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DAVIS-BESSE EMERGENCY ACTION LEVEL BASIS DOCUMENT

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Emergency Response Manager

5/16/16
Date

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Table of Contents

GENERAL INFORMATION

The reference material captures the basis for the emergency action levels (EALs) as described in RA-EP-01500, Emergency Classification.

December 19, 2008 the NRC approved the upgraded of the Davis-Besse EALs to NEI 99-01, Revision 05

September 30, 2009 Davis-Besse implemented the NEI 99-01, Revision 05 EALs.

Emergency Classification Levels (ECLs)

Nuclear power plant emergencies are separated into four Emergency Classification Levels (ECLs): UNUSUAL EVENT, ALERT, SITE AREA EMERGENCY, and GENERAL EMERGENCY. The ECLs are escalated from least severe to most severe according to relative threat to the health and safety of the public and emergency workers. An ECL is determined to be met by identifying abnormal conditions and then comparing them to INITIATING CONDITIONS (ICs) through EMERGENCY ACTION LEVELS (EAL) and Fission Product Barrier (FPB) threshold values as discussed below. When multiple EALs are met, event declaration is based in the highest ECL reached.

UNUSUAL EVENT: 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 offsite response or monitoring are expected unless further degradation of safety systems occurs.

ALERT: 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 Guide exposure levels.

SITE AREA EMERGENCY: Events are in progress or have occurred which involve 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 act: 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 Guide exposure levels beyond the site boundary.

GENERAL EMERGENCY: 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 Guide exposure levels offsite for more than the immediate site area.

A state or phase called RECOVERY may be entered prior to returning to a normal organization and operation. Recovery provides dedicated resources and organizational structure in support of restoration and communication activities following the termination of the emergency event.

INITIATING CONDITIONS (ICs)

The ICs provide a general description emergency conditions that are organized beneath the broader categories of the ECLs. The IC can be a continuous, measurable condition that is outside Technical Specifications, or it can encompass events such as FIRES or system/equipment failures.

Each IC is given a unique identification code consisting of two letters and one number. The first letter identifies the recognition category, the second letter identifies the ECL, and the number identifies the sequence of the IC within the recognition category. The EAL identification codes are developed as follows:

Recognition Categories

- "F" FISSION PRODUCT BARRIER DEGRADATION
- "R" RADIOLOGICAL EFFLUENT/ABNORMAL RADIATION LEVELS
- "H" HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
- "E" DRY FUEL STORAGE FACILITY (DFSF)
- "S" SYSTEM MALFUNCTIONS - HOT
- "C" SYSTEM MALFUNCTIONS - COLD

Emergency Classification Levels

- "G" GENERAL EMERGENCY
- "S" SITE AREA EMERGENCY
- "A" ALERT
- "U" UNUSUAL EVENT

EMERGENCY ACTION LEVELS (EALs) and Fission Product Barriers (FPBs)

EALs are predetermined, site specific, observable conditions below the ICs that place the state of the plant in a given ECL.

EALs are individually identified by the IC identification code followed by the EAL number, such as RG1.1 for a major effluent release or HA3.2 for high winds.

Fission Product Barriers (FPBs) are given unique character identification codes and are further subdivided into loss and potential loss categories. Since meeting or exceeding one or more FPBs can result in various ECLs, the first two letters simply identify the particular barrier by abbreviation. The number in the FPB identification code associates it with a particular FPB recognition category. The FPB identification codes are developed as follows:

Barrier Abbreviation

- "FC" FUEL CLAD
- "RC" REACTOR COOLANT
- "CT" CONTAINMENT

FPB Recognition Categories

- "1" CRITICAL SAFETY FUNCTION STATUS
- "2" CONTAINMENT RADIATION MONITORING
- "3" CORE TEMPERATURE
- "4" RPV LEVEL
- "5" RCS LEAK RATE
- "6" SG TUBE LEAKAGE / RUPTURE
- "7" RCS ACTIVITY
- "8" CONTAINMENT CONDITIONS
- "9" CONTAINMENT ISOLATION
- "10" ED JUDGMENT

FPBs are treated the same as EALs in that they are applicable only as long as the condition(s) that meet or exceed their thresholds exist. This is in contrast to ECLs which once declared, remain in place until termination or recovery.

For EALs that contain time imbedded criterion, the Emergency Director should not wait until the applicable time period has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Once any EAL is verified to be met, the Emergency Director should not invest additional time determining if other EALs of the same or lower emergency classifications exist. However, the Emergency Director should check the applicability of EALs associated with higher classifications to ensure that he doesn't inadvertently under-classify the emergency.

Equipment used for monitoring and evaluating plant conditions include routine instrumentation, backup or redundant instrumentation, and the use of other parameter

instrumentation that can provide indirect indication.

- When an EAL or FPB refers to a specific instrument or indication that is unavailable prior to an event, alternate indication must be identified to compensate for the loss until the primary indication is restored for the applicable operating mode. Instrumentation used to classify events cannot be removed from service without also implementing adequate compensatory measures.

- When an EAL or FPB refers to a specific instrument or indication that is known to be inaccurate or becomes unavailable during an event (such as off scale high or low), other direct or indirect instrumentation must be used whenever possible. If there are no other direct or indirect means available, then the EAL or FPB can be assumed to have been exceeded consistent with its previous valid trend.

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 site's Technical Specifications. Activities which cause the site to operate beyond that allowed by the site's 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 specific operating license. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

All EALs and FBPs assume 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.

Emergency Action Levels with Embedded Time Requirements

Some EALs have embedded time requirements. For example, "HU4 – FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA." Declaration must be made as soon as the Emergency Director recognizes that the fire will not be extinguished within 15 minutes, but not to exceed 15 minutes from receiving the initial valid fire alarm. Therefore for EALs with time embedded requirements the time for emergency declaration starts with the initial alarm or indication of the event, not after the embedded time has elapsed.

For EALs with longer embedded time requirements the 15 minute clock for declaration begins with recognition that the assigned time limit will be exceeded. For example, the EAL for SG1 "Restoration of either the C-1 bus or the D-1 bus within 4 hours is not

likely." If 20 minutes after loss of the C-1 bus and the D-1 bus it becomes apparent that restoration of either bus will not likely occur within 4 hours the Emergency Director must immediately declare the General Emergency.

Operating Mode Applicability

For purposes of event classification, the following operating mode applicability definitions establish the conditions when the EAL or FPB thresholds represent a threat:

Mode	Reactivity Condition (K_{eff})	% Rated Thermal Power*	Average Reactor Coolant Temperature (°F)
1) Power Operation	≥ 0.99	> 5%	N/A
2) Startup	≥ 0.99	≤ 5%	N/A
3) Hot Standby	< 0.99	N/A	≥ 280
4) Hot Shutdown	< 0.99	N/A	280 > Tavg > 200
5) Cold Shutdown	< 0.99	N/A	≤ 200
6) Refueling	One or more vessel head closure bolts less than fully tensioned.		
D) Defueled	All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).		

* Excluding decay heat.

The Operating Mode Applicability table is based on the Technical Specifications definition of Operational Mode. ICs are based on the operating mode that exists at the time the event occurred, prior to any protective system or operator action initiated in response.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that have Cold Shutdown or Refueling for mode applicability, even if Hot Shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the FPB threshold values are applicable only to events that initiate in Hot Shutdown or higher. If there is a change in operating mode following an event declaration, any subsequent events involving EALs outside of the current declaration escalation path will be evaluated on the mode of the plant at the time the subsequent events occur.

EAL Technical Basis Manual Content

Definitions

A list of definitions is provided for terms having specific meaning to the EALs. EAL terminology definitions are provided with the intent to be used for a particular IC or EAL/FPB threshold value and may not be applicable to other uses of that term in other procedures outside the Emergency Preparedness Program. Refer to Section 1.0 of the Emergency Plan for a comprehensive list of definitions that are used throughout the Emergency Plan.

EAL Matrix Table

The EAL Technical Basis Manual contains six EAL matrix tables, one for each of the different EAL recognition categories.

The EAL matrix is designed as an evaluation tool that organizes the ECLs from the highest (GENERAL EMERGENCY) on the left to the lowest (UNUSUAL EVENT) on the right. Evaluating the EALs for each ECL from highest to lowest reduces the possibility that an event will be under classified. All recognition categories are to be reviewed for applicability prior to event declaration.

Other user aids such as wallboards may be developed from the matrix table to support evaluation of abnormal conditions in other human factored formats.

EAL Documentation Format

Each EAL within the technical bases manual is documented in the following manner:

- IC Identification Number
- INITIATING CONDITION
- Operating Mode Applicability
- EALs or FBP Threshold Value(s)
- Basis
    - Generic
    - Site Specific
- Basis Reference(s)

Davis-Besse to NEI 99-01, Revision 5 IC Cross Reference Tables

NEI to Davis-Besse

NEI	DB	NEI	DB
FU1	FU1	SU1	SU1
FA1	FA1	SU2	SU5
FS1	FS1	SU3	SU4
FG1	FG1	SU4	SU9
		SU5	SU7
FC1	N/A	SU6	SU6
FC2	FC7	SU8	SU3
FC3	FC3	SA2	SA3
FC4	FC4	SA4	SA4
FC6	FC2	SA5	SA1
FC7	N/A	SS1	SS1
FC8	FC10	SS2	SS3
		SS3	SS2
RC1	N/A	SS6	SS4
RC2	RC5	SG 1	SG 1
RC4	RC6	SG2	SG3
RC6	RC2		
RC7	N/A	CU1	CU7
RC8	RC10	CU2	CU8
		CU3	CU1
CT1	N/A	CU4	CU10
CT2	CT8	CU6	CU6
CT3	CT3	CU7	CU2
CT4	CT6	CU8	CU3
CT5	CT9	CA1	CA7
CT6	CT2	CA3	CA1
CT7	N/A	CA4	CA10
CT8	CT10	CS1	CS7
		CG1	CG7
AU1	RU1		
AU2	RU2	HU1	HU3
AA1	RA1	HU2	HU4
AA2	RA2	HU3	HU5
AA3	RA3	HU4	HU1
AS1	RS1	HU5	HU6
AG 1	RG 1	HA1	HA3
		HA2	HA4
		HA3	HA5
		HA4	HA1
		HA5	HA2
		HA6	HA7
		HS2	HS2
		HS3	HS6
		HS4	HS1
		HG1	HG1
		HG2	HG6
		E-HU1	E-HU1

Davis-Besse to NEI

DB	NEI	DB	NEI
FG1	FG1	SG1	SG1
FS1	FS1	SSI	SS1
FA1	FA1	SA1	SA5
FU1	FU1	SU1	SU1
		SS2	SS3
N/A	FC1	SG3	SG2
FC2	FC6	SS3	SS2
FC3	FC3	SA3	SA2
FC4	FC4	SU3	SU8
FC7	FC2	SS4	SS6
N/A	FC7	SA4	SA4
FC10	FC8	SU4	SU3
		SU5	SU2
N/A	RC1	SU6	SU6
RC2	RC6	SU7	SU5
RC5	RC2	SU9	SU4
RC6	RC4		
N/A	RC7	CA1	CA3
RC10	RC8	CU1	CU3
		CU2	CU7
N/A	CT1	CU3	CU8
CT2	CT6	CU6	CU6
CT3	CT3	CG7	CG1
CT6	CT4	CS7	CS1
N/A	CT7	CA7	CA1
CT8	CT2	CU7	CU1
CT9	CT5	CU8	CU2
CT10	CT8	CA10	CA4
		CU10	CU4
RG1	AG1		
RS1	AS1	HG1	HG1
RA1	AA1	HS1	HS4
RU1	AU1	HA1	HA4
RA2	AA2	HU1	HU4
RU2	AU2	HS2	HS2
RA3	AA3	HA2	HA5
		HA3	HA1
		H U 3	H U 1
		HA4	HA2
		HU4	HU2
		HA5	HA3
		HU5	HU3
		E-HU1	E-HU1
		HG6	HG2
		HS	HS3
		HA6	HA6
		HU6	HU5

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: An explosive device suspected of having sufficient force to damage plant systems or structures.

CIVIL DISTURBANCE: A group of persons violently protesting station operations or activities at the site.

COMPENSATORY INDICATIONS: Plant Process Computer, SPDS, and PI Data (Process Book).

CONFINEMENT BOUNDARY: The barrier(s) between areas containing radioactive substances and the environment.

CONTAINMENT CLOSURE: The procedurally defined actions taken to secure primary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions.

CREDIBLE SOURCE: The NRC or other governmental agency such as Federal Bureau of Investigation (FBI), Federal Aviation Administration (FAA), North American Aerospace Defense Command (NORAD), Military Authorities, Law enforcement agencies.

EMERGENCY ACTION LEVELS (EALs): A predetermined, site specific, observable threshold for a plant INITIATING CONDITION that places the plant in a given emergency classification level. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (onsite or offsite); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency classification level.

EXPLOSION: A rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

EXTORTION: An attempt to cause an action at the station by threat of force.

FAULTED: In a steam generator, the existence of secondary side leakage that results in an uncontrolled drop in steam generator pressure or the steam generator being completely depressurized.

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

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

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 intimidates 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 the nuclear power plant. 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).

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.

INTRUSION / INTRUDER: A person(s) present in a specified area without authorization. Discovery of a bomb in a specified area is indication of intrusion into that area by a HOSTILE FORCE.

NORMAL LEVELS: The highest reading in the past twenty-four hours excluding the current peak value.

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.

OWNER CONTROLLED AREA: The property associated with the station and owned by the company. Access is normally limited to persons entering for official business.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: An area that normally encompasses all controlled areas within the security protected area fence.

RUPTURED: In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

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 one or more of the following: 1) automatic runback > 25% thermal reactor power, 2) electrical load rejection > 25% full electrical load, 3) Reactor Trip, 4) Safety Injection Actuation, or 5) thermal power oscillations > 10%.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on management. 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, the reasons for which may be known or unknown, that is not the result of an intended evolution and requires corrective or mitigating actions.

VALID: An indication, report, or condition, is considered to be VALID when it is verified by 1) an instrument channel check, 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 AND is sufficient to cause concern regarding the continued availability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREA: Any area, normally within the PROTECTED AREA that 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.

ACRONYMS

AC ..... Alternating Current
AFW ..... Auxiliary Feed Water
APRM ..... Average Power Range Meter
ATWS ..... Anticipated Transient Without Scram
B&W ..... Babcock and Wilcox
BWST ..... Borated Water Storage Tank
BWR ..... Boiling Water Reactor
CCW ..... Component Cooling Water
CDE ..... Committed Dose Equivalent
CE ..... Combustion Engineering
CFR ..... Code of Federal Regulations
CRM ..... Containment Radiation Monitor
CSF ..... Critical Safety Function
CSFST ..... Critical Safety Function Status Tree
DC ..... Direct Current
DFSF ..... Dry Fuel Storage Facility
DHR ..... Decay Heat Removal
DOT. ..... Department of Transportation
EAL ..... Emergency Action Level
ECCS ..... Emergency Core Cooling System
ECL ..... Emergency Classification Level
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
ERG ..... Emergency Response Guideline
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
GE ..... General Emergency
HPCI ..... High Pressure Coolant Injection
HPSI ..... High Pressure Safety Injection
IC ..... Initiating Condition
IPEEE ..... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI ..... Independent Spent Fuel Storage Installation
ITS ..... Improved Technical Specifications
Keff ..... Effective Neutron Multiplication Factor
LCO ..... Limiting Conditions for Operation
LER ..... Licensee Event Report
LOCA ..... Loss of Coolant Accident

LPSI ..... Low Pressure Safety Injection
LWR ..... Light Water Reactor
MFW ..... Main Feed Water
mR ..... milliRoentgen
MSIV ..... Main Steam Isolation Valve
MSL ..... Main Steam Line
MSSV ..... Main Steam Safety Valve
MU-HPI ..... Makeup High Pressure Injection
MW ..... Megawatt
NEI ..... Nuclear Energy Institute
NESP ..... National Environmental Studies Project
NORAD ..... North American Aerospace Defense Command
NPP ..... Nuclear Power Plant
NRC ..... Nuclear Regulatory Commission
NSSS ..... Nuclear Steam Supply System
NUMARC ..... Nuclear Management and Resources Council
OBE ..... Operating Basis Earthquake
OCA ..... Owner Controlled Area
ODCM/ODAM ..... Offsite Dose Calculation (Assessment) Manual
ORO ..... Offsite Response Organization
PA ..... Protected Area
POAH ..... Point of Adding Heat
PORV ..... Power Operated Relief Valve
PRA/PSA ..... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR ..... Pressurized Water Reactor
PSIG ..... Pounds per Square Inch Gauge
R ..... Roentgen
RCC ..... Reactor Control Console
RCDT ..... Reactor Coolant Drain Tank
RCIC ..... Reactor Core Isolation Cooling
RCS ..... Reactor Coolant System
REM ..... Roentgen Equivalent Man
RPS ..... Reactor Protection System
RPV ..... Reactor Pressure Vessel
RVLIS ..... Reactor Vessel Level Indicating System
RWCU ..... Reactor Water Cleanup
SBGTS ..... Stand-By Gas Treatment System
SBO ..... Station Blackout
SBODG ..... Station Blackout Diesel Generator
SCBA ..... Self-Contained Breathing Apparatus
SG ..... Steam Generator
SI ..... Safety Injection
SPDS ..... Safety Parameter Display System
SRO ..... Senior Reactor Operator
SSE ..... Safe Shutdown Earthquake
TEDE ..... Total Effective Dose Equivalent
TOAF ..... Top of Active Fuel

TSC ..... Technical Support Center
TRM ..... Technical Requirements Manual
UE ..... Unusual Event
WE ..... Westinghouse Electric
WOG ..... Westinghouse Owners Group

Refer to Section 1.0 of the Emergency Plan for a comprehensive listing of Acronyms that are used throughout the Emergency Plan.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FG1

INITIATING CONDITION:

Loss of any two barriers and loss or potential loss of the third barrier.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

Refer to fission product barrier loss and potential loss threshold values to determine barrier status.

Basis:

Generic

Fuel cladding, RCS and containment comprise the fission product barriers.

At the GENERAL EMERGENCY classification level each barrier is weighted equally.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, Tables 5-F-1 and 5-F-3

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FS1

INITIATING CONDITION:

Loss or potential loss of any two barriers.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

Refer to fission product barrier loss and potential loss threshold values to determine barrier status.

Basis:

Generic

Fuel cladding, RCS and containment comprise the fission product barriers.

At the SITE AREA EMERGENCY classification level, each barrier is weighted equally.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, Tables 5-F-1 and 5-F-3

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FA1

INITIATING CONDITION:

Any loss or any potential loss of either fuel clad or RCS.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

Refer to fission product barrier loss and potential loss threshold values to determine barrier status.

Basis:

Generic

Fuel cladding, RCS and containment comprise the fission product barriers.

The fuel cladding and RCS barriers are weighted more heavily than the containment barrier. Unlike the containment barrier, loss or potential loss of either the fuel cladding 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 containment barrier in combination with loss or potential loss of either fuel cladding or RCS barrier results in declaration of a SITE AREA EMERGENCY under FS1.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, Tables 5-F-1 and 5-F-3

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FU1

INITIATING CONDITION:

Any loss or any potential loss of containment.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

Refer to fission product barrier loss and potential loss threshold values to determine barrier status.

Basis:

Generic

Fuel cladding, RCS and containment comprise the fission product barriers.

Unlike the Fuel cladding and RCS barriers, the loss of either of which results in an ALERT under FA1, loss of the 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 containment barrier in combination with the loss or potential loss of either the fuel cladding or RCS barrier results in declaration of a SITE AREA EMERGENCY under FSI.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, Tables 5-F-1 and 5-F-3

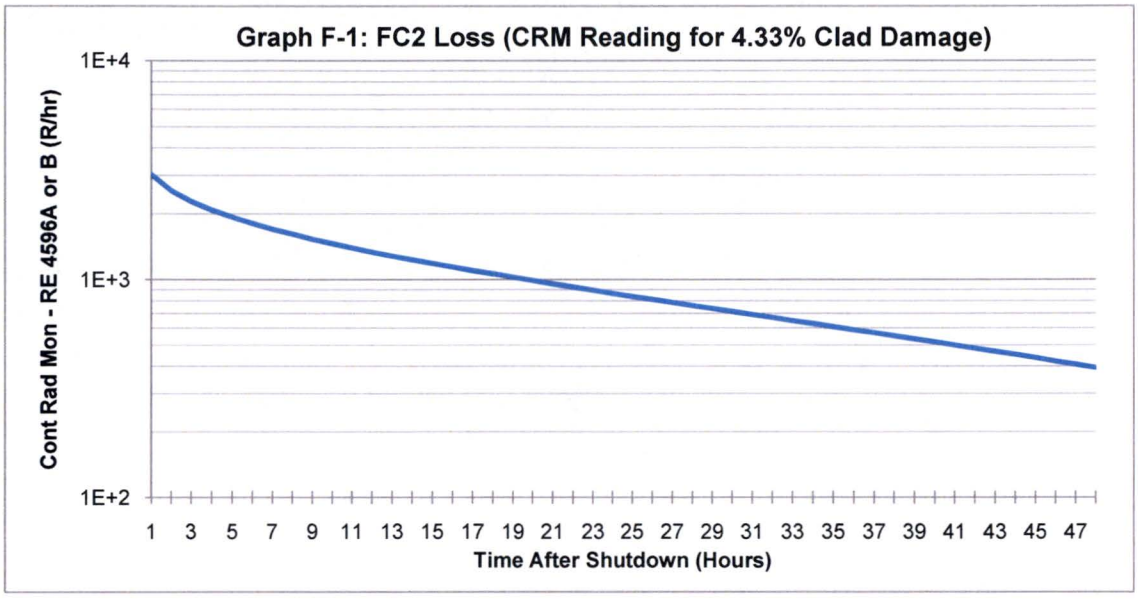
RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC2

Containment Radiation Monitoring

Loss:

1. Containment Radiation Monitor (RE 4596A or B) > Graph F-I.



Potential Loss:

None

Basis:

Generic

The site specific reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment.

The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I -131 into the containment atmosphere.

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

FISSION PRODUCT BARRIER DEGRADATION
Containment Radiation Monitoring

FC2 (continued)

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 #1. Thus, this threshold indicates a loss of both the Fuel Clad Barrier and RCS Barrier that appropriately escalates the emergency classification level to a SITE AREA EMERGENCY.

There is no Potential Loss threshold associated with this item.

Site Specific

A reactor coolant concentration of 300 μCi/gm dose equivalent I -131 is equivalent to 4.33% fuel clad gap damage per EP-EALCALC-DB-0701 Rev 0.

The containment radiation monitor readings specified in Graph F-1 were derived using 4.33% clad damage, no containment sprays in operation, assuming a LOCA depressurized system. The graph is calculated assuming the instantaneous release and dispersal of the above reactor coolant noble gas and iodine inventory into the containment atmosphere.

When evaluating fission product barrier integrity using Graph F-1 and F-2, time after shutdown should be confirmed with the Control Room as the time that the reactor is tripped, and Reactor Power is verified to be lowering on the Intermediate Range (this is consistent with guidance contained in DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE).

When evaluating fission product barriers using the F-1, F-2 curves and time after shutdown is less than one hour (or the reactor is still critical), the one hour after shutdown point should be chosen. This is conservative as it represents sufficient time for plant conditions to deteriorate to the point that core damage may occur, and the activity released from the RCS into the containment atmosphere to reach equilibrium mixing throughout containment.

During a main steam line break in containment or LOCA with temperature > 170 F, there is a potential to induce transient errors (positive and negative) into the output of RE4596A and RE4596B during the peak rate of temperature change. Consult alternate indications. If the main steam line break or LOCA is accompanied by core damage this error is insignificant. Reference: NRC Information Notice 97-45 Supplement 1.

Basis Reference(s):

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

1. NEI 99-01 Rev 5, Table 5-F-3
2. EP-EALCALC-DB-0701 Rev 0, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss and CT2 Potential Loss)
3. NRC Information Notice 97-45 Supplement 1 - Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails
4. 09-53277, Containment High Range Rad Monitor Function and Current EALS
5. 09-53278, Containment High Range Rad Monitor Function and NEI 99-01 EAL Submittal
6. 07-31108, Potential For Thermally Induced Currents In Containment HRRM
7. 09-55171, Containment High Range Rad Monitors Engineering Assistance Requested

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC3

Core Temperature Loss:

1. Calculated Clad Temperature in **Region 3 or higher** (DB-OP-02000 Figure 2).

Potential Loss:

1. Calculated Clad Temperature in **Region 2** (DB-OP-02000 Figure 2).

Basis:

Generic

Loss Threshold #1

The site specific reading should correspond to significant superheating of the coolant.

This value typically corresponds to the core exit temperature reading that indicates core cooling - RED for plants with CSFST, which is usually about 1200° F.

Potential Loss Threshold #1

The site specific reading should correspond to loss of subcooling.

This value typically corresponds to the core exit temperature reading that indicates core cooling - ORANGE for plants with CSFST, which is usually about 700° to 900° F.

Site Specific

WCAP-14969-A states, "Analyses performed for the WOG ERGs for indication of inadequate core cooling concluded that the temperature indicated by the core exit thermocouples, especially during transient heatup conditions, is always several hundred degrees lower than the fuel rod cladding temperatures. Thus, an indicated temperature of 1200°F can be translated to a peak cladding temperature on the order of 1400°F."

Loss Threshold #1

The average incore thermocouple temperature and RCS pressure is used to determine whether Calculated Clad Temperature is in Region 3. This corresponds to a region of significant RCS superheating (T_{clad} = 1400° F threshold curve).

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC3 (continued)

Potential Loss Threshold #1

The average incore thermocouple temperature and RCS pressure is used to determine whether Calculated Clad Temperature is in Region 2. This corresponds to a loss of RCS subcooling with clad temperatures remaining below the point where damage is likely (T_{clad} approximately 900° to 1100° F).

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3
2. WCAP-14969-A, Westinghouse Owners Group Core Damage Assessment Guidance
3. DB-OP-02000 Figure 2, Incore T/C Temperature vs. RC Pressure for Inadequate Core Cooling

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC4

RPV Level

Loss:

None

Potential Loss:

1. a. RCS level (hot leg level monitoring) **< 92 inches.**

   AND

   b. Calculated Clad Temperature in **Region 2 or higher** (DB-OP-02000 Figure 2).

Basis:

Generic

There is no Loss threshold associated with this item.

The site specific value for the Potential Loss threshold corresponds to the top of the active fuel.

Site Specific

The RCS level is the lowest accurate hot leg level reading representing 10% of scale. This combined with a loss of subcooling as indicated by Calculated Clad Temperature in Region 2 or higher are the site specific indications indicative of RCS level at the top of active fuel.

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3
2. DB-OP-02000 Figure 2, Incore TIC Temperature vs. RC Pressure for Inadequate Core Cooling
3. C-ICE-064.02-008 Rev 0, Uncertainties for Reactor Coolant Hot Leg Monitoring System

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC7

RCS Activity

Loss:

1. Coolant activity > 300 μCi/gm dose equivalent I -131.

Potential Loss:

None

Basis:

Generic

The site specific value corresponds to 300 μCi/gm I -131 equivalent. Assessment by the EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no Potential Loss threshold associated with this item.

Site Specific

300 μCi/gm Dose Equivalent I -131 corresponds to 4.33% fuel cladding damage per EP-EALCALC-DB-0701 Rev 0.

There is no mechanism to determine reactor coolant activity in real time. Therefore, once a sample is analyzed to exceed 300 μCi/gm Dose Equivalent I -131, this threshold is met (and will continue to be met) until a subsequent sample indicates activity is reduced below the limit.

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3
2. EP-EALCALC-DB-0701 Rev 0, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss and CT2 Potential Loss)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC10

Emergency Director Judgment

Loss:

1. Any condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier.

Potential Loss:

1. Potential Loss: Any condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier.

Basis:

Generic

These thresholds address any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost or potentially lost, In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC2

Containment Radiation Monitoring

Loss:

1. Containment Radiation Monitor (RE 4596A or B) **> 15 R/hr**.

Potential Loss:

None

Basis:

Generic

The site specific reading is a value which indicates the release of reactor coolant to the containment.

The reading should be 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 containment atmosphere.

This reading will be less than that specified for FC2(L)1. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier threshold, fuel damage would also be indicated.

There is no potential loss threshold associated with this item.

Site Specific

The containment monitor reading was calculated assuming an instantaneous release and dispersal of the USAR maximum (no clad damage) RCS coolant activity into the containment atmosphere. Maximum coolant activity, vice normal coolant activity, was used to provide; USAR documented initial source term values, bounding for a full loss of barrier and, sufficient margin (approximately 10x) above the downscale reading.

During a main steam line break in containment or LOCA with temperature > 170 F, there is a potential to induce transient errors (positive and negative) into the output of RE4596A and RE4596B during the peak rate of temperature change. Consult alternate indications. If the main steam line break or LOCA is accompanied by core damage this error is insignificant. Reference: NRC Information Notice 97-45 Supplement 1.

Since fuel cladding degradation and/or failures could also result in high Containment Area Radiation levels, a reading of >15 R/hr may be obtained without a physical loss of the RCS barrier. However, this threshold should be declared as being met if the Containment Area Radiation levels are >15 R/hr even if there are no other indications of a RCS leak or

physical loss. In this case it would also be prudent to evaluate the fuel clad fission product barrier thresholds.

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3
2. EP-EALCALC-DB-0702 Rev 0, Containment Radiation Monitor Reading Following a LOCA (RC2 Loss)
3. NRC Information Notice 97-45 Supplement 1 - Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails
4. 09-53277, Containment High Range Rad Monitor Function and Current EALS
5. 09-53278, Containment High Range Rad Monitor Function and NEI 99-01 EAL Submittal
6. 07-31108, Potential For Thermally Induced Currents In Containment HRRM
7. 09-55171, Containment High Range Rad Monitors Engineering Assistance Requested

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC5

RCS Leak Rate

Loss:

1. RCS leakage results in **< 20°F** subcooling margin.

Potential Loss:

1. RCS leak rate **> 250 gpm.**

Basis:

Generic

Loss Threshold #1

This threshold addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

Potential Loss Threshold #1

This threshold is based on the apparent inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered to be the flow rate equivalent to one charging/makeup pump discharging to the header. Isolating letdown is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path such as a CVCS leak exists. The intent of this condition is met if attempts to isolate Letdown are NOT successful. Additional charging/makeup pumps being required is indicative of a substantial RCS leak.

Site Specific

Loss Threshold #1

RCS subcooling value corresponds to the limit specified in DB-OP-02000.

Potential Loss Threshold #1

This threshold is based on the capacity of a single makeup injection line in service with letdown isolated.

When PORV cooling is initiated, the 250 gpm RCS leak rate will likely be exceeded and the conditions for this EAL will be met.

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC5 (continued)

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3
2. DB-OP-02000; RPS, SFAS, SFRCS Trip, or SG Tube Rupture
3. Condition Report 09-67382

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC6

SG Tube Leakage / Rupture

Loss:

1. RUPTURED SG results in an SFAS actuation.

Potential Loss:

None

Basis:

Generic

This threshold addresses the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment Barrier Loss thresholds. It addresses RUPTURED SG(s) for which the leakage is large enough to cause actuation of ECCS (SI). This is consistent to the RCS leak rate barrier Potential Loss threshold.

There is no Potential Loss threshold associated with this item.

Site Specific

ECCS systems are actuated by Safety Features Actuation System (SFAS).

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC10

Emergency Director Judgment

Loss:

1. Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier.

Potential Loss:

1. Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier.

Basis:

Generic

These thresholds address any other factors that are to be used by the Emergency Director in determining whether the RCS Barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

Site Specific

None

Basis Reference:

1. NEI 99-01 Rev 5, Table 5-F-3

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT2

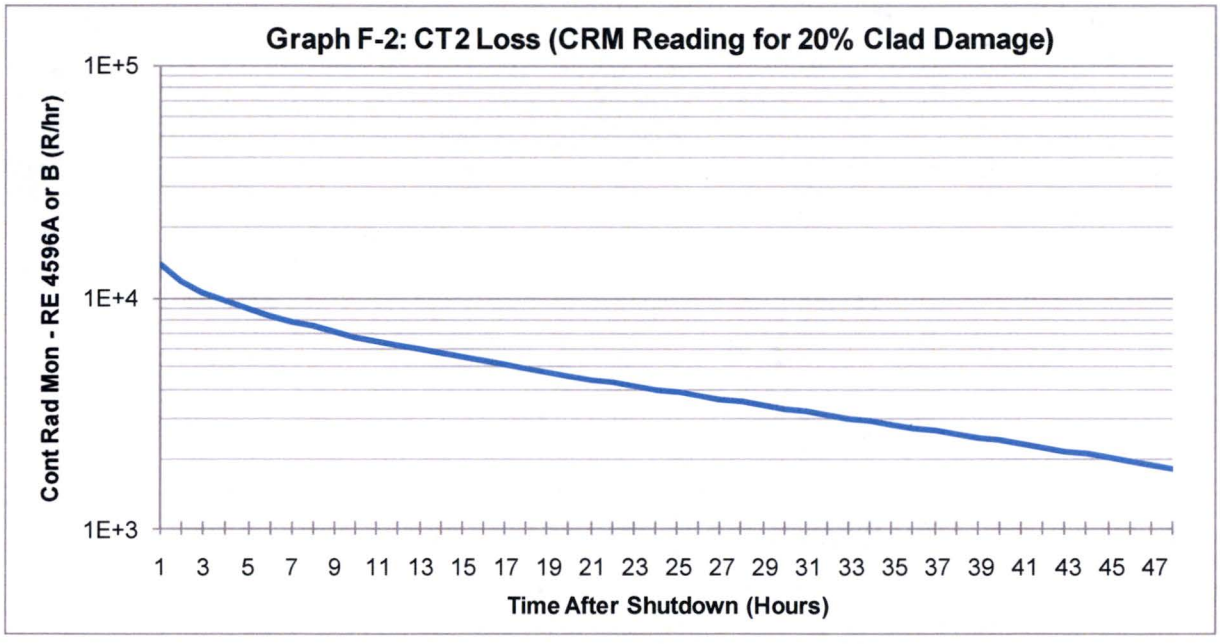
Containment Radiation Monitoring

Loss:

1. None

Potential Loss:

Containment Radiation Monitor (RE 4596A or B) **> Graph F-2.**



Basis:

Generic

There is no Loss threshold associated with this item.

The site specific reading is a value which indicates significant fuel damage well in excess of the thresholds associated with both loss of fuel clad and loss of RCS Barriers. As stated in Section 3.8 of NEI 99-01 Rev 5, 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.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT2 (continued)

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 toss 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%.

Site Specific

The containment radiation monitor readings specified in Graph F-2 were derived using 20% clad damage, no containment sprays in operation, assuming a LOCA depressurized system. The graph is calculated assuming the instantaneous release and dispersal of the above reactor coolant noble gas and iodine inventory into the containment atmosphere.

When evaluating fission product barrier integrity using Graph F-1 and F-2, time after shutdown should be confirmed with the Control Room as the time that the reactor is tripped, and Reactor Power is verified to be lowering on the Intermediate Range (this is consistent with guidance contained in DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE).

When evaluating fission product barriers using the F-1, F-2 curves and time after shutdown is less than one hour (or the reactor is still critical), the one hour after shutdown point should be chosen. This is conservative as it represents sufficient time for plant conditions to deteriorate to the point that core damage may occur, and the activity released from the RCS into the containment atmosphere to reach equilibrium mixing throughout containment.

During a main steam line break in containment or LOCA with temperature > 170 F, there is a potential to induce transient errors (positive and negative) into the output of RE4596A and RE4596B during the peak rate of temperature change. Consult alternate indications. If the main steam line break or LOCA is accompanied by core damage this error is insignificant. Reference: NRC Information Notice 97-45 Supplement 1.

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3
2. EP-EALCALC-DB-0701 Rev 0, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss and CT2Potential Loss)
3. NRC Information Notice 97-45 Supplement 1 - Environmental Qualification Deficiency

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

for Cables and Containment Penetration Pigtails

4. 09-53277, Containment High Range Rad Monitor Function and Current EALS
5. 09-53278, Containment High Range Rad Monitor Function and NEI 99-01 EAL Submittal
6. 07-31108, Potential For Thermally Induced Currents In Containment HRRM
7. 09-55171, Containment High Range Rad Monitors Engineering Assistance Requested

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT3

Core Temperature

Loss:

None

Potential Loss:

1. **a.** Calculated Clad Temperature in Region 4 - **Severe Accident Region** (DB-OP-02000 Figure 2).

   AND

   b. Restoration procedures not effective within **15 minutes.**

OR

2. **a.** Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2).

   AND

   b. RCS level (hot leg level monitoring) **< 92 inches.**

   AND

   c. Restoration procedures not effective within **15 minutes.**

Basis:

Generic

There is no Loss threshold associated with this item.

The conditions in these thresholds represent an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the Core Cooling and RCS Leakage criteria in the Fuel and RCS Barrier columns, this threshold would result in the declaration of a GENERAL EMERGENCY - loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.

The function restoration procedures are those Emergency Operating Procedures (EOPs) that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT3 (continued)

function restoration procedures to arrest the core melt sequence.

Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

Site Specific

Potential Loss Threshold #1

WCAP-14969-A Table 1 provides 2400° F as where a very rapid release of volatile fission products from fuel pellets occurs (departure from failure of fuel rod cladding) and Table 2 establishes core outlet temperatures of > 2000° F as indication of core melt.

It also states, "Analyses performed for the WOG ERGs for indication of inadequate core cooling concluded that the temperature indicated by the core exit thermocouples, *especially* during transient heatup conditions, is always several hundred degrees lower than the fuel rod cladding temperatures. Thus, an indicated temperature of 1200°F can be translated to a peak cladding temperature on the order of 1400°F."

The average incore thermocouple temperature and RCS pressure is used to determine whether Calculated Clad Temperature is in Region 4. This corresponds to clad temperatures, T_{clad} = 1800° F, reaching the site specific threshold indicative of imminent core damage.

Potential Loss Threshold #2

The RCS level is the lowest accurate hot leg level reading representing 10% of scale. This combined with significant RCS superheating as indicated by Calculated Clad Temperature in Region 3 are the site specific thresholds representative of imminent core melting.

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3
2. WCAP-14969-A, Westinghouse Owners Group Core Damage Assessment Guidance
3. DB-OP-02000 Figure 2, Incore T/C Temperature vs. RC Pressure for Inadequate Core Cooling
4. C-ICE-064.02-008 Rev 0, Uncertainties for Reactor Coolant Hot Leg Monitoring System

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT6

SG Tube Leakage / Rupture

Loss:

1. RUPTURED SG is also FAULTED outside of containment.

   OR

2. a. Primary-to-Secondary leak rate **>10 gpm**.

   AND

   b. UNISOLABLE steam release from affected SG to the environment.

Potential Loss:

None

Basis:

Generic

The loss threshold recognizes that SG tube leakage can represent a bypass of the Containment Barrier as well as a loss of the RCS Barrier.

Users should realize that the two loss thresholds could be considered redundant. This was recognized during the development process. The inclusion of a threshold that uses Emergency Procedure commonly used terms like "RUPTURED and FAULTED" adds to the ease of the classification process and has been included based on this human factor concern.

This threshold results in an UNUSUAL EVENTS for smaller breaks that: 1) do not exceed the normal charging [make] up capacity threshold in RCS leak rate barrier Potential Loss threshold, or 2) do not result in ECCS actuation in RCS SG Tube Rupture Barrier Loss threshold. For larger breaks, RCS Barrier threshold criteria would result in an ALERT. For SG tube ruptures which may involve multiple steam generators or UNISOLABLE secondary line breaks, this threshold would exist in conjunction with RCS Barrier thresholds and would result in a SITE AREA EMERGENCY.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT6 (continued)

Loss Threshold #1

This threshold addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and Containment Barriers and is a subset of the second threshold. In conjunction with RCS leak rate barrier loss threshold, this would always result in the declaration of a SITE AREA EMERGENCY.

Loss Threshold #2

This threshold addresses SG tube leaks that exceed 10 gpm in conjunction with an UNISOLABLE release path to the environment from the affected steam generator. The threshold for establishing the UNISOLABLE secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SG tube rupture with concurrent loss of offsite power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of an UNISOLABLE release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT8

Containment Pressure

Loss:

1. A containment pressure rise followed by a rapid UNPLANNED drop in containment pressure.

   OR

2. Containment pressure or sump level response not consistent with LOCA conditions.

Potential Loss:

1. Containment pressure **> 54.4 psia** and rising.

   OR

2. Containment hydrogen **> 4%.**

   OR

3. a. Containment pressure **> 40.0 psia.**

      AND

   b. Containment cooling is less than **ANY** of the **Table CT-1** conditions:

Table CT-1	
SPRAY	**COOLERS**
2	0
1	1
0	2

Basis:

Generic

Loss Thresholds #1 and #2

Rapid UNPLANNED loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure increase from a primary or secondary high energy line break indicates a loss of containment integrity. Containment pressure and sump levels should increase as a result of mass and energy release into containment from

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT8 (continued)

a LOCA. Thus, sump level or pressure not increasing indicates containment bypass and a loss of containment integrity.

This indicator relies on operator recognition of an UNPLANNED response for the condition and therefore does not have a specific value associated with it. The UNPLANNED response is important because it is the indicator for a containment bypass condition.

Potential Loss Threshold #1

The site specific pressure is based on the containment design pressure.

Potential Loss Threshold #2

Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists.

Potential Loss Threshold #3

This threshold represents a potential loss of containment in that the containment heat removal/depressurization system (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the set point at which the equipment was supposed to have actuated.

Site Specific

Potential Loss Threshold #1

This threshold is the containment design pressure of 40.0 psig (14.4 psi added to convert to psia) and is above the value projected from the design basis loss of coolant accident. The worst case scenario is a double ended hot leg guillotine break at reactor vessel outlet which produces a peak pressure of 52.3 psia.

Potential Loss Threshold #3

Containment integrity is ensured by two full capacity, independent, pressure reducing systems operating on different principles (the spray system and the containment air cooler system). The spray system, in conjunction with the emergency cooling system, limits and maintains post accident conditions to less than the containment design values. One cooler and spray path are operated concurrently to prevent pressure from exceeding design limits over the entire spectrum of RCS break sizes and to rapidly reduce the driving

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT8 (continued)

force for leakage of radioactive materials from the containment. The spray system is at least equivalent in heat removal capacity to the containment air coolers. In combination with containment air coolers, it affords redundant alternative methods to reduce containment pressure after a LOCA and maintain pressure at a reduced level.

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3
2. USAR Table 6.2.1, Peak Containment Pressures for various LOCA's
3. USAR 6.2.1.2.1, Design Parameters
4. C-NSA-060.05-010 Rev 06

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT9

Containment Isolation Failure

Loss:

1. a. Failure of ALL valves in any one line to close.

   AND

   b. Direct downstream pathway to the environment exists after containment isolation signal.

Potential Loss:

None

Basis:

Generic

This threshold addresses 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.

There is no Potential Loss threshold associated with this item.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-4

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT10

Emergency Director Judgment

Loss:

1. Any condition in the opinion of the Emergency Director that indicates loss of the containment barrier.

Potential Loss

1. Any condition in the opinion of the Emergency Director that indicates potential loss of the containment barrier.

Basis:

Generic

These thresholds address any other factors that are to be used by the Emergency Director in determining whether the Containment Barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

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 progress (Loss or Potential Loss of either fuel clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, Table 5-F-3

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RG1

INITIATING CONDITION:

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 1000 mRem TEDE or 5000 mRem Child Thyroid CDE for the actual or projected duration of the release using actual meteorology.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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.

1. Station Vent Channel 1 Noble Gas (RE 4598) **> 2.86E +01 µCi/cc** for **15 minutes** or longer.

   OR

2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER** of the following:

   - **> 1000 mRem** TEDE.
   - **> 5000 mRem** CDE Child Thyroid.

   OR

3. Field survey results at or beyond the site boundary indicate EITHER of the following:

   - Gamma (closed window) dose rate **> 1000 mR/hr** for **60 minutes** or longer.
   - Air sample analysis **> 5000 mRem** CDE Child Thyroid for one hour of inhalation.

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS
RG1

Basis:

Generic

This IC 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.

The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "...sum of EDE and CEDE .... "

The TEDE dose is set at the EPA PAG, while the 5000 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

EAL #1

The site specific monitor list in EAL #1 should include effluent monitors on all potential release pathways.

The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site boundary (or beyond) dose of 1000 mRem whole body or 5000 mRem thyroid in one hour, whichever is more limiting (as was done for EALs #2 and #3). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.

The meteorology used should be the same as those used for determining RU1 and RA1 monitor reading EALs. The same source term (noble gases, particulates, and halogens) may also be used as long as it maintains a realistic and near linear escalation between the EALs for the four classifications. If proper escalations do not result from the use of the same source term, if the calculated values are unrealistically high, or if correlation between the values and dose assessment values does not exist, then consider using an accident source term for RS1 and RG1 calculations.

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 should call for the timely performance of dose

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RG1

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.

Site Specific

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

EAL #1

The station vent is the primary effluent monitor release point. Other pathways such as Safety and relief valves, as well as AFW and MSSV, do not have direct effluent release monitors. Releases from these points would be verified and monitored by RG1.3 survey readings.

The monitor value is based on reaching the limiting EPA PAG at site boundary under the prescribed accident, release, and meteorological conditions. The accident damage nuclide mix is based on a DBA LOCA yield limited to a fuel gap activity release. Complete assumptions and inputs are documented in calculation EP-EALCALC-DB-0703 Rev 1.

EALs #2 and #3

The 'site boundary' is defined at a minimum exclusion distance of 0.75 miles. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

Basis Reference(s):

1. NEI 99-01 REV 5, AG1
2. EP-EALCALC-DB-0703 Rev 1, Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1, and RU1)
3. USAR 1.2.1.1, Site

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RS1

INITIATING CONDITION:

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 100 mRem TEDE or 500 mRem Child Thyroid CDE for the actual or projected duration of the release.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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.

1. Station Vent Channel 1 Noble Gas (RE 4598) **> 2.86E+00 µCi/cc** for **15 minutes** or longer.

   OR

2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER** of the following:

   > 100 mRem TEDE.

   > 500 mRem CDE Child Thyroid.

   OR

3. Field survey results at or beyond the site boundary indicate **EITHER** of the following:

   - Gamma (closed window) dose rate **> 100 mR/hr** for **60 minutes** or longer.
   - Air sample analysis **> 500 mRem** CDE Child Thyroid for one hour of inhalation.

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RS1

Basis:

Generic

This IC 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.

The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "...sum of EDE and CEDE .... "

The TEDE dose is set at 10% of the EPA PAG, while the 500 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

EAL #1

The site specific monitor list in EAL #1 should include effluent monitors on all potential release pathways.

The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site boundary (or beyond) dose of 100 mRem whole body or 500 mRem thyroid in one hour, whichever is more limiting (as was done for EALs #2 and #3). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.

The meteorology used should be the same as those used for determining RU1 and RA1 monitor reading EALs. The same source term (noble gases, particulates, and halogens) may also be used as long as it maintains a realistic and near linear escalation between the EALs for the four classifications. If proper escalations do not result from the use of the same source term, if the calculated values are unrealistically high, or if correlation between the values and dose assessment values does not exist, then consider using an accident source term for RS1 and RG1 calculations.

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 should 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

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RS1

classification level), the dose assessment results override the monitor reading EAL.

Site Specific

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

EAL #1

The station vent is the primary effluent monitor release point. Other pathways such as safety and relief valves, as well as AFW and MSSV, do not have direct effluent release monitors. Releases from these points would be verified and monitored by RS1.3 survey readings.

The monitor value is based on reaching 1/10 the limiting EPA PAG at site boundary under the prescribed accident, release, and meteorological conditions. The accident damage nuclide mix is based on a DBA LOCA yield limited to a fuel gap activity release. Complete assumptions and inputs are documented in calculation EP-EALCALC-DB-0703 Rev 1.

EALs #2 and #3

The 'site boundary' is defined at a minimum exclusion distance of 0.75 miles. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

Basis Reference(s):

1. NEI 99-01 Rev 5, AS1
2. EP-EALCALC-DB-0703 Rev 1, Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1, and RU1)
3. USAR 1.2.1.1, Site

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RA1

INITIATING CONDITION:

Any release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM limit for 15 minutes or longer.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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.

1. Station Vent Channel 1 Noble Gas (RE 4598) **> 2.29E-01 µCi/cc** for **15 minutes** or longer.

   OR

2. ANY of the following effluent monitors **> 200 times the high alarm setpoint, not to exceed 8 E+6 CPM,** as established by a current radioactivity discharge permit for **15 minutes** or longer:

   - Waste Gas System Outlet (RE 1822A or B).
   - Clean Waste System Outlet (RE 1770A or B).
   - Miscellaneous Waste System Outlet (RE 1878A or B).
   - Discharge permit specified monitor.

   OR

3. Confirmed sample analysis for gaseous or liquid releases **> 200 times the ODCM limit** for **15 minutes** or longer.

Basis:

Generic

This IC 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

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RA1

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. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODCM multiples are specified in RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite 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.

Releases should not be prorated or averaged. For example, a release exceeding 600x ODCM for 5 minutes does not meet the threshold.

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.

EAL #1

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.

EAL #2

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity Discharge permit. This value may be associated with a planned batch release, or a continuous release path.

In either case, the value is established by the ODCM to warn of a release that is not in compliance. Indexing the EAL to the ODCM setpoints in this manner insures that the EAL will never be less than the setpoint established by a specific Discharge permit.

EAL #3

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.

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RA1

Site Specific

EAL #1

A multiple of 200 cannot be used for the gaseous effluent value as it would result in the Alert value being higher than the SITE AREA EMERGENCY. Therefore, to provide a straight logarithmic progression for the EALs, the RA1.1 value is set half way between the RS1.1 value and the RU1.1 value. Assumptions and calculation inputs are provided in EP-EALCALC-DB-0703 Rev 1.

EAL #2

The threshold of >200 times the high alarm setpoint is limited to a maximum valve of 8E+6 CPM to assure an on scale readable value. An elevated monitor reading while the effluent flow path is isolated is not considered a valid reading.

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity Discharge permit. This value may be associated with a planned batch release, or a continuous release path.

In either case, the value is established by the ODCM to warn of a release that is not in compliance. Indexing the EAL to the ODCM setpoints in this manner insures that the EAL will never be less than the setpoint established by a specific Discharge permit.

EAL #3

Grab samples are used to: determine release concentrations or release rates, confirm meter readings, or indicate the need for sampling when the effluent monitors are not in service or other alarms occur. The maximum instantaneous release rate limits are calculated in accordance with the ODCM. These are indicated on approved discharge permit release packages.

Basis Reference(s):

1. NEI 99-01 Rev 5, AA1
2. EP-EALCALC-DB-0703 Rev 1, Radiological Gaseous Effluent EAL Values EALs RG1, RS1, RA1, and RU1)

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RU1

INITIATING CONDITION:

Any release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM limit for 60 minutes or longer.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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.

1. Station Vent Channel 1 Noble Gas (RE 4598) **> 1.84E-02 µCi/cc** for **60 minutes** or longer.

   OR

2. **ANY** of the following effluent monitors **> 2 times the high alarm setpoint** established by a current radioactivity discharge permit for **60 minutes** or longer:

   - Waste Gas System Outlet (RE 1822A or B).
   - Clean Waste System Outlet (RE 1770A or B).
   - Miscellaneous Waste System Outlet (RE 1878A or B).
   - Discharge permit specified monitor.

   OR

3. Confirmed sample analysis for gaseous or liquid releases **> 2 times the ODCM limit** for **60 minutes** or longer.

Basis:

Generic

This IC 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

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RU1

effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODCM multiples are specified in RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite 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.

Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold.

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.

EAL #1

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

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.

The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. This EAL is determined using this methodology.

EAL #2

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path.

In either case, the value is established by the ODCM to warn of a release that is not in compliance. Indexing the EAL to the ODCM setpoints in this manner insures that the EAL will never be less than the setpoint established by a specific Discharge permit.

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RU1

EAL #3

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.

Site Specific

EAL #1

The gaseous effluent value of 2 times the ODCM setpoint was determined using formulas, isotopic dose conversion factors and meteorology data as specified in the ODCM based on a normal operating isotopic mixture (no clad damage condition). Assumptions and calculation inputs are provided in EP-EALCALC-DB-0703 Rev 1.

EAL #2

An elevated monitor reading while the effluent flow path is isolated is not considered a VALID reading.

The effluent monitors listed are those normally used for planned discharges. If a discharge is performed using a different flow path or effluent monitor (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the Discharge permit.

EAL #3

Grab samples are used to: determine release concentrations or release rates, confirm meter readings, or indicate the need for sampling when the effluent monitors are not in service or other alarms occur. The maximum instantaneous release rate limits are calculated in accordance with the ODCM. These are indicated on approved Discharge permit release packages.

Basis Reference(s):

1. NEI 99-01 Rev 5, AU1
2. EP-EALCALC-DB-0703 Rev 1, Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1, and RU1)

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RA2

INITIATING CONDITION:

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.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. A water level drop in the spent fuel pool or refueling canal that will result in irradiated fuel becoming uncovered.

   OR

2. **>1000 mR/hr** on **ANY** of the following due to damage to irradiated fuel or loss of water level:

   - Fuel Handling Area (RE 8417 or RE 8418).
   - Equipment Hatch (RE 8425).
   - Spent Fuel Area (RE 8426 or RE 8427).

Basis:

Generic

This IC 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.

These events escalate from RU2 in that fuel activity has been released, or is anticipated due to fuel heatup. This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

EAL #1

Site specific indications may include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. If available, video cameras may allow remote observation. Depending on available level instrumentation, the declaration threshold may need to be based on indications of water makeup rate or decrease in water storage tank level.

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RA2

EAL #2

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.

For example, a refueling bridge radiation monitor 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 be combined with another indicator (or personnel report) of water loss.

Site Specific

EAL #1

Water level is normally monitored by:

- Spent fuel pool level LI1600.
- Refueling canal level LI 1627.

Symptoms for a loss of inventory may also be indicated by (DB-OP-02527):

- Annunciator Alarms.
- Computer Alarms.
- Level Indication.
- Erratic Pump Amps.
- Upscale indication on local RE monitors.

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RA2

EAL #2

NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82," (July, 1987) indicates that even if corrective actions are not taken when spent fuel becomes uncovered, no prompt fatalities are predicted and the risk of injury is low. Therefore, a period of time will be available to take corrective actions prior to the actual onset of fuel damage.

Visual observation of spent fuel uncovery represents a major ALARA concern in that radiation levels could exceed 10,000 R/hr on the refuel bridge when uncovery occurs. The value of 1000 mR/hr was conservatively chosen for classification purposes.

Basis Reference(s):

1. NEI 99-01 Rev 5, AA2
2. Information Notice No. 90-08, KR-85 Hazards from Decayed Fuel
3. DP-OP-02530, Fuel Handling Accident
4. DB-OP-02527, Loss of Decay Heat Removal

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RU2

INITIATING CONDITION:

UNPLANNED rise in plant radiation levels.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. a. UNPLANNED water level drop in the spent fuel pool or refueling canal as indicated by ANY of the following:

   - Spent Fuel Pool Level (LI 1600) < 19.0 **feet.**
   - Refueling Canal Level (LI 1627) < 19.0 **feet.**
   - Report of visual observation.

   AND

   b. Area radiation monitor reading rise on ANY of the following:

   - Fuel Handling Area (RE 8417 or RE 8418).
   - Equipment Hatch (RE 8425).
   - Spent Fuel Area (RE 8426 or RE 8427).

   OR

2. UNPLANNED area radiation monitor or radiation survey > 1000 times **NORMAL LEVELS.**

Basis:

Generic

This IC 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.

EAL #1

Site specific indications may include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. If available, video cameras

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RU2

may allow remote observation. Depending on available level instrumentation, the declaration threshold may need to be based on indications of water makeup rate or decrease in water storage tank level.

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 radiation monitor 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 be 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 CU8.

EAL #2

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 waste materials.

Site Specific

EAL #1

19 feet is the level at which the refuel bridge will be abandoned and the area evacuated prior to fuel uncover. It considers the worse case condition of a withdrawn bundle while maintaining submergence.

Water level is normally monitored by:

- Spent fuel pool level LI 1600.
- Refueling canal level LI 1627.

Symptoms for a loss of inventory may also be indicated by (DB-OP-02527):

- Annunciator Alarms.
- Computer Alarms.

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RU2

- Level Indication.
- Erratic Pump Amps.
- Upscale indication on local RE monitors.

EAL #2

For the purpose of this EAL, reference to radiation survey results of >1000x NORMAL LEVELS is in reference to whole body dose rates (or 30 cm) and not contact dose rates.

Basis Reference(s):

1. NEI 99-01 Rev 5, AU2
2. Information Notice No. 90-08, KR-85 Hazards from Decayed Fuel
3. DP-OP-02530, Fuel Handling Accident
4. DP-OP-00030, Fuel Handling Operations

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RA3

INITIATING CONDITION:

Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. Dose rate **> 15 mR/hr** in **ANY** of the following areas requiring continuous occupancy to maintain plant safety functions:

   - Control Room.
   - Central Alarm Station

Basis:

Generic

This IC addresses increased radiation levels that impede 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 IC. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other IC may be involved.

As used here, 'impede' includes hindering or interfering, provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant.

The value of 15mR/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 mR/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 Control Room and, as appropriate to the site, any other control stations that are staffed continuously, such as a Radwaste Control Room or a Security Alarm Station.

RECOGNITION CATEGORY
RADIOLOGICAL EFFLUENTS / ABNORMAL RADIATION LEVELS

RA3

Site Specific

Areas requiring continuous occupancy include the Control Room and Central Alarm Station (CAS). CAS is included in this EAL because it controls security doors and barriers allowing operators access to plant equipment needed to maintain plant safety functions.

Basis Reference(s):

1. NEI 99-01 Rev 5, AA3

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1

INITIATING CONDITION:

HOSTILE ACTION resulting in loss of physical control of the facility.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions listed below:

   - Reactivity Control (ability to shut down the reactor and keep it shutdown).
   - RCS Inventory (ability to cool the core).
   - Secondary Heat Removal (ability to maintain heat sink).

   OR

2. A HOSTILE ACTION has caused failure of spent fuel cooling systems and IMMINENT fuel damage is likely.

Basis:

Generic

EAL #1

This EAL 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.

Typically, these safety functions are reactivity control (ability to shutdown the reactor and *keep* it shutdown), RCS inventory (ability to cool the core), and secondary 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. 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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

EAL #2

This EAL 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.

Site Specific

EAL #1

Any one of the three bullets satisfies the condition of the EAL.

Basis Reference(s):

1. NEI 99-01 Rev 5, HG1
2. C-NF-062.02-007 Rev 0

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1

INITIATING CONDITION:

HOSTILE ACTION within the PROTECTED AREA.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.

Basis:

Generic

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 Offsite Response Organization (ORO) 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. 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 OROs 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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1

notification.

Site Specific

HOSTILE ACTION is considered to be within the PROTECTED AREA when the results of the HOSTILE ACTION are manifested inside the inner PROTECTED AREA fence, even if the HOSTILE ACTION originated outside of this fence.

Basis Reference(s):

1. NEI 99-01 Rev 5, HS4

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

INITIATING CONDITION:

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.

   OR

2. A validated notification from the NRC of an airliner attack threat within **30 minutes** of the site.

Basis:

Generic

Note: Timely and accurate communication between security shift supervision and the control room is crucial for the implementation of effective security EALs.

These EALs 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 onsite evacuation, dispersal or sheltering).

EAL #1

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 OWNER CONTROLLED AREA. Those events are adequately addressed by other EALs.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Although nuclear plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for the Offsite Response Organization (ORO) 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.

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.

EAL #2

This EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this EAL is to ensure that notifications for the airliner attack threat are made in a timely manner and that OROs 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 is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. 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 (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.

Site Specific

EAL #2

Airliner is the site specific term used for airliner.

Basis Reference(s):

1. NEI 99-01 Rev 5, HA4
2. DB-OP-02544, Security Events or Threats

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

INITIATING CONDITION:

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor.

   OR

2. A credible site specific Security Threat notification.

   OR

3. A validated notification from the NRC providing information of an aircraft threat.

Basis:

Generic

Note: Timely and accurate communication between security shift supervision and the control room is crucial for the implementation of 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 HA1, HS1 and HG1.

A higher initial classification could be made based upon the nature and timing of the Security Threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the site's Safeguards Contingency Plan and Emergency Plan.

EAL #1

Reference is made to site specific security shift supervision because these individuals are the designated personnel onsite 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 Safeguards Contingency Plan.

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU1

This threshold is based on site specific security plans. Site specific Safeguards Contingency Plans are based on guidance provided by NEI 03-12.

EAL #2

This threshold is included 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 site specific Safeguards Contingency Plan.

EAL #3

The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that Offsite Response Organizations (OROs) and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

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.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, HU4
2. DB-OP-02544, Security Events or Threats

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS2

INITIATING CONDITION:

Control Room evacuation has been initiated and plant control cannot be established.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. a. Control Room evacuation has been initiated.

   AND

   b. Control of the plant cannot be established within 15 minutes.

Basis:

Generic

The intent of this IC 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 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), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within the site specific time for transfer that the licensee has control of the plant from the remote shutdown panel.

The site specific time for transfer is based on analysis or assessments as to how quickly control must be reestablished without core uncovering and/or core damage. This time should not exceed 15 minutes without additional justification.

Site Specific

The 15 minute time for transfer is based on analysis or assessments as to how quickly control must be reestablished without core uncovering and/or core damage. The 15 minute time period starts when either 1) control of the plant is no longer maintained in the Control Room or 2) the last operator has left the Control Room, whichever comes first.

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS2

Basis Reference(s):

1. NEI 99-01 Rev 5, HS2
2. DB-OP-02508, Control Room Evacuation
3. DB-OP-02519, Serious Control Room Fire

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

INITIATING CONDITION:

Control Room evacuation has been initiated.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. Control Room evacuation has been initiated.

Basis:

Generic

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facilities may be necessary.

Site Specific

DB-OP-02519, Serious Control Room Fire, and DB-OP-02508, Control Room Evacuation, specify conditions under which Control Room evacuation may be necessary.

This EAL is only applicable when the decision has been made to evacuate the Control Room, and not when conditions that may require evacuation are being evaluated by referring to either DB-OP-02519 or DB-OP-02508.

Basis Reference(s):

1. NEI 99-01 Rev 5, HA5
2. DB-OP-02508, Control Room Evacuation
3. DB-OP-02519, Serious Control Room Fire

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

INITIATING CONDITION:

Natural or destructive phenomena affecting VITAL AREAS.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. a. Seismic event > **Operating Basis** Earthquake (OBE) as indicated by OBE alarm on seismic alarm panel C5764A.

   AND

   b. Earthquake confirmed by **ANY** of the following:

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

   OR

2. Tornado or high winds **> 90 mph** resulting in **EITHER** of the following:

   - VISIBLE DAMAGE to **ANY** structures in **Table H-1** areas containing safety systems or components.
   - Control Room indication of degraded performance of those safety systems.

   OR

3. Internal flooding in **Table H-1** areas resulting in **EITHER** of the following:

   - Electrical shock hazard that precludes access to operate or monitor safety equipment.
   - Control Room indication of degraded performance of those safety systems.

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

OR

4. Lake level **> 578 feet** on the land side of the dike resulting in **EITHER** of the following:

   - VISIBLE DAMAGE to **ANY** structures in **Table H-1** areas containing safety systems or components.
   - Control Room indication of degraded performance of those safety systems.

   OR

5. Turbine failure-generated PROJECTILES resulting in **EITHER** of the following:

   - VISIBLE DAMAGE to or penetration of **ANY** structures in **Table H-1** areas containing safety systems or components.
   - Control Room indication of degraded performance of those safety systems.

   OR

6. Vehicle crash resulting in **EITHER** of the following:

   - VISIBLE DAMAGE to **ANY** structures in **Table H-1** areas containing safety systems or components.
   - Control Room indication of degraded performance of those safety systems.

Table H-1: Safe Shutdown Vital Areas
• Control Room • Containment • Auxiliary Building • Intake Structure • Borated Water Storage Tank

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

Basis:

Generic

These EALs escalate from HU3 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 Room 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.

EALs #2 - #6

These EALs should specify site specific structures or areas that contain safety system, or component and functions required for safe shutdown of the plant. Site specific Safe Shutdown Analysis should be consulted for equipment and plant areas required to establish or maintain safe shutdown.

EAL #1

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.

This threshold should be based on site specific FSAR design basis. See EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, for information on seismic event categories,

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

EAL #2

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.

The high wind value should be based on site specific FSAR design basis as long as it is within the range of the instrumentation available for wind speed.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

EAL #3

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.

The site specific areas include those areas that contain systems required for safe shutdown of the plant, which are not designed to be partially or fully submerged. The plant's IPEEE may provide insight into areas to be considered when developing this EAL.

EAL #4

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 hurricane, flood, or seiche) that can also be precursors of more serious events.

EAL #5

This EAL addresses the threat to safety related equipment imposed by PROJECTILES generated by main turbine rotating component failures. Therefore, this EAL is consistent with the definition of an ALERT in that the potential exists for actual or substantial potential degradation of the level of safety of the plant.

The site specific list of areas should include all areas containing safety structure, system, or component, their controls, and their power supplies.

EAL #6

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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

Site Specific

Safe Shutdown Vital Areas are areas that house equipment that is needed to ensure safe shutdown of the plant. Personnel access to Safe Shutdown Vital Areas may be an important factor in monitoring and controlling equipment operability. Safe Shutdown Vital Areas include structures that are in contact with or immediately adjacent to the areas that actually contains the equipment of concern.

EAL #1

The Maximum Probable Earthquake is 0.08g. It is the conservatively determined earthquake and associated ground motion that might reasonably or probably be expected to occur at the nuclear plant site. The Maximum Probable Earthquake is similar to the Operating Basis Earthquake (OBE) terminology used by the NRC.

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 inventory 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."

De-energizing the Network Control Center makes the entire Seismic Monitoring System NON FUNCTIONAL. De-energizing an individual Motion Recorder makes that instrument NON-FUNCTIONAL. Refer to the following for applicable compensatory measures required to be in place prior to de-energization:

- TRM 8.3.3, Seismic Instrumentation
- RA-EP-02820, Earthquake
- DBRM-EMER-5003, Equipment Important to Emergency Response"

EAL #2

The wind speed threshold is based on station structural wind load design criteria for a wind velocity of 90 mph at 30 feet above the ground (the "fastest mile" American Society of Civil Engineers estimation for a 100 year recurrence interval).

Wind speed is obtained from meteorological data in the Control Room that is averaged over a 15 minute period to prevent instantaneous wind gusts or fluctuations from affecting the measurement.

EAL #4

A land side flood level of 578' (Flood Emergency Level) is the point at which station isolation procedures would be implemented due to access restrictions. This level remains below the

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

DBAB floor level of 579' and the station floor level of 585'.

EAL #6
This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes, or impact of PROJECTILES into a plant Safe Shutdown Vital Area.

Basis Reference(s):

1. NEI 99-01 Rev 5, HA1
2. RA-EP-02830 - Flooding, Attachment 1, DBNPS Elevation Profile
3. USAR 3.2.1, Seismic Classification
4. USAR 3.2.1.1, Definitions and 3.2.1.2, Seismic Class I Structures, Systems, and Equipment
5. USAR 3.3.1, Wind Criteria
6. USAR 3.4.1.1, Seismic Category I Systems and Equipment Below El. 583.5'
7. TRM 3.3.3.3, Seismic Instrumentation
8. RA-EP-02880, Internal Flooding

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU3

INITIATING CONDITION:

Natural or destructive phenomena affecting the PROTECTED AREA.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. a. Seismic event as indicated by **seismic instrument trigger annunciator** STA SEISMIC INSTR ON (7-6-A).

   AND

   b. Earthquake confirmed by **EITHER** of the following:

   • Earthquake felt in plant.

   • National Earthquake Center.

   OR

2. a. Tornado within the PROTECTED AREA.

   OR

   b. High winds **> 90 mph**.

   OR

3. Internal flooding in **Table H-1** areas that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode.

   OR

4. Lake level **> 578 feet** on the land side of the dike.

   OR

5. Turbine failure resulting in casing penetration or damage to turbine or generator seals.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU3

Table H-1: Safe Shutdown Vital Areas
• Control Room • Containment • Auxiliary Building • Intake Structure • Borated Water Storage Tank

Basis:

Generic

These EALs are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

EAL #1

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."

For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01g. This EAL should be developed on site specific basis. The method of detection can be based on instrumentation, validated by a reliable source, or operator assessment.

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

EAL #2

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

The high wind value should be based on site specific FSAR design basis as long as it is within the range of the instrumentation available for wind speed.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU3

EAL #3

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

The site specific areas include those areas that contain systems required for safe shutdown of the plant, which are not designed to be partially or fully submerged. The plant's IPEEE may provide insight into areas to be considered when developing this EAL.

EAL #4

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

EAL #5

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 HA4, HU4, and HU5.

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.

Site Specific

EAL #1

This threshold is based on the strong-motion seismograph actuation level which is the sensed earthquake threshold of 0.01 g.

EAL #2

The wind speed threshold is based on station structural wind load design criteria for a wind velocity of 90 mph at 30 feet above the ground (the "fastest mile" American Society of Civil Engineers estimation for a 100 year recurrence interval).

Wind speed is obtained from meteorological data in the Control Room that is averaged over a 15 minute period to prevent instantaneous wind gusts or fluctuations from affecting the

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU3

measurement.

EAL #4

A land side flood level of 578' (Flood Emergency Level) is the point at which station isolation procedures would be implemented due to access restrictions. This level remains below the DBAB floor level of 579' and the station floor level of 585'.

Basis Reference(s):

1. NEI 99-01 Rev 5, HU1
2. RA-EP-02830 - Flooding, Attachment 1, DBNPS Elevation Profile
3. USAR 3.3.1, Wind Criteria
4. USAR 3.4.1.1, Seismic Category I Systems and Equipment Below El. 583.5'
5. DP-OP-02007, Radwaste Alarm Panel 7 Annunciators
6. TRM 3.3.3.3, Seismic Instrumentation
7. RA-EP-02880, Internal Flooding

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

INITIATING CONDITION:

FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. FIRE or EXPLOSION resulting in EITHER of the following:

   - VISIBLE DAMAGE to **ANY** structures in **Table H-1** areas containing safety systems or components.

   - Control Room indication of degraded performance of those safety systems.

Table H-1: Safe Shutdown Vital Areas
• Control Room • Containment • Auxiliary Building • Intake Structure • Borated Water Storage Tank

Basis:

Generic

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.

This EAL should specify site specific structures or areas that contain safety system, or

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

component and functions required for safe shutdown of the plant. Site specific Safe Shutdown Analysis should be consulted for equipment and plant areas required to establish or maintain safe shutdown.

Site Specific

The definition of EXPLOSION includes any "catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components."

Based on this definition, a catastrophic failure of a steam line that causes damage affecting the operability of plant safety systems structures, systems, or components in any Table H-1 area is classifiable under this EAL.

Additionally, a catastrophic failure of a high pressure gas system (line breaks, tank ruptures, etc.) that causes damage affecting the operability of plant safety systems structures, systems, or components in any Table H-1 area is classifiable under this EAL.

NOTE: The lifting of a relief valve or the bursting of an installed rupture disc without any other damage is considered a normal system function and is not an EXPLOSION.

Finally, a catastrophic failure of energized electrical equipment such as breakers, transformers, etc. that causes damage affecting the operability of plant safety systems structures, systems, or components in any Table H-1 area (beyond that which is normally expected for an arc flash) is classifiable under this EAL.

Basis Reference(s):

1. NEI 99-01 Rev 5, HA2
2. Condition Report 09-69475, WHITE FINDING IDENTIFIED FOR INADEQUATE EMERGENCY CLASSIFICATION OF EVENT

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU4

INITIATING CONDITION:

FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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.

1. FIRE not extinguished within 15 minutes of Control Room notification or verification of a Control Room FIRE alarm in actual contact with or immediately adjacent to ANY of the Table H-1 areas.

Table H-1: Safe Shutdown Vital Areas
• Control Room • Containment • Auxiliary Building • Intake Structure • Borated Water Storage Tank

OR

2. EXPLOSION within the PROTECTED AREA.

Basis:

Generic

This EAL addresses the magnitude and extent of FIRES or EXPLOSIONS that may be potentially significant precursors of damage to safety systems. It addresses the FIRE or 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.

EAL #1

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU4

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).

The site specific list should be limited and applies to buildings and areas in actual contact with or immediately adjacent to VITAL AREAS or other significant buildings or areas. The intent of this IC is not to include buildings (i.e., warehouses) or areas that are not in actual contact with or immediately adjacent to VITAL AREAS. This excludes FIRES within administration buildings, waste-basket FIRES, and other small FIRES of no safety consequence. Immediately adjacent implies that the area immediately adjacent contains or may contain equipment or cabling that could impact equipment located in VITAL AREAS or the fire could damage equipment inside VITAL AREAS or that precludes access to VITAL AREAS.

EAL #2

This EAL addresses only those EXPLOSIONS of sufficient force to damage permanent structures or equipment within the PROTECTED AREA.

No attempt is made to assess the actual magnitude of the 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.

Site Specific

For the purposes of declaring an emergency event, the term "extinguished" means no visible flames.

The 15 minute time period starts either with:

1. A credible notification to the Control Room that a FIRE is occurring in the PROTECTED AREA, (verbal notification from the scene) or
2. Control Room indication of a fire detection system alarm/actuation (any single alarm or a single actuation of a system)

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU4

One of the following 4 things must occur within 15 minutes of the original FIRE indication / notification:

1. The alarm must be proven to be spurious (there is no FIRE),
2. The FIRE must be extinguished,
3. The FIRE must be proven not to be in or immediately adjacent to a Table H-1 area, or
4. The Unusual Event must be declared.

Because a FIRE detection alarm is assumed to be an indication of a FIRE unless it is proven to be spurious within the 15-minute period by personnel observation, and because the Containment is not normally accessible within this 15-minute period while operating at power, a single Containment FIRE detection alarm will require declaration of an UNUSUAL EVENT unless access can be gained to disprove the sensor alarm within 15 minutes, even without other confirmatory indications.

Areas directly associated with a FIRE that may be considered to have a hazardous environment (due to smoke or administrative controls awaiting ventilation and/or testing) do not warrant declaration of an ALERT under HA5. However, a hazardous atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under EAL HA5.

The definition of EXPLOSION includes any "catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components."

Based on this definition, a catastrophic failure of a steam line that causes damage to permanent structures, systems, or components inside the Protected Area is classifiable under this EAL.

Additionally, a catastrophic failure of a high pressure gas system (line breaks, tank ruptures, etc.) that cause damage to permanent structures, systems, or components inside the Protected Area is classifiable under this EAL.

NOTE: The lifting of a relief valve or the bursting of an installed rupture disc without any other damage is considered a normal system function and is not an EXPLOSION.

Finally, a catastrophic failure of energized electrical equipment such as breakers or transformers that cause damage to permanent structures, systems, or components inside the Protected Area (beyond that which is normally expected for an arc flash) is classifiable under this EAL.

Basis Reference(s):

1. NEI 99-01 Rev 5, HU2

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU4

2. Condition Report 09-66994, EMERGENCY ACTION LEVEL CLARIFICATION
3. Condition Report 09-69475, WHITE FINDING IDENTIFIED FOR INADEQUATE EMERGENCY CLASSIFICATION OF EVENT

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5

INITIATING CONDITION:

Access to a VITAL AREA is prohibited due to release of toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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 safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

1. 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.

Basis:

Generic

Gases in a VITAL 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 personnel ill effects from exposure, or operating experience with the hazards.

If the equipment in the stated area is 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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5

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 gases, 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.

Site Specific

During normal operating conditions when Containment is closed access is not required for any normal, abnormal or emergency operations. Therefore the Containment is not considered when evaluating this EAL for the presence of an asphyxiant. An asphyxiant, although dangerous to personnel because it displaces Oxygen, does not adversely affect the operations of equipment located in the Containment that is important to plant safety, whereas the presence of corrosive or flammable gases may adversely impact equipment operations. This clarification was written in response to questions about a Nitrogen leak into Containment during normal power operations.

A steam leak of significant magnitude that prevents access to the area should be considered an asphyxiant.

Basis Reference(s):

1. NEI 99-01 Rev 5, HA3
2. Condition Report 09-66994, EMERGENCY ACTION LEVEL CLARIFICATION

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

INITIATING CONDITION:

Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

1. Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS.

   OR

2. Report by local, county or state officials for evacuation or sheltering of site personnel based on an offsite event.

Basis:

Generic

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 IC 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.

Site Specific

During normal operating conditions when Containment is closed access is not required for any normal, abnormal or emergency operations. Therefore the Containment is not considered when evaluating this EAL for the presence of an asphyxiant. An asphyxiant, although dangerous to personnel because it displaces Oxygen, does not adversely affect the operations of equipment located in the Containment that is important to plant safety, whereas the presence of corrosive or flammable gases may adversely impact equipment

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

operations. This clarification was written in response to questions about a Nitrogen leak into Containment during normal power operations.

A steam leak of significant magnitude that prevents access to the area should be considered an asphyxiant.

Basis Reference(s):

1. NEI 99-01 Rev 5, HU3
2. Condition Report 09-66994, EMERGENCY ACTION LEVEL CLARIFICATION

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG6

INITIATING CONDITION:

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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 Guide exposure levels offsite for more than the immediate site area.

Basis:

Generic

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 classification level description for GENERAL EMERGENCY.

Site Specific

If another General Emergency EAL is met, then use that EAL to classify instead of this EAL.

Basis Reference(s):

1. NEI 99-01 Rev 5, HG2
2. EPA-400, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS6

INITIATING CONDITION:

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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 Guide exposure levels beyond the site boundary.

Basis:

Generic

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 classification level description for SITE AREA EMERGENCY.

Site Specific

If another Site Area Emergency EAL is met, then use that EAL to classify instead of this EAL.

Basis Reference(s):

1. NEI 99-01 Rev 5, HS3
2. EPA-400, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6

INITIATING CONDITION:

Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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 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 Guide exposure levels.

Basis:

Generic

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 classification level.

Site Specific

If another Alert EAL is met, then use that EAL to classify instead of this EAL.

Basis Reference(s):

1. NEI 99-01 Rev 5, HA6
2. EPA-400, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6

INITIATING CONDITION:

Other conditions exist which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4, 5, 6, D

EAL:

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 offsite response or monitoring are expected unless further degradation of safety systems occurs.

Basis:

Generic

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 classification level.

Site Specific

If another Unusual Event EAL is met, then use that EAL to classify instead of this EAL.

Basis Reference(s):

1. NEI 99-01 Rev 5, HU5

RECOGNITION CATEGORY
DRY FUEL STORAGE FACILITY (DFSF)

E-HU1

INITIATING CONDITION:

Damage to a loaded cask CONFINEMENT BOUNDARY.

OPERATING MODE APPLICABILITY:

Not Applicable

EAL:

1. Damage to a loaded cask CONFINEMENT BOUNDARY.

Basis:

Generic

An UNUSUAL EVENT in this IC 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.

The results of the ISFSI Safety Analysis Report (SAR) per NUREG 1536 or SAR referenced in the cask's Certificate of Compliance and the related NRC Safety Evaluation *Report* identify natural phenomena events and accident conditions that could potentially effect the CONFINEMENT BOUNDARY. This EAL addresses a dropped cask, a tipped over cask, EXPLOSION, PROJECTILE damage, FIRE damage or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.).

Site Specific

The Davis-Besse site specific term for ISFSI is Dry Fuel Storage Facility (DFSF).

The definition of CONFINEMENT BOUNDARY is "The barrier(s) between areas containing radioactive substances and the environment" and is not considered to be the concrete structure which is not a barrier since it is vented to the environment.

Basis Reference(s):

1. NEI 99-01 Rev 5, E-HU1
2. Davis-Besse Certified Safety Analysis Report (CSAR)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SG1

INITIATING CONDITION:

Prolonged loss of all offsite and all onsite AC power to emergency busses.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

1. a. Loss of **ALL** offsite and **ALL** onsite AC power to **BOTH** C-1 and D-1 busses.

   AND

   b. EITHER of the following:

   - Restoration of either the C-1 bus or the D-1 bus within **4 hours** is not likely.
   - Calculated Clad Temperature in **Region 4 - Severe Accident Region** (DB-OP-02000 Figure 2).

Basis:

Generic

Loss of all AC power to emergency busses 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 emergency busses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a GENERAL EMERGENCY.

The hours to restore AC power can be based on a site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout," as available. Appropriate allowance for offsite emergency response including evacuation of surrounding areas should be considered. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

This IC is specified to assure that in the unlikely event of a prolonged station blackout, 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 emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SG1

public protective actions.

In addition, under these conditions, Fission Product Barrier monitoring capability may be degraded.

Site Specific

Onsite power sources include the SBODG, which has no automatic start features, but must be capable of being manually started and aligned within 10 minutes following a station blackout event to supply power to C-1 or D-1 from the Control Room in order to be considered available. Therefore, even if it is not currently supplying power, if the SBODG and the associated switchgear needed to supply power to C-1 or D-1 are available, the SBODG should be considered capable of supplying power to C-1 or D-1 within 15 minutes.

The purpose of the Station Blackout Diesel Generator (SBODG) is to provide AC power to all systems required for coping with a station blackout as defined in 10 CFR 50.2. The SBODG meets the criteria for alternate AC power source as specified in Appendix B of NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiative Addressing Station Blackout at Light Water Reactors."

The SBODG fuel oil supply is separate from fuel oil supply for the station's EDGs. The SBODG fuel oil supply tank capacity is based on an eight hour supply with the SBODG at rated load. A minimum supply of four hours must be stored in this tank to meet the site's station blackout duration analysis conditions.

WCAP-14969-A Table 1 provides 2400° F as where a very rapid release of volatile fission products from fuel pellets occurs (departure from failure of fuel rod cladding) and Table 2 establishes core outlet temperatures of > 2000 F as indication of core melt.

It also states, "Analyses performed for the WOG ERGs for indication of inadequate core cooling concluded that the temperature indicated by the core exit thermocouples, especially during transient heatup conditions, is always several hundred degrees lower than the fuel rod cladding temperatures. Thus, an indicated temperature of 1200°F can be translated to a peak cladding temperature on the order of 1400°F."

The average incore thermocouple temperature and RCS pressure is used to determine whether Calculated Clad Temperature is in Region 4. This corresponds to clad temperatures, T_{clad} = 1800 F, reaching the site specific threshold indicative of imminent core damage.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SG1

Basis Reference(s):

1. NEI 99-01 Rev 5, SG1
2. USAR 8.3.1.1.4.2, Alternate AC Source - Station Blackout Diesel Generator
3. DB-OP-02000 Figure 2, Incore T/C Temperature vs. RC Pressure for Inadequate Core Cooling

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SS1

INITIATING CONDITION:

Loss of all offsite and all onsite AC power to emergency busses for 15 minutes or longer.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

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.

1. Loss of **ALL** offsite and **ALL** onsite AC power to **BOTH** C-1 and D-1 busses for **15 minutes** or longer.

Basis:

Generic

Loss of all AC power to emergency busses 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 emergency busses will lead to loss of fuel clad, RCS, and containment, thus this event can escalate to a GENERAL EMERGENCY.

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

Site Specific

Onsite power sources include the SBODG, which has no automatic start features, but must be capable of being manually started and aligned within 10 minutes following a station blackout event to supply power to C-1 or D-1 from the Control Room in order to be considered available. Therefore, even if it is not currently supplying power, if the SBODG and the associated switchgear needed to supply power to C-1 or D-1 are available, the SBODG should be considered capable of supplying power to C-1 or D-1 within 15 minutes.

Basis Reference(s):

1. NEI 99-01 Rev 5, SS1
2. Condition Report 09-67360, NEW EMERGENCY ACTION LEVELS: USE OF THE SBODG IMPROVEMENT OPPORTUNITY

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SA1

INITIATING CONDITION:

AC power capability to emergency busses reduced to a single source for 15 minutes or longer such that an additional single power source failure will result in loss of power to the emergency busses.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

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.

1. a. AC power to C-1 and D-1 busses is reduced to a single source for **15 minutes** or longer.

   AND

   b. Any additional single power source failure will result in loss of **ALL** power to **BOTH** C-1 and D-1 busses.

Basis:

Generic

The condition indicated by this IC is the degradation of the offsite and onsite AC power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power and loss of onsite emergency generators with only one train of emergency busses being backfed from the unit main generator, or the loss of onsite emergency generators with only one train of emergency busses being backfed from offsite power.

The subsequent loss of this single power source would escalate the event to a SITE AREA EMERGENCY in accordance with SSI.

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

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SA1

Site Specific

Onsite power sources include the SBODG, which has no automatic start features, but must be capable of being manually started and aligned within 10 minutes following a station blackout event to supply power to C-1 or D-1 from the Control Room in order to be considered available. Therefore, even if it is not currently supplying power, if the SBODG and the associated switchgear needed to supply power to C-1 or D-1 are available, the SBODG should be considered capable of supplying power to C-1 or D-1 within 15 minutes.

Basis Reference(s):

1. NEI 99-01 Rev 5, SA5
2. Condition Report 09-67360, NEW EMERGENCY ACTION LEVELS: USE OF THE SBODG IMPROVEMENT OPPORTUNITY

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU1

INITIATING CONDITION:

Loss of all offsite AC power to emergency busses for 15 minutes or longer.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

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.

1. Loss of **ALL** offsite AC power to **BOTH** C-1 and D-1 busses for **15 minutes** or longer.

Basis:

Generic

Prolonged loss of offsite 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 to emergency busses.

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

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, SU1

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SS2

INITIATING CONDITION:

Loss of all vital DC power for 15 minutes or longer.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

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.

1. **< 105 VDC** on the vital 125 VDC busses for **15 minutes** or longer.

Basis:

Generic

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.

Site specific bus voltage should be based on the minimum bus voltage necessary for the operation of safety related equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed. Typically the value for the entire battery set is approximately 105 VDC. For a 60 cell string of batteries the cell voltage is typically 1.75 Volts per cell. For a 58 string battery set the minimum voltage is typically 1.81 Volts per cell.

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

Site Specific

The 125/250 VDC Power Distribution is composed of the following [SD-007 2.1.1]:

- four approximately 1500 amp-hour 125 VDC batteries
- six 125 VDC 750 amp battery chargers
- two 125/250 VDC motor control centers (MCCs)
- two 125 VDC motor control centers (MCCs)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SS2

• four 125 VDC essential distribution panels

The system also supports a 120 VAC instrumentation power distribution system which is powered from 125 VDC / 120 VAC inverters.

The 125/250 VDC and 120 VAC Instrumentation System are designed to provide redundant and reliable power to components and systems that are essential to plant safety including the Reactor Protection System (RPS), the Safety Features Actuation System (SFAS), and the Anticipatory Reactor Trip System (ARTS) (SD-007 1.1.2.1).

The station batteries supply essential and nonessential 250/125 VDC loads and distribution panels during a loss of the battery charger supply. The batteries are sized to supply the station DC and AC instrument loads for a period of 1 hour without AC power (USAR 8.3.2.1.2).

The 60 cell station batteries are rated at approximately 1500 amp-hour capacity at 8-hour discharge rate to an end voltage of 1.75 volts per cell, i.e., 105 VDC battery voltage (SD-007 4.6.24).

Basis Reference(s):

1. NEI 99-01 Rev 5, SS3
2. USAR 8.3.2.1.2, DC Power Systems Description
3. SD-007, 125/250 VDC and 120 V Instrument AC System

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SG3

INITIATING CONDITION:

Automatic trip and all manual actions failed to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists.

OPERATING MODE APPLICABILITY:

1, 2

EAL:

1. a. An automatic reactor trip failed to shutdown the reactor as indicated by reactor power **> 5%.**

   AND

   b. **ALL** manual trip actions failed to shutdown the reactor as indicated by reactor power **> 5%.**

   AND

   c. **EITHER** of the following has occurred due to continued power generation:

   - Calculated Clad Temperature in **Region 3 or higher** (DB-OP-02000 Figure 2).
   - MFW, AFW, and MU-HPI PORV Cooling are unavailable.

Basis:

Generic

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 reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the safety systems are designed (typically 3 to 5% power).

For PWRs, the extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200° F or that the reactor vessel water level is below the top of active fuel.

Another consideration is the inability to initially remove heat during the early stages of this

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SG3

sequence. For PWRs, if emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist.

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 offsite intervention time.

Site Specific

Each of the redundant Aux Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions.

The >5% power portion of this EAL threshold was chosen to be an easily recognizable, on-scale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range. If the reactor is at very low power (<5%) and a reactor trip occurs, the operators must use indications other than reactor power being >5% to determine if the trip failed to shutdown the reactor.

EAL #1 .c bullet #1

WCAP-14969-A states, "Analyses performed for the WOG ERGs for indication of inadequate core cooling concluded that the temperature indicated by the core exit thermocouples, especially during transient heatup conditions, is always several hundred degrees lower than the fuel rod cladding temperatures. Thus, an indicated temperature of 1200°F can be translated to a peak cladding temperature on the order of 1400°F."

The average incore thermocouple temperature and RCS pressure is used to determine whether Calculated Clad Temperature is in Region 3. This corresponds to an extreme challenge to the ability to cool the core (T_{clad} = 1400° F threshold curve).

EAL #1.c bullet #2

An extreme challenge to heat removal is defined as a complete loss of MFW, AFW and MU-HPI PORV Cooling.

Basis Reference(s):

1. NEI 99-01 Rev 5, SG2
2. WCAP-14969-A, Westinghouse Owners Group Core Damage Assessment Guidance

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SG3

3. SD-15 2.1.2.2, Auxiliary Feedwater Pumps and Turbines

4. DB-OP-02000 Figure 2, Incore T/C Temperature vs. RC Pressure for Inadequate Core Cooling

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SS3

INITIATING CONDITION:

Automatic trip failed to shutdown the reactor and manual actions taken within the Controls Area (CA) failed to shutdown the reactor.

OPERATING MODE APPLICABILITY:

1, 2

EAL:

1. a. An automatic reactor trip failed to shutdown the reactor as indicated by reactor power **> 5%.**

   AND

   b. Manual trip actions taken within the Controls Area (CA) failed to shutdown the reactor as indicated by reactor power **> 5%.**

Basis:

Generic

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.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the safety systems are designed (typically 3 to 5% power).

Manual scram (trip) 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 (trip) actions are not considered successful if action away from the reactor control console is required to scram (trip) 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.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SS3

Site Specific

An automatic reactor trip signal is defined as reaching any one of the setpoints that should actuate the Reactor Protection System (RPS) including depressing either of the manual RPS trip pushbuttons. If either of the manual RPS trip pushbuttons is depressed or an RPS setpoint is reached and reactor power remains greater than 5% power, then an automatic reactor trip signal has failed to shutdown the reactor.

The Anticipatory Reactor Trip System (ARTS) also automatically generates a reactor trip signal (de-energizes CRD breakers) in a manner similar to the RPS. However, ARTS actuation setpoints are chosen such that ARTS actuation is not needed to ensure that Technical Specification Safety Limits are met (ARTS is designed to prevent lifting of the PORV on a secondary plant upset).. Therefore, a failure of ARTS alone to automatically trip the reactor does not necessarily challenge the Technical Specification Safety Limits and therefore is NOT classifiable under this EAL.

The manual trip actions taken in the Controls Area (as outlined in procedure DB-OP-02000) are depressing the manual RPS trip pushbuttons and de-energizing busses E2 and F2 from Control Room panel C5715. If any additional action outside the Control Room is required to shut down the reactor after an automatic trip signal is generated, then this EAL is met. Each of the redundant Aux Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions.

Manual scram (trip) actions are not considered successful if action away from the Controls Area is required to scram (trip) the reactor.

The >5% power portion of this EAL threshold was chosen to be an easily recognizable, on-scale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range. If the reactor is at very low power (<5%) and a reactor trip occurs, the operators must use indications other than reactor power being >5% to determine if the trip failed to shutdown the reactor.

Basis Reference(s):

1. NEI 99-01 Rev 5, SS2
2. SD-15 2.1.2.2, Auxiliary Feedwater Pumps and Turbines
3. NOP-OP-1002, Conduct of Operations

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SA3

INITIATING CONDITION:

Automatic trip failed to shutdown the reactor.

OPERATING MODE APPLICABILITY:

1, 2

EAL:

1. a. An automatic reactor trip failed to shutdown the reactor.

   AND

   b. Manual actions taken at the Controls Area (CA) successfully shut down the reactor as indicated by reactor power **< 5%.**

Basis:

Generic

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the safety systems are designed (typically 3 to 5% power).

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.

If the manual scram (trip) switches/pushbuttons on the Control Room console panels are considered an automatic input into the Reactor Protection System, a failure to scram (trip) without any other automatic input would make this threshold applicable.

This condition indicates failure of the automatic protection system to scram (trip) 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.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SA3

Site Specific

An automatic reactor trip signal is defined as reaching any one of the setpoints that should actuate the Reactor Protection System (RPS) including depressing either of the manual RPS trip pushbuttons. If either of the manual RPS trip pushbuttons is depressed or an RPS setpoint is reached and reactor power remains greater than 5% power, then an automatic reactor trip signal has failed to shutdown the reactor.

The Anticipatory Reactor Trip System (ARTS) also automatically generates a reactor trip signal (de-energizes CRD breakers) in a manner similar to the RPS. However, ARTS actuation setpoints are chosen such that ARTS actuation is not needed to ensure that Technical Specification Safety Limits are met (ARTS is designed to prevent lifting of the PORV on a secondary plant upset). Therefore, a failure of ARTS alone to automatically trip the reactor does not necessarily challenge the Technical Specification Safety Limits and therefore is NOT classifiable under this EAL.

The manual trip actions taken in the Controls Area (as outlined in procedure DB-OP-02000) are depressing the manual RPS trip pushbuttons and de-energizing busses E2 and F2 from Control Room panel C5715. Manual trip actions are not considered to be successful if any additional action outside the Control Room is required to shut down the reactor.

Each of the redundant Aux Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions.

The >5% power portion of this EAL threshold was chosen to be an easily recognizable, on-scale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range. If the reactor is at very low power (<5%) and a reactor trip occurs, the operators must use indications other than reactor power being >5% to determine if the trip failed to shutdown the reactor.

Basis Reference(s):

1. NEI 99-01 Rev 5, SA2
2. SD-15 2.1.2.2, Auxiliary Feedwater Pumps and Turbines
3. NOP-OP-1002, Conduct of Operations

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU3

INITIATING CONDITION:

Inadvertent criticality.

OPERATING MODE APPLICABILITY:

3, 4

EAL:

1. UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

Generic

This IC addresses inadvertent criticality events. This IC indicates a potential degradation of the level of safety of the plant, warranting an UNUSUAL EVENT emergency classification level. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

This condition can be identified using the startup rate monitor. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned control rod movements (such as shutdown bank withdrawal). These short term positive startup rates are the result of the increase In neutron population due to subcritical multiplication.

Site Specific

This condition can be identified using:

- Source Range startup rate channels for NI-1 or NI-2
- Intermediate Range channels NI-3 or NI-4
- Neutron flux wide range NI-5874B or NI-5875B
- Post Accident Monitoring System (PAMS) Gamma Metrics - Provides measure of flux level from shutdown (0.1 cps) through 200% power level

Basis Reference(s):

1. NEI 99-01 Rev 5, SU8
2. Regulatory Guide 8.12, Criticality Accident Alarm Systems
3. DB-OP-06912, Approach to Criticality

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SS4

INITIATING CONDITION:

Inability to monitor a SIGNIFICANT TRANSIENT in progress.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

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.

1. a. Loss of **>75%** of the following for **15 minutes** or longer:

   - Control Room Safety System annunciator panels 1 - 16 and annunciator printer.

   OR

   - Control Room Safety System indication on Post Accident Monitoring panels (C5798 and C5799).

   AND

   b. A SIGNIFICANT TRANSIENT (Table S-1) is in progress.

Table S-1: Significant Transients
• Automatic runback > 25% thermal reactor power
• Electrical load rejection > 25% full electrical load
• Reactor trip
• Safety Injection actuation
• Thermal power oscillations > 10%

   AND

   c. COMPENSATORY INDICATIONS (computer points) are unavailable.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SS4

Basis:

Generic

This IC 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.

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 Supervisor 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 UNUSUAL EVENT is based on SU5, "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 annunciators for this EAL should be limited to include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.)

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.

The specific indications should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled, to maintain the Reactor Coolant System intact, maintain the spent fuel cooled, and to maintain containment intact.

COMPENSATORY INDICATIONS in this context includes computer based information such as SPDS. This should include all computer systems available for this use depending on

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SS4

specific plant design and subsequent retrofits.

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

Site Specific

Control Room Safety System Design annunciators are defined as annunciator panels 1 through 16 and the annunciator printer.

Control Room Safety System indication is provided on Post Accident Monitoring panels C5798 and C5799.

Basis Reference(s):

1. NEI 99-01 Rev 5, SS6

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SA4

INITIATING CONDITION:

Loss of safety system annunciation or indication in the Control Room with either: (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY INDICATIONS are unavailable.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

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.

1. a. Loss of **>75%** of the following for 15 minutes or longer:

   - Control Room Safety System annunciator panels 1 - 16 and annunciator printer.

   OR

   - Control Room Safety System indication on Post Accident Monitoring panels (C5798 and C5799).

   AND

   b. EITHER of the following:

   • A SIGNIFICANT TRANSIENT **(Table S-1)** is in progress.

   • COMPENSATORY INDICATIONS (computer points) are unavailable.

Table S-1: Significant Transients
• Automatic runback > 25% thermal reactor power
• Electrical load rejection > 25% full electrical load
• Reactor trip
• Safety Injection actuation
• Thermal power oscillations > 10%

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SA4

Basis:

Generic

This IC 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.

Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

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 Supervisor 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 UNUSUAL EVENT is based on SU5, "Inability to Reach Required Operating Mode Within Technical Specification Limits."

Site specific annunciators or indicators for this EAL must include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent tad monitors, etc.).

COMPENSATQRY INDICATIONS in this context includes computer based information such as 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.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SA4

Site Specific

Control Room Safety System Design annunciators are defined as annunciator panels 1 through 16 and the annunciator printer.

Control Room Safety System indication is provided on Post Accident Monitoring panels C5798 and C5799.

Basis Reference(s):

1. NEI 99-01 Rev 5, SA4

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU4

INITIATING CONDITION:

Loss of safety system annunciation or indication in the Control Room for 15 minutes or longer.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

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.

1. Loss of > 75% of the following for 15 minutes or longer:

   - Control Room Safety System annunciator panels 1 - 16 and annunciator printer.

     OR

   - Control Room Safety System indication on Post Accident Monitoring panels (C5798 and C5799).

Basis:

Generic

This IC and its associated EAL are 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 (e.g., SPDS, plant computer, etc.).

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

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU4

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 CFR50.72. If the shutdown is not in compliance with the Technical Specification action, the UNUSUAL EVENT is based on SU5, "Inability to Reach Required Operating Mode Within Technical Specification Limits."

Site specific annunciators or indicators for this EAL must include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

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

Site Specific

Control Room Safety System Design annunciators are defined as annunciator panels 1 through 16 and the annunciator printer.

Control Room Safety System indication is provided on Post Accident Monitoring panels C5798 and C5799.

Basis Reference(s):

1. NEI 99-01 Rev 5, SU3

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU5

INITIATING CONDITION:

Inability to reach required operating mode within Technical Specification limits.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

1. Plant is not brought to required operating mode within Technical Specification LCO action statement time.

Basis:

Generic

Limiting Conditions for Operation (LCOs) require the plant to be brought to a required operating 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.

Site Specific

For cases were ""Enforcement Discretion" is received for a Technical Specification LCO, the Technical Specification LCO time applies as written for purpose of declaring the event.

If an "Emergency Technical Specifications" change is granted then apply the emergency Technical Specification LCO as approved for purpose of declaring the event.

Basis Reference(s):

1. NEI 99-01 Rev 5, SU2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU6

INITIATING CONDITION:

Loss of all onsite or offsite communications capabilities.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

1. Loss of ALL of the following onsite communication methods affecting the ability to perform routine operations:

   - Radios.
   - Plant page.
   - Plant telephone System (hardwired).

   OR

2. Loss of ALL of the following offsite communications methods affecting the ability to perform offsite notifications:

   - 4-Way Ringdown Circuit.
   - NRC Emergency Notification System - ENS.
   - NRC Health Physics Network - HPN.
   - Commercial telephones (hardwired and wireless).

Basis:

Generic

The purpose of this IC and its associated EALs 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 issues 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 Federal, State, and local authorities of plant problems. This EAL is intended to be used only

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU6

when extraordinary means (e.g., relaying of information from non-routine radio transmissions, individuals being sent to offsite locations, etc.) are being used to make communications possible.

Site specific list for onsite communications loss must encompass the loss of all means of communications [e.g., commercial telephones, sound powered phone systems, page party system (Gaitronics) and radios / walkie talkies] routinely used for operations.

Site specific list for offsite communications loss must encompass the loss of all means of communications with offsite authorities. This should include the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems that are routinely used for offsite emergency notifications.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, SU6

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU7

INITIATING CONDITION:

RCS leakage

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

1. Unidentified or pressure boundary leakage **> 10 gpm.**

   OR

2. Identified leakage **> 25 gpm.**

Basis:

Generic

This IC 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 IC. However, a relief valve that operates and fails to close per design should be considered applicable to this IC 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.

Site Specific

For reactor coolant system leaks of >250 gpm refer to the Fission Product Barrier Matrix EALs for RC – Reactor Coolant System.

Basis Reference(s):

1. NEI 99-01 Rev 5, SU5

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU9

INITIATING CONDITION:

Fuel clad degradation.

OPERATING MODE APPLICABILITY:

1, 2, 3, 4

EAL:

1. Letdown Monitor (RE 1998) **> 2.0E+06 cpm**.

   OR

2. RCS activity > **Technical Specification limits** as indicated by **ANY** of the following:

   - Dose equivalent 1-131 in the unacceptable region of Technical Specification 3.4.16 **Figure 3.4.16-1**.
   - **> 1 μCi/gm** dose equivalent 1-131 for **> 48 hours**.
   - **> 100 / $\bar{E}$μCi/gm** gross specific coolant activity.

Basis:

Generic

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.

EAL #1

This threshold addresses site specific radiation monitor readings that provide indication of a degradation of fuel clad integrity.

EAL #2

This threshold addresses coolant samples exceeding coolant Technical Specifications for transient iodine spiking limits.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - HOT

SU9

Site Specific

EAL #1

The Davis-Besse Nuclear Power Station historical highest failed fuel level of 0.007% corresponds to a reading of 1.4E+05 cpm on the Failed Fuel Monitor (RE 1998). The top of scale for RE 1998 is 1.0E+07. A monitor value of 2.0E+06 (~0.1% clad damage) was chosen for its ability to be recognized even though exceeding Technical Specification Limits could potentially result in much higher readings.

EAL #2

An UNUSUAL EVENT is only warranted when actual fuel clad damage is the cause of the elevated coolant sample (as determined by laboratory confirmation). However, fuel clad damage should be assumed to be the cause of elevated RCS activity unless another cause is known, e.g., RCS chemical decontamination evolution during shutdown results in high activity levels. This EAL and its associated applicability are based on Technical Specifications.

See also Fission Product Barrier Matrix EALs for FC – Fuel Clad.

Basis Reference(s):

1. NEI 99-01 Rev 5, SU4
2. Radiation Monitor Setpoint Manual, RE 1998 Failed Fuel Detector Rev 33
3. Improved Technical Specifications LCO 3.4.16, RCS Specific Activity

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CA1

INITIATING CONDITION:

Loss of all offsite and all onsite AC power to emergency busses for 15 minutes or longer.

OPERATING MODE APPLICABILITY:

5, 6, D

EAL:

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.

1. Loss of **ALL** offsite and **ALL** onsite AC power to **BOTH** C-1 and D-1 busses for **15 minutes** or longer.

Basis:

Generic

Loss of all AC power compromises all plant safety systems requiring electric power including Residual Heat Removal, 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 emergency busses, relative to that specified for the SITE AREA EMERGENCY EAL.

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

Site Specific

Onsite power sources include the SBODG, which has no automatic start features, but must be capable of being manually started and aligned within 10 minutes following a station blackout event to supply power to C-1 or D-1 from the Control Room in order to be considered available. Therefore, even if it is not currently supplying power, if the SBODG and the associated switchgear needed to supply power to C-1 or D-1 are available, the SBODG should be considered capable of supplying power to C-1 or D-1 within 15 minutes.

Consideration should be given to available loads necessary to remove decay heat or provide RCS makeup capability when evaluating loss of AC power to emergency busses.

Even though an emergency bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RCS makeup capability) are not available on

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CA1

the energized bus, the bus should not be considered available.

Basis Reference(s):

1. NEI 99-01 Rev 5, CA3
2. Condition Report 09-67360, NEW EMERGENCY ACTION LEVELS: USE OF THE SBODG IMPROVEMENT OPPORTUNITY

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU1

INITIATING CONDITION:

AC power capability to emergency busses reduced to a single source for 15 minutes or longer such that an additional single power source failure will result in loss of power to the emergency busses.

OPERATING MODE APPLICABILITY:

5, 6

EAL:

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.

1. a. AC power to C-1 and D-1 busses is reduced to a single source for **15 minutes** or longer.

   AND

   b. Any additional single power source failure will result in loss of **ALL** power to **BOTH** C-1 and D-1 busses.

Basis:

Generic

The condition indicated by this IC is the degradation of the offsite and onsite AC power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. The subsequent loss of this single power source would escalate the event to an ALERT in accordance with CAI.

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

Site Specific

Emergency Generators include the Station Blackout Diesel Generator.

Basis Reference(s):

1. NEI 99-01 Rev 5, CU3
2. Condition Report 09-67360, NEW EMERGENCY ACTION LEVELS: USE OF THE SBODG IMPROVEMENT OPPORTUNITY

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU2

INITIATING CONDITION:

Loss of required DC power for 15 minutes or longer.

OPERATING MODE APPLICABILITY:

5, 6

EAL:

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.

1. **< 105 VDC** on the required 125 VDC busses for **15 minutes** or longer.

Basis:

Generic

The purpose of this IC and its associated 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.

Plants will routinely perform maintenance on a train related basis during shutdown periods. It is intended that the loss of the operating (operable) train is to be considered.

Site specific bus voltage should be based on the minimum bus voltage necessary for the operation of safety related equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed. Typically the value for the entire battery set is approximately 105 VDC. For a 60 cell string of batteries the cell voltage is typically 1.75 Volts per cell. For a 58 string battery set the minimum voltage is typically 1.81 Volts per cell.

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

Site Specific

The 125/250 VDC Power Distribution is composed of the following [SD-007 2.1.1]:

- four approximately 1500 amp-hour 125 VDC batteries
- six 125 VDC 750 amp battery chargers
- two 125/250 VDC motor control centers (MCCs)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU2

• two 125 VDC motor control centers (MCCs)

• four 125 VDC essential distribution panels

The system also supports a 120 VAC Instrumentation Power Distribution System which is powered from 125 VDC / 120 VAC inverters.

The 125/250 VDC and 120 VAC Instrumentation System are designed to provide redundant and reliable power to components and systems that are essential to plant safety including the Reactor Protection System (RPS), the Safety Features Actuation System (SFAS), and the Anticipatory Reactor Trip System (ARTS) (SD-007 1.1.2.1).

The station batteries supply essential and nonessential 250/125 VDC loads and distribution panels during a loss of the battery charger supply. The batteries are sized to supply the station DC and AC instrument loads for a period of 1 hour without AC power (USAR 8.3.2.1.2).

The 60 cell station batteries are rated at approximately 1500 amp-hour capacity at 8-hour discharge rate to an end voltage of 1.75 volts per cell, i.e., 105 VDC battery voltage (SD-007 4.6.24).

Basis Reference(s):

1. NEI 99-01 Rev 5, CU7
2. USAR 8.3.2.1.2, DC Power Systems Description
3. SD-007, 125/250 VDC and 120 V Instrument AC System

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU3

INITIATING CONDITION:

Inadvertent criticality.

OPERATING MODE APPLICABILITY:

5, 6

EAL:

1. UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

Generic

This IC addresses inadvertent criticality events. This IC indicates a potential degradation of the level of safety of the plant, warranting an UNUSUAL EVENT emergency classification level. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

This condition can be identified using the startup rate monitor. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned control rod movements (such as shutdown bank withdrawal). These short term positive startup rates are the result of the increase in neutron population due to subcritical multiplication.

Site Specific

This condition can be identified using:

- Source Range startup rate channels for NI-1 or NI-2
- Intermediate Range channels NI-3 or NI-4
- Neutron flux wide range NI-5874B or NI-5875B
- Post Accident Monitoring System (PAMS) Gamma Metrics - Provides measure of flux level from shutdown (0.1 cps) through 200% power level

Basis Reference(s):

1. NEI 99-01 Rev 5, CU8
2. Regulatory Guide 8.12, Criticality Accident Alarm Systems
3. DB-OP-06912, Approach to Criticality

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU6

INITIATING CONDITION:

Loss of all onsite or offsite communications capabilities.

OPERATING MODE APPLICABILITY:

5, 6, D

EAL:

1. Loss of ALL of the following onsite communication methods affecting the ability to perform routine operations:

   - Radios.
   - Plant page.
   - Plant telephone System (hardwired).

   OR

2. Loss of ALL of the following offsite communications methods affecting the ability to perform offsite notifications:

   - 4-Way Ringdown Circuit.
   - NRC Emergency Notification System - ENS.
   - NRC Health Physics Network- HPN.
   - Commercial telephones (hardwired and wireless).

Basis:

Generic

The purpose of this IC and its associated EALs 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 issues 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 Federal, State, and local authorities of plant issues. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions,

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU6

individuals being sent to off-site locations, etc.) are being used to make communications possible.

Site specific list for onsite communications loss must encompass the loss of all means of communications [e.g., commercial telephones, sound powered phone systems, page party system (Gaitronics) and radios / walkie talkies].

Site specific list for offsite communications loss must encompass the loss of all means of communications with offsite authorities. This should include the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

Site Specific

None

Basis Reference(s):

1. NEI 99-01 Rev 5, CU6

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CG7

INITIATING CONDITION:

Loss of RPV inventory affecting fuel clad integrity with containment challenged.

OPERATING MODE APPLICABILITY:

5, 6

EAL:

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.

1. a. RPV level **< 567 feet 7 inches** (top of active fuel) for **30 minutes** or longer.

   AND

   b. **ANY Table C-1** containment challenge indications.

   OR

2. a. RCS level cannot be monitored with core uncovery for 30 minutes or longer.

   AND

   b. Loss of RPV inventory as indicated by **ANY** of the following:

   - Containment Radiation Monitor (RE 4596A or B) **> 16 R/hr.**
   - Refueling bridge portable area radiation monitor **> 30 R/hr.**
   - Erratic source range monitor indication.
   - UNPLANNED level rise in Containment sumps, Aux Building sumps, BWST or RCDT.

   AND

(Continued Next Page)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CG7

(Continued From Previous Page)

c. ANY **Table C-1** containment challenge indications.

Table C-1: Containment Challenge Indications
• CONTAINMENT CLOSURE not established. • Hydrogen concentration > 4% inside containment. • UNPLANNED rise in containment pressure.

Basis:

Generic

This IC represents the inability to restore and maintain RPV level to above the top of active fuel with containment challenged. Fuel damage is probable if RPV level cannot be restored, as available decay heat will cause boiling, further reducing the RPV level. With the 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. The GENERAL EMERGENCY is declared on the occurrence of the loss or IMMINENT loss of function of all three barriers.

These EALs are based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal," SECY 91-283, "Evaluation of Shutdown and Low Power Risk Issues," NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," and NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management."

A number of variables can have a significant impact on heat removal capability challenging the Fuel Clad Barrier. Examples include: mid-loop, reduced level/flange level, head in place, cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, and U-tube draining.

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

If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute core uncovery time limit, then escalation to GENERAL EMERGENCY would not occur.

Site shutdown contingency plans typically provide for re-establishing CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory functions.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CG7

core uncovery could result in an explosive mixture of dissolved gases in containment. However, 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.

EAL #2

Sump and tank 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.

In the Cold Shutdown mode, normal RCS level and 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 RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank 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.

As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

This EAL should conservatively estimate a site specific dose rate setpoint indicative of core uncovery (i.e., level at TOAF).

For PWRs, 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.

Site Specific

EAL #2.b bullet #1
The Containment Radiation Monitor reading is based on calculation EP-EALCALC-DB-0704 Rev 0.

EAL #2.b bullet #2
The Refueling bridge portable area radiation monitor is installed as soon as practical after entering Mode 5 and its reading is based on calculation EP-EALCALC-DB-0704 Rev 0.

Radiation monitor AMP-100 or equivalent is required on the Main Fuel Handling Bridge in Containment during cold shutdown and refueling modes by DB-HP-01152, High Exposure Work

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SYSTEM MALFUNCTIONS - COLD

CG7

EAL #2.b bullet #3

Erratic Source Range Monitors indication can be identified using:

- Source Range startup rate channels for NI-1 or NI-2
- Intermediate Range channels NI-3 or NI-4
- Neutron flux wide range NI-5874B or NI-5875B
- Post Accident Monitoring System (PAMS) Gamma Metrics - Provides measure of flux level from shutdown (0.1 cps) through 200% power level

Table C-1

If hydrogen concentration reaches or exceeds 4% in an oxygen rich environment, a potentially combustible mixture exists.

Hydrogen monitors although available at all times are not in service during normal operations. They are started per DB-OP-02000 Table 3. Action is required if the measured concentration reaches 0.6% (notify TSC) and 3% (place systems in service to control concentration).

The hydrogen monitor measurement range is 0 - 10 volume percent.

Basis Reference(s):

1. NEI 99-01 Rev 5, CG1
2. EP-EALCALC-DB-0704 Rev 0, Radiation Monitor Readings for Core Uncovery During Refueling (EALs CG7 and CS7)
3. DB-OP-02000 Table 3, Containment Monitoring and Control
4. DB-HP-01152, High Exposure Work

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CS7

INITIATING CONDITION:

Loss of RPV inventory affecting core decay heat removal capability.

OPERATING MODE APPLICABILITY:

5, 6

EAL:

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.

1. a. CONTAINMENT CLOSURE not established.

   AND

   b. Loss of RPV inventory as indicated by RCS level (LI 10596) **< 0.4 feet.**

   OR

2. a. CONTAINMENT CLOSURE established.

   AND

   b. RPV level **< 567 feet 7 inches** (top of active fuel).

   OR

3. a. RCS level cannot be monitored for 30 minutes or longer.

   AND

   b. Loss of RPV inventory as indicated by **ANY** of the following:

   - Containment Radiation Monitor (RE 4596A or B) **> 16 R/hr.**
   - Refueling bridge portable area radiation monitor **> 30 R/hr.**
   - Erratic source range monitor indication.
   - UNPLANNED level rise in Containment sumps, Aux Building sumps, BWST or RCDT.

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CS7

Basis:

Generic

Under the conditions specified by this IC, continued decrease in RCS/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.

EAL #1

6'i below the bottom ID of the RCS Loop should be the level equal to 6" below the bottom of the RPV loop penetration (not the low point of the loop). PWRs unable to measure this level should choose the first observable point below the bottom ID of the loop as the EAL value. If a water level instrument is not available such that the PWR EAL value cannot be determined, then EAL 3 should be used to determine if the IC has been met.

EAL #3

In the Cold Shutdown mode, normal RCS level and 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 RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank 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.

As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

This EAL should conservatively estimate a site specific dose rate setpoint indicative of core uncovery (i.e., level at TOAF).

For PWRs, 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.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CS7

Site Specific

EAL #1 .b

RCS level cannot be measured below the 571 feet elevation (0.4 feet on LI 10596) which is centerline of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually.

EAL #3.b bullet #1
The Containment Radiation Monitor reading is based on calculation EP-EALCALC-DB-0704 Rev 0.

EAL #3.b bullet #2

The Refueling bridge portable area radiation monitor is installed as soon as practical after entering Mode 5 and its reading is based on calculation EP-EALCALC-DB-0704 Rev 0.

Radiation monitor AMP-100 or equivalent is required on the Main Fuel Handling Bridge in Containment during cold shutdown and refueling modes by DB-HP-01152, High Exposure Work

EAL #3.b bullet #3

Erratic Source Range Monitors indication can be identified using:

- Source Range startup rate channels for NI-1 or NI-2
- Intermediate Range channels NI-3 or NI-4
- Neutron flux wide range NI-5874B or NI-5875B
- Post Accident Monitoring System (PAMS) Gamma Metrics - Provides measure of flux level from shutdown (0.1 cps) through 200% power level

Basis Reference(s):

1. NEI 99-01 Rev 5, CS1
2. EP-EALCALC-DB-0704 Rev 0, Radiation Monitor Readings for Core Uncovery During Refueling (EALs CG7 and CS7)
3. DB-HP-01152, High Exposure Work

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CA7

INITIATING CONDITION:

Loss of RPV inventory.

OPERATING MODE APPLICABILITY:

5, 6

EAL:

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.

1. Loss of RPV inventory as indicated by RCS level (LI 10596) **< 0.4 feet**.

   OR

2. a. RCS level cannot be monitored for 15 minutes or longer.

      AND

   b. Loss of RPV inventory as indicated by UNPLANNED level rise in Containment sumps, Aux Building sumps, BWST or RCDT.

Basis:

Generic

These EALs serve as precursors to a loss of ability to adequately cool the fuel. 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 emergency classification level of an ALERT.

EAL #1

The PWR Bottom ID of the RCS Loop Setpoint was chosen because at this level, remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The Bottom ID of the RCS Loop Setpoint should be the level equal to the bottom of the RPV loop penetration (not the low point of the loop).

The inability to restore and maintain level after reaching this setpoint would be indicative of a failure of the RCS barrier.

EAL #2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CA7

In the Cold Shutdown mode, normal RCS level and 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 RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank 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 CS7 duration. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CG7 basis. Therefore this EAL meets the definition for an ALERT.

Site Specific

EAL #1

RCS level cannot be measured below the 571 feet elevation (0.4 feet on LI 10596) which is centerline of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually.

Basis Reference(s):

1. NEI 99-01 Rev 5, CA1

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU7

INITIATING CONDITION:

RCS leakage.

OPERATING MODE APPLICABILITY:

5

EAL:

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.

1. RCS leakage results in the inability to maintain or restore RCS level **> Procedure Established Minimum Level** for **15 minutes** or longer.

Basis:

Generic

This IC 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.

The difference between CU7 and CU8 deals with the RCS conditions that exist between Cold Shutdown and Refueling modes. In the Refueling mode the RCS is not intact and RPV level and inventory are monitored by different means. In Cold Shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available.

Site Specific

Normally, the RCS level band is established by DB-OP-06904. If the pressurizer level decrease is not attributable to a malfunction of pressurizer level control or the makeup system and decay heat removal is in service, operators enter DB-OP-02527. Otherwise, operators may enter DB-OP-02522.

Basis Reference(s):

1. NEI 99-01 Rev 5, CU1
2. DB-OP-02004, Reactor Coolant Alarm Panel 4 Annunciators
3. DB-OP-02522, Small RCS Leaks
4. DB-OP-02527, Loss of Decay Heat Removal
5. DB-OP-06904, Shutdown Operations

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU8

INITIATING CONDITION:

UNPLANNED loss of RCS inventory.

OPERATING MODE APPLICABILITY:

6

EAL:

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.

1. UNPLANNED RCS level as indicated by either of the following:

   - RCS water level drop below the RPV flange **(< 80 inches)** for **15 minutes** or longer when the RCS level band is established above the RPV flange.

   - RCS water level drop below the RCS level band for **15 minutes** or longer when the RCS level band is established **below** the RPV flange.

OR

2. a. RCS level cannot be monitored.

   AND

   b. Loss of RPV inventory as indicated by UNPLANNED level rise in Containment sumps, Aux Building sumps, BWST or RCDT.

Basis:

Generic

This IC 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 RCS 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 RCS water level for the given evolution (if the planned RCS 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

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU8

should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists.

The difference between CU7 and CU8 deals with the RCS conditions that exist between Cold Shutdown and Refueling modes. In Cold Shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode the RCS is not intact and RPV level and inventory are monitored by different means.

EAL #1

This EAL involves a decrease in RCS level below the top of 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 RU2.1 until such time as the level decreases to the level of the vessel flange.

If RCS level continues to decrease and reaches the Bottom ID of the RCS loop, then escalation to CA7.

EAL #2

This EAL addresses conditions in the Refueling mode when normal means of core temperature indication and RCS level indication may not be available. Redundant means of RCS 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 RCS inventory event, the operators would need to determine that RCS inventory loss was occurring by observing sump and tank level changes. Sump and tank 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.

Site Specific

EAL #1

The Reactor Vessel flange is at 577 feet 7 inches (-80 inches indicated). RCS level is normally monitored using the following instruments:

- LI 10596
- LI 10577A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

Reactor vessel level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure.

Modes: 1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling, D – Defueled

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU8

EAL #2

There is no redundant means of RPV level indication normally installed when Mode 6 is entered.

Basis Reference(s):

1. NEI 99-01 Rev 5, CU2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CA10

INITIATING CONDITION:

Inability to maintain plant in Cold Shutdown.

OPERATING MODE APPLICABILITY:

5, 6

EAL:

1. UNPLANNED RCS temperature **> 200° F** for the specified duration on **Table C-2**.

Table C-2: RCS Reheat Duration Thresholds		
RCS	**Containment Closure**	**Duration**
Intact (but not RCS Reduced Inventory)	N/A	> 60 minutes*
Not Intact or RCS Reduced Inventory	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, then this EAL is not applicable.		

OR

2. RCS pressure rise **> 10 psig** due to a loss of RCS cooling (this EAL does not apply in solid plant conditions).

Basis:

Generic

For PWRs, this IC and its associated EALs are based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as pressurization, vortexing, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show that there are sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

A loss of Technical Specification components alone is not intended to constitute an ALERT. The same is true of a momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CA10

EAL #1

The RCS Reheat Duration Threshold table 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. RCS should be considered intact when the RCS pressure boundary is established (e.g., no freeze seals, nozzle dams installed or SG manways removed). The status of CONTAINMENT CLOSURE in this condition 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 RCS Reheat Duration Threshold table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during Refueling and Cold Shutdown modes when CONTAINMENT CLOSURE is established but RCS is not intact. As discussed above, RCS should be assumed to be intact when the RCS pressure boundary is established (e.g., no freeze seals, nozzle dams installed or SG manways removed). 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," (discussed earlier in this basis) and is believed to be conservative given that a low pressure Containment Barrier to fission product release is established.

Finally, the RCS Reheat Duration Threshold table also addresses the complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when neither CONTAINMENT CLOSURE is established nor RCS is intact. RCS is intact when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or nozzle dams). No delay time is allowed because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

The note (*) indicates that this EAL is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the specified time frame.

EAL #2

The 10 psi pressure increase addresses situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. The RCS pressure setpoint chosen should be 10 psi or the lowest pressure that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psi.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CA10

Site Specific

Under worst case conditions (shutdown 2 days and RCS level 26" above the centerline of the hot leg) time to boil is ~ 15 minutes per C-NSA-049.02-020. Core uncovery can occur in as little as 90 minutes per DB-OP-02527.

EAL #1

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification Cold Shutdown temperature limit of (200° F):

- Selected incore temperature or temporary thermocouples
- TIRC4B2, REACTOR COOLANT T-COLD WIDE RANGE LOOP 1
- TIRC4A2, REACTOR COOLANT T-COLD WIDE RANGE LOOP 2
- TIRC3B5, REACTOR COOLANT T-HOT LOOP 1
- TIRC3B6, REACTOR COOLANT T-HOT LOOP 1
- TIRC3A5, REACTOR COOLANT T-HOT LOOP 2
- TIRC3A6, REACTOR COOLANT T-HOT LOOP 2
- DHR display on SPDS

EAL #2

The following instrumentation is capable of providing indication of a 10 psig increase in RCS pressure:

- PI 6365B, LOOP 1 PRESSURE
- PI 6365A, LOOP 2 PRESSURE
- PRS RC2A1, LOOP 2 WIDE RANGE PRESSURE
- PI RC2A6, LOW RANGE PRESSURE

Basis Reference(s):

1. NEI 99-01 Rev 5, CA4
2. DB-OP-06904, Shutdown Operations
3. DB-OP-02527, Loss of Decay Heat Removal
4. C-NSA-049.02-020

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU10

INITIATING CONDITION:

Loss of decay heat removal capability.

OPERATING MODE APPLICABILITY:

5, 6

EAL:

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.

1. RCS temperature **> 200° F** due to a loss of decay heat removal capability.

   OR

2. Loss of ALL RCS temperature and RCS level indication for **15 minutes** or longer.

Basis:

Generic

This IC is 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 RCS inventory. Since the RCS usually remains intact in the Cold Shutdown mode a large inventory of water is available to keep the core covered.

Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for typically 100 hours (site specific) or longer 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 will occur if required.

During refueling the level in the RPV will normally be maintained above the RPV flange. Refueling evolutions that decrease water level below the RPV 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.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU10

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 of Refueling modes, EAL #2 would result in declaration of an UNUSUAL EVENT if both temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication.

Site Specific

Under worst case conditions (shutdown 2 days and RCS level 26" above the centerline of the hot leg) time to boil is - 15 minutes per C-NSA-049.02-020. Core uncovery can occur in as little as 90 minutes per DB-OP-02527.

EAL #1

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification Cold Shutdown temperature limit of (200° F):

- Selected incore temperature or temporary thermocouples
- TIRC4B2, REACTOR COOLANT T-COLD WIDE RANGE LOOP 1
- TIRC4A2, REACTOR COOLANT T-COLD WIDE RANGE LOOP 2
- TIRC3B5, REACTOR COOLANT T-HOT LOOP 1
- TIRC3B6, REACTOR COOLANT T-HOT LOOP 1
- TIRC3A5, REACTOR COOLANT T-HOT LOOP 2
- TIRC3A6, REACTOR COOLANT T-HOT LOOP 2
- DHR display on SPDS

EAL #2

The Reactor Vessel flange is at 577 feet 7 inches (-80 inches indicated). RCS level is normally monitored using the following instruments:

- LI 10596
- LI 10577A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS - COLD

CU10

Reactor vessel level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure.

- There is no redundant means of RPV level indication normally installed when Mode 6 is entered.

Basis Reference(s):

1. NEI 99-01 Rev 5, CU4
2. DB-OP-06904, Shutdown Operations
3. DB-OP-02527, Loss of Decay Heat Removal
4. C-NSA-049.02-020

Summary of Changes

Revision 0 – Initial Issuance – September 30, 2009

Revision 1 – February 4, 2010

1. Added new section "Summary of Changes" to capture changes made to the Basis Document.
2. Page 7 – Corrected Operating Mode Applicability table to reflect that Cold Shutdown, Reactivity Condition is <0.99 instead of "N/A". Corrective Action 09-65851-01
3. Page 32 – RC5 – Added statement that when PORV cooling is initiated that the condition for this EAL will likely be met. Corrective Action 09-67382-01
4. Page 33 – RC5 – Added to the Bases Reference section Condition Report 09-67382.
5. Page 89 – HA4 – Provide additional information describing events that would be considered an explosion in the Site Specific section to reflect lesson learned from the June 25, 2009, After-the-Fact Alert declaration. Condition Report 09-69475
6. Page 89 – HA4 - Added to the Bases Reference section Condition Report 09-69475.
7. Page 91 – HU4 - Revised wording associated with a steam line break Corrective Action 09-66994-02 lessons learned during operator training.
8. Page 91 – HU4 - Provide additional information describing events that would be considered an explosion in the Site Specific section to reflect lesson learned from the June 25, 2009, After-the-Fact Alert declaration. Condition Report 09-69475.
9. Page 92 – HU4 - Added to the Bases Reference section Condition Reports 09-66994 and 09-69475.
10. Page 94 – HA5 - Added wording to reflect that a significant steam leak that prevents access to the area should be considered an asphyxiant.
11. Page 94 – HA5 – Added to the Bases Reference section Condition Report 09-66994.
12. Page 96 – HU5 – Added wording to reflect that a significant steam leak that prevents access to the area should be considered an asphyxiant.
13. Page 96 – HU5 – Added to the Bases Reference section Condition Report 09-66994.
14. Page 106 – SA1 – Added to the Site Specific section words that clarify that the Station Blackout Diesel Generator (SBODG) is an emergency generator. Corrective Action 09-67360-01
15. Page 107 – SA1 – Added to the Bases Reference section Condition Report 09-67360.
16. Page 133 – CU1 – Added to the Site Specific section words that clarify that the Station Blackout Diesel Generator (SBODG) is an emergency generator. Corrective Action 09-67360-01
17. Page 133 – CU1 – Added to the Bases Reference section Condition Report 09-67360.
18. Pages 154 (CA10), 156 (CU10) and 157 (CU10) removed revision number from the C-NSA-049.02-020 calculation reference. These changes were made both the Site Specific and Bases Reference sections. This calculation is updated periodically and as part of the engineering process evaluates the impact on the emergency action levels.
19. Removed footer that provided Mode information from the cover and index pages.

Revision 2

1. Page 2 Added new section "Emergency Action Levels with Embedded Time Requirements" to Table of Contents.
2. Pages 3 -16 removed footer describing Modes.
3. Page 6 – Clarified expectation that once a higher emergency classification EAL is exceed time should not be expended on evaluating EALs for lower classifications. This was added in response to a lesson learned during 2010 EAL training.
4. Page 7 - 8 Added new section "Emergency Action Levels with Embedded Time Requirements" to discussion expectation for declaring EALs with "embedded" time requirements. This was added in response to OE30423 and Condition Report 10-74069.
5. Page 16 – Added Station Blackout Diesel Generator (SBODG) to the list of acronyms to reflect use in basis document.
6. Pages 97 – 100 added to the Site Specific section a statement that if another EAL is met, then use that EAL to classify instead of this EAL. This was added in response to a lesson learned during 2010 EAL training.
7. Page 101 – Site Specific information was added to clarify that the concrete vault that houses the container for the fuel assembly is not the CONFINEMENT BOUNDARY. This was added in response to a lesson learned during 2010 EAL training.
8. Page 103 and 105 information was added to the Site Specific section of SG1 and SS1 that the SBODG is considered an emergency diesel generator. This was added in response to a lesson learned during 2010 EAL training.
9. Page 105 and 107 Site Specific information found in SG1 that consideration should be given to available loads necessary to remove decay heat or provide RCS makeup capability when evaluating loss of AC power to emergency busses. Even though an emergency bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RCS makeup capability) are not available on the energized bus, the bus should not be considered available. This is also applicable to SS1 and SA1 was added to their Site Specific information. This was added in response to a lesson learned during 2010 EAL training.
10. Page 129 – For EAL SU7 words were added to the Site Specific section to direct individual to the Fission Procedure Barrier tab if RCS leakage is >250 gpm. This was added in response to a lesson learned during 2010 EAL training.
11. Page 131 - For EAL SU9 words were added to the Site Specific section to direct individual to the Fission Procedure Barrier tab if continued degradation of the fuel clad occurs. This was added in response to a lesson learned during 2010 EAL training.

Revision 3

1. Page 9 – changed five to six EAL matrix tables to account for the Dry Fuel Storage recognition category.
2. Page 28 – Added wording that Dose Equivalent I-131 is not a real time measurement and that once the threshold is met it will continue until proven otherwise.
3. Page 30 and 31 – Words were added to the Site Specific section to clarify that radiation levels greater that 15 R/hr may occur without physical loss of the RCS barrier. This was a lesson learned from the April 5, 2011 Dry Run.

4. Page 70 – Provided description of what constitutes the Protected Area boundary for EAL HS1. This was added as a result of feedback from operator training.
5. Page 75 – Clarified the wording used to describe when the time starts for evacuation of the Control Room by adding numbers to the two conditions.
6. Page 77 added DB-OP-02519, Serious Control Room Fire as a reference in the Site Specific and Reference Sections. This change was based on feedback from operator training.
7. Page 82 added to the Site Specific section for EAL HA3 "When the Seismic Monitoring Cabinet key locked switch is placed in OFF, all OBE and SSE alarms and cabinet recording functions (TNC 8.3.3 functions 1 and 3 and the DBRM-EMER-5003 Seismic Event Detection function) are nonfunctional. Therefore, the Compensatory Measures listed in DBRM-EMER-5003 for a loss of the Seismic Event Detection function are required to be employed by the operators to determine if an earthquake occurs and to determine the magnitude of the event in a timely manner." (CA 2012-04838-08)
8. Pages 83 and 87 – Added RA-EP-02880, Internal Flooding to the Basis Reference sections for EALs HA3 and HU3.
9. Page 91 – Added words explaining that a single Containment fire alarm will require declaration of an UNUSUAL EVENT if it can't be disproved in <15 minutes.
10. Pages 105, 106, 108, 133 and 134 - Added clarification on when to consider the SBODG as available to provide AC power to busses C-1 and D-1.
11. Page 107 – Added words that were missing from the INITIATING CONDITION.
12. Pages 115 and 117 – Added clarification as to what constitutes an "automatic trip" to EALs SS3 and SA3.
13. Pages 161 and 162 - Updated Summary of Changes to reflect those changes made in Revision 3.

Revision 4

1. Page 23 - FC2, and Page 37 - CT2, added guidance to Site Specific Basis to clarify time after reactor shutdown to be used when evaluating fission product barrier integrity using Graph F-1 and F-2 (to be consistent with guidance in DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE). (CA 2012-14109-1)
2. Page 23 - FC2, and Page 37 - CT2, added guidance to Site Specific Basis that when evaluating fission product barrier integrity using Graph F-1 and F-2 , if time after reactor trip is less than one hour (or reactor is still critical), to evaluate using the one hour after shutdown point on the curve. (CA 2012-14109-2)
3. Page 65, under RU2, clarification was added to EAL #2, that radiation survey results of >1000x NORMAL LEVELS is in reference to whole body dose rates (or 30 cm) and not contact dose rates. (Notification 600801552)
4. Pages 104, 106, 109 and 110 in the Site Specific Sections of SG1, SS1, SA1 and SU1 added a new first paragraph:

   The reason busses C-1 and D-1 are specified in this EAL is because they supply the electrical power to the Emergency Core Cooling Systems (ECCSs) needed to keep the Fuel Cladding Fission Product Barrier intact. Therefore, when determining if this EAL is met, consideration must also be given to the availability of these ECCS components. Using this logic, if either High Pressure Injection (HPI) Pump 1 or Low Pressure

Injection (LPI) Pump 1 is not capable of providing water to the reactor pressure vessel (RPV), then bus C-1 must be considered lost (since any power on bus C-1 does not help mitigate the potential loss of the Fuel Clad). Similarly, if HPI 2 or LPI 2 are not capable of providing water to the RPV, bus D-1 must be considered lost for this EAL regardless of any power supplies being available to the bus.

The reason LPI is included is because HPI by itself can not take a suction on the long-term water source (the Containment) without the LPI Pump supplying its suction source. Also, Makeup Pumps are not considered as a water source to the RPV for Fuel Clad protection purposes when considering this EAL.

This action was taken to incorporate feedback from periodic operator training.

5. Page 109 added to the Site Specific section of SA1:

   It is important to note that a loss of the ECCS function for one train alone is not sufficient to declare an Alert using this EAL since more than one power supply remains available to the other train. In this case, the operators will comply with the Technical Specifications for the equipment loss alone.

   This action was taken to incorporate feedback from periodic operator training.

6. Pages 115, 118, 119 and 120 added to the Site Specific sections the following paragraph:

   The >5% power portion of this EAL threshold was chosen to be an easily recognizable, on-scale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range. If the reactor is at very low power (<5%) and a reactor trip occurs, the operators must use indications other than reactor power being >5% to determine if the trip failed to shutdown the reactor.

   This action was taken to incorporate feedback from periodic operator training.

7. Page 159 and 161 – Corrected typographical error under CA10 and CU10, EAL #1. Under listed indications, "TIRC3A56" was corrected to "TIRC3A5", REACTOR COOLANT T-HOT LOOP 2, on both pages. (Notification 600801552)

8. Pages 164 and 165 - Updated Summary of Changes to reflect those changes made in Revision 4.

Revision 5

1. Page 92 and 93 added the following clarifying statement to HU4 based on issues identified during training.

   The 15 minute time period starts either with:

1. A credible notification to the Control Room that a FIRE is occurring in the PROTECTED AREA, (verbal notification from the scene) or
2. Control Room indication of a fire detection system alarm/actuation (any single alarm or a single actuation of a system)

One of the following 4 things must occur within 15 minutes of the original FIRE indication / notification:

1. The alarm must be proven to be spurious (there is no FIRE),
2. The FIRE must be extinguished,
3. The FIRE must be proven not to be in or immediately adjacent to a Table H-1 area, or
4. The Unusual Event must be declared.

2. Alterations #4 and #5 made in Revision 4 to DBRM-EMER-1500A, were deleted on pages 105, 107, 108 and 109. Subsequent review/use found this information resulted in the EAL being misinterpreted. The Site Specific information was restored to that found in Revision 3.

3. Pages 165 and 166 - Updated Summary of Changes to reflect those changes made in Revision 5.

Revision 6

1. Page 160 and 161 corrected site specific information for EAL #2. EAL #2 had information on RCS pressure when the correct information should be on RCS level. This latent issue dates back to revision 0. CR 2014-04532