable primary system leakage outside the primary containment are addressed after a containment isolation. The indicators should be confirmed to be caused by RCS leakage.

There is no Potential Loss threshold associated with this item.

**CNS Basis:**

The Maximum Safe Operating values define this primary containment barrier threshold because they are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside primary containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-5A, Secondary Containment Control, Tables 9 and 10 (see below).

In general, multiple indications should be used to determine if a primary system is discharging outside primary containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment. As indicated by Note 5 in EOP-5A Table 10, RP surveys and ARM teledosimetry system may be used for these indications.

**EOP-5A Table 9 – Secondary Containment Temperatures**

| 9 | SECONDARY CONTAINMENT TEMPERATURES SPDS 16 | | | |
|---|---|---|---|---|
| Maximum Normal Operating Value | | Maximum Safe Operating Value | | |
| Area | Any Temp. Switch Alarmed | Area | Value (°F) | Actual Value |
| NE Quad | RCIC-TS-77A<br>RCIC-TS-77C | NE Quad | 195 | |
| SE Quad | RWCU-TS-117F | SE Quad | 195 | |
| NW Quad | RHR-TS-99C | NW Quad | 195 | |
| SW Quad<br>and<br>HPCI Room | RHR-TS-99G<br>HPCI-TS-105B<br>HPCI-TS-105D | SW Quad<br>and<br>HPCI Room | 195 | |
| 1001' EL<br>976' EL<br>958' EL | RWCU-TS-117B | 1001' EL<br>976' EL<br>958' EL | 195 | |
| 903' EL<br>and<br>931' EL | RHR-TS-99A<br>RHR-TS-99E<br>MS-TS-126A<br>MS-TS-126C<br>RWCU-TS-117E<br>RWCU-TS-117A<br>HPCI-TS-105A | 903' EL<br>and<br>931' EL | 195 | |

## EOP-5A Table 10 – Secondary Containment Radiation Levels

| 10 | SECONDARY CONTAINMENT RADIATION LEVELS SPDS 15 ⑤ | | | | | |
|---|---|---|---|---|---|---|
| | Maximum Normal Operating Value | | | Maximum Safe Operating Value | | |
| Area | | Any ARM Alarmed | Range (mR/hr) | Area | Value (mR/hr) | Actual Value |
| FUEL POOL AREA | | RMA-RA-1 | $100 - 10^6$ | 1001' EL | 1000 | |
| FUEL POOL AREA | | RMA-RA-2 | .01 - 100 | 1001' EL | | |
| RWCU PRECOAT AREA | | RMA-RA-4 | 0.1 - 1000 | 958' EL | | |
| RWCU SLUDGE AND DECANT PUMP AREA | | RMA-RA-5 | 0.1 - 1000 | 931' EL | 1000 | |
| CRD HYDRAULIC EQUIP AREA (SOUTH) | | RMA-RA-8 | .01 - 100 | 903' EL | | |
| CRD HYDRAULIC EQUIP AREA (NORTH) | | RMA-RA-9 | .01 - 100 | | | |
| HPCI PUMP ROOM | | RMA-RA-10 | .01 - 100 | HPCI Room | | |
| RHR PUMP ROOM, (SOUTHWEST) | | RMA-RA-11 | .01 - 100 | SW Quad | 1000 | |
| TORUS HPV AREA (SOUTHWEST) | | RMA-RA-27 | 1.0 - 10000 | SW Torus | | |
| RHR PUMP ROOM, (NORTHWEST) | | RMA-RA-12 | .01 - 100 | NW Quad | 1000 | |
| RCIC/CORE SPRAY PUMP ROOM, (NORTHEAST) | | RMA-RA-13 | .01 - 100 | NE Quad | 1000 | |
| CORE SPRAY PUMP ROOM, (SOUTHEAST) | | RMA-RA-14 | .01 - 100 | SE Quad | 1000 | |

⑤
Area radiation levels can be monitored by RP surveys or ARM teledosimetry system

### CNS Basis Reference(s):

1. EOP-5A Secondary Containment Control

**Barrier:** Primary Containment

**Category:** D. ERD

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Primary Containment

**Category:** D. ERD

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Primary Containment

**Category:** E. Rad

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Primary Containment

**Category:** E. Rad

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 26. Drywell radiation monitor (RMA-RM-40A/B) > 5.00E+04 Rem/hr |

**NEI 99-01 Basis:**

50,000 Rem/hr is a value which indicates significant fuel damage well in excess of that required for loss of RCS and Fuel Clad. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a Potential Loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

**CNS Basis:**

Procedure 5.7.17 Attachment 7 provides a method of calculating percent fuel clad damage and fuel melt based on drywell radiation. A reading of 2.44E+6 Rem/hr corresponds to 100% core melt on RMA-RM-40A/B. A value of 4.88E+4 Rem/hr (rounded to 5.00E+04 Rem/hr) yields 20% fuel clad damage using this method (ref. 1).

**CNS Basis Reference(s):**

1. EPIP Procedure 5.7.17 Dose Assessment, Attachment 7

**Barrier:** Primary Containment

**Category:** F. Judgment

**Degradation Threat:** Loss

**Threshold:**

21. **Any** condition in the opinion of the Emergency Director that indicates loss of the PC barrier

**NEI 99-01 Basis:**

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Primary Containment barrier is lost. In addition, the inability to monitor the barrier should also be considered as a factor in Emergency Director judgment that the barrier may be considered lost.

**CNS Basis**:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC

power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

**CNS Basis Reference(s):**

None

**Barrier:** Primary Containment

**Degradation Threat:** Potential Loss

**Category:** F. Judgment

**Threshold:**

| |
|---|
| 27. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier |

**NEI 99-01 Basis:**

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Primary Containment barrier is potentially lost. In addition, the inability to monitor the barrier should also be considered as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

**CNS Basis**:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC

power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

**CNS Basis Reference(s):**

None

# Attachment 2

## Emergency Action Level Comparison Matrix

## Cooper Nuclear Station, Docket No. 50-298, DPR-46

NEBRASKA
PUBLIC
POWER
DISTRICT

## Cooper Nuclear Station

# EAL Comparison Matrix

Revision 0

## Table of Contents

**Introduction**

This document provides a line-by-line comparison of the Initiating Conditions (ICs), Mode Applicability and Emergency Action Levels (EALs) in NEI 99-01 Revision 5, Methodology for Development of Emergency Action Levels, and the CNS ICs, Mode Applicability and EALs. This document provides a means of assessing CNS differences and deviations from the NRC endorsed guidance given in NEI 99-01. Discussion of CNS EAL bases and lists of source document references are given in the EAL Technical Bases Document. It is, therefore, advisable to reference the EAL Technical Bases Document for background information while using this document.

**Comparison Matrix Format**

The ICs and EALs discussed in this document are grouped according to NEI 99-01 Recognition Categories. Within each Recognition Category, the ICs and EALs are listed in tabular format according to the order in which they are given in NEI 99-01. Generally, each row of the comparison matrix provides the following information:

- NEI EAL/IC identifier
- NEI EAL/IC wording
- CNS EAL/IC identifier
- CNS EAL/IC wording
- Description of any differences or deviations

**EAL Wording**

In Section 4.2, NEI recommends the following: "The method of presentation should be one with which the operations staff are comfortable. As is the case for emergency procedures, bases for steps should be in separate (or separable) document suitable for training and for reference by emergency response personnel and off-site agencies. Each nuclear plant should already have presentation and human factors standards as part of its procedure writing guidance. EALs that are consistent with those procedure writing standards (in particular, emergency operating procedures which most closely correspond to the conditions under which EALs must be used) should be the norm for each utility."

To assist the Emergency Director, the CNS EALs have been written in a clear and concise style (to the extent that the differences from the NEI EAL wording could be reasonably documented and justified). As a result, unnecessary words have been removed from the CNS EALs to reduce EAL-user reading burden to the extent practicable.

The wording reduction gained from elimination of a few characters in a given EAL may not appear to be advantageous within the context of one EAL. When applied to the composite set of EALs, however, significant gains are realized and reading efficiency is improved. This supports timely and accurate classification in the tense atmosphere of an emergency event. The EAL differences introduced to reduce reading burden comprise almost all of the differences justified in this document.

Guidance for improving the NEI 99-01 EAL scheme is based on the CNS EOP Writer's Guide as well as other CNS documents and industry guidance related to development of written instructions.

**EAL Emphasis Techniques**

Due to the width of the table columns and table formatting constraints in this document, line breaks and indentation may differ slightly from the appearance of comparable wording in the source documents. NEI 99-01 is the source document for the NEI EALs; the CNS EAL Technical Bases Document for the CNS EALs.

Development of the CNS IC/EAL wording has attempted to minimize inconsistencies and apply sound human factors principles as required by the sections of NEI 99-01 discussed above. As a result, differences occur between NEI and CNS ICs/EALs for these reasons alone. When such difference may infer a technical difference in the IC/EAL, the difference is identified and a justification provided.

The print and paragraph formatting conventions summarized below guide presentation of the CNS EALs. Space restrictions in the EAL table of this document sometimes override this guidance in cases when following the guidance would introduce undesirable complications in the EAL layout.

- Upper case print is reserved for system abbreviations, logic terms (and, or, etc. when not used as a conjunction), annunciator window engravings.

- Bold font is used for logic terms, negative terms (**not**, **cannot**, etc.), **any**, **all**.
- Underscore is avoided as it can interfere with text in narrow line spacing.
- Three or more items in a list are normally introduced with "**any** of the following" or "**all** of the following." Items of the list begin with bullets when a priority or sequence is not inferred.
- The use of **AND/OR** logic within the same EAL has been avoided when possible. When such logic cannot be avoided, indentation and separation of subordinate contingent phrases is employed.

**Global Differences**

The differences listed below generally apply throughout the set of EALs. The global differences do not decrease the effectiveness or the intent of NEI 99-01.

1. The NEI phrase "Notification of Unusual Event" has been changed to "Unusual Event" to reduce EAL-user reading burden.
2. The term "**valid**" where used generically in the NEI 99-01 Revision 5 example EAL wording has been deleted since assessment of all indications, reports and conditions are intended to be based on "valid" indications, reports and conditions as defined in the definition section. The use of the term "valid" in qualifying use of indications infers that some classifications may be based on "invalid" indications, reports or conditions.
3. NEI 99-01 IC Example EALs are implemented in separate plant EALs to improve clarity and readability. For example, NEI lists all IC HU1 Example EALs under one IC. The corresponding CNS EALs appear as unique EALs (e.g., HU1.1 through HU1.5).
4. Mode applicability identifiers (numbers/letter) modify the NEI 99-01 mode applicability names as follows: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling, D - Defueled. NEI 99-01 defines Defueled as follows: "Reactor Vessel contains no irradiated fuel (full core off-load during refueling or extended outage)." CNS Technical Specifications do not define a Hot Standby mode.
5. Symbols >, <, ≥ and ≤ for greater than, less than, greater than or equal to, and less than or equal to, respectively, have been adopted to minimize reading burden.
6. "min." has been used as the standard abbreviation for "minutes" and is used to reduce EAL user reading burden
7. IC/EAL identification:
   - NEI 99-01 defines the thresholds requiring emergency classification (example EALs) and assigns them to ICs which, in turn, are grouped in "Recognition Categories." The Recognition Categories, however, are so broad and the IC descriptions are so varied that an EAL is difficult to locate in a timely manner when the EAL-user must refer to a set of EALs with the NEI organization and identification scheme. The NEI document clearly states that the EAL/IC/Recognition Category scheme is **not** intended to be the plant-specific EAL scheme for any plant, and appropriate human factors principles should be applied to development of an EAL scheme that helps the EAL-user make timely and accurate classifications. CNS endeavors to improve upon the NEI EAL organization and identification scheme to enhance usability of the plant-specific EAL set. To this end, the CNS IC/EAL scheme includes the following features:
     a. Division of the NEI EAL set into three groups:
        - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
        - EALs applicable only under Mode 1, 2 or 3 – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Startup or Run mode.
        - EALs applicable only under Mode 4, 5 or DEF – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

       The purpose of the groups is to avoid review of EALs that cannot be applicable in the current operating mode of the plant. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL-user for a given plant condition, reduces EAL-user reading burden and, thereby, speeds identification of the EAL that applies to the emergency.

b. Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The CNS EAL categories/subcategories and their relationship to NEI Recognition Categories are listed in Table 1.

c. Unique identification of each EAL – Four characters comprise the EAL identifier as illustrated in Figure 1.

**Figure 1 – EAL Identifier**

**XXX.X**

Category (A, H, S, F, C, E)
Emergency classification (G, S, A, U)
Subcategory number (1 if no subcategory)
Sequential number within subcategory/classification

The first character is a letter associated with the category in which the EAL is located. The second character is a letter associated with the emergency classification level (G for General Emergency, S for Site Area Emergency, A for Alert, and U for Notification of Unusual Event). The third character is a number associated with one or more subcategories within a given category. Subcategories are sequentially numbered beginning with the number "1". If a category does not have a subcategory, this character is assigned the number "1". The fourth character is a number preceded by a period for each EAL within a subcategory. EALs are sequentially numbered within the emergency classification level of a subcategory beginning with the number "1".

The EAL identifier is designed to fulfill the following objectives:

- Uniqueness – The EAL identifier ensures that there can be no confusion over which EAL is driving the need for emergency classification.
- Speed in locating the EAL of concern – When the EALs are displayed in a matrix format, knowledge of the EAL identifier alone can lead the EAL-user to the location of the EAL within the classification matrix. The identifier conveys the category, subcategory and classification level. This assists ERO responders (who may not be in the same facility as the Emergency Director) to find the EAL of concern in a timely manner without the need for a word description of the classification threshold.
- Possible classification upgrade – The category/subcategory/identifier scheme helps the EAL-user find higher emergency classification EALs that may become active if plant conditions worsen.

Table 2 lists the CNS ICs and EALs that correspond to the NEI ICs/Example EALs when the above EAL/IC organization and identification scheme is implemented.

**Differences and Deviations**

In accordance NRC Regulatory Issue Summary (RIS) 2003-18 "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels" Supplements 1 and 2, a difference is an EAL change in which the basis scheme guidance differs in wording but agrees in meaning and intent, such that classification of an event would be the same, whether using the basis scheme guidance or the CNS EAL. A deviation is an EAL change in which the basis scheme guidance differs in wording and is altered in meaning or intent, such that classification of the event could be different between the basis scheme guidance and the CNS proposed EAL.

Administrative changes that do not actually change the textual content are neither differences nor deviations. Likewise, any format change that does not alter the wording of the IC or EAL is considered neither a difference nor a deviation.

The following are examples of differences:

- Choosing the applicable EAL based upon plant type (i.e., BWR vs. PWR).
- Using a numbering scheme other than that provided in NEI 99-01 that does not change the intent of the overall scheme.
- Where the NEI 99-01 guidance specifically provides an option to not include an EAL if equipment for the EAL does not exist at CNS (e.g., automatic real-time dose assessment capability).
- Pulling information from the bases section up to the actual EAL that does not change the intent of the EAL.
- Choosing to state ALL Operating Modes are applicable instead of stating N/A, or listing each mode individually under the Abnormal Rad Level/Radiological Effluent and Hazard and Other Conditions Affecting Plant Safety sections.
- Using synonymous wording (e.g., greater than or equal to vs. at or above, less than or equal vs. at or below, greater than or less than vs. above or below, etc.)
- Adding CNS equipment/instrument identification and/or noun names to EALs.
- Changing the format of the EALs to conform to the CNS EAL writing guidance (e.g., numbering individual EALs, re-ordering individual EALs within an IC that does not affect the logic, etc.).
- Combining like ICs that are exactly the same but have different operating modes as long as the intent of each IC is maintained and the overall progression of the EAL scheme is not affected.
- Any change to the IC and/or EAL, and/or basis wording, as stated in NEI 99-01, that does not alter the intent of the IC and/or EAL, i.e., the IC and/or EAL continues to:
    - Classify at the correct classification level.
    - Logically integrate with other EALs in the EAL scheme.
    - Ensure that the resulting EAL scheme is complete (i.e., classifies all potential emergency conditions).

The following are examples of deviations:

- Use of altered mode applicability.
- Altering key words or time limits.
- Changing words of physical reference (protected area, safety-related equipment, etc.).
- Eliminating an IC. This includes the removal of an IC from the Fission Product Barrier Degradation category as this impacts the logic of Fission Product Barrier ICs.
- Changing a Fission Product Barrier from a Loss to a Potential Loss or vice-versa.
- Not using NEI 99-01definitions as the intent is for all NEI 99-01 users to have a standard set of defined terms as defined in NEI 99-01. Differences due to plant types are permissible (BWR or PWR). Verbatim compliance to the wording in NEI 99-01 is not necessary as long as the intent of the defined word is maintained. Use of the wording provided in NEI 99-01 is encouraged since the intent is for all users to have a standard set of defined terms as defined in NEI 99-01.
- Any change to the IC and/or EAL, and/or basis wording as stated in NEI 99-01 that does alter the intent of the IC and/or EAL, i.e., the IC and/or EAL:
    - Does not classify at the classification level consistent with NEI 99-01.
    - Is not logically integrated with other EALs in the EAL scheme.
    - Results in an incomplete EAL scheme (i.e., does not classify all potential emergency conditions).

The "Difference/Deviation Justification" columns in the remaining sections of this document identify each difference between the NEI 99-01 IC/EAL wording and the CNS IC/EAL wording. An explanation that justifies the reason for each difference is then provided. If the difference is determined to be a deviation, a statement is made to that affect and explanation is given that states why classification may be different from the NEI 99-01 IC/EAL and the reason for its acceptability. In all cases, however, the differences and deviations do not decrease the effectiveness of the intent of NEI 99-01 Revision 5. A summary list of CNS EAL deviations from NEI 99-01 is given in Table 3.

## Table 1 – CNS EAL Categories/Subcategories

| CNS EALs | | NEI Recognition Category |
|---|---|---|
| **Category** | **Subcategory** | |
| Group: Any Operating Mode | | |
| A – Abnormal Rad Release / Rad Effluent | 1 – Offsite Rad Conditions<br>2 – Onsite Rad Conditions & Spent Fuel Pool Events | Abnormal Rad Levels/Radiological Effluent |
| H – Hazards | 1 – Natural or Destructive Phenomena<br>2 – Fire or Explosion<br>3 – Hazardous Gas<br>4 – Security<br>5 – Control Room Evacuation<br>6 – Judgment | Hazards and Other Conditions Affecting Plant Safety |
| E – ISFSI | None | ISFSI Malfunction |
| Group: Mode 1, 2 or 3 | | |
| S – System Malfunction | 1 – Loss of Power<br>2 – ATWS / Criticality<br>3 – Inability to Reach or Maintain Shutdown Conditions<br>4 – Instrumentation / Communications<br>5 – Fuel Clad Degradation<br>6 – RCS Leakage | System Malfunction |
| F – Fission Product Barrier Degradation | None | Fission Product Barrier Degradation |

| CNS EALs | | NEI Recognition Category |
|---|---|---|
| **Category** | **Subcategory** | |
| Group: Mode 4, 5 or DEF | | |
| C – Cold Shutdown / Refuel System Malfunction | 1 – Loss of Power<br>2 – RPV Level<br>3 – RCS Temperature<br>4 – Communications<br>5 – Inadvertent Criticality | Cold Shutdown./ Refueling System Malfunction |

# Table 2 – NEI / CNS EAL Identification Cross-Reference

| NEI | | CNS | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| AU1 | 1 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AU1.1 |
| AU1 | 2 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AU1.2 |
| AU1 | 3 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AU1.3 |
| AU1 | 4 | N/A | N/A |
| AU1 | 5 | N/A | N/A |
| AU2 | 1 | A – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Pool Events | AU2.1 |
| AU2 | 2 | A – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Pool Events | AU2.2 |
| AA1 | 1 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AA1.1 |
| AA1 | 2 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AA1.2 |
| AA1 | 3 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AA1.3 |
| AA1 | 4 | N/A | N/A |
| AA1 | 5 | N/A | N/A |
| AA2 | 1 | A – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Pool Events | AA2.2 |
| AA2 | 2 | A – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Pool Events | AA2.1 |

| NEI | | CNS | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| AA3 | 1 | A – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Pool Events | AA2.3 |
| AS1 | 1 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AS1.1 |
| AS1 | 2 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AS1.2 |
| AS1 | 3 | N/A | N/A |
| AS1 | 4 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AS1.3 |
| AG1 | 1 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AG1.1 |
| AG1 | 2 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AG1.2 |
| AG1 | 3 | N/A | N/A |
| AG1 | 4 | A – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions | AG1.3 |
| CU1 | 1 | C – Cold SD/ Refuel System Malfunction, 6 – RCS Leakage | CU2.1 |
| CU2 | 1 | C – Cold SD/ Refuel System Malfunction, 2 – RPV Level | CU2.2 |
| CU2 | 2 | C – Cold SD/ Refuel System Malfunction, 2 – RPV Level | CU2.3 |
| CU3 | 1 | C – Cold SD/ Refuel System Malfunction, 1 – Loss of Power | CU1.1 |
| CU4 | 1 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Temperature | CU3.1 |
| CU4 | 2 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Temperature | CU3.2 |
| CU6 | 1, 2 | C – Cold SD/ Refuel System Malfunction, 4 – Communications | CU4.1 |
| CU7 | 1 | C – Cold SD/ Refuel System Malfunction, 1 – Loss of Power | CU1.2 |
| CU8 | 1 | C – Cold SD/ Refuel System Malfunction, 5 – Inadvertent Criticality | CU5.1 |

| NEI | | CNS | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| CU8 | 2 | N/A | N/A |
| CA1 | 1, 2 | C – Cold SD/ Refuel System Malfunction, 2 – RPV Level | CA2.1 |
| CA3 | 1 | C – Cold SD/ Refuel System Malfunction, 1 – Loss of Power | CA1.1 |
| CA4 | 1, 2 | C – Cold SD/ Refuel System Malfunction, 3 – RCS Temperature | CA3.1 |
| CS1 | 1 | C – Cold SD/ Refuel System Malfunction, 2 – RPV Level | CS2.1 |
| CS1 | 2 | C – Cold SD/ Refuel System Malfunction, 2 – RPV Level | CS2.2 |
| CS1 | 3 | C – Cold SD/ Refuel System Malfunction, 2 – RPV Level | CS2.3 |
| CG1 | 1 | C – Cold SD/ Refuel System Malfunction, 2 – RPV Level | CG2.1 |
| CG1 | 2 | C – Cold SD/ Refuel System Malfunction, 2 – RPV Level | CG2.2 |
| D-AU1<br>D-AU2<br>D-SU1<br>D-HU1<br>D-HU2<br>D-HU3<br>D-AA1<br>D-AA2<br>D-HA1<br>D-HA2 | | N/A | N/A |
| E-HU1 | 1 | E - ISFSI | EU1.1 |
| FU1 | 1 | F – Fission Product Barrier Degradation | FU1.1 |

| NEI | | CNS | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| FA1 | 1 | F – Fission Product Barrier Degradation | FA1.1 |
| FS1 | 1 | F – Fission Product Barrier Degradation | FS1.1 |
| FG1 | 1 | F – Fission Product Barrier Degradation | FG1.1 |
| HU1 | 1 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.1 |
| HU1 | 2 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.2 |
| HU1 | 3 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.4 |
| HU1 | 4 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.3 |
| HU1 | 5 | H – Hazards, 1 – Natural or Destructive Phenomena | HU1.5 |
| HU2 | 1 | H – Hazards, 2 – Fire or Explosion | HU2.1 |
| HU2 | 2 | H – Hazards, 2 – Fire or Explosion | HU2.2 |
| HU3 | 1 | H – Hazards, 3 – Hazardous Gas | HU3.1 |
| HU3 | 2 | H – Hazards, 3 – Hazardous Gas | HU3.2 |
| HU4 | 1, 2, 3 | H – Hazards, 4 – Security | HU4.1 |
| HU5 | 1 | H – Hazards, 6 – Judgment | HU6.1 |
| HA1 | 1 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.1 |
| HA1 | 2 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.2 |
| HA1 | 3 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.4 |
| HA1 | 4 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.3 |

| NEI | | CNS | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| HA1 | 5 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.6 |
| HA1 | 6 | H – Hazards, 1 – Natural or Destructive Phenomena | HA1.5 |
| HA2 | 1 | H – Hazards, 2 – Fire or Explosion | HA2.1 |
| HA3 | 1 | H – Hazards, 3 – Hazardous Gas | HA3.1 |
| HA4 | 1, 2 | H – Hazards, 4 – Security | HA4.1 |
| HA5 | 1 | H – Hazards, 5 – Control Room Evacuation | HA5.1 |
| HA6 | 1 | H – Hazards, 6 –Judgment | HA6.1 |
| HS2 | 1 | H – Hazards, 5 – Control Room Evacuation | HS5.1 |
| HS3 | 1 | H – Hazards, 6 – Judgment | HS6.1 |
| HS4 | 1 | H – Hazards, 4 – Security | HS4.1 |
| HG1 | 1, 2 | H – Hazards, 4 – Security | HG4.1 |
| HG2 | 1 | H – Hazards, 6 – Judgment | HG6.1 |
| SU1 | 1 | S – System Malfunction, 1 – Loss of Power | SU1.1 |
| SU2 | 1 | S – System Malfunction, 3 – Inability to Reach or Maintain Shutdown Conditions | SU3.1 |
| SU3 | 1 | S – System Malfunction, 4 – Instrumentation / Communications | SU4.1 |
| SU4 | 1 | S – System Malfunction, 5 – Fuel Clad Degradation | SU5.1 |
| SU4 | 2 | S – System Malfunction, 5 – Fuel Clad Degradation | SU5.2 |
| SU5 | 1, 2 | S – System Malfunction, 6 – RCS Leakage | SU6.1 |

| NEI | | CNS | |
|---|---|---|---|
| IC | Example EAL | Category and Subcategory | EAL |
| SU6 | 1, 2 | S – System Malfunction, 4 – Instrumentation / Communications | SU4.2 |
| SU8 | 1 | S – System Malfunction, 2 – ATWS / Criticality | SU2.1 |
| SU8 | 2 | N/A | N/A |
| SA2 | 1 | S – System Malfunction, 2 – ATWS / Criticality | SA2.1 |
| SA4 | 1 | S – System Malfunction, 4 – Instrumentation / Communications | SA4.1 |
| SA5 | 1 | S – System Malfunction, 1 – Loss of Power | SA1.1 |
| SS1 | 1 | S – System Malfunction, 1 – Loss of Power | SS1.1 |
| SS2 | 1 | S – System Malfunction, 2 – ATWS / Criticality | SS2.1 |
| SS3 | 1 | S – System Malfunction, 1 – Loss of Power | SS1.2 |
| SS6 | 1 | S – System Malfunction, 4 – Instrumentation / Communications | SS4.1 |
| SG1 | 1 | S – System Malfunction, 1 – Loss of Power | SG1.1 |
| SG2 | 1 | S – System Malfunction, 2 – ATWS / Criticality | SG2.1 |

## Table 3 – Summary of Deviations

| NEI | | CNS EAL | Description |
|---|---|---|---|
| IC | Example EAL | | |
| SU5 | 2 | SU6.1 | The NEI identified leakage threshold of 25 gpm has been raised to 30 gpm for consistency with CNS Technical Specifications LCO 3.4.4, which is not exceeded unless unidentified leakage exceeds 5 gpm, total leakage exceeds 30 gpm, or identified leakage exceeds a 2 gpm increase in the past 24 hours. The total leakage limit at many BWRs is 25 gpm and is thus comparable to the NEI threshold for identified leakage. There is no safety analysis that assumes a total leakage at the Technical Specifications limit; rather, it is based on RCS makeup capacity and drywell floor sump capacity. This change is necessary because, unlike most BWRs with a 25 gpm total leakage Technical Specification limit, CNS could be required to declare an Unusual Event before exceeding the Technical Specification total leakage limit of 30 gpm. For example, if identified leakage reached 26 gpm with no unidentified or pressure boundary leakage, the NEI threshold would be exceeded without exceeding the CNS Technical Specification limit. NEI 99-01 Section 3.7 states an Unusual Event represents "Potential degradation of the level of safety of the plant...indicated primarily by exceeding plant technical specification Limiting Condition of Operation (LCO)..." If CNS were to implement the NEI identified leakage threshold, the EAL would not be compatible with the NEI definition of an Unusual Event. |
| CS1 | 3 | CS2.3<br>CG2.2 | The NEI 99-01 example EALs include the use of radiation monitor readings corresponding to those expected for core uncovery in the Refueling Mode (vessel head removed). The generic bases states that the use of radiation monitoring as an EAL input may not be appropriate for some BWRs. Consistent with the bases, the CNS Containment High Range Radiation Monitors cannot be utilized for this purpose because of their location relative to the reactor vessel and core. Additionally, no other installed radiation monitoring system exists that can be utilized for the function. However, CNS does have extensive redundant RPV level monitoring capability available to assess core uncovery in the Refueling Mode. Consistent with indicators used in the EALs derived from generic IC CS1, unexplained RPV leakage indications; Table C-1 has been incorporated and expanded with other site-specific indicators of inventory loss. Therefore this generic indicator is not incorporated into the applicable site-specific EALs. |

# Category A

# Abnormal Rad Levels / Radiological Effluent

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AU1 | Any release of gaseous or liquid radioactivity to the environment greater than 2 times the Radiological Effluent Technical Specifications/ODCM for 60 minutes or longer.<br>MODE: All | AU1 | **Any** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODAM limits for 60 minutes or longer<br>MODE: All | CNS ODAM limits provide the site-specific Radiological Effluent Technical Specifications. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.<br>VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 60 minutes or longer:<br>(site specific monitor list and threshold values) | AU1.1 | **Any** gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)<br>Note 2: The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | Gaseous release is emphasized in this EAL to be consistent with the NEI basis, which states ""Some sites may find it advantageous to address gaseous and liquid releases with separate initiating conditions and EALs."<br>The NEI phrase "VALID reading on **ANY** of the following radiation monitors greater than the reading shown …" has been replaced with "**Any** gaseous monitor reading > Table A-1 column "UE"…" The CNS radiation monitors that detect radioactivity effluent release to the environment are the ERP, Rx Bldg Vent, TG Bldg Vent, and RW/ARW Vent.<br>Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix. |
| 2 | **Note:** The Emergency Director should not wait until the | AU1.2 | **Any** liquid effluent monitor reading > Table A-1 column "UE" for ≥ 60 min. | Liquid release is emphasized in this EAL to be consistent with the NEI basis, which states: "Some sites may find it |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| | applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.<br>VALID reading on any effluent monitor reading greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer. | | (Note 2)<br>**AND**<br>Effluent discharge is **not** isolated<br>Note 2: The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | advantageous to address gaseous and liquid releases with separate initiating conditions and EALs."<br>The NEI phrase "VALID reading on any effluent monitor reading greater than 2 times the alarm setpoint established by a current radioactivity discharge permit ..." has been replaced with "**Any** liquid effluent monitor reading > Table A-1 column "UE"".<br>The CNS radiation monitors that detect liquid radioactivity effluent release to the environment are the Radwaste effluent monitor and the Service Water Effluent monitor. The value of "2 x calculated alarm values" for the Radwaste Effluent Monitor and the listed value for the Service Water effluent is consistent with the NEI bases, and represents two times the ODAM release limits for liquid release. The alarm setpoints for both liquid and gaseous effluent monitors are conservatively set to ensure the ODAM release limits are not exceeded.<br>The phrase "AND...Effluent discharge is **not** isolated" has been added to the CNS EAL. At low classification levels, NEI states in the AU1/AA1 bases that the concern for classification is the continuing, uncontrolled release of radioactivity and not the magnitude of the release. When the liquid release is isolated, the release is no longer continuing nor is it uncontrolled. Therefore, the classification is not appropriate when the liquid release is isolated.<br>Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix. |
| 3 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the | AU1.3 | Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODAM limits for ≥ 60 min. (Note 2)<br>Note 2: The Emergency Director should **not** wait until the applicable | The CNS ODAM is the site-specific effluent Technical Specifications.<br>The NEI phrase "two times" has been replaced with phrase "2 x" to reduce EAL user reading burden. The phrases have the same meaning.<br>Reference to the NEI note is included in the EAL wording |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| | applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.<br>Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 2 times (site specific RETS values) for 60 minutes or longer | | time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | "(Note 2)." Numbering the note facilitates referencing in the EAL matrix. |
| 4 | VALID reading on perimeter radiation monitoring system greater than 0.10 mR/hr above normal background sustained for 60 minutes or longer [for sites having telemetered perimeter monitors] | N/A | N/A | Deleted NEI Example EAL #4 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in Table A-1 and dose assessment capabilities. |
| 5 | VALID indication on automatic real-time dose assessment capability greater than (site-specific value) for 60 minutes or longer [for sites having such capability] | N/A | N/A | Deleted NEI Example EAL #5 because the plant is not equipped with real-time dose assessment. This threshold is properly addressed by the radiation monitors listed in Table A-1 and dose assessment capabilities. |

| Table A-1 Effluent Monitor Classification Thresholds | | | | | |
|---|---|---|---|---|---|
| | **Monitor** | **GE** for ≥ 15 min. | **SAE** for ≥ 15 min. | **ALERT** for ≥ 15 min. | **UE** for ≥ 60 min. |
| **GASEOUS** | ERP | 3.50E+08 μCi/sec | 3.50E+07 μCi/sec | 2.80E+06 μCi/sec | 2.24E+05 μCi/sec |
| | Rx Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.45E+05 μCi/sec | 8.48E+04 μCi/sec |
| | Turb Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.62E+05 μCi/sec | 9.02E+04 μCi/sec |
| | RW / ARW Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.64E+05 μCi/sec | 9.08E+04 μCi/sec |
| **LIQUID** | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| | Service Water Effluent | ----- | ----- | 4.80E-04 μCi/cc | 4.80E-06 μCi/cc |

* with effluent discharge **not** isolated

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AU2 | UNPLANNED rise in plant radiation levels.<br>MODE: All | AU2 | Unplanned rise in plant radiation levels<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. UNPLANNED water level drop in a reactor refueling pathway as indicated by (site specific level or indication).<br>**AND**<br>b. VALID Area Radiation Monitor reading rise on (site specific list). | AU2.1 | Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by **any** of the following:<br>• LI-86 (calibrated to 1001' elev.)<br>• Spent fuel pool low level alarm<br>• Visual observation<br>**AND**<br>Area radiation monitor reading rise on RMA-RA-1 or RMA-RA-2 (or by survey) | The reactor refueling pathway consists of the reactor cavity and spent fuel pool. Since the fuel transfer canal is either directly connected to (and cannot be isolated from) the reactor cavity or spent fuel pool, its inclusion in the CNS EAL is unnecessary.<br>The CNS "... (site-specific level or indication)" of an unplanned water level drop are: the reactor cavity level instrument (NBI-LI-86 if calibrated to read to the elevation of the Refueling Floor, 1001' elev.), spent fuel low level (determined by Annunciator 9-4-2/A-3, "Fuel Pool Cooling Trouble," alarming due to Annunciator Panel 25-15, "Fuel Pool Low Level at 4" below normal," and visual observation.<br>RMA-RA-1 and RMA-RA-2 are the site-specific radiation monitors most likely to detect a water level drop in the reactor refueling pathway. The parenthetical phrase "or by survey" has been added to ensure this source of information is not excluded from consideration of this threshold. |
| 2 | UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels.<br>*Normal levels can be considered as the highest reading in the past twenty-four | AU2.2 | Unplanned area radiation monitor readings or survey results rise by a factor of 1,000 over normal levels*<br>*Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value | The NEI term "twenty-four" has been replaced with Arabic numerals for clarification.<br>The term "indicate a" has been deleted for proper English. |

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| | hours excluding the current peak value. | | | |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AA1 | Any release of gaseous or liquid radioactivity to the environment greater than 200 times the Radiological Effluent Technical Specifications/ODCM for 15 minutes or longer.<br>MODE: All | AA1 | **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODAM limits for 15 minutes or longer<br>MODE: All | The CNS ODAM limits provide the site-specific Radiological Effluent Technical Specifications. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.<br>VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer:<br>(site-specific list monitor list and threshold values) | AA1.1 | **Any** gaseous monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)<br>Note 2: The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | Gaseous release is emphasized in this EAL to be consistent with the NEI basis, which states ""Some sites may find it advantageous to address gaseous and liquid releases with separate initiating conditions and EALs."<br>The NEI phrase "VALID reading on **ANY** of the following radiation monitors greater than the reading shown …" has been replaced with "**Any** gaseous monitor reading > Table A-1 column "Alert"…" The CNS radiation monitors that detect radioactivity effluent release to the environment are the ERP, Rx Bldg Vent, TG Bldg Vent, and RW/ARW Vent.<br>If the Alert threshold were to be two-hundred times the ODAM limit, the resultant value would exceed the SAE threshold, which is based on 10% of the PAG limits. For this reason, the Alert threshold is the log-average of the UE (ODAM) and the SAE threshold (CNSDOSE dose assessment calculation). This provides a reasonable escalation in classification from the UE to the Alert and SAE thresholds.<br>Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix. |
| 2 | **Note:** The Emergency Director | AA1.2 | **Any** liquid effluent monitor reading > | The NEI phrase "VALID reading on any effluent monitor |

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|---|---|---|---|---|
| | should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.<br>VALID reading on any effluent monitor reading greater than 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer. | | Table A-1 column "Alert" for ≥ 15 min. (Note 2)<br>**AND**<br>Effluent discharge is **not** isolated<br>Note 2: The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | reading greater than 200 times the alarm setpoint established by a current radioactivity discharge permit..." has been replaced with "**Any** liquid effluent monitor reading > Table A-1 column "Alert"".<br>Liquid release is emphasized in this EAL to be consistent with the NEI basis, which states "Some sites may find it advantageous to address gaseous and liquid releases with separate initiating conditions and EALs."<br>The CNS radiation monitors that detect liquid radioactivity effluent release to the environment are the Radwaste effluent monitor and the Service Water Effluent monitor. The value of "200 x calculated alarm values" for the Radwaste Effluent Monitor and the calculated Service Water monitor value is consistent with the NEI bases, and represents two hundred times the ODAM release limits for liquid release. The alarm setpoints for both liquid and gaseous effluent monitors are conservatively set to ensure the ODAM release limits are not exceeded.<br>The phrase "AND...Effluent discharge is **not** isolated" has been added to the CNS EAL. At low classification levels, NEI states in the AU1/AA1 bases that the concern for classification is the continuing, uncontrolled release of radioactivity and not the magnitude of the release. When the liquid release is isolated, the release is no longer continuing nor is it uncontrolled. Therefore, the classification is not appropriate when the liquid release is isolated.<br>Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix. |
| 3 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, | AA1.3 | Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM limits for ≥ 15 min. (Note 2)<br>Note 2: The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined | The CNS ODAM is the site-specific effluent Technical Specifications.<br>The NEI phrase "200 times" has been replaced with phrase "200 x" to reduce EAL-user reading burden. The phrases have the same meaning.<br>Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the |

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| | declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.<br>Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 200 times (site specific RETS values) for 15 minutes or longer) | | that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown. | EAL matrix. |
| 4 | VALID reading on perimeter radiation monitoring system greater than 10.0 mR/hr above normal background sustained for 15 minutes or longer [for sites having telemetered perimeter monitors] | N/A | N/A | Deleted NEI Example EAL #4 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in Table A-1 and dose assessment capabilities. |
| 5 | VALID indication on automatic real-time dose assessment capability greater than (site-specific value) for 15 minutes or longer [for sites having such capability] | N/A | N/A | Deleted NEI Example EALs #5 because the plant is not equipped with and real-time dose assessment. This threshold is properly addressed by the radiation monitors listed in Table A-1 and dose assessment capabilities. |

| Table A-1 Effluent Monitor Classification Thresholds | | | | | | |
|---|---|---|---|---|---|---|
| Monitor | | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
| GASEOUS | ERP | 3.50E+08 μCi/sec | 3.50E+07 μCi/sec | 2.80E+06 μCi/sec | 2.24E+05 μCi/sec |
| | Rx Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.45E+05 μCi/sec | 8.48E+04 μCi/sec |
| | Turb Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.62E+05 μCi/sec | 9.02E+04 μCi/sec |
| | RW / ARW Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.64E+05 μCi/sec | 9.08E+04 μCi/sec |
| LIQUID | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| | Service Water Effluent | ----- | ----- | 4.80E-04 μCi/cc | 4.80E-06 μCi/cc |

* with effluent discharge **not** isolated

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AA2 | Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel.<br>MODE: All | AA2a | Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the RPV<br>MODE: All | Replaced the term "Reactor Vessel" with "RPV" as this is the common terminology for BWRs. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered | AA2.2 | A water level drop in the reactor cavity or spent fuel pool that will result in irradiated fuel becoming uncovered | Since the fuel transfer canal is either directly connected to (and cannot be isolated from) the reactor cavity or spent fuel pool, its inclusion in the CNS EAL is unnecessary. |
| 2 | A VALID alarm or (site specific elevated reading) on **ANY** of the following due to damage to irradiated fuel or loss of water level.<br>(site specific radiation monitors) | AA2.1 | Damage to irradiated fuel **OR** loss of water level (uncovering irradiated fuel outside the RPV) that causes **EITHER** of the following:<br>RMA-RA-1 Fuel Pool Area Rad reading > 50 R/hr<br>**OR**<br>RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm | Included the IC wording in the EAL to clarify that the rise in radiation levels is the result of damage or uncovering of irradiated fuel.<br>The NEI phrase "… (site-specific) alarm or reading…" has been replaced with "…alarm…" to reflect the indication provided by the site-specific list of radiation monitors.<br>The listed radiation monitors and specified indications represent the site-specific equivalents. The value of 50 R/hr on RMA-RA-1 represents 100 times the high alarm setpoint and is unambiguously indicative of spent fuel damage or uncovery. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AA3 | Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions.<br>MODE: All | AA2b | Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Dose rate greater than 15 mR/hr in **ANY** of the following areas requiring continuous occupancy to maintain plant safety functions:<br>(Site-specific list) | AA2.3 | Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:<br>Main Control Room (RM-RA-20)<br>**OR**<br>CAS | The NEI units "mR/hr" have been changed to "mRem/hr" to agree with units of measure used in other Category A EALs.<br>The NEI word "ANY" has been replaced with "EITHER" since there are less than three CNS areas requiring continuous occupancy.<br>The phrase "Main Control Room ...OR CAS ..." has been added to the plant EAL to clarify the meaning of areas requiring continuous occupancy. These areas are specifically identified in the NEI IC AA3 basis discussion. The CNS Radwaste Control Room is not required to be continuously occupied in order to maintain plant safety functions.<br>The listed ARM identifies the Control Room area radiation monitor. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AS1 | Off-site dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 100 mrem TEDE or 500 mrem Thyroid CDE for the actual or projected duration of the release.<br>MODE: All | AS1 | Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 0.1 Rem TEDE or 0.5 Rem thyroid CDE for the actual or projected duration of the release<br>MODE: All | NEI doses given in "mrem" have been changed to "Rem" to agree with the units uses in the CNS dose assessment program. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.<br>VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer:<br>(site specific monitor list and threshold values) | AS1.1 | **Any** gaseous monitor reading > Table A-1 column "SAE" for ≥ 15 min. (Note 1)<br>Note 1: The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. (See EAL AS1.2) Do **not** delay declaration awaiting dose assessment results. | The CNS radiation monitors that detect radioactivity effluent release to the environment are the ERP, Rx Bldg Vent, TG Bldg Vent, and RW/ARW Vent.<br>The Table A-1 Site Are Emergency thresholds have been determined using CNS-DOSE dose projection calculations.<br>Reference to the NEI note is included in the EAL wording "(Note 1)." Numbering the note facilitates referencing in the EAL matrix.<br>Added the sentence "See EAL AS1.2" to Note 1 for clarification of the EAL that pertains to dose projections. |

| | | | | |
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| 2 | Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary | AS1.2 | Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem thyroid CDE at or beyond the site boundary | The symbol ">" for "greater than" has been added to the CNS EAL for clarification.<br>NEI doses given in "mrem" have been changed to "Rem" to agree with the units uses in the CNS dose assessment program. |
| 3 | A VALID reading sustained for 15 minutes or longer on perimeter radiation monitoring system greater than 100 mR/hr. [for sites having telemetered perimeter monitors] | N/A | N/A | Deleted NEI Example EAL #3 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitor listed in EAL #1 and dose assessment capabilities. |
| 4 | Field survey results indicate closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation, at or beyond the site boundary. | AS1.3 | Field survey indicates closed window dose rate > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary<br>**OR**<br>Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary | Split the example into two logical conditions separated by the "OR" logical connector for usability.<br>The NEI abbreviation "R" has been replaced with the plant abbreviation "Rem" to agree with units of measure given in the EPA PAGs.<br>NEI doses given in "mrem" have been changed to "Rem" to agree with the units used in the CNS dose assessment program.<br>The NEI phrase "one hour" has been abbreviated "1 hr" to reduce EAL-user reading burden. |

| Table A-1 Effluent Monitor Classification Thresholds | | | | | |
|---|---|---|---|---|---|
| | Monitor | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
| GASEOUS | ERP | 3.50E+08 µCi/sec | 3.50E+07 µCi/sec | 2.80E+06 µCi/sec | 2.24E+05 µCi/sec |
| GASEOUS | Rx Bldg Vent | 3.50E+07 µCi/sec | 3.50E+06 µCi/sec | 5.45E+05 µCi/sec | 8.48E+04 µCi/sec |
| GASEOUS | Turb Bldg Vent | 3.50E+07 µCi/sec | 3.50E+06 µCi/sec | 5.62E+05 µCi/sec | 9.02E+04 µCi/sec |
| GASEOUS | RW / ARW Bldg Vent | 3.50E+07 µCi/sec | 3.50E+06 µCi/sec | 5.64E+05 µCi/sec | 9.08E+04 µCi/sec |
| LIQUID | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| LIQUID | Service Water Effluent | ----- | ----- | 4.80E-04 µCi/cc | 4.80E-06 µCi/cc |

* with effluent discharge **not** isolated

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| AG1 | Off-site dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 1000 mrem TEDE or 5000 mrem Thyroid CDE for the actual or projected duration of the release using actual meteorology.<br>MODE: All | AG1 | Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1 Rem TEDE or 5 Rem thyroid CDE for the actual or projected duration of the release using actual meteorology<br>MODE: All | NEI doses given in "mrem" have been changed to "Rem" to agree with the units uses in the CNS dose assessment program. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.<br>VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer:<br>(site specific monitor list and threshold values) | AG1.1 | **Any** gaseous monitor reading > Table A-1 column "GE" for ≥ 15 min. (Note 1)<br>Note 1: The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (See EAL AG1.2.) Do **not** delay declaration awaiting dose assessment results. | The CNS radiation monitors that detect radioactivity effluent release to the environment are the ERP, Rx Bldg Vent, TG Bldg Vent, and RW/ARW Vent.<br>The Table A-1 General Emergency thresholds have been determined using CNS-DOSE dose projection calculations.<br>Reference to the NEI note is included in the EAL wording "(Note 1)." Numbering the note facilitates referencing in the EAL matrix.<br>Added the sentence "See EAL AG1.2" to Note 1 for clarification of the EAL that pertains to dose projections. |

| | | | | |
|---|---|---|---|---|
| 2 | Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the site boundary. | AG1.2 | Dose assessment using actual meteorology indicates doses > 1 Rem TEDE or > 5 Rem thyroid CDE at or beyond the site boundary | The symbol ">" has been added to modify "5 Rem" for clarification.<br>NEI doses given in "mrem" have been changed to "Rem" to agree with the units uses in the CNS dose assessment program. |
| 3 | VALID perimeter radiation monitoring system reading greater than 1000 mR/hr for 15 minutes or longer. [for sites having telemetered perimeter monitors] | N/A | N/A | Deleted NEI Example EAL #3 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in EAL #1 and dose assessment capabilities. |
| 4 | Field survey results indicate closed window dose rates greater than 1000 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 5000 mrem for one hour of inhalation, at or beyond site boundary. | AG1.3 | Field survey results indicate closed window dose rates > 1 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary<br>**OR**<br>Analyses of field survey samples indicate thyroid CDE > 5 Rem for 1 hr of inhalation at or beyond the site boundary | Split the example into two logical conditions separated by the "OR" logical connector for usability.<br>The NEI abbreviation "mR" has been replaced with the plant abbreviation "Rem" to agree with units of measure given in the EPA PAGs and CNS procedures.<br>NEI doses given in "mrem" have been changed to "Rem" to agree with the units uses in the CNS dose assessment program.<br>Added the greater than symbol to the 5 Rem threshold for completeness as it is NEI intent that the classification be made if the analyses of field survey samples exceed the threshold. Otherwise, classification based on field survey samples would only apply if the results were to be exactly 5 Rem.<br>The NEI phrase "one hour" has been abbreviated "1 hr" to reduce EAL-user reading burden. |

| Table A-1 Effluent Monitor Classification Thresholds | | | | | |
|---|---|---|---|---|---|
| Monitor | | GE<br>for ≥ 15 min. | SAE<br>for ≥ 15 min. | ALERT<br>for ≥ 15 min. | UE<br>for ≥ 60 min. |
| GASEOUS | ERP | 3.50E+08 μCi/sec | 3.50E+07 μCi/sec | 2.80E+06 μCi/sec | 2.24E+05 μCi/sec |
| GASEOUS | Rx Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.45E+05 μCi/sec | 8.48E+04 μCi/sec |
| GASEOUS | Turb Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.62E+05 μCi/sec | 9.02E+04 μCi/sec |
| GASEOUS | RW / ARW Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.64E+05 μCi/sec | 9.08E+04 μCi/sec |
| LIQUID | Rad Waste Effluent | ----- | ----- | 200 x calculated alarm values* | 2 x calculated alarm values* |
| LIQUID | Service Water Effluent | ----- | ----- | 4.80E-04 μCi/cc | 4.80E-06 μCi/cc |

* with effluent discharge **not** isolated

# Category C

# Cold Shutdown / Refueling System Malfunction

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU1 | RCS leakage.<br>MODE: Cold Shutdown | CU2a | RCS leakage<br>MODE: 4 - Cold Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br><br>1. RCS leakage results in the inability to maintain or restore RPV level greater than (site specific low level RPS actuation setpoint) for 15 minutes or longer. [*BWR*]<br><br>1. RCS leakage results in the inability to maintain or restore level within (site specific pressurizer or RCS/RPV level target band) for 15 minutes or longer. [*PWR*] | CU2.1 | RPV level **cannot** be restored and maintained > +3 in. for ≥ 15 min. (Note 3) due to RCS leakage<br><br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | The NEI phrase "the inability to maintain or restore RPV level" has been changed to "RPV level **cannot** be restored and maintained" for conciseness, clarity and agreement with similar terminology found in CNS EOPs. The phrase "due to RCS leakage" has been added to implement the NEI phrase "RCS leakage results in."<br><br>+3 in. is the site specific low level RPS actuation setpoint.<br><br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix.<br><br>The PWR portion of the NEI EAL has not been implemented because CNS is a BWR. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU2 | UNPLANNED loss of RCS/RPV inventory.<br>MODE: Refueling | CU2b | Unplanned loss of RPV inventory<br>MODE: 5 - Refueling | The NEI acronym "RCS" has been replaced with "RPV" to use terminology commonly accepted at BWRs. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | UNPLANNED RCS/RPV level drop as indicated by either of the following:<br>• RCS/RPV water level drop below the RPV flange for 15 minutes or longer when the RCS/RPV level band is established above the RPV flange.<br>• RCS/RPV water level drop below the RCS level band for 15 minutes or longer when the RCS/RPV level band is established below the RPV flange.<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | CU2.2 | Unplanned RPV level drop for ≥ 15 min (Note 3) below **EITHER**:<br>RPV flange (LI-86: 206 in. normal calibration, 113.75 in. elevated indication)<br>**OR**<br>RPV level band when the RPV level band is established below the RPV flange<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | The NEI phrase "RCS" has been replaced with "RPV" to use terminology commonly accepted at BWRs.<br>Reformatted for readability.<br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix.<br>206 in. normal calibration or 113.75 in. elevated indication on the Shutdown RPV water level instrument (LI-86) corresponds to the RPV flange. |

| | | | | |
|---|---|---|---|---|
| 2 | RCS/RPV level cannot be monitored with a loss of RCS/RPV inventory as indicated by an unexplained level rise in (site specific sump or tank) | CU2.3 | RPV level **cannot** be monitored with **any** unexplained RPV leakage indication, Table C-1 | The NEI phrase "Loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase" has been changed to "with **any** unexplained RCS leakage indication, Table C-1" to reduce EAL-user reading burden and thereby promote timely and accurate emergency classifications. This change also aligns the syntax of the CNS EAL with that used in EAL CA2.1.<br><br>Table C-1 lists the site-specific conditions that could be indicative of a loss of inventory from the RPV. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory. |

| **Table C-1 RPV Leakage Indications** |
|---|
| • Drywell equipment drain sump level rise<br>• Drywell floor drain sump level rise<br>• Reactor Building equipment drain sump level rise<br>• Reactor Building floor drain sump level rise<br>• Torus water level rise<br>• RPV make-up rate rise<br>• Observation of unisolable RCS leakage |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU3 | AC power capability to emergency busses reduced to a single power source for 15 minutes or longer such that any additional single failure would result in station blackout.<br>MODE: Cold Shutdown, Refueling | CU1a | AC power capability to critical buses reduced to a single power source for 15 minutes or longer such that **any** additional single failure would result in loss of **all** AC power to critical buses<br>MODE: 4 - Cold Shutdown, 5 - Refueling | The term "station blackout" was replaced with "loss of all AC power to critical buses" as this clearly describes the intended condition leading to the Alert threshold in CU1a. Station Blackout is not an operationally defined term for loss of all AC to critical buses.<br>The CNS "critical buses" are the NEI "emergency buses." |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>1. a. AC power capability to (site specific emergency busses) reduced to a single power source for 15 minutes or longer.<br>**AND**<br>b. Any additional single power source failure will result in station blackout. | CU1.1 | AC power capability to critical 4160V buses 1F and 1G reduced to a single power source (Table C-4) for ≥ 15 min. such that **any** additional single failure would result in loss of **all** AC power to critical buses (Note 3)<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | The NEI example EAL contingent "...any additional single failure will result in station blackout" has been simplified consistent with the IC wording.<br>The term "station blackout" was replaced with "loss of all AC power to critical buses" as this clearly describes the intended condition leading to the Alert threshold in CA1.1. Station Blackout is not an operationally defined term for loss of all AC to critical buses.<br>The NEI phrase "...AND...will..." has been changed to "...such that...would..." to clarify the consequences if the single remaining power source were to be lost.<br>Table C-4 provides a list of AC power sources.<br>Critical 4160V buses 1F and 1G are the CNS emergency buses.<br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix. |

| Table C-4 AC Power Sources |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU4 | UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV.<br>MODE: Cold Shutdown, Refueling | CU3 | Unplanned loss of decay heat removal capability with irradiated fuel in the RPV<br>MODE: 4 - Cold Shutdown, 5 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | An UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit. | CU3.1 | **Any** unplanned event results in RCS temperature > 212°F due to loss of decay heat removal capability | The NEI indefinite article "an" has been replaced with the term "any" to improve precision and avoid ambiguity. The term "any" means "one or more." The indefinite article "an" means one single event but lacks specificity if more than one unplanned event occurs. The CNS wording resolves this concern.<br>The NEI phrase "...exceeding the Technical Specification cold shutdown temperature limit" has been replaced with "> 212ºF." >212°F is the Technical Specification cold shutdown temperature limit and is specified in the EAL instead of the NEI wording to reduce EAL-user reading burden.<br>Added the words from the IC above "...due to loss of decay heat removal capability" to clearly indicate classification is based on loss of decay heat removal capability. |
| 2 | Loss of all RCS temperature and RCS/RPV level indication for 15 minutes or longer.<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | CU3.2 | Loss of **all** RCS temperature and RPV level indication for ≥ 15 min. (Note 3)<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely | Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix. |

| | | | | |
|---|---|---|---|---|
| | | | exceed the applicable time. | |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU6 | Loss of all On-site or Off-site communications capabilities.<br>MODE: Cold Shutdown, Refueling, Defueled | CU4 | Loss of **all** onsite or offsite communications capabilities<br>MODE: 4 - Cold Shutdown, 5 - Refueling, 6 - Defueled | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss of all of the following on-site communication methods affecting the ability to perform routine operations:<br>(site specific list of communications methods) | CU4.1 | Loss of **all** Table C-2 onsite (internal) communication methods affecting the ability to perform routine operations<br>**OR**<br>Loss of **all** Table C-2 offsite (external) communication methods affecting the ability to perform offsite notifications | CU4.1 implements Example EALs #1 and #2. These were combined for improved usability.<br>The NEI example EALs specify site-specific lists of onsite and offsite communications methods. The CNS EAL lists these methods in Table C-2 because the number of communications methods is too long to include within the text of the EAL.<br>The adjectives "(internal)" and "(external)" have been added to the CNS EAL for clarification. The terms "onsite/offsite" could be interpreted as the location in which the communication originates instead of the location to which communication is directed. |
| 2 | Loss of all of the following off-site communication methods affecting the ability to perform offsite notifications:<br>(site specific list of communications methods) | | | |

| Table C-2 Communications Systems | | |
|---|---|---|
| **System** | **Onsite** (internal) | **Offsite** (external) |
| Station Intercom System "Gaitronics" | X | |
| Site UHF Radio Paging System | X | |
| Alternate Intercom | X | |
| CNS On-Site Cell Phone System | X | X |
| Telephone system (PBX) | X | X |
| Federal Telecommunications System (FTS 2001) | | X |
| Microwave Telephone Network | | X |
| Local Telephones (C.O. Lines) | | X |
| CNS State Notification Telephones | | X |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU7 | Loss of required DC power for 15 minutes or longer.<br>MODE: Cold Shutdown, Refueling | CU1b | Loss of required DC power for 15 minutes or longer<br>MODE: 4 - Cold Shutdown, 5 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>1. Less than (site specific bus voltage indication) on required (site specific Vital DC busses) for 15 minutes or longer. | CU1.2 | < 105 VDC bus voltage indications on **all** Technical Specification required 125 VDC buses for ≥ 15 min. (Note 3)<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | "105 VDC" is the site-specific bus voltage.<br>The "site-specific Vital DC busses" are all Technical Specification required 125 VDC buses.<br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CU8 | Inadvertent criticality.<br>MODE: Cold Shutdown, Refueling | CU5 | Inadvertent criticality<br>MODE: 4 - Cold Shutdown, 5 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | An UNPLANNED sustained positive period observed on nuclear instrumentation. (BWR) | CU5.1 | An unplanned sustained positive period observed on nuclear instrumentation | None |
| 1 | An UNPLANNED sustained positive startup rate observed on nuclear instrumentation. (PWR) | N/A | N/A | This Example EAL has not been implemented because it applies only to PWR plants. CNS is a BWR. BWRs are not equipped with startup rate meters. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CA1 | Loss of RCS/RPV inventory.<br>MODE: Cold Shutdown, Refueling | CA2 | Loss of RPV inventory<br>MODE: 4 - Cold Shutdown, 5 - Refueling | The NEI phrase "RPV" is the used terminology commonly accepted at BWRs. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss of RCS/RPV inventory as indicated by level less than (site specific level).<br>*[Low-Low ECCS actuation setpoint / Level 2 (BWR)]*<br>*[Bottom ID of the RCS loop (PWR)]* | CA2.1 | RPV level < -42 in.<br>**OR**<br>RPV level **cannot** be monitored for ≥ 15 min. (Note 3) with **any** unexplained RPV leakage indication, Table C-1<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | The NEI phrase "RPV" is the used terminology commonly accepted at BWRs.<br>CA2.1 implements both Example EALs #1 and #2. The NEI Example EALs have been combined in one CNS EAL to improve usability.<br>The NEI phrase "Loss of RCS inventory as indicated by..." has been deleted because it is obvious from the low-low ECCS actuation setpoint water level (-42 in.) that inventory in the RCS has been lost. This change has been made to reduce EAL-user reading burden and thereby promote timely and accurate emergency classifications. The remainder of the plant EAL clearly associates the threshold parameters with the inventory in the RPV.<br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix.<br>CNS is a BWR and is not equipped with the PWR RCS loop hot leg penetration.<br>The NEI phrase "loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase" has been changed to "with **any** unexplained RCS leakage indication, Table C-1" to reduce EAL-user reading burden and thereby promote timely and accurate emergency classifications. This change also aligns the syntax of the CNS EAL with that used in EAL CU2.3.<br>Table C-1 lists the conditions that could be indicative of a loss of inventory from the RPV. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. With RHR System operating in the Shutdown Cooling |
| 2 | RCS/RPV level cannot be monitored for 15 minutes or longer with a loss of RCS/RPV inventory as indicated by an unexplained level rise in (site specific sump or tank).<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | | | |

| | | | | |
|---|---|---|---|---|
| | | | | mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory. |

| Table C-1 RPV Leakage Indications |
|---|
| • Drywell equipment drain sump level rise<br>• Drywell floor drain sump level rise<br>• Reactor Building equipment drain sump level rise<br>• Reactor Building floor drain sump level rise<br>• Torus water level rise<br>• RPV make-up rate rise<br>• Observation of unisolable RCS leakage |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CA3 | Loss of all Off-site and all On-Site AC power to emergency busses for 15 minutes or longer.<br>MODE: Cold Shutdown, Refueling, Defueled | CA1 | Loss of **all** offsite and **all** onsite AC power to critical buses for 15 minutes or longer<br>MODE: 4 - Cold Shutdown, 5 - Refueling, D - Defueled | The CNS "critical buses" are the NEI "emergency buses." |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>Loss of all Off-Site and all On-Site AC Power to (site specific emergency busses) for 15 minutes or longer. | CA1.1 | Loss of **all** offsite and **all** onsite AC power (Table C-4) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3)<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | Table C-4 provides a list of AC power sources.<br>Critical 4160V buses 1F and 1G are the CNS essential buses.<br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix. |

| Table C-4 AC Power Sources |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CA4 | Inability to maintain plant in cold shutdown.<br>MODE: Cold Shutdown, Refueling | CA3 | Inability to maintain plant in cold shutdown<br>MODE: 4 - Cold Shutdown, 5 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | An UNPLANNED event results in RCS temperature greater than (site specific Technical Specification cold shutdown temperature limit) for greater than the specified duration on table. | CA3.1 | **Any** unplanned event results in **EITHER**:<br>RCS temperature > 212°F for > Table C-3 duration (Note 4)<br>**OR**<br>RPV pressure increase > 10 psig due to a loss of RCS cooling<br>Note 4: Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety. | CA3.1 implements NEI EALs #1 and #2. With one EAL, redundant wording is deleted; thus, reducing EAL-user reading burden.<br>The NEI indefinite article "an" in example EAL #1 has been replaced with the term "any" to improve precision and avoid ambiguity. The term "any" means "one or more." The indefinite article "an" means one single event but lacks specificity if more than one unplanned event occurs. The CNS wording resolves this concern.<br>The CNS Technical Specification cold shutdown temperature limits 212°F.<br>NEI criteria associated with RCS temperature exceeding the Technical Specification cold shutdown temperature limit have been listed in Table C-3.<br>The NEI criteria "...or RCS inventory reduced.." has not been included in the associated Table C-3 thresholds as this is only applicable to PWRs. CNS does not define a "reduced inventory" condition while in Cold Shutdown or Refueling Modes.<br>Note 4 has been added to ensure the definition of Containment Closure is understood.<br>The CNS pressure of 10 psig is the site-specific RCS pressure. |
| 2 | An UNPLANNED event results in RCS pressure increase greater than 10 psi due to a loss of RCS cooling. (PWR-This EAL does not apply in Solid Plant conditions.) | | | |

| Table: RCS Reheat Duration Thresholds | | |
|---|---|---|
| **RCS** | **Containment Closure** | **Duration** |
| Intact (but not RCS Reduced Inventory [PWR]) | N/A | 60 minutes* |
| Not intact or RCS Reduced Inventory *(PWR)* | Established | 20 minutes* |
| | Not Established | 0 minutes |
| * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable. | | |

| **Table C-3 RCS Reheat Duration Thresholds** | |
|---|---|
| * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable | |
| 1. RCS intact (Containment Closure N/A) | 60 min.* |
| 2. Containment Closure established **AND** RCS **not** intact | 20 min.* |
| 3. Containment Closure **not** established **AND** RCS **not** intact | 0 min. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CS1 | Loss of RCS/RPV inventory affecting core decay heat removal capability.<br>MODE: Cold Shutdown, Refueling | CS2 | Loss of RPV inventory affecting core decay heat removal capability<br>MODE: 4 - Cold Shutdown, 5 - Refueling | The NEI phrase "RPV" is the used terminology commonly accepted at BWRs. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | With CONTAINMENT CLOSURE not established, RCS/RPV level less than (site specific level).<br>[*6" below the bottom ID of the RCS loop (PWR)*]<br>[6" below the low-low ECCS actuation setpoint (BWR)]<br><br>OR | CS2.1 | With Containment Closure **not** established (Note 4), RPV level < -48 in.<br>Note 4: Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety. | The NEI phrase "RPV" is the used terminology commonly accepted at BWRs.<br>CNS is a BWR and is not equipped with the PWR RCS loop setpoint.<br>-48 in. is 6 in. below the low-low ECCS actuation setpoint (-42 in.).<br>Note 4 has been added to ensure the definition of Containment Closure is understood. |
| 2 | With CONTAINMENT CLOSURE established, RCS/RPV level less than (site specific level for TOAF).<br><br>OR | CS2.2 | With Containment closure established, RPV level < -158 in. (Note 4)<br>Note 4: Containment Closure is the action taken to secure primary or secondary containment and its associated | -158 in. is the CNS EOP RPV water level for top of active fuel.<br>Note 4 has been added to ensure the definition of Containment Closure is understood. |

| | | | | |
|---|---|---|---|---|
| | | | structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety. | |
| 3 | RCS/RPV level cannot be monitored for 30 minutes or longer with a loss of RCS/RPV inventory as indicated by **ANY** of the following:<br>• (Site specific radiation monitor) reading greater than (site specific value).<br>• Erratic Source Range Monitor Indication<br>• UNPLANNED level rise in (site specific sump or tank).<br>**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | CS2.3 | RPV level **cannot** be monitored for ≥ 30 min. (Note 3) with a loss of inventory as indicated by **EITHER:**<br>Unexplained RPV leakage indication, Table C-1<br>**OR**<br>Erratic Source Range Monitor indication<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | Applicable RPV leakage indications have been listed in Table C-1<br>The NEI 99-01 example EALs include the use of radiation monitor readings corresponding to those expected for core uncovery in the Refueling Mode (vessel head removed). The generic bases states that the use of radiation monitoring as an EAL input may not be appropriate for some BWRs. Consistent with the bases, the CNS Containment High Range Radiation Monitors cannot be utilized for this purpose because of their location relative to the reactor vessel and core. Additionally, no other installed radiation monitoring system exists that can be utilized for the function. However, CNS does have extensive redundant RPV level monitoring capability available to assess core uncovery in the Refueling Mode. Consistent with indicators used in the EALs derived from generic IC CS1, unexplained RPV leakage indications; Table C-1 has been incorporated and expanded with other site-specific indicators of inventory loss. Therefore this generic indicator is not incorporated into the applicable site-specific EALs. **This is a deviation from NEI 99-01 Revision 5**.<br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix. |

| Table C-1 RPV Leakage Indications |
|---|
| • Drywell equipment drain sump level rise<br>• Drywell floor drain sump level rise<br>• Reactor Building equipment drain sump level rise<br>• Reactor Building floor drain sump level rise<br>• Torus water level rise<br>• RPV make-up rate rise<br>• Observation of unisolable RCS leakage |

| NEI IC# | NEI IC Wording* | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CG1 | Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged.<br>MODE: Cold Shutdown, Refueling | CG2 | Loss of RPV inventory affecting fuel clad integrity with containment challenged<br>MODE: 4 - Cold Shutdown, 5 - Refueling | The NEI phrase "RPV" is the used terminology commonly accepted at BWRs. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.<br>a. RCS/RPV level less than (site specific level for TOAF) for 30 minutes or longer.<br>**AND**<br>b. **ANY** containment challenge indication (see Table): | CG2.1 | RPV level < -158 in. for ≥ 30 min. (Note 3)<br>**AND**<br>**Any** Containment Challenge indication, Table C-5<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | The NEI phrase "RPV" is the used terminology commonly accepted at BWRs.<br>-158 in. is the CNS EOP RPV water level for top of active fuel.<br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix.<br>Table C-5 lists the Containment Challenge indications.<br>The NEI threshold "Explosive mixture inside containment" has been changed to "Deflagration concentrations exist inside PC..." for completeness and clarity. The BWROG EPGs/SAGs specifically define the limits associated with explosive mixtures in terms of deflagration concentrations of hydrogen and oxygen inside the drywell and wetwell. For Mk I containments like CNS, the deflagration limits are 6% hydrogen and 5% oxygen concentrations in the drywell or torus.<br>The NEI phrase "Secondary Containment radiation monitors above {site-specific} value (BWR only)" has been changed to "Secondary Containment area radiation > 1000 mR/hr (EOP-5A Table 10)" to agree with the syntax employed in the CNS fission product barrier PC Loss C.5. The Maximum Safe Operating Value of 1000 R/hr instead of alarm setpoints are used to be consistent with the Containment Challenge threshold of NEI IC CG1 and Table 10 of EOP-5A, Secondary Containment Control. This is consistent with the NEI 99-01 IC CG1 basis which states "the site-specific radiation monitor values should be based on the EOP "maximum |

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|---|---|---|---|---|
| | | | | safe values" because these values are easily recognizable and have an emergency basis."<br>Reference to Note 4 has been added to Table C-5 to ensure the definition of Containment Closure is understood. |
| 2 | a. RCS/RPV level cannot be monitored with core uncovery indicated by **ANY** of the following for 30 minutes or longer.<br>• (Site specific radiation monitor) reading greater than (site specific setpoint).<br>• Erratic source range monitor indication<br>• UNPLANNED level rise in (site specific sump or tank).<br>• *[Other site specific indications]*<br>**AND**<br>b. **ANY** containment challenge indication (see Table):<br>**Table: Containment Challenge Indications**<br>• CONTAINMENT CLOSURE not established.<br>• (Site specific explosive mixture) inside containment. | CG2.2 | RPV level **cannot** be monitored for ≥ 30 min. (Note 3) with core uncovery indicated by **EITHER**:<br>Unexplained RPV leakage indication, Table C-1<br>**OR**<br>Erratic Source Range Monitor Indication<br>**AND**<br>**Any** Containment Challenge indication, Table C-5<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | The NEI phrase "RPV" is the used terminology commonly accepted at BWRs.<br>The term logic term "EITHER" is used instead of "ANY" since only two contingencies are used.<br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix.<br>Table C-1 lists the conditions that could be indicative of a loss of inventory from the RPV. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.<br>The NEI 99-01 example EALs include the use of radiation monitor readings corresponding to those expected for core uncovery in the Refueling Mode (vessel head removed). The generic bases states that the use of radiation monitoring as an EAL input may not be appropriate for some BWRs. Consistent with the bases, the CNS Containment High Range Radiation Monitors cannot be utilized for this purpose because of their location relative to the reactor vessel and core. Additionally, no other installed radiation monitoring system exists that can be utilized for the function. However, CNS does have extensive redundant RPV level monitoring capability available to assess core uncovery in the Refueling Mode. Consistent with indicators used in the EALs derived from generic IC CS1, unexplained RPV leakage indications; Table C-1 has been incorporated and expanded with other site-specific indicators of inventory loss. Therefore this generic indicator is not incorporated |

| | | | | |
|---|---|---|---|---|
| | • UNPLANNED rise in containment pressure.<br>• Secondary containment radiation monitor reading above (site specific value). [*BWR only*] | | | into the applicable site-specific EALs. **This is a deviation from NEI 99-01 Rev. 5**.<br>Table C-5 lists the Containment Challenge indications.<br>The NEI threshold "Explosive mixture inside containment" has been changed to "Deflagration concentrations exist inside PC…" for completeness and clarity. The BWROG EPGs/SAGs specifically define the limits associated with explosive mixtures in terms of deflagration concentrations of hydrogen and oxygen inside the drywell and wetwell. For Mk I containments like CNS, the deflagration limits are 6% hydrogen and 5% oxygen concentrations in the drywell or torus.<br>The NEI phrase "Secondary Containment radiation monitors above {site-specific} value (BWR only)" has been changed to "Secondary Containment area radiation > 1000 mR/hr (EOP-5A Table 10)" to agree with the syntax employed in the CNS fission product barrier PC Loss C.5. The Maximum Safe Operating Value of 1000 mR/hr instead of alarm setpoints are used to be consistent with the Containment Challenge threshold of NEI IC CG1 and Table 10 of EOP-5A, Secondary Containment Control. This is consistent with the NEI 99-01 IC CG1 basis which states "the site-specific radiation monitor values should be based on the EOP "maximum safe values" because these values are easily recognizable and have an emergency basis."<br>Reference to Note 4 has been added to Table C-5 to ensure the definition of Containment Closure is understood. |

| Table C-1 RPV Leakage Indications |
|---|
| • Drywell equipment drain sump level rise<br>• Drywell floor drain sump level rise<br>• Reactor Building equipment drain sump level rise<br>• Reactor Building floor drain sump level rise<br>• Torus water level rise<br>• RPV make-up rate rise<br>• Observation of unisolable RCS leakage |

| Table C-5 Containment Challenge Indications |
|---|
| • Containment Closure **not** established (Note 4)<br>• Deflagration concentrations exist inside PC<br>$\geq$ 6% $H_2$ in drywell or torus<br>**AND**<br>$\geq$ 5% $O_2$ in drywell or torus<br>• Unplanned rise in PC pressure<br>• Secondary Containment area radiation > 1000 mR/hr (EOP-5A Table 10) |

# Category D

# Permanently Defueled Station Malfunction

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| D-AU1<br>D-AU2<br>D-SU1<br>D-HU1<br>D-HU2<br>D-HU3<br>D-AA1<br>D-AA2<br>D-HA1<br>D-HA2 | Recognition Category D<br>Permanently Defueled Station Malfunction | N/A | N/A | NEI Recognition Category D ICs and EALs are applicable only to permanently defueled stations. CNS is not a defueled station. |

# Category E

## Events Related to Independent Spent Fuel Storage Installations

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| E-HU1 | Damage to a loaded cask CONFINEMENT BOUNDARY.<br>MODE: Not applicable | EU1 | Damage to a loaded cask confinement boundary<br>MODE: N/A | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Damage to a loaded cask confinement BOUNDARY. | EU1.1 | Damage to a loaded cask confinement boundary | None |

# Category F

# Fission Product Barrier Degradation

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FU1 | ANY Loss or ANY Potential Loss of Containment<br>MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FU1 | **Any** loss or **any** potential loss of Primary Containment<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | Added 'Primary' to 'Containment' as this is proper terminology for the intended structure for BWRs |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | ANY Loss or ANY Potential Loss of Containment. | FU1.1 | **Any** loss or **any** potential loss of Primary Containment (Table F-1) | Added "Primary" to "Containment" as this is proper terminology for the intended structure for BWRs.<br>Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-2. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FA1 | ANY Loss or ANY Potential Loss of EITHER Fuel Clad or RCS.<br>MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FA1 | **Any** loss or **any** potential loss of either Fuel Clad or RCS<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | ANY Loss or ANY Potential Loss of EITHER Fuel Clad or RCS. | FA1.1 | **Any** loss or **any** potential loss of either Fuel Clad or RCS (Table F-1) | Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-2. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FS1 | Loss or Potential Loss of ANY two barriers.<br>MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FS1 | Loss or potential loss of **any** two barriers<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss or Potential Loss of ANY two barriers. | FS1.1 | Loss or potential loss of **any** two barriers (Table F-1) | Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-2. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FG1 | Loss of ANY Two Barriers AND Loss or Potential Loss of third barrier.<br>MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FG1 | Loss of **any** two barriers and loss or potential loss of third barrier<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss of ANY Two Barriers AND Loss or Potential Loss of third barrier. | FG1.1 | Loss of **any** two barriers<br>**AND**<br>Loss or potential loss of third barrier (Table F-1) | Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-2. |

| NEI Ex. EAL # | NEI Table 5-F-1 Notes | CNS EAL # | CNS Table F-1 EAL Notes | Difference/Deviation Justification |
|---|---|---|---|---|
| N/A | The logic used for these initiating conditions reflects the following considerations:<br>• The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier (See Sections 3.4 and 3.8). NOUE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.<br>• At the Site Area Emergency | FU1.1<br>FA1.1<br>FS1.1<br>FG1.1 | The logic used for these initiating conditions reflects the following considerations:<br>• The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Primary Containment barrier. UE EALs associated with RCS and Fuel Clad barriers are addressed under System Malfunction EALs.<br>• At the Site Area Emergency level, there must be some ability to dynamically assess | First bullet: The NEI parenthetical phrase "See Sections 3.4 and 3.8" has been deleted because it refers to NEI EAL developmental information.<br>First bullet: The NEI acronym "NOUE" has been implemented as "UE" for simplification.<br>The NEI abbreviation "ICs" has been changed to "EALs" for clarification.<br>Added 'Primary' to 'Containment' as this is proper terminology for the intended structure for BWRs |

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| | level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" EALs existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.<br>• The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.<br>• The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in | | how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS barrier "loss" EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was **no** immediate need to escalate to a General Emergency.<br>• The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.<br>• The Primary Containment barrier should **not** be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Primary Containment barrier. When **no** event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Primary | |

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| | progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications. | | Containment barrier status is addressed by Technical Specifications | |

| NEI Ex. EAL # | NEI Table 5-F-4 Notes | CNS EAL # | CNS Table F-1 EAL Notes | Difference/Deviation Justification |
|---|---|---|---|---|
| N/A | *Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also, multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is IMMINENT. In this IMMINENT loss situation use judgment and classify as if the thresholds are exceeded. | FU1.1<br>FA1.1<br>FS1.1<br>FG1.1 | Determine which combinations of the three barriers are lost or have a potential loss and use FU1.1, FA1.1, FS1.1 and FG1.1 to classify the event. Also an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent. In this imminent loss situation use judgment and classify as if the thresholds are exceeded. | The NEI phrase "...the following key..." has been replaced with "...FU1.1, FA1.1, FS1.1 and FG1.1..." to adapt the NEI wording to the syntax of a note. These CNS EALs are equivalent to the NEI key. |

## Table F-1 Fission Product Barrier Matrix

| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Primary Containment Barrier | |
|---|---|---|---|---|---|---|
| | Loss | Potential Loss | Loss | Potential Loss | Loss | Potential Loss |
| A. RPV Level | 1. PC flooding is required due to **any** of the following:<br>• RPV water level **cannot** be restored and maintained > -183 in.<br>• RPV water level **cannot** be restored and maintained ≥ -209 in. and **no** core spray subsystem flow can be restored and maintained ≥ 4,750 gpm<br>• RPV water level **cannot** be determined and core damage is occurring | 5. RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined | 7. RPV level cannot be restored and maintained > -158 in. or **cannot** be determined | None | None | 22. PC Flooding required |
| B. PC Pressure / Temperature | None | None | 8. PC pressure > 1.84 psig due to RCS leakage | None | 16. PC pressure rise followed by a rapid unexplained drop in PC pressure<br>17. PC pressure response **not** consistent with LOCA conditions | 23. PC pressure > 56 psig and rising<br>24. Deflagration concentrations exist inside PC<br>≥ 6% $H_2$ in drywell or torus<br>(or **cannot** be determined)<br>**AND**<br>≥ 5% $O_2$ in drywell or torus<br>(or **cannot** be determined)<br>25. Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7) |
| C. Isolation | None | None | 9. Release pathway exists outside primary containment resulting from isolation failure in **any** of the following (excluding normal process system flowpaths from an unisolable system):<br>• Main steam line<br>• HPCI steam line<br>• RCIC steam line<br>• RWCU<br>• Feedwater | 13. RCS leakage > 50 gpm inside the drywell<br>14. Unisolable primary system discharge outside primary containment as indicated by exceeding **any** secondary containment Maximum Normal Operating temperature or radiation value (EOP-5A Tables 9 and 10) | 18. Failure of **all** valves in **any** one line to close<br>**AND**<br>Direct downstream pathway to the environment exists after PC isolation signal<br>19. Intentional PC venting per EOPs<br>20. Unisolable primary system discharge outside PC as indicated by exceeding **any** secondary containment Maximum Safe Operating temperature or radiation value (EOP-5A Tables 9 and 10) | None |
| D. ERD | None | None | 10. Emergency RPV depressurization is required | None | None | None |
| E. Rad | 2. Drywell radiation monitor (RMA-RM-40A/B) > 2.50E+03 Rem/hr<br>3. Primary coolant activity > 300 µCi/gm dose equivalent I-131 | None | 11. Drywell radiation monitor (RMA-RM-40A/B) > 2.40E+02 Rem/hr | None | None | 26. Drywell radiation monitor (RMA-RM-40A/B) > 5.00E+04 Rem/hr |
| F. Judgment | 4. **Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier | 6. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier | 12. **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier | 15. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier | 21. **Any** condition in the opinion of the Emergency Director that indicates loss of the PC barrier | 27. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier |

**Fuel Clad Fission Product Barrier Degradation Thresholds**

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FC Loss 1 | **Primary Coolant Activity Level**<br>A. Primary coolant activity greater than (site specific value). | FC Loss 3 | Primary coolant activity > 300 μCi/gm dose equivalent I-131 | > 300 μCi/gm dose equivalent I-131 is the site-specific coolant activity. |
| FC Loss 2 | **Reactor Vessel Water Level**<br>A. RPV water level cannot be restored and maintained above (site specific RPV water level corresponding to the requirement for primary containment flooding). | FC Loss 1 | Primary Containment Flooding is required due to **any** of the following:<br>• RPV water level **cannot** be restored and maintained > -183 in.<br>• RPV water level **cannot** be restored and maintained ≥ -209 in. and **no** core spray subsystem flow can be restored and maintained ≥ 4,750 gpm<br>• RPV water level **cannot** be determined and core damage is occurring | The site-specific thresholds corresponding to the requirement for primary containment flooding are given in the CNS EOPs and dictate entry to the SAGs. |
| FC Loss 3 | **Not Applicable** | N/A | N/A | |

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FC Loss 4 | **Primary Containment Radiation Monitoring**<br>A. Primary containment radiation monitor reading greater than (site specific value). | FC Loss 2 | Drywell radiation monitor (RMA-RM-40A/B) > 2.50E+03 Rem/hr | The NEI phrase "Primary containment" has been changed to "Drywell" to reflect the area in which the specified radiation monitors are located.<br>The NEI term "reading" has been deleted because it is only possible to detect drywell/torus radiation through a radiation monitor reading. The verbiage is therefore redundant. It is not necessary to specify the monitor by which the parameter is read as well as the parameter. The CNS threshold is clear, concise and reduces EAL-user reading burden.<br>Procedure 5.7.17 Attachment 7 provides a method of calculating percent fuel clad damage and fuel melt based on drywell radiation. Under LOCA conditions, a reading of 2.44E+6 Rem/hr corresponds to 100% core melt on RMA-RM-40A/B. A value of 2.44E+3 Rem/hr (rounded to 2.50E+03 Rem/hr) yields 1% fuel clad damage using this method. |
| FC Loss 5 | **Other (Site-Specific) Indications**<br>A. (Site-specific) as applicable. | N/A | N/A | Other site-specific indications of Fuel Clad loss have not been identified. |
| FC Loss 6 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier. | FC Loss 4 | **Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier | None |
| FC P-Loss 1 | **Primary Coolant Activity Level**<br>Not Applicable | N/A | N/A | |

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| FC P-Loss 2 | **Reactor Vessel Water Level**<br>A. RPV water level cannot be restored and maintained above (site specific RPV water level corresponding to the top of active fuel) or cannot be determined. | FC P-Loss 5 | RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined | The NEI phrase "RPV Water Level…" has been replaced with "RPV level" for simplification. It is clear from the context in which it is used that the level of concern is RPV water level.<br>-158 in. is the top of active fuel which corresponds to the intended site-specific water level value. |
| FC P-Loss 3 | **Not Applicable** | N/A | N/A | |
| FC P-Loss 4 | **Primary Containment Radiation Monitoring**<br>Not Applicable | N/A | N/A | |
| FC P-Loss 5 | **Other (Site-Specific) Indications**<br>A. (Site-specific) as applicable. | N/A | N/A | Other site-specific indications of Fuel Clad potential loss have not been identified. |
| FC P-Loss 6 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier. | FC P-Loss 6 | **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier | None |

**RCS Fission Product Barrier Degradation Thresholds**

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| RCS Loss 1 | **Primary Containment Pressure**<br>A. Primary containment pressure greater than (site specific value) due to RCS leakage. | RCS Loss 8 | PC pressure > 1.84 psig due to RCS leakage | 1.84 psig is the Primary Containment (PC) high pressure scram setpoint. |
| RCS Loss 2 | **Reactor Vessel Water Level**<br>A. RPV water level cannot be restored and maintained above (site specific RPV water level corresponding to the top of active fuel) or cannot be determined. | RCS Loss 7 | RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined | The NEI phrase "RPV Water Level…" has been replaced with "RPV level" for simplification. It is clear from the context in which it is used that the level of concern is RPV water level.<br>-158 in. is the top of active fuel which corresponds to the intended RPV water level site-specific value. |
| RCS Loss 3 | **RCS Leak Rate**<br>A. (site specific Indication of an UNISOLABLE Main Steamline, HPCI, Feedwater, RWCU, or RCIC break)<br>**OR**<br>B. Emergency RPV Depressurization is required. | RCS Loss 9 | Release pathway exists outside primary containment resulting from isolation failure in **any** of the following (excluding normal process system flowpaths from an unisolable system):<br>• Main steam line<br>• HPCI steam line<br>• RCIC steam line<br>• RWCU<br>• Feedwater | NEI threshold "A" is implemented by RCS Loss 9.<br>The NEI threshold "(site specific Indication of an UNISOLABLE Main Steamline, HPCI, Feedwater, RWCU, or RCIC break)" has been changed to "Release pathway exists outside primary containment resulting from isolation failure in **any** of the following (excluding normal process system flowpaths from an unisolable system):…Main steam line…HPCI steam line…RCIC steam line…RWCU…Feedwater" to clarify NEI intent.<br>Since BWR EOPs may specify continued use of one of these paths to achieve safe plant shutdown and cooldown, it is appropriate to exclude from emergency classification during normal process system operations that coincidentally use flow through an unisolable system. In these cases, other EALs and fission product barrier thresholds (e.g., ATWS events, requirements for emergency RPV depressurization, etc.) ensure the appropriate emergency classification is reached. |

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| | | RCS Loss 10 | Emergency RPV Depressurization is required | NEI threshold "B" is implemented by RCS Loss 10. |
| RCS Loss 4 | **Primary Containment Radiation Monitoring**<br>A. Primary containment radiation monitor reading greater than (site specific value). | RCS Loss 11 | Drywell radiation monitor (RMA-RM-40A/B) > 2.40E+02 Rem/hr | The NEI phrase "Primary containment" has been changed to "Drywell" to reflect the area in which the specified radiation monitors are located.<br>The NEI term "reading" has been deleted because it is only possible to detect drywell/torus radiation through a radiation monitor reading. The verbiage is therefore redundant. It is not necessary to specify the monitor by which the parameter is read as well as the parameter. The CNS threshold is clear, concise and reduces EAL-user reading burden.<br>Procedure 5.7.17 Attachment 7 provides a method of calculating percent fuel clad damage and fuel melt based on drywell radiation. A reading of 2.44E+6 Rem/hr corresponds to 100% core melt on RMA-RM-40A/B. A value of 2.44E+2 Rem/hr (rounded to 2.40E+02 Rem/hr) yields 0.1%% fuel clad damage using this method. |
| RCS Loss 5 | **Other (Site-Specific) Indications**<br>A. (site-specific) as applicable | N/A | N/A | Other site-specific indications of RCS loss have not been identified. |
| RCS Loss 6 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicate Loss of the RCS Barrier | RCS Loss 12 | **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier | None |
| RCS P-Loss 1 | **Primary Containment Pressure**<br>Not applicable | N/A | N/A | |
| RCS P-Loss 2 | **Reactor Vessel Water Level**<br>Not applicable | N/A | N/A | |

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| RCS P-Loss 3 | **RCS Leak Rate**<br>A. RCS leakage greater than 50 gpm inside the drywell.<br>OR<br>B. UNISOLABLE primary system leakage outside primary containment as indicated by exceeding EITHER of the following:<br>a. Max Normal Operating Temperature.<br>**OR**<br>b. Max Normal Area Radiation. | RCS P-Loss 13 | RCS leakage > 50 gpm inside the drywell | None |
| | | RCS P-Loss 14 | Unisolable primary system discharge outside primary containment as indicated by exceeding **any** secondary containment Maximum Normal Operating temperature or radiation value (EOP-5A Tables 9 and 10) | The term "any" has been added to the CNS threshold and the NEI term "EITHER" has been deleted for clarification because there are multiple Maximum Normal Operating temperatures and radiation levels.<br>The phrase "value (EOP-5A Tables 9 and 10)" has been added to the CNS threshold for consistency with terminology used in the EOPs. The referenced tables list the area temperatures and radiation levels. |
| RCS P-Loss 4 | **Primary Containment Radiation Monitoring**<br>Not applicable | N/A | N/A | |
| RCS P-Loss 5 | **Other (Site-Specific) Indications**<br>A. (site-specific) as applicable. | N/A | N/A | Other site-specific indications of RCS potential loss have not been identified. |
| RCS P-Loss 6 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier. | RCS P-Loss 15 | **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier | None |

**Primary Containment Fission Product Barrier Degradation Thresholds**

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CMT Loss 1 | **Primary Containment Conditions**<br>A. Primary containment pressure rise followed by a rapid unexplained drop in primary containment pressure,<br>OR<br>B. Primary containment pressure response not consistent with LOCA conditions. | PC Loss 16 | PC pressure rise followed by a rapid unexplained drop in PC pressure | None |
| | | PC Loss 17 | PC pressure response **not** consistent with LOCA conditions | None |
| CMT Loss 2 | **Reactor Vessel Water Level**<br>Not Applicable | N/A | N/A | |
| CMT Loss 3 | **Primary Containment Isolation Failure or Bypass**<br>A. Failure of all valves in any one line to close.<br>**AND**<br>Direct downstream pathway to the environment exists after primary containment isolation signal. | PC Loss 18 | Failure of **all** valves in **any** one line to close<br>**AND**<br>Direct downstream pathway to the environment exists after PC isolation signal | None |
| | | PC Loss 19 | Intentional PC venting per EOPs | None |

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| | **OR**<br>B. Intentional primary containment venting per EOPs.<br>**OR**<br>C. UNISOLABLE primary system leakage outside primary containment as indicated by exceeding EITHER of the following:<br>a. Max Safe Operating Temperature.<br>**OR**<br>b. Max Safe Area Radiation. | PC Loss 20 | Unisolable primary system discharge outside PC as indicated by exceeding **any** secondary containment Maximum Safe Operating temperature or radiation value (EOP-5A Tables 9 and 10) | The term "any" has been added to the CNS threshold and the NEI term "EITHER" has been deleted for clarification because there are multiple Maximum Normal Operating temperatures and radiation levels.<br>The phrase "value (EOP-5A Tables 9 and 10)" has been added to the CNS threshold for consistency with terminology used in the EOPs. The referenced tables list the area temperatures and radiation levels. |
| CMT Loss 4 | **Primary Containment Radiation Monitoring**<br>Not Applicable | N/A | N/A | |
| CMT Loss 5 | **Other (Site-Specific) Indications**<br>A. (site-specific) as applicable. | N/A | N/A | Other site-specific indications of Containment loss have not been identified. |
| CMT Loss 6 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier. | PC Loss 21 | **Any** condition in the opinion of the Emergency Director that indicates loss of the PC barrier | Replaced 'Containment' with 'PC' as this is proper terminology for the intended structure for BWRs. |
| CMT P-Loss 1 | **Primary Containment Conditions**<br>A. Primary containment pressure greater than (site specific value) | PC P-Loss 23 | PC pressure > 56 psig and rising | 56 psig is the CNS primary containment internal design pressure. |

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| | and rising.<br>**OR**<br>B. Explosive mixture exists inside primary containment.<br>**OR**<br>C. RPV pressure and suppression pool temperature cannot be maintained below the HCTL. | PC P-Loss 24 | Deflagration concentrations exist inside PC<br>≥ 6% $H_2$ in drywell or torus (or **cannot** be determined)<br>**AND**<br>≥ 5% $O_2$ in drywell or torus (or **cannot** be determined) | The NEI threshold "Explosive mixture exists inside primary containment" has been changed to "Deflagration concentrations exist inside PC…" for completeness and clarity. The BWROG EPGs/SAGs specifically define the limits associated with explosive mixtures in terms of deflagration concentrations of hydrogen and oxygen inside the drywell and wetwell. For Mk I containments like CNS, the deflagration limits are 6% hydrogen and 5% oxygen concentrations in the drywell or torus. Since the EPGs/SAGs require deflagration concentration actions to be performed when hydrogen and oxygen concentrations cannot be determined, this phrase has been added to the meaning of explosive mixtures. |
| | | PC P-Loss 25 | Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7) | Replaced "suppression pool" with "torus water" for consistency with CNS EOPs.<br>Added the parenthetical reference to EOP/SAG Graph 7 to assist the end-user in locating the HCTL curves. |
| CMT P-Loss 2 | **Reactor Vessel Water Level**<br>A. Primary containment flooding required. | PC P-Loss 22 | PC Flooding required | The NEI phrase "Primary containment flooding required" has been changed to "PC Flooding required." to use terminology consistent with the CNS EOPs. |
| CMT P-Loss 3 | **Primary Containment Isolation Failure or Bypass**<br>Not Applicable | N/A | N/A | |

| NEI FPB # | NEI IC Wording | CNS FPB # | CNS FPB Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| CMT P-Loss 4 | **Primary Containment Radiation Monitoring**<br>A. Primary containment radiation monitor reading greater than (site specific value), | PC P-Loss 26 | Drywell radiation monitor (RMA-RM-40A/B) > 5.00E+04 Rem/hr | The NEI phrase "Primary containment" has been changed to "Drywell" to reflect the area in which the specified radiation monitors are located.<br>The NEI term "reading" has been deleted because it is only possible to detect drywell/torus radiation through a radiation monitor reading. The verbiage is therefore redundant. It is not necessary to specify the monitor by which the parameter is read as well as the parameter. The CNS threshold is clear, concise and reduces EAL-user reading burden.<br>Procedure 5.7.17 Attachment 7 provides a method of calculating percent fuel clad damage and fuel melt based on drywell radiation. A reading of 2.44E+6 Rem/hr corresponds to 100% core melt on RMA-RM-40A/B. A value of 4.88E+4 Rem/hr (rounded to 5.00E+04 Rem/hr) yields 20% fuel clad damage using this method |
| CMT P-Loss 5 | **Other (Site-Specific) Indications**<br>A. (site-specific) as applicable. | N/A | N/A | Other site-specific indications of Containment potential loss have not been identified. |
| CMT P-Loss 6 | **Emergency Director Judgment**<br>A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment barrier. | PC P-Loss 27 | **Any** condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier | Replaced 'Containment' with 'PC' as this is proper terminology for the intended structure for BWRs. |

# Category H

## Hazards and Other Conditions Affecting Plant Safety

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU1 | Natural or destructive phenomena affecting the PROTECTED AREA.<br>MODE: All | HU1 | Natural or destructive phenomena affecting the Protected Area<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Seismic event identified by **ANY** 2 of the following:<br>• Seismic event confirmed by (site specific indication or method)<br>• Earthquake felt in plant<br>• National Earthquake Center | HU1.1 | Seismic event identified by **any two** of the following:<br>• SMA-3 Strong Motion Accelograph actuated or Alarm B-3/B-1 SEISMIC EVENT<br>• Earthquake felt in plant<br>• National Earthquake Information Center | Receipt of the specified indicators is the site-specific method of indicating a felt earthquake. CNS seismic instrumentation actuates at 0.01 g.<br>Added the word "Information" to National Earthquake Center as that is the proper name. |
| 2 | Tornado striking within PROTECTED AREA boundary or high winds greater than (site specific mph). | HU1.2 | Tornado striking within Protected Area boundary<br>**OR**<br>Sustained high winds > 100 mph | Sustained 100 mph wind is the site-specific wind speed.<br>Divided the EAL into two contingents for clarification. |
| 3 | Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in **ANY** of the following areas:<br>(site specific area list) | HU1.4 | Flooding in **any** Table H-1 area that has the potential to affect safety-related equipment needed for the current operating mode | The NEI term "Internal" has been deleted from the plant EAL because any source of flooding (whether a source internal or external to the specified area) can have the potential to affect safety-related systems.<br>The CNS (site-specific) areas of the plant are listed in Table H-1. |
| 4 | Turbine failure resulting in casing penetration or damage to turbine | HU1.3 | Main turbine failure resulting in casing penetration or damage to | Added the word "Main" to avoid confusion with other turbines at CNS (e.g., feedwater turbines, HPCI, RCIC). |

| | | | | |
|---|---|---|---|---|
| | or generator seals. | | turbine or generator seals | |
| 5 | (Site-Specific) occurrences affecting the PROTECTED AREA. | HU1.5 | High river/forebay water level > 899' MSL<br>**OR**<br>Low river level/forebay < 870' MSL | Both high and low river/forebay water levels have been identified for CNS as other site-specific natural or destructive phenomena appropriate for classification at the Unusual Event level. |

| **Table H-1 Safe Shutdown Areas** |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU2 | FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA.<br>MODE: All | HU2 | Fire within the Protected Area not extinguished within 15 minutes of detection or explosion within Protected Area boundary<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed, the applicable time.<br><br>FIRE not extinguished within 15 minutes of control room notification or verification of a control room FIRE alarm in **ANY** of the following areas:<br><br>(site specific area list) | HU2.1 | Fire in **any** Table H-1 area **not** extinguished within 15 min. of Control Room notification or receipt of a valid Control Room alarm due to fire (Note 3)<br><br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | The NEI phrase "not extinguished within 15 minutes" follows the areas in which fire is a concern to improve readability and understandability.<br><br>The NEI phrase "...verification of a control room alarm" has been changed to "...receipt of a valid Control Room alarm due to fire" to be consistent with the generic bases which states: "Validation of a fire detection system alarm includes actions that can be taken within the Control Room or other nearby location to ensure that the alarm is not spurious." Alarms caused by smoke (e.g., slipping drive belts) are not valid alarms and should not be classified unless due to fire.<br><br>Added "...**any** Table H-1 area..." to be consistent with the generic NEI 99-01 bases that the EAL is intended to address fires in areas contiguous to areas "containing functions and systems required for safe shutdown".<br><br>Reference to the NEI note is included in the EAL wording "(Note 3)." Numbering the note facilitates referencing in the EAL matrix.<br><br>Note 3 implements the generic HU2 note though the wording has been made consistent with the similar note utilized in the Cold Shutdown category by deleting the phrase "...has exceeded or,". This phrase is unnecessary because, by definition, if the condition has existed for the allowed duration, the classification threshold has been met. |

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| 2 | EXPLOSION within the PROTECTED AREA. | HU2.2 | Explosion within the Protected Area | None |

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU3 | Release of toxic, corrosive, asphyxiant, or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS.<br>MODE: All | HU3 | Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS. | HU3.1 | Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect normal plant operations | None |
| 2 | Report by local, county or state officials for evacuation or sheltering of site personnel based on an off-site event. | HU3.2 | Recommendation by local, county or state officials to evacuate or shelter site personnel based on an offsite event | The NEI phrase "Report…for evacuation or sheltering of "has been changed to "Recommendation to evacuate or shelter" for clarification and readability. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU4 | Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.<br>MODE: All | HU4 | Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the (site-specific security shift supervision). | HU4.1 | A security condition that does **not** involve a hostile action as reported by the Security Shift Supervisor<br>**OR**<br>A credible site-specific security threat notification<br>**OR**<br>A validated notification from NRC providing information of an aircraft threat | The NEI Example EALs have been combined in one plant EAL.<br>The "Security Shift Supervisor" is the site-specific security supervision. |
| 2 | A credible site-specific security threat notification. | | | |
| 3 | A validated notification from NRC providing information of an aircraft threat. | | | |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HU5 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE.<br>MODE: All | HU6 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE<br>MODE: All | The NEI acronym "NOUE" has been implemented as "UE" for simplification. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs. | HU6.1 | Other conditions exist which in the judgment of the Emergency Director indicate that **EITHER**:<br>Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant<br>**OR**<br>A security threat to facility protection has been initiated<br>**No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. | Added "EITHER...OR" and indentation to improve readability. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA1 | Natural or destructive phenomena affecting VITAL AREAS.<br>MODE: All | HA1 | Natural or destructive phenomena affecting Vital Areas<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. Seismic event greater than Operating Basis Earthquake (OBE) as indicated by (site specific seismic instrumentation) reading (site specific OBE limit).<br>**AND**<br>b. Earthquake confirmed by **ANY** of the following:<br>• Earthquake felt in plant<br>• National Earthquake Center<br>• Control Room indication of degraded performance of systems required for the safe shutdown of the plant. | HA1.1 | Seismic event > 0.1g as indicated by SMA-3 Strong Motion Accelograph or Alarm B-3/A-1 EMERGENCY SEISMIC HIGH LEVEL<br>**AND**<br>Earthquake confirmed by **any** of the following:<br>• Earthquake felt in plant<br>• National Earthquake Information Center<br>• Control Room indication of degraded performance of systems required for the safe shutdown of the plant | 0.1g is the site-specific Operating Basis Earthquake.<br>The specified indicators are the site-specific methods of indicating an OBE seismic event.<br>Added the word "Information" to National Earthquake Center as that is the proper name. |
| 2 | Tornado striking or high winds greater than (site specific mph) resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems: | HA1.2 | Tornado striking or high winds > 100 mph within Protected Area boundary resulting in **EITHER**:<br>Visible damage to **any** Table H-1 area structure containing safety systems or components<br>**OR**<br>Control Room indication of | 100 mph wind is the site-specific wind speed.<br>The term "**EITHER**" has been added to the plant EAL to highlight the subsequent contingents and clarify with indentation the logic of the EAL.<br>Added "...**any** Table H-1..." to be consistent with the generic NEI 99-01 bases that the EAL is intended to address visible damage to structures "containing functions and systems required for safe shutdown." |

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| | (site specific structure list) | | degraded performance of safety systems | Deleted the word "those" as it is clear from Table H-1 and the context in which safety systems are used that the safety systems of concern are those associated with Table H-1 areas. |
| 3 | Internal flooding in **ANY** of the following areas resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment **OR** control room indication of degraded performance of those safety systems:<br><br>(site specific area list) | HA1.4 | Flooding in **any** Table H-1 area resulting in **EITHER**:<br>An electrical shock hazard that precludes access to operate or monitor safety equipment<br>**OR**<br>Control Room indication of degraded performance of safety systems | The NEI term "Internal" has been deleted from the plant EAL because any source of flooding (whether a source internal or external to the specified area) can have the potential to affect safety-related systems.<br>The term "**EITHER**" has been added to the plant EAL to highlight the subsequent contingents and clarify with indentation the logic of the EAL.<br>Added "…**any** Table H-1 area..." to be consistent with the generic NEI 99-01 bases that the EAL is intended to address flooding in areas "containing functions and systems required for safe shutdown". The Table H-1 areas have been identified as the CNS specific area susceptible to internal flooding.<br>Deleted the word "those" as it is clear from Table H-1 and the context in which safety systems is used that the safety systems of concern are those associated with Table H-1 areas. |
| 4 | Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems:<br>(site specific structure list) | HA1.3 | Main turbine failure-generated projectiles result in **EITHER**:<br>Visible damage to or penetration of **any** Table H-1 area structure containing safety systems or components<br>**OR**<br>Control Room indication of degraded performance of safety systems | Added the word "Main" to avoid confusion with other turbines at CNS (e.g., feedwater turbines, HPCI, RCIC).<br>The term "**EITHER**" has been added to the plant EAL to highlight the subsequent contingents and clarify with indentation the logic of the EAL.<br>Added "…**any** Table H-1 area..." to be consistent with the generic NEI 99-01 bases that the EAL is intended to address visible damage to structures "containing functions and systems required for safe shutdown."<br>Deleted the word "those" as it is clear from Table H-1 and the context in which safety systems are used that the safety systems of concern are those associated with Table H-1 areas. |
| 5 | Vehicle crash resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems | HA1.6 | Vehicle crash resulting in **EITHER**:<br>Visible damage to **any** Table | The term "**EITHER**" has been added to the plant EAL to highlight the subsequent contingents and clarify with indentation the logic of the EAL. |

| | | | | |
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| | or components **OR** control room indication of degraded performance of those safety systems:<br>(site specific structure list) | | H-1 area structure containing safety systems or components<br>**OR**<br>Control Room indication of degraded performance of safety systems | Added "...**any** Table H-1..." to be consistent with the generic NEI 99-01 bases that the EAL is intended to address vehicle crash into structures/equipment "containing functions and systems required for safe shutdown".<br>Deleted the word "those" as it is clear from Table H-1 and the context in which safety systems are used that the safety systems of concern are those associated with Table H-1 areas. |
| 6 | (Site specific occurrences) resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems:<br>(site specific structure list) | HA1.5 | High river/forebay water level > 902' MSL<br>**OR**<br>Low river/forebay level < 865' MSL | Both high and low river/forebay water levels have been identified as other site-specific natural or destructive phenomena appropriate for classification at the Alert level.<br>The NEI phrase "resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems" has been deleted. River/forebay water levels exceeding the threshold values, by definition, lead to visible damage of Table H-1 structures or control room indication of degraded performance of those systems.<br>River level of 902' requires reactor shutdown and represents the maximum possible (10,000 year) flood.<br>865' MSL is the Safe Shutdown low river level. |

| Table H-1 Safe Shutdown Areas |
| --- |
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA2 | FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown.<br>MODE: All | HA2 | Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | FIRE or EXPLOSION resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems:<br>(site specific structure list) | HA2.1 | Fire or explosion resulting in **EITHER**:<br>Visible damage to **any** Table H-1 area containing safety systems or components<br>**OR**<br>Control Room indication of degraded performance of safety systems | The term "**EITHER**" has been added to the plant EAL to highlight the subsequent contingents and clarify with indentation the logic of the EAL.<br>Added "…**any** Table H-1 area..." to be consistent with the generic NEI 99-01 bases that the EAL is intended to address fire or explosion in areas "containing functions and systems required for safe shutdown".<br>Deleted the word "those" as it is clear from Table H-1 and the context in which safety systems are used that the safety systems of concern are those associated with Table H-1 areas. |

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA3 | Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor.<br>MODE: All | HA3 | Access to a vital area is prohibited due to release of toxic, corrosive, asphyxiant or flammable gases which jeopardizes operation of operable equipment required to maintain safe operations or safely shutdown the reactor<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.<br>Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor. | HA3.1 | Access to **any** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shut down the reactor (Note 7)<br>Note 7: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should **not** be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event | Table H-1 provides the site-specific list of vital areas containing equipment necessary for safe shutdown. The table has been added to this EAL for clarification.<br>The NEI phrase "a VITAL" has been replaced with "**any** Table H-1 area" for consistency with other Hazards EALs. The indefinite adjective "a" is ambiguous and could infer that the threshold is met only when one area (but not more than one area) is affected. The term **any** means one or more. It is apparent from the title at the top of Table H-1 that the areas are vital areas.<br>Reference to the NEI note is included in the EAL wording "(Note 7)." Numbering the note facilitates referencing in the EAL matrix. |

| Table H-1 Safe Shutdown Areas |
|---|
| • Reactor Building<br>• Control Building<br>• Service Water Pump Room<br>• Diesel Generator Building<br>• Cable Expansion Room |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA4 | HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat.<br>MODE: All | HA4 | Hostile action within the owner controlled area or airborne attack threat | None |

<table>
<tr><th>NEI Ex. EAL #</th><th>NEI Example EAL Wording</th><th>CNS EAL #</th><th>CNS EAL Wording</th><th>Difference/Deviation Justification</th></tr>
<tr><td>1</td><td>A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLED AREA as reported by the (site specific security shift supervision).</td><td rowspan="2">HA4.1</td><td rowspan="2">A hostile action is occurring or has occurred within the Owner Controlled Area as reported by the Security Shift Supervisor<br>OR<br>A validated notification from NRC of an airliner attack threat within 30 min. of the site</td><td rowspan="2">Example EALs #1 and #2 have been combined into a single EAL for usability.<br>“Security Shift Supervisor” is the site-specific title for security shift supervision.</td></tr>
<tr><td>2</td><td>A validated notification from NRC of an airliner attack threat less within 30 minutes of the site</td></tr>
</table>

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA5 | Control room evacuation has been initiated.<br>MODE: All | HA5 | Control Room evacuation has been initiated<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | (Site-specific procedure) requires control room evacuation. | HA5.1 | Control Room evacuation has been initiated | The IC wording has been utilized since the intent is to classify the Alert based on Control Room evacuation, regardless whether the associated procedure has been entered or executed. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HA6 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.<br>MODE: All | HA6 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels. | HA6.1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:<br>An actual or potential substantial degradation of the level of safety of the plant<br>**OR**<br>A security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action<br>**Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels beyond the site boundary. | Added the words "beyond the site boundary" to clarify where the impact of the release applies for consistency with the SAE definition.<br>Added "EITHER...OR" and indentation to improve readability. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HS2 | Control room evacuation has been initiated and plant control cannot be established.<br>MODE: All | HS5 | Control Room evacuation has been initiated and plant control **cannot** be established<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. Control room evacuation has been initiated.<br>**AND**<br>b. Control of the plant cannot be established within (site specific minutes). | HS5.1 | Control Room evacuation has been initiated<br>**AND**<br>Control of the plant **cannot** be established within 15 min. | 15 minutes is the site-specific period to establish plant control. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HS3 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.<br>MODE: All | HS6 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of Site Area Emergency<br>MODE: All | None. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary. | HS6.1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:<br>An actual or likely major failures of plant functions needed for protection of the public<br>**OR**<br>Hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public<br>**Any** releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary. | EPA PAG values have been added for clarification.<br>Added "EITHER…OR" and indentation to improve readability. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HS4 | HOSTILE ACTION within the PROTECTED AREA.<br>MODE: All | HS4 | Hostile action within the Protected Area<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | A HOSITLE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-security shift supervision). | HS4.1 | A hostile action is occurring or has occurred within the **Protected Area** as reported by the Security Shift Supervisor | The "Security Shift Supervisor" is the title of the site-specific security shift supervision. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HG1 | HOSTILE ACTION resulting in loss of physical control of the facility.<br>MODE: All | HG4 | Hostile action resulting in loss of physical control of the facility<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. | HG4.1 | A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions<br>**OR**<br>A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool | None |
| 2 | A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool. | | | |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| HG2 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.<br>MODE: All | HG6 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of General Emergency<br>MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels off-site for more than the immediate site area. | HG6.1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:<br>Actual or imminent substantial core degradation or melting with potential for loss of containment integrity<br>**OR**<br>Hostile action that results in an actual loss of physical control of the facility<br>Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary. | EPA PAG values have been added for clarification.<br>The NEI phrase "immediate area" has been replaced with "site boundary" as the site-specific term used for immediate plant area.<br>Replaced "...offsite for more than the immediate site area" with "beyond the site boundary" to be consistent with the SAE definition.<br>Added "EITHER...OR" and indentation to improve readability. |

# Category S

# System Malfunction

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU1 | Loss of all Off-site AC power to emergency busses for 15 minutes or longer.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU1 | Loss of **all** offsite AC power to critical buses for 15 minutes or longer<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | The CNS "critical buses" are the NEI "emergency buses." |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>Loss of all off-site AC power to (site specific emergency busses) for 15 minutes or longer. | SU1.1 | Loss of **all** offsite AC power (Table S-3) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3)<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | Table S-3 provides a list of AC power sources.<br>Critical 4160V buses 1F and 1G are the CNS essential buses.<br>Note 3 implements the generic SU1 note though the wording has been made consistent with the similar note utilized in the Cold Shutdown category by deleting the phrase "...has exceeded or,". This phrase is unnecessary because, by definition, if the condition has existed for the allowed duration, the classification threshold has been met. |

| Table S-3 AC Power Sources |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU2 | Inability to reach required shutdown within Technical Specification limits.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU3 | Inability to reach required shutdown within Technical Specification limits<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time. | SU3.1 | Plant is **not** brought to required operating mode within Technical Specifications LCO action statement time | None |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU3 | UNPLANNED loss of safety system annunciation or indication in the control room for 15 minutes or longer.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU4a | Unplanned loss of safety system annunciation or indication in the Control Room for 15 minutes or longer<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>UNPLANNED Loss of greater than approximately 75% of the following for 15 minutes or longer:<br>a. (Site specific control room safety system annunciation)<br>**OR**<br>b. (Site specific control room safety system indication) | SU4.1 | Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition or will likely exceed the applicable time. | Control Room Panels 9-3, 9-4, 9-5, and C display the site-specific annunciators and indicators associated with safety systems.<br>Note 3 implements the generic SU3 note though the wording has been made consistent with the similar note utilized in the Cold Shutdown category by deleting the phrase "...has exceeded or,". This phrase is unnecessary because, by definition, if the condition has existed for the allowed duration, the classification threshold has been met. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU4 | Fuel Clad degradation.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU5 | Fuel clad degradation<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | (Site specific radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits.) | SU5.1 | SJAE monitor > 1.58E+3 mR/hr | Steam Jet Air Ejectors (SJAEs) remove all non-condensable gases from the condensers including air in-leakage and disassociated products originating in the reactor and exhausts them to the offgas holdup volume. A rise in offgas activity could therefore indicate damage to the fuel cladding, a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The Technical Specification allowable limit is ≤ 1 Ci/sec. The SJAE monitor Hi-Hi radiation alarm setpoint is set at 50% of the instantaneous release limit and represents approximately 0.1% fuel cladding damage. The Offgas system isolates at 1.58E+3 mR/hr (SJAE monitor Hi-Hi setpoint) after a 15 minute time delay.<br>The NEI phrase "...radiation...readings indicating fuel clad degradation greater than Technical Specification allowable limits" has been deleted because it is commonly understood that the SJAE radiation monitor HI-HI setpoint is indicative of fuel clad degradation associated with the Technical Specification limit. |
| 2 | (Site specific coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits.) | SU5.2 | Coolant activity ≥ 4.0 μCi/gm dose equivalent I-131 | The NEI phrase "...sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits" has been deleted because it is commonly understood that:<br>• Coolant activity is determined from analysis of coolant samples. Deletion of the phrase in this EAL maintains consistency with the Fuel Clad barrier Loss threshold. NEI does not use the word "sample" in Fuel Clad barrier Loss threshold #1 "Coolant Activity GREATER THAN (site-specific) |

| | | | | |
|---|---|---|---|---|
| | | | | Value."<br>• The specified coolant activity is a Technical Specification limit related to fuel clad degradation as given in Technical Specification LCO 3.4.6.<br>The specified coolant activity is given in Technical Specifications 3.4.6. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU5 | RCS leakage.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU6 | RCS leakage<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Unidentified or pressure boundary leakage greater than 10 gpm. | SU6.1 | Unidentified or pressure boundary leakage > 10 gpm<br>**OR**<br>Identified leakage > 30 gpm (Note 6)<br><br>Note 6: See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage | SU6.1 implements Example EALs #1 and #2. These were combined for improved usability.<br>The NEI identified leakage threshold of 25 gpm has been raised to 30 gpm for consistency with CNS Technical Specifications LCO 3.4.4, which is not exceeded unless unidentified leakage exceeds 5 gpm, total leakage exceeds 30 gpm, or identified leakage exceeds a 2 gpm increase in the past 24 hours. The total leakage limit at many BWRs is 25 gpm and is thus comparable to the NEI threshold for identified leakage. There is no safety analysis that assumes a total leakage at the Technical Specifications limit; rather, it is based on RCS makeup capacity and drywell floor sump capacity. This change is necessary because, unlike most BWRs with a 25 gpm total leakage Technical Specification limit, CNS could be required to declare an Unusual Event before exceeding the Technical Specification total leakage limit of 30 gpm. For example, if identified leakage reached 26 gpm with no unidentified or pressure boundary leakage, the NEI threshold would be exceeded without exceeding the CNS Technical Specification limit. NEI 99-01 Section 3.7 states an Unusual Event represents "Potential degradation of the level of safety of the plant...indicated primarily by exceeding plant technical specification Limiting Condition of Operation (LCO)..." If CNS were to implement the NEI identified leakage threshold, the EAL would not be compatible with the NEI definition of an Unusual Event. **This change is a deviation from NEI 99-01 Revision 5.**<br>The note has been added to remind the EAL-user to review fission product barrier Table F-1 for possible escalation to higher emergency |
| 2 | Identified leakage greater than 25 gpm. | | | |

| | | | | |
|---|---|---|---|---|
| | | | | classifications due to RCS leakage. This addition is the result of EAL feedback received during EAL validation exercises. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU6 | Loss of all On-site or Off-site communications capabilities.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU4b | Loss of **all** onsite or offsite communications capabilities<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | Loss of all of the following on-site communication methods affecting the ability to perform routine operations.<br>(site specific list of communications methods) | SU4.2 | Loss of **all** Table S-2 onsite (internal) communication methods affecting the ability to perform routine operations<br>**OR**<br>Loss of **all** Table S-2 offsite (external) communication methods affecting the ability to perform offsite notifications | SU4.2 implements Example EALs #1 and #2. These were combined for improved usability.<br>The NEI example EALs specify site-specific lists of onsite and offsite communications methods. The CNS EAL lists these methods in Table S-2 because the number of communications methods is too long to include within the text of the EAL.<br>The adjectives "(internal)" and "(external)" have been added to the CNS EAL for clarification. The terms "onsite/offsite" could be interpreted as the location in which the communication originates instead of the location to which communication is directed. |
| 2 | Loss of all of the following off-site communication methods affecting the ability to perform offsite notifications.<br>(site specific list of communications methods) | | | |

| Table S-2 Communications Systems | | |
|---|---|---|
| **System** | **Onsite** (internal) | **Offsite** (external) |
| Station Intercom System "Gaitronics" | X | |
| Site UHF Radio Paging System | X | |
| Alternate Intercom | X | |
| CNS On-Site Cell Phone System | X | X |
| Telephone system (PBX) | X | X |
| Federal Telecommunications System (FTS 2001) | | X |
| Microwave Telephone Network | | X |
| Local Telephones (C.O. Lines) | | X |
| CNS State Notification Telephones | | X |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SU8 | Inadvertent criticality.<br>MODE: Hot Standby, Hot Shutdown | SU2 | Inadvertent criticality<br>MODE: 3 - Hot Shutdown | Deleted Hot Standby mode per NEI 99-01 Revision 5 Section 3.14 and Technical Specifications Table 1.1-1. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | UNPLANNED sustained positive period observed on nuclear instrumentation. [*BWR*] | SU2.1 | An unplanned sustained positive period observed on nuclear instrumentation | Added the preposition "An" to align wording with the NEI 99-01 CU8 example EAL wording and CNS EAL CU5.1. |
| 2 | UNPLANNED sustained positive startup rate observed on nuclear instrumentation. [*PWR*] | N/A | N/A | NEI Example EAL #2 has not been implemented because it applies only to PWR plants. CNS is a BWR. BWRs are not equipped with startup rate meters. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SA2 | Automatic Scram (Trip) fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor.<br>MODE: Power Operation, Startup | SA2 | Automatic scram fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor.<br>MODE: 1 - Power Operation, 2 - Startup | Deleted "(Trip)" as the term is applicable to PWRs. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. An automatic scram (trip) failed to shutdown the reactor.<br>**AND**<br>b. Manual actions taken at the reactor control console successfully shutdown the reactor as indicated by (site specific indications of plant shutdown). | SA2.1 | An automatic scram failed to shut down the reactor<br>**AND**<br>Manual actions taken at the reactor control console successfully shut down the reactor as indicated by reactor power < 3% | Deleted "(trip)" as the term is applicable to PWRs.<br>The site-specific indication of plant shutdown is reactor power less than the APRM downscale trip setpoint (3%). This is a threshold commonly used in CNS EOPs to differentiate between ATWS and shutdown conditions. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SA4 | UNPLANNED Loss of safety system annunciation or indication in the control room with EITHER (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory indicators unavailable.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SA4 | Unplanned loss of safety system annunciation or indication in the Control Room with **EITHER** (1) a significant transient in progress, or (2) compensatory indicators unavailable<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>a. UNPLANNED loss of greater than approximately 75% of the following for 15 minutes or longer:<br>• (Site specific control room safety system annunciation)<br>**OR**<br>• (Site specific control room safety system indication)<br>b. **EITHER** of the following:<br>• A SIGNIFICANT | SA4.1 | Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)<br>**AND EITHER**:<br>**Any** significant transient is in progress, Table S-1<br>**OR**<br>Compensatory indications are unavailable<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that | Control Room Panels 9-3, 9-4, 9-5, and C display the site-specific annunciators and indicators associated with safety systems.<br>Table S-1 provides the list of events that constitute a "significant transient" as specified in the NEI Section 5.4 definition of significant transient.<br>Note 3 implements the generic SA4 note though the wording has been made consistent with the similar note utilized in the Cold Shutdown category by deleting the phrase "...has exceeded or,". This phrase is unnecessary because, by definition, if the condition has existed for the allowed duration, the classification threshold has been met. |

| | | | | | |
|---|---|---|---|---|---|
| | TRANSIENT is in progress.<br>• Compensatory indications are unavailable. | | the condition will likely exceed the applicable time. | |

| Table S-1 Significant Transients |
|---|
| Reactor scram |
| Runback > 25% thermal power |
| Electrical load rejection > 25% full electrical load |
| ECCS injection |
| Thermal power oscillations > 10% |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SA5 | AC power capability to emergency busses reduced to a single power source for 15 minutes or longer such that any additional single failure would result in station blackout.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SA1 | AC power capability to critical buses reduced to a single power source for 15 minutes or longer such that **any** additional single failure would result in loss of **all** AC power to critical buses<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | The term "station blackout" was replaced with "loss of all AC power to critical buses" as this clearly describes the intended condition leading to the SAE threshold in SS1.1. Station Blackout is not an operationally defined term for loss of all AC to critical buses.<br>The CNS "critical buses" are the NEI "emergency buses." |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>a. AC power capability to (site-specific emergency busses) reduced to a single power source for 15 minutes or longer.<br>**AND**<br>b. Any additional single power source failure will result in station blackout. | SA1.1 | AC power capability to critical 4160V buses 1F and 1G reduced to a single power source (Table S-3) for ≥ 15 min. such that **any** additional single failure would result in loss of all AC power to critical buses (Note 3)<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | The NEI example EAL contingent "...any additional single failure will result in station blackout" has been simplified consistent with the IC wording.<br>The term "station blackout" was replaced with "loss of all AC power to critical buses" as this clearly describes the intended condition leading to the Alert threshold. Station Blackout is not an operationally defined term for loss of all AC to critical buses.<br>The NEI phrase "...AND...will..." has been changed to "...such that...would..." to clarify the consequences if the single remaining power source were to be lost.<br>Table S-3 provides a list of AC power sources.<br>Critical 4160V buses 1F and 1G are the CNS essential buses.<br>Note 3 implements the generic SA5 note though the wording has been made consistent with the similar note utilized in the Cold Shutdown category by deleting the phrase "...has exceeded or,". This phrase is unnecessary because, by definition, if the condition has existed for the allowed duration, the classification threshold has been met. |

| Table S-3 AC Power Sources |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SS1 | Loss of all Off-site and all On-Site AC power to emergency busses for 15 minutes or longer.<br><br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SS1a | Loss of **all** offsite and **all** onsite AC power to critical buses for 15 minutes or longer<br><br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | The CNS "critical buses" are the NEI "emergency buses." |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br><br>1. Loss of all Off-Site and all On-Site AC power to (site specific emergency busses) for 15 minutes or longer. | SS1.1 | Loss of **all** offsite and **all** onsite AC power (Table S-3) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3)<br><br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | Table S-3 provides a list of AC power sources.<br><br>Critical 4160V buses 1F and 1G are the CNS essential buses.<br><br>Note 3 implements the generic SS1 note though the wording has been made consistent with the similar note utilized in the Cold Shutdown category by deleting the phrase "...has exceeded or,". This phrase is unnecessary because, by definition, if the condition has existed for the allowed duration, the classification threshold has been met. |

| Table S-3 AC Power Sources |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SS2 | Automatic Scram (Trip) fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting down the reactor.<br>MODE: Power Operation, Startup | SS2 | Automatic scram fails to shut down the reactor and manual actions taken from the reactor control console are **not** successful in shutting down the reactor<br>MODE: 1 - Power Operation, 2 - Startup | Deleted "(Trip)" as the term is PWR specific. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. An automatic scram (trip) failed to shutdown the reactor.<br>**AND**<br>b. Manual actions taken at the reactor control console do not shutdown the reactor as indicated by (site specific indications of reactor not shutdown). | SS2.1 | An automatic scram failed to shut down the reactor<br>**AND**<br>Manual actions taken at the reactor control console (Note 5) do **not** shut down the reactor as indicated by reactor power ≥ 3%<br>Note 5: Manual scram methods for EAL SS2.1 are the following:<br>• Reactor Scram push buttons<br>• Reactor Mode switch in SHUTDOWN<br>• Manual or auto actuation of ARI | Deleted "(Trip)" as the term is PWR specific.<br>Note 5 has been added to clarify the acceptable methods for initiating a manual scram. Auto actuation of ARI is included in the list of methods because the operator, by procedure, always ensures actuation of ARI has occurred if the ARI actuation setpoints are exceeded. This means action to depress the ARI pushbuttons is taken if the automatic ARI actuation setpoints are exceeded but failed to actuate. If ARI properly actuates automatically, the ARI pushbuttons are not depressed.<br>The site-specific indication of plant shutdown is reactor power less than the APRM downscale trip setpoint (3%). This is a threshold commonly used in CNS EOPs to differentiate between ATWS and shutdown conditions. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SS3 | Loss of all vital DC power for 15 minutes or longer.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SS1b | Loss of **all** vital DC power for 15 minutes or longer<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>1. Less than (site specific bus voltage indication) on all (site specific Vital DC busses) for 15 minutes or longer. | SS1.2 | < 105 VDC bus voltage indications on **all** vital 125 VDC buses (1A and 1B) for ≥ 15 min. (Note 3)<br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. | 125 VDC SWGR busses 1A and 1B are the divisional site-specific vital DC buses. "< 105 VDC" is the site-specific bus voltage.<br>Note 3 implements the generic SS1 note though the wording has been made consistent with the similar note utilized in the Cold Shutdown category by deleting the phrase "...has exceeded or,". This phrase is unnecessary because, by definition, if the condition has existed for the allowed duration, the classification threshold has been met. |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SS6 | Inability to Monitor a SIGNIFICANT TRANSIENT in Progress<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SS4 | Inability to monitor a significant transient in progress<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.<br>a. Loss of greater than approximately 75% of the following for 15 minutes or longer:<br>• (Site specific control room safety system annunciation)<br>**OR**<br>• (Site specific control room safety system indication)<br>**AND**<br>b. A SIGNIFICANT TRANSIENT is in progress.<br>**AND**<br>c. Compensatory indications | SS4.1 | Loss of > approximately 75% of the annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)<br>**AND**<br>**Any** significant transient is in progress, Table S-1<br>**AND**<br>Compensatory indications are unavailable<br><br>**Note 3:** The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the | Control Room Panels 9-3, 9-4, 9-5, and C display the site-specific annunciators and indicators associated with safety systems.<br>Note 3 implements the generic SS6 note though the wording has been made consistent with the similar note utilized in the Cold Shutdown category by deleting the phrase "...has exceeded or,". This phrase is unnecessary because, by definition, if the condition has existed for the allowed duration, the classification threshold has been met.<br>The NEI phrase "SIGNIFICANT TRANSIENT in progress" has been replaced with "**Any** significant transient is in progress, Table S-1" to improve specificity and clarity, and avoid ambiguity. The term "any" means "one or more." Table S-1 provides the list of events that constitute a "significant transient" as specified in the NEI Section 5.4 definition of significant transient. |

| | | | | |
|---|---|---|---|---|
| | are unavailable. | | applicable time. | |

| Table S-1 Significant Transients |
|---|
| Reactor scram<br>Runback > 25% thermal power<br>Electrical load rejection > 25% full electrical load<br>ECCS injection<br>Thermal power oscillations > 10% |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SG1 | Prolonged loss of all Off-site and all On-Site AC power to emergency busses.<br>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SG1 | Prolonged loss of **all** offsite and **all** onsite AC power to critical buses<br>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown | The CNS "critical buses" are the NEI "emergency buses." |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. Loss of all off-site and all on-site AC power to (site specific emergency busses).<br>**AND**<br>b. **EITHER** of the following:<br>• Restoration of at least one emergency bus in less than (site specific hours) is not likely.<br>• (Site specific indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.) | SG1.1 | Loss of **all** offsite and **all** onsite AC power (Table S-3) to critical 4160V buses 1F and 1G<br>**AND EITHER**:<br>Restoration of at least one critical bus in < 4 hours is **not** likely<br>**OR**<br>RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined | Table S-3 provides a list of AC power sources.<br>Critical 4160V buses 1F and 1G are the CNS emergency buses.<br>The NEI phrase "...of the following: ..." has been deleted. It is evident from the subsequent paragraphs and indentation applied to the CNS EAL that they follow the previous paragraph.<br>Four hours are the "(site-specific)" hours for station blackout coping. The four-hour interval to restore AC power is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155.<br>The NEI phrase "...(Site-Specific) Indication of continuing degradation of core cooling based on Fission Product Barrier monitoring" has been replaced with "RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined" for clarification and agreement with the Fuel Clad barrier and RCS barrier potential loss thresholds. This threshold represents the NEI conditions consistent with the corresponding Fuel Clad barrier Potential Loss and RCS barrier Loss thresholds for RPV water level in Table F-1. |

| Table S-3 AC Power Sources |
|---|
| **Offsite** |
| • Startup Station Service Transformer<br>• Emergency Station Service Transformer<br>• Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer |
| **Onsite** |
| • DG-1<br>• DG-2 |

| NEI IC# | NEI IC Wording | CNS IC#(s) | CNS IC Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| SG2 | Automatic Scram (Trip) and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists.<br>MODE: Power Operation, Startup | SG2 | Automatic scram and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists<br>MODE: 1 - Power Operation, 2 - Startup | Deleted "(Trip)" as the term is PWR specific. |

| NEI Ex. EAL # | NEI Example EAL Wording | CNS EAL # | CNS EAL Wording | Difference/Deviation Justification |
|---|---|---|---|---|
| 1 | a. An automatic scram (trip) failed to shutdown the reactor.<br>**AND**<br>b. All manual actions do not shutdown the reactor as indicated by (site specific indications of reactor not shutdown).<br>**AND**<br>c. **EITHER** of the following exist or have occurred due to continued power generation:<br>• (Site specific indication that core cooling is extremely challenged.)<br>• (Site specific indication that heat removal is extremely challenged.) | SG2.1 | Automatic and **all** manual scrams were **not** successful<br>**AND**<br>Reactor power ≥ 3%<br>**AND EITHER** of the following exist or have occurred due to continued power generation:<br>RPV level **cannot** be restored and maintained > -183 in. or **cannot** be determined<br>**OR**<br>Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7) | The phrase "Reactor power ≥ 3%" has been added to clarify the conditions under which an automatic and manual reactor scram would be determined to be unsuccessful. Following a reactor scram, nuclear flux is expected to drop promptly below the power range and then decay off at a fixed rate to a normal shutdown level. Core heat production after several minutes should be limited to that from radioactive decay of fission products, rather than from the fission process itself. The APRM downscale trip setpoint of 3% is a minimum reading on the power range scale that indicates power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. The basis for the NEI example EAL states that this condition occurs if "...the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed." The EOP-1A reactor power entry condition, therefore, satisfies the NEI requirement.<br>The NEI example EAL specifies "...Indication(s) exists that the core cooling is extremely challenged...OR...Indication(s) exists that heat removal is extremely challenged." To clarify the intent of the EAL, the CNS EAL includes the specific RPV water level (-183 in. or cannot be determined) that represents an extreme challenge to core cooling and the Heat Capacity Temperature Limit (HCTL) exceeded that |

| | | | | |
|---|---|---|---|---|
| | | | | represents an extreme challenge to heat removal capabilities.<br><br>-183 in. is the Minimum Steam Cooling RPV Water Level (MSCRWL). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence.<br><br>The Heat Capacity Temperature Limit (HCTL) is the highest torus temperature from which Emergency RPV Depressurization will not raise torus pressure above the Primary Containment Pressure Limit (PCPL), while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent. The HCTL is a function of RPV pressure and torus level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. Plant parameters in excess of the HCTL could be a precursor of primary containment failure.<br><br>Added the parenthetical reference to EOP/SAG Graph 7 to assist the end-user in locating the HCTL curves. |

**Attachment 3**

**Emergency Action Level Charts**
**for**
**Modes 1 through 5 and Defuel Mode**

**Cooper Nuclear Station, Docket No. 50-298, DPR-46**

# GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

## A — Abnorm. Rad Release / Rad Effluent

### 1 — Offsite Rad Conditions

**AG1.1** 1 2 3 4 5 DEF
Any gaseous monitor reading > Table A-1 column "GE" for ≥ 15 min. (Note 1)

**AS1.1** 1 2 3 4 5 DEF
Any gaseous monitor reading > Table A-1 column "SAE" for ≥ 15 min. (Note 1)

**AA1.1** 1 2 3 4 5 DEF
Any gaseous monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

**AU1.1** 1 2 3 4 5 DEF
Any gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

**AG1.2** 1 2 3 4 5 DEF
Dose assessment using actual meteorology indicates doses > 1 Rem TEDE or > 5 Rem thyroid CDE at or beyond the site boundary

**AS1.2** 1 2 3 4 5 DEF
Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem thyroid CDE at or beyond the site boundary

**AA1.2** 1 2 3 4 5 DEF
Any liquid effluent monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)
AND
Effluent discharge is not isolated

**AU1.2** 1 2 3 4 5 DEF
Any liquid effluent monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)
AND
Effluent discharge is not isolated

**AG1.3** 1 2 3 4 5 DEF
Field survey results indicate closed window dose rates > 1 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary
OR
Analyses of field survey samples indicate thyroid CDE > 5 Rem for 1 hr of inhalation at or beyond the site boundary

**AS1.3** 1 2 3 4 5 DEF
Field survey indicates closed window dose rate > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary
OR
Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary

**AA1.3** 1 2 3 4 5 DEF
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM limits for ≥ 15 min. (Note 2)

**AU1.3** 1 2 3 4 5 DEF
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODAM limits for ≥ 60 min. (Note 2)

### 2 — Onsite Rad Conditions & Spent Fuel Pool Events

**Table A-1 Effluent Monitor Classification Thresholds**

| | Monitor | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
|---|---|---|---|---|---|
| GASEOUS | ERP | 3.50E+08 μCi/sec | 3.50E+07 μCi/sec | 2.80E+06 μCi/sec | 2.24E+05 μCi/sec |
| GASEOUS | Rx Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.45E+05 μCi/sec | 8.48E+04 μCi/sec |
| GASEOUS | Turb Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.62E+05 μCi/sec | 9.02E+04 μCi/sec |
| GASEOUS | RW / ARW Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.64E+05 μCi/sec | 9.08E+04 μCi/sec |
| LIQUID | Rad Waste Effluent | — | — | 200 x calculated alarm values* | 2 x calculated alarm values* |
| LIQUID | Service Water Effluent | — | — | 4.80E-04 μCi/cc | 4.80E-06 μCi/cc |

* with effluent discharge not isolated

**AA2.1** 1 2 3 4 5 DEF
Damage to irradiated fuel OR loss of water level (uncovering irradiated fuel outside the RPV) that causes EITHER of the following:
RMA-RA-1 Fuel Pool Area Rad reading > 50 R/hr
OR
RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm

**AU2.1** 1 2 3 4 5 DEF
Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by any of the following:
- LI-86 (calibrated to 1001' elev.)
- Spent fuel pool low level alarm
- Visual observation

AND
Area radiation monitor reading rise on RMA RA 1 or RMA-RA-2 (or by survey)

**AA2.2** 1 2 3 4 5 DEF
A water level drop in the reactor cavity or spent fuel pool that will result in irradiated fuel becoming uncovered

**AA2.3** 1 2 3 4 5 DEF
Dose rates > 15 mRem/hr in EITHER of the following areas requiring continuous occupancy to maintain plant safety functions:
Main Control Room (RM-RA-20)
OR
CAS

**AU2.2** 1 2 3 4 5 DEF
Unplanned area radiation monitor reading or survey results rise by a factor of 1,000 over normal levels*

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

## H — Hazards

### 1 — Natural or Destructive Phenomena

**Table H-1 Safe Shutdown Areas**
- Reactor Building
- Control Building
- Service Water Pump Room
- Diesel Generator Building
- Cable Expansion Room

**HA1.1** 1 2 3 4 5 DEF
Seismic event > 0.1g as indicated by SMA-3 Strong Motion Accelograph or Alarm B-3/A-1 EMERGENCY SEISMIC HIGH LEVEL
AND
Earthquake confirmed by any of the following:
- Earthquake felt in plant
- National Earthquake Information Center
- Control Room indication of degraded performance of systems required for the safe shutdown of the plant

**HU1.1** 1 2 3 4 5 DEF
Seismic event identified by any two of the following:
- SMA-3 Strong Motion Accelograph actuated or Alarm B-3/B-1 SEISMIC EVENT
- Earthquake felt in plant
- National Earthquake Information Center

**HA1.2** 1 2 3 4 5 DEF
Tornado striking or high winds > 100 mph within Protected Area boundary resulting in EITHER:
Visible damage to any Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

**HU1.2** 1 2 3 4 5 DEF
Tornado striking within Protected Area boundary
OR
Sustained high winds > 100 mph

**HA1.3** 1 2 3 4 5 DEF
Main turbine failure-generated missiles result in EITHER:
Visible damage to or penetration of any Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

**HU1.3** 1 2 3 4 5 DEF
Main turbine failure resulting in casing penetration or damage to turbine or generator seals

**HA1.4** 1 2 3 4 5 DEF
Flooding in any Table H-1 area resulting in EITHER:
An electrical shock hazard that precludes access to operate or monitor safety equipment
OR
Control Room indication of degraded performance of safety systems

**HU1.4** 1 2 3 4 5 DEF
Flooding in any Table H-1 area that has the potential to affect safety-related equipment needed for the current operating mode

**HA1.5** 1 2 3 4 5 DEF
High river/forebay water level > 902' MSL
OR
Low river/forebay level < 865' MSL

**HU1.5** 1 2 3 4 5 DEF
High river/forebay water level > 899' MSL
OR
Low river level/forebay < 870' MSL

**HA1.6** 1 2 3 4 5 DEF
Vehicle crash resulting in EITHER:
Visible damage to any Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

**Notes**
1. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
   If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values.
   (See EAL AS1.2/AG1.2.) Do not delay declaration awaiting dose assessment results.
2. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
3. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
4. Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.
5. Manual scram methods for EAL SS2.1 are the following:
   - Reactor Scram push buttons
   - Reactor Mode switch in SHUTDOWN
   - Manual or auto actuation of ARI
6. See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage.
7. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

### 2 — Fire or Explosion

**HA2.1** 1 2 3 4 5 DEF
Fire or explosion resulting in EITHER:
Visible damage to any Table H-1 area containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

**HU2.1** 1 2 3 4 5 DEF
Fire in any Table H-1 area not extinguished within 15 min. of Control Room notification or receipt of a valid Control Room alarm due to fire (Note 3)

**HU2.2** 1 2 3 4 5 DEF
Explosion within the Protected Area

### 3 — Hazardous Gas

**HA3.1** 1 2 3 4 5 DEF
Access to any Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shut down the reactor (Note 7)

**HU3.1** 1 2 3 4 5 DEF
Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect normal plant operations

**HU3.2** 1 2 3 4 5 DEF
Recommendation by local, county or state officials to evacuate or shelter site personnel based on an offsite event

### 4 — Security

**HG4.1** 1 2 3 4 5 DEF
A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions
OR
A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool

**HS4.1** 1 2 3 4 5 DEF
A hostile action is occurring or has occurred within the Protected Area as reported by the Security Shift Supervisor

**HA4.1** 1 2 3 4 5 DEF
A hostile action is occurring or has occurred within the Owner Controlled Area as reported by the Security Shift Supervisor
OR
A validated notification from NRC of an airliner attack threat within 30 min. of the site

**HU4.1** 1 2 3 4 5 DEF
A security condition that does not involve a hostile action as reported by the Security Shift Supervisor
OR
A credible site-specific security threat notification
OR
A validated notification from NRC providing information of an aircraft threat

### 5 — Control Room Evacuation

General Emergency: None

**HS5.1** 1 2 3 4 5 DEF
Control Room evacuation has been initiated
AND
Control of the plant cannot be established within 15 min.

**HA5.1** 1 2 3 4 5 DEF
Control Room evacuation has been initiated

Unusual Event: None

### 6 — Judgment

**HG6.1** 1 2 3 4 5 DEF
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve EITHER:
Actual or imminent substantial core degradation or melting with potential for loss of containment integrity
OR
Hostile action that results in an actual loss of physical control of the facility

Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary

**HS6.1** 1 2 3 4 5 DEF
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve EITHER:
An actual or likely major failures of plant functions needed for protection of the public
OR
Hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public

Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary

**HA6.1** 1 2 3 4 5 DEF
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve EITHER:
An actual or potential substantial degradation of the level of safety of the plant
OR
A security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action

Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels beyond the site boundary

**HU6.1** 1 2 3 4 5 DEF
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate EITHER:
Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant
OR
A security threat to facility protection has been initiated

No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs

## E — ISFSI

General Emergency: None | Site Area Emergency: None | Alert: None

**EU1.1** N/A
Damage to a loaded cask confinement boundary

Modes: 1 Power Operation | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | DEF Defueled

Cooper Nuclear Station
Emergency Action Level Matrix
EPIP 5.7.1 Attachment 1, Rev. 37

Approved: ____________ Manager, Emergency Preparedness ____ Date

# GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

## S — System Malfunct.

### 1 — Loss of Power

**SG1.1** 1 2 3
Loss of all offsite and all onsite AC power (Table S-3) to critical 4160V buses 1F and 1G
AND EITHER:
Restoration of at least one critical bus in < 4 hours is not likely
OR
RPV level cannot be restored and maintained > -158 in. or cannot be determined

**SS1.1** 1 2 3
Loss of all offsite and all onsite AC power (Table S-3) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3)

**SS1.2** 1 2 3
< 105 VDC bus voltage indications on all vital 125 VDC buses (1A and 1B) for ≥ 15 min. (Note 3)

**SA1.1** 1 2 3
AC power capability to critical 4160V buses 1F and 1G reduced to a single power source (Table S-3) for ≥ 15 min. such that any additional single failure would result in loss of all AC power to critical buses (Note 3)

**SU1.1** 1 2 3
Loss of all offsite AC power (Table S-3) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3)

### 2 — ATWS Criticality

**SG2.1** 1 2
Automatic and all manual scrams were not successful
AND
Reactor power ≥ 3%
AND EITHER of the following exist or have occurred due to continued power generation:
RPV level cannot be restored and maintained > -183 in. or cannot be determined
OR
Average torus water temperature and RPV pressure cannot be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7)

**SS2.1** 1 2
An automatic scram failed to shut down the reactor
AND
Manual actions taken at the reactor control console (Note 5) do not shut down the reactor as indicated by reactor power ≥ 3%

**SA2.1** 1 2
An automatic scram failed to shut down the reactor
AND
Manual actions taken at the reactor control console successfully shut down the reactor as indicated by reactor power < 3%

**SU2.1** 3
An unplanned sustained positive period observed on nuclear instrumentation

### 3 — Inability to Reach Shutdown Conditions

General Emergency: None | Site Area Emergency: None | Alert: None

**SU3.1** 1 2 3
Plant is not brought to required operating mode within Technical Specifications LCO action statement time

### 4 — Inst. / Comm.

General Emergency: None

**SS4.1** 1 2 3
Loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)
AND
Any significant transient is in progress, Table S-1
AND
Compensatory indications are unavailable

**SA4.1** 1 2 3
Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)
AND EITHER:
Any significant transient is in progress, Table S-1
OR
Compensatory indications are unavailable

**SU4.1** 1 2 3
Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)

**SU4.2** 1 2 3
Loss of all Table S-2 onsite (internal) communication methods affecting the ability to perform routine operations
OR
Loss of all Table S-2 offsite (external) communication methods affecting the ability to perform offsite notifications

### 5 — Fuel Clad Degradation

General Emergency: None | Site Area Emergency: None | Alert: None

**SU5.1** 1 2 3
SJAE monitor > 1.58E+3 mR/hr

**SU5.2** 1 2 3
Coolant activity ≥ 4.0 μCi/gm dose equivalent I-131

### 6 — RCS Leakage

General Emergency: None | Site Area Emergency: None

**SU6.1** 1 2 3
Unidentified or pressure boundary leakage > 10 gpm
OR
Identified leakage > 30 gpm
(Note 6)

## F — Fission Product Barriers

**FG1.1** 1 2 3
Loss of any two barriers
AND
Loss or potential loss of third barrier (Table F-1)

**FS1.1** 1 2 3
Loss or potential loss of any two barriers (Table F-1)

**FA1.1** 1 2 3
Any loss or any potential loss of either Fuel Clad or RCS (Table F-1)

**FU1.1** 1 2 3
Any loss or any potential loss of Primary Containment (Table F-1)

## Table F-1 Fission Product Barrier Matrix

| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Primary Containment Barrier | |
|---|---|---|---|---|---|---|
| | Loss | Potential Loss | Loss | Potential Loss | Loss | Potential Loss |
| A. RPV Level | 1. PC flooding is required due to any of the following:<br>• RPV water level cannot be restored and maintained > -183 in.<br>• RPV water level cannot be restored and maintained ≥ -209 in. and no core spray subsystem flow can be restored and maintained ≥ 4,750 gpm<br>• RPV water level cannot be determined and core damage is occurring | 5. RPV level cannot be restored and maintained > -158 in. or cannot be determined | 7. RPV level cannot be restored and maintained > -158 in. or cannot be determined | None | None | 22. PC Flooding required |
| B. PC Pressure / Temperature | None | None | 8. PC pressure > 1.84 psig due to RCS leakage | None | 16. PC pressure rise followed by a rapid unexplained drop in PC pressure<br>17. PC pressure response not consistent with LOCA conditions | 23. PC pressure > 56 psig and rising<br>24. Deflagration concentrations exist inside PC<br>≥ 6% $H_2$ in drywell or torus (or cannot be determined)<br>AND<br>≥ 5% $O_2$ in drywell or torus (or cannot be determined)<br>25. Average torus water temperature and RPV pressure cannot be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7) |
| C. Isolation | None | None | 9. Release pathway exists outside primary containment resulting from isolation failure in any of the following (excluding normal process system flowpaths from an unisolable system):<br>• Main steam line<br>• HPCI steam line<br>• RCIC steam line<br>• RWCU<br>• Feedwater | 13. RCS leakage > 50 gpm inside the drywell<br>14. Unisolable primary system discharge outside primary containment as indicated by exceeding any secondary containment Maximum Normal Operating temperature or radiation value (EOP-5A Tables 9 and 10) | 18. Failure of all valves in any one line to close<br>AND<br>Direct downstream pathway to the environment exists after PC isolation signal<br>19. Intentional PC venting per EOPs<br>20. Unisolable primary system discharge outside PC as indicated by exceeding any secondary containment Maximum Safe Operating temperature or radiation value (EOP-5A Tables 9 and 10) | None |
| D. ERD | None | None | 10. Emergency RPV depressurization is required | None | None | None |
| E. Rad | 2. Drywell radiation monitor (RMA-RM-40A/B) > 2.50E+03 Rem/hr<br>3. Primary coolant activity > 300 μCi/gm dose equivalent I-131 | None | 11. Drywell radiation monitor (RMA-RM-40A/B) > 2.40E+02 Rem/hr | None | None | 26. Drywell radiation monitor (RMA-RM-40A/B) > 5.00E+04 Rem/hr |
| F. Judgment | 4. Any condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier | 6. Any condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier | 12. Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier | 15. Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier | 21. Any condition in the opinion of the Emergency Director that indicates loss of the PC barrier | 27. Any condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier |

**Table S-1 Significant Transients**
- Reactor scram
- Runback > 25% thermal power
- Electrical load rejection > 25% full electrical load
- ECCS injection
- Thermal power oscillations > 10%

**Table S-2 Communications Systems**

| System | Onsite (internal) | Offsite (external) |
|---|---|---|
| Station Intercom System "Gaitronics" | X | |
| Site UHF Radio Paging System | X | |
| Alternate Intercom | X | |
| CNS On-Site Cell Phone System | X | X |
| Telephone system (PBX) | X | X |
| Federal Telecommunications System (FTS 2001) | | X |
| Microwave Telephone Network | | X |
| Local Telephones (C.O. Lines) | | X |
| CNS State Notification Telephones | | X |

**Table S-3 AC Power Sources**

Offsite
- Startup Station Service Transformer
- Emergency Station Service Transformer
- Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer

Onsite
- DG-1
- DG-2

**Notes**
1. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
   If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values.
   (See EAL AS1.2/AG1.2.) Do not delay declaration awaiting dose assessment results.
2. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
3. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
4. Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.
5. Manual scram methods for EAL SS2.1 are the following:
   - Reactor Scram push buttons
   - Reactor Mode switch in SHUTDOWN
   - Manual or auto actuation of ARI
6. See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage.
7. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

**EAL Identifier**

XXX.X
- Category (A, H, S, F, C, E)
- Emergency classification (G, S, A, U)
- Subcategory number (1 if no subcategory)
- Sequential number within subcategory/classification

# MODE 1, 2 or 3

# General Emergency | Site Area Emergency | Alert | Unusual Event

## A — Abnorm. Rad Release / Rad Effluent

### 1 — Offsite Rad Conditions

**AG1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Any gaseous monitor reading > Table A-1 column "GE" for ≥ 15 min. (Note 1)

**AG1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Dose assessment using actual meteorology indicates doses > 1 Rem TEDE or > 5 Rem thyroid CDE at or beyond the site boundary

**AG1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Field survey results indicate closed window dose rates > 1 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary
OR
Analyses of field survey samples indicate thyroid CDE > 5 Rem for 1 hr of inhalation at or beyond the site boundary

**AS1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Any gaseous monitor reading > Table A-1 column "SAE" for ≥ 15 min. (Note 1)

**AS1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem thyroid CDE at or beyond the site boundary

**AS1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Field survey indicates closed window dose rate > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary
OR
Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary

**AA1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Any gaseous monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

**AA1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Any liquid effluent monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)
AND
Effluent discharge is not isolated

**AA1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM limits for ≥ 15 min. (Note 2)

**AU1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Any gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

**AU1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Any liquid effluent monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)
AND
Effluent discharge is not isolated

**AU1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODAM limits for ≥ 60 min. (Note 2)

### 2 — Onsite Rad Conditions & Spent Fuel Pool Events

**Table A-1 Effluent Monitor Classification Thresholds**

| | Monitor | GE for ≥ 15 min. | SAE for ≥ 15 min. | ALERT for ≥ 15 min. | UE for ≥ 60 min. |
|---|---|---|---|---|---|
| GASEOUS | ERP | 3.50E+08 μCi/sec | 3.50E+07 μCi/sec | 2.80E+06 μCi/sec | 2.24E+05 μCi/sec |
| | Rx Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.45E+05 μCi/sec | 8.48E+04 μCi/sec |
| | Turb Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.62E+05 μCi/sec | 9.02E+04 μCi/sec |
| | RW / ARW Bldg Vent | 3.50E+07 μCi/sec | 3.50E+06 μCi/sec | 5.64E+05 μCi/sec | 9.08E+04 μCi/sec |
| LIQUID | Rad Waste Effluent | — | — | 200 x calculated alarm values* | 2 x calculated alarm values* |
| | Service Water Effluent | — | — | 4.80E-04 μCi/cc | 4.80E-06 μCi/cc |

* with effluent discharge not isolated

**AA2.1** [1 | 2 | 3 | 4 | 5 | DEF]
Damage to irradiated fuel OR loss of water level (uncovering irradiated fuel outside the RPV) that causes EITHER of the following:
RMA-RA-1 Fuel Pool Area Rad reading > 50 R/hr
OR
RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm

**AA2.2** [1 | 2 | 3 | 4 | 5 | DEF]
A water level drop in the reactor cavity or spent fuel pool that will result in irradiated fuel becoming uncovered

**AA2.3** [1 | 2 | 3 | 4 | 5 | DEF]
Dose rates > 15 mRem/hr in EITHER of the following areas requiring continuous occupancy to maintain plant safety functions:
Main Control Room (RM-RA-20)
OR
CAS

**AU2.1** [1 | 2 | 3 | 4 | 5 | DEF]
Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by any of the following:
- LI-86 (calibrated to 1001' elev.)
- Spent fuel pool low level alarm
- Visual observation

AND
Area radiation monitor reading rise on RMA RA 1 or RMA-RA-2 (or by survey)

**AU2.2** [1 | 2 | 3 | 4 | 5 | DEF]
Unplanned area radiation monitor reading or survey results rise by a factor of 1,000 over normal levels*

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

## H — Hazards

### 1 — Natural or Destructive Phenomena

**Table H-1 Safe Shutdown Areas**
- Reactor Building
- Control Building
- Service Water Pump Room
- Diesel Generator Building
- Cable Expansion Room

**HA1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Seismic event > 0.1g as indicated by SMA-3 Strong Motion Accelograph or Alarm B-3/A-1 EMERGENCY SEISMIC HIGH LEVEL
AND
Earthquake confirmed by any of the following:
- Earthquake felt in plant
- National Earthquake Information Center
- Control Room indication of degraded performance of systems required for the safe shutdown of the plant

**HA1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Tornado striking or high winds > 100 mph within Protected Area boundary resulting in EITHER:
Visible damage to any Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

**HA1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Main turbine failure-generated missiles result in EITHER:
Visible damage to or penetration of any Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

**HA1.4** [1 | 2 | 3 | 4 | 5 | DEF]
Flooding in any Table H-1 area resulting in EITHER:
An electrical shock hazard that precludes access to operate or monitor safety equipment
OR
Control Room indication of degraded performance of safety systems

**HA1.5** [1 | 2 | 3 | 4 | 5 | DEF]
High river/forebay water level > 902' MSL
OR
Low river/forebay level < 865' MSL

**HA1.6** [1 | 2 | 3 | 4 | 5 | DEF]
Vehicle crash resulting in EITHER:
Visible damage to any Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

**HU1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Seismic event identified by any two of the following:
- SMA-3 Strong Motion Accelograph actuated or Alarm B-3/B-1 SEISMIC EVENT
- Earthquake felt in plant
- National Earthquake Information Center

**HU1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Tornado striking within Protected Area boundary
OR
Sustained high winds > 100 mph

**HU1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Main turbine failure resulting in casing penetration or damage to turbine or generator seals

**HU1.4** [1 | 2 | 3 | 4 | 5 | DEF]
Flooding in any Table H-1 area that has the potential to affect safety-related equipment needed for the current operating mode

**HU1.5** [1 | 2 | 3 | 4 | 5 | DEF]
High river/forebay water level > 899' MSL
OR
Low river level/forebay < 870' MSL

### Notes

1. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
   If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values.
   (See EAL AS1.2/AG1.2.) Do not delay declaration awaiting dose assessment results.
2. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
3. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
4. Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.
5. Manual scram methods for EAL SS2.1 are the following:
   - Reactor Scram push buttons
   - Reactor Mode switch in SHUTDOWN
   - Manual or auto actuation of ARI
6. See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage.
7. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

### 2 — Fire or Explosion

**HA2.1** [1 | 2 | 3 | 4 | 5 | DEF]
Fire or explosion resulting in EITHER:
Visible damage to any Table H-1 area containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

**HU2.1** [1 | 2 | 3 | 4 | 5 | DEF]
Fire in any Table H-1 area not extinguished within 15 min. of Control Room notification or receipt of a valid Control Room alarm due to fire (Note 3)

**HU2.2** [1 | 2 | 3 | 4 | 5 | DEF]
Explosion within the Protected Area

### 3 — Hazardous Gas

**HA3.1** [1 | 2 | 3 | 4 | 5 | DEF]
Access to any Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shut down the reactor (Note 7)

**HU3.1** [1 | 2 | 3 | 4 | 5 | DEF]
Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect normal plant operations

**HU3.2** [1 | 2 | 3 | 4 | 5 | DEF]
Recommendation by local, county or state officials to evacuate or shelter site personnel based on an offsite event

### 4 — Security

**HG4.1** [1 | 2 | 3 | 4 | 5 | DEF]
A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions
OR
A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool

**HS4.1** [1 | 2 | 3 | 4 | 5 | DEF]
A hostile action is occurring or has occurred within the Protected Area as reported by the Security Shift Supervisor

**HA4.1** [1 | 2 | 3 | 4 | 5 | DEF]
A hostile action is occurring or has occurred within the Owner Controlled Area as reported by the Security Shift Supervisor
OR
A validated notification from NRC of an airliner attack threat within 30 min. of the site

**HU4.1** [1 | 2 | 3 | 4 | 5 | DEF]
A security condition that does not involve a hostile action as reported by the Security Shift Supervisor
OR
A credible site-specific security threat notification
OR
A validated notification from NRC providing information of an aircraft threat

### 5 — Control Room Evacuation

General Emergency: None

**HS5.1** [1 | 2 | 3 | 4 | 5 | DEF]
Control Room evacuation has been initiated
AND
Control of the plant cannot be established within 15 min.

**HA5.1** [1 | 2 | 3 | 4 | 5 | DEF]
Control Room evacuation has been initiated

Unusual Event: None

### 6 — Judgment

**HG6.1** [1 | 2 | 3 | 4 | 5 | DEF]
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve EITHER:
Actual or imminent substantial core degradation or melting with potential for loss of containment integrity
OR
Hostile action that results in an actual loss of physical control of the facility

Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary

**HS6.1** [1 | 2 | 3 | 4 | 5 | DEF]
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve EITHER:
An actual or likely major failures of plant functions needed for protection of the public
OR
Hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public

Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary

**HA6.1** [1 | 2 | 3 | 4 | 5 | DEF]
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve EITHER:
An actual or potential substantial degradation of the level of safety of the plant
OR
A security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action

Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels beyond the site boundary

**HU6.1** [1 | 2 | 3 | 4 | 5 | DEF]
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate EITHER:
Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant
OR
A security threat to facility protection has been initiated

No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs

## E — ISFSI

General Emergency: None

Site Area Emergency: None

Alert: None

**EU1.1** [N/A]
Damage to a loaded cask confinement boundary

Modes: 1 Power Operation | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | DEF Defueled

Cooper Nuclear Station
Emergency Action Level Matrix
EPIP 5.7.1 Attachment 1, Rev. 37

Approved: ______________ Manager, Emergency Preparedness ______ Date

# General Emergency | Site Area Emergency | Alert | Unusual Event

## C — Cold SD/ Refuel System Malfunct.

### 1 — Loss of Power

General Emergency: None

Site Area Emergency: None

**CA1.1** [4 | 5 | DEF]
Loss of all offsite and all onsite AC power (Table C-4) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3)

**CU1.1** [4 | 5]
AC power capability to critical 4160V buses 1F and 1G reduced to a single power source (Table C-4) for ≥ 15 min. such that any additional single failure would result in loss of all AC power to critical buses (Note 3)

**CU1.2** [4 | 5]
< 105 VDC bus voltage indications on all Technical Specification required 125 VDC buses for ≥ 15 min. (Note 3)

### 2 — RPV Level

**CG2.1** [4 | 5]
RPV level < -158 in. for ≥ 30 min. (Note 3)
AND
Any Containment Challenge indication, Table C-5

**CG2.2** [4 | 5]
RPV level cannot be monitored for ≥ 30 min. (Note 3) with core uncovery indicated by EITHER:
Unexplained RPV leakage indication, Table C-1
OR
Erratic Source Range Monitor indication
AND
Any Containment Challenge indication, Table C-5

**CS2.1** [4 | 5]
With Containment Closure not established, RPV level < -48 in. (Note 4)

**CS2.2** [4 | 5]
With Containment Closure established, RPV level < -158 in. (Note 4)

**CS2.3** [4 | 5]
RPV level cannot be monitored for ≥ 30 min. (Note 3) with a loss of inventory as indicated by EITHER:
Unexplained RPV leakage indication, Table C-1
OR
Erratic Source Range Monitor indication

**CA2.1** [4 | 5]
RPV level < -42 in.
OR
RPV level cannot be monitored for ≥ 15 min. (Note 3) with any unexplained RPV leakage indication, Table C-1

**CU2.1** [4]
RPV level cannot be restored and maintained > +3 in. for ≥ 15 min. (Note 3) due to RCS leakage

**CU2.2** [5]
Unplanned RPV level drop for ≥ 15 min (Note 3) below EITHER:
RPV flange (LI-86: 206 in. normal calibration, 113.75 in. elevated calibration)
OR
RPV level band when the RPV level band is established below the RPV flange

**CU2.3** [5]
RPV level cannot be monitored with any unexplained RPV leakage indication, Table C-1

### 3 — RCS Temp.

General Emergency: None

Site Area Emergency: None

**CA3.1** [4 | 5]
Any unplanned event results in EITHER:
RCS temperature > 212°F for > Table C-3 duration (Note 4)
OR
RPV pressure increase > 10 psig due to loss of RCS cooling

**CU3.1** [4 | 5]
Any unplanned event results in RCS temperature > 212°F due to loss of decay heat removal capability

**CU3.2** [4 | 5]
Loss of all RCS temperature and RPV level indication for ≥ 15 min. (Note 3)

### 4 — Comm.

General Emergency: None

Site Area Emergency: None

Alert: None

**CU4.1** [4 | 5 | DEF]
Loss of all Table C-2 onsite (internal) communication methods affecting the ability to perform routine operations
OR
Loss of all Table C-2 offsite (external) communication methods affecting the ability to perform offsite notifications

### 5 — Inadvertent Criticality

General Emergency: None

Site Area Emergency: None

Alert: None

**CU5.1** [4 | 5]
An unplanned sustained positive period observed on nuclear instrumentation

**Table C-1 RPV Leakage Indications**
- Drywell equipment drain sump level rise
- Drywell floor drain sump level rise
- Reactor Building equipment drain sump level rise
- Reactor Building floor drain sump level rise
- Torus water level rise
- RPV make-up rate rise
- Observation of unisolable RCS leakage

**Table C-5 Containment Challenge Indications**
- Containment Closure not established (Note 4)
- Deflagration concentrations exist inside PC
  - ≥ 6% $H_2$ in drywell or torus
  - AND
  - ≥ 5% $O_2$ in drywell or torus
- Unplanned rise in PC pressure
- Secondary Containment area radiation > 1000 mR/hr (EOP-5A Table 10)

**Table C-2 Communications Systems**

| System | Onsite (internal) | Offsite (external) |
|---|---|---|
| Station Intercom System "Gaitronics" | X | |
| Site UHF Radio Paging System | X | |
| Alternate Intercom | X | |
| CNS On-Site Cell Phone System | X | X |
| Telephone system (PBX) | X | X |
| Federal Telecommunications System (FTS 2001) | | X |
| Microwave Telephone Network | | X |
| Local Telephones (C.O. Lines) | | X |
| CNS State Notification Telephones | | X |

**Table C-3 RCS Reheat Duration Thresholds**

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable

| | |
|---|---|
| 1. RCS intact (Containment Closure N/A) | 60 min.* |
| 2. Containment Closure established AND RCS not intact | 20 min.* |
| 3. Containment Closure not established AND RCS not intact | 0 min. |

**Table C-4 AC Power Sources**

Offsite
- Startup Station Service Transformer
- Emergency Station Service Transformer
- Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer

Onsite
- DG-1
- DG-2


### Notes

1. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
   If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values.
   (See EAL AS1.2/AG1.2.) Do not delay declaration awaiting dose assessment results.
2. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
3. The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
4. Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.
5. Manual scram methods for EAL SS2.1 are the following:
   - Reactor Scram push buttons
   - Reactor Mode switch in SHUTDOWN
   - Manual or auto actuation of ARI
6. See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage.
7. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.


**EAL Identifier**

XXX.X

- Category (A, H, S, F, C, E)
- Emergency classification (G, S, A, U)
- Subcategory number (1 if no subcategory)
- Sequential number within subcategory/classification

# MODE 4, 5 or DEF