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RS-07-163

10 CFR 50.4
10 CFR 50 Appendix E

November 28, 2007

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Dresden Nuclear Power Station, Units 1, 2, and 3
Facility Operating License No. DPR-2,
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-010, 50-237, and 50-249

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Limerick Generating Station, Units 1 and 2
Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Peach Bottom Atomic Power Station, Units 1, 2, and 3
Facility Operating License No. DPR-12,
Renewed Facility Operating License Nos. DPR-44, and DPR-56
NRC Docket Nos. 50-171, 50-277, and 50-278

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Three Mile Island Station, Unit 1
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Revisions to the Exelon Nuclear Standardized Radiological Emergency Plan Implementing Procedures

In accordance with 10 CFR 50, Appendix E, Section V, "Implementing Procedures," Exelon Generating Company, LLC (EGC) and AmerGen Energy Company, LLC (AmerGen) are submitting changes to the following Radiological Emergency Plan procedures.

Procedure No.	Revision	Title
EP-AA-1001	20	Exelon Nuclear Radiological Emergency Plan Annex For Braidwood Station
EP-AA-1002	21	Exelon Nuclear Radiological Emergency Plan Annex For Byron Station
EP-AA-1003	11	Exelon Nuclear Radiological Emergency Plan Annex For Clinton Station
EP-AA-1004	23	Exelon Nuclear Radiological Emergency Plan Annex For Dresden Station
EP-AA-1005	25	Exelon Nuclear Radiological Emergency Plan Annex For LaSalle Station
EP-AA-1006	25	Exelon Nuclear Radiological Emergency Plan Annex For Quad Cities Station
EP-AA-110-301	4	Core Damage Assessment (BWR)
EP-AA-110-302	2	Core Damage Assessment (PWR)
EP-AA-113-F-17	D	Braidwood Assembly, Accountability And Evacuation Guidelines
EP-AA-113-F-19	C	Dresden Assembly, Accountability And Evacuation Guidelines

These proposed changes were evaluated under the requirements of 10 CFR 50.54(q) and were determined not to result in a decrease in the effectiveness of the Emergency Plan. The revised procedures were implemented on October 30, 2007, and are being submitted within 30 days of implementation as required by 10 CFR 50 Appendix E. Copies of the revised procedures are provided in Attachments 1 through 10 of this letter.

There are no commitments in this letter. If you have any questions concerning this letter, please contact Mitchel Mathews at (630) 657-2819.

Respectfully,

Jeffrey L. Hansen
Manager – Licensing

Attachments:

1. EP-AA-1001, "Exelon Nuclear Standardized Radiological Emergency Plan Annex for Braidwood Station, " Revision 20
2. EP-AA-1002, "Exelon Nuclear Standardized Radiological Emergency Plan Annex for Byron Station, " Revision 21
3. EP-AA-1003, "Exelon Nuclear Standardized Radiological Emergency Plan Annex for Clinton Station, " Revision 11
4. EP-AA-1004, "Exelon Nuclear Standardized Radiological Emergency Plan Annex for Dresden Station, " Revision 23
5. EP-AA-1005, "Exelon Nuclear Standardized Radiological Emergency Plan Annex for LaSalle Station, " Revision 25
6. EP-AA-1006, "Exelon Nuclear Standardized Radiological Emergency Plan Annex for Quad Cities Station, " Revision 25
7. EP-AA-110-301, "Core Damage Assessment (BWR), " Revision 4
8. EP-AA-110-302, "Core Damage Assessment (PWR), " Revision 2
9. EP-AA-113-F-17, "Braidwood Assembly, Accountability and Evacuation Guidelines, " Revision D
10. EP-AA-113-F-19, "Dresden Assembly, Accountability and Evacuation Guidelines, " Revision C

Attachment 1

EP-AA-1001

"Exelon Nuclear Standardized Radiological Emergency Plan Annex for Braidwood Station"

Revision 20

Exelon Nuclear

EP-AA-1001
Revision 20

EXELON NUCLEAR

RADIOLOGICAL EMERGENCY PLAN ANNEX FOR BRAIDWOOD STATION

Submitted: Kevin Appel Date: 10/10/07
Midwest Region Emergency Preparedness Manager

Authorized: Jim Meister Date: 10/12/07
Vice President – Operations Support

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REVISION HISTORY

Revision 1; March 1986	Revision 3l: October 16, 1998	Revision 17, March 2006
Revision 2a; June 1987	Revision 3m: March 05, 1999	Revision 18, October 2006
Revision 2b; May 1988	Revision 4: May 13, 1999	Revision 19, April 2007
Revision 3; January 1991	Revision 5: June 23, 1999	
Revision 3a; November 1992	Revision 6: January 08, 2001	
Revision 3b; December 1993	Revision 7: October 08, 2001	
Revision 3c; January 1994	Revision 8: October 31, 2001	
Revision 3d; November 1994	Revision 9: January 03, 2002	
Revision 3e; December 1994	Revision 10: July 08, 2002	
Revision 3f; November 1995	Revision 11: August 30, 2002	
Revision 3g; June 1996	Revision 12: November 15, 2002	
Revision 3h; June 1996	Revision 13: May 16, 2003	
Revision 3i; June 1997	Revision 14, December 2004	
Revision 3j: January 05, 1998	Revision 15, May 2005	
Revision 3k: August 14, 1998	Revision 16, January 2006	

Section 1: Introduction

As required in the conditions set forth by the Nuclear Regulatory Commission (NRC) for the operating licenses for the Exelon Nuclear Stations, the management of Exelon recognizes its responsibility and authority to operate and maintain the nuclear power stations in such a manner as to provide for the safety of the general public.

The Exelon Emergency Preparedness Program consists of the Exelon Nuclear Radiological Emergency Plan, Station Annexes, emergency plan implementing procedures, and associated program administrative documents. The Exelon Nuclear Radiological Emergency Plan outlines the basis for response actions that would be implemented in an emergency. Planning efforts common to all Exelon Nuclear generating stations are encompassed within the Exelon Nuclear Radiological Emergency Plan.

This document serves as the Braidwood Station Annex and contains information and guidance that is unique to the station. This includes Emergency Action Levels (EALs), and facility geography and location for a full understanding and representation of the station’s emergency response capabilities. The Station Annex is subject to the same review and audit requirements as the Exelon Nuclear Radiological Emergency Plan.

1.1 Facility Description

The Braidwood Power Station - Units 1 & 2 (Braidwood Station) is located in northern Illinois, approximately 20.0 miles south-southwest of the City of Joliet and 3.0 miles west of the Kankakee River, in Will County. The site is situated in an area composed of flat agricultural farmland that has been scarred from coal strip mining.

The station site is roughly rectangular in shape, with the plant structures occupying the northwest portion of the site.

At its closest approach, the Kankakee River is approximately 3.0 miles east of the northeastern site boundary.

Braidwood Station occupies approximately 4454 acres of land. This area includes the main site area and the cooling lake. The main site area occupies approximately 1917 acres, and the cooling lake occupies the remaining 2537 acres.

Figure 1-1 shows the general location of Braidwood Station. More specific information on station siting may be found in the Updated Final Safety Analysis Report (UFSAR).

The plant consists of two identical pressurized water reactor (PWR) nuclear steam supply systems (NSSS) and turbine-generators furnished by Westinghouse Electric Corporation. Each nuclear steam supply system is designed for a power output of 3586.6 MWt. Cooling for the plant is provided by a cooling lake of 2537 acres with an average depth of approximately 10 feet.

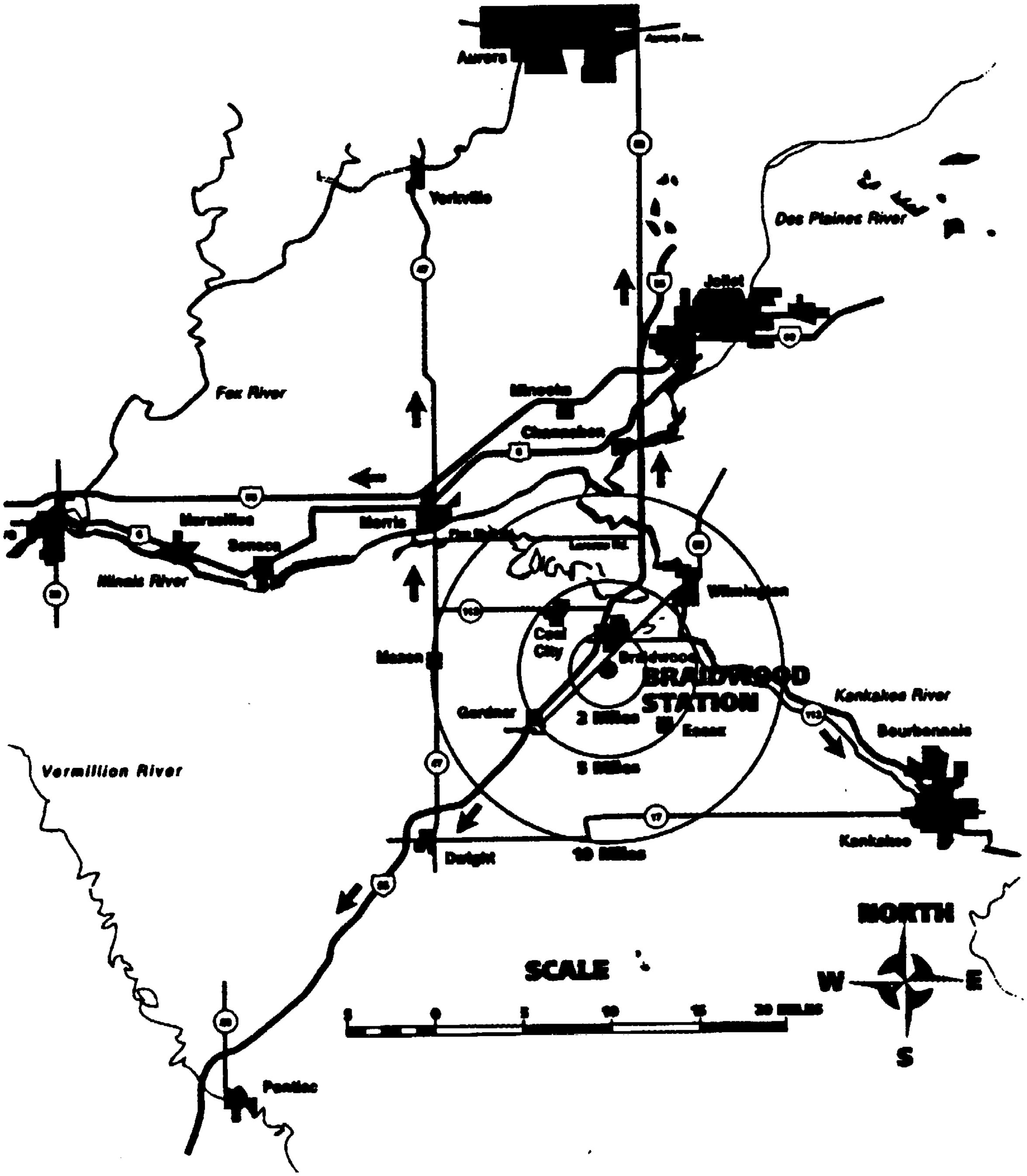
Braidwood Station has two release points for gaseous radioactive effluents, the two Auxiliary Building ventilation stacks. The top of each stack rises 200 feet above the grade elevation. Braidwood Station has one release point for liquid radioactive effluents, the Kankakee River. Liquid radioactive wastes are stored and sampled prior to release to the Kankakee River. A radiation monitor in the discharge line will automatically terminate releases if radioactivity levels exceed predetermined values.

1.2 Emergency Planning Zone

The plume exposure Emergency Planning Zone (EPZ) for Braidwood Station shall be an area surrounding the Station with a radius of about ten miles. (Exact boundaries are determined by the State of Illinois). Refer to Figure 1-1.

The ingestion pathway EPZ for Braidwood Station shall be an area surrounding the Station with a radius of about 50 miles.

Figure 1-1: Braidwood Station Location and 10 Mile EPZ



Section 2: Organizational Control of Emergencies

Initial response to any emergency is by the normal plant organization present at the site. This organization includes positions that are onsite 24 hours per day and is described in Section B.1 of the Exelon Nuclear Radiological Emergency Plan.

Once an emergency is declared, the Emergency Response Organization (ERO) is activated as described in Section B of the Exelon Nuclear Radiological Emergency Plan.

2.1 Non-Exelon Nuclear Support Groups

Exelon Nuclear has contractual agreements with several companies whose services would be available in the event of a radiological emergency. These agencies and their available services are listed in Appendix 3 of the Exelon Nuclear Radiological Emergency Plan.

Emergency response coordination with governmental agencies and other support organizations is discussed in Section A of the Exelon Nuclear Radiological Emergency Plan.

Agreements exist on file at Braidwood Station with several support agencies. These agencies and their support roles are listed in Appendix 2, Station Letters of Agreement.

Section 3: Classification of Emergencies

3.1 General

Section D of the Exelon Nuclear Standardized Emergency Plan divides the types of emergencies into four Emergency Classification Levels (ECLs). The first four are the UNUSUAL EVENT, ALERT, SITE AREA EMERGENCY, and GENERAL EMERGENCY. These ECLs are entered by meeting the Emergency Action Level (EAL) Threshold Values provided in this section of the Annex. 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. Depending on the severity of an event, prior to returning to a standard day-to-day organization, a state or phase called RECOVERY may be entered to provide dedicated resources and organization in support of restoration and communication activities following the termination of the emergency.

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 Guideline 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 acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

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

RECOVERY: Recovery can be considered as a phase of the emergency and is entered by meeting emergency termination criteria provided in EP-AA-111 Emergency Classification and Protective Action Recommendations.

An emergency is classified by assessing plant conditions and comparing abnormal conditions to Initiating Conditions and Threshold Values for each Emergency Action Level.

Individuals responsible for the classification of events will refer to the Initiating Condition and Threshold Values on the matrix of the appropriate station Standardized Emergency Plan Annex (this document). This matrix will contain Initiating Conditions, EAL Threshold Values, Mode Applicability Designators, appropriate EAL numbering system, and additional guidance necessary to classify events. It may be provided as a user aid.

The matrix is set up in four Recognition Categories. The first is designated as "R" and relates to Abnormal Radiological Conditions / Abnormal Radiological Effluent Releases. The second is designated as "F" and relates to Fission Product Barrier Degradation. The third is designated as "M" and relates to System Malfunctions. The fourth is designated as "H" and relates to Hazards and Other Conditions.

The matrix is designed to provide an evaluation of the Initiating Conditions from the worst conditions (General Emergencies) on the left to the relatively less severe conditions on the right (Unusual Events). Evaluating conditions from left to right will reduce the possibility that an event will be under classified. All Recognition Categories should be reviewed for applicability prior to classification.

The Initiating Conditions are coded with a two letter and one number code. The first letter is the Recognition Category designator, the second letter is the Classification Level, “U” for (NOTIFICATION OF) UNUSUAL EVENT, “A” for ALERT, “S” for SITE AREA EMERGENCY and “G” for GENERAL EMERGENCY. The EAL number is a sequential number for that Recognition Category series. All Initiating Conditions that are describing the severity of a common condition (series) will have the same number.

The EAL number may then be used to reference a corresponding page(s), which provides the basis information pertaining to the Initiating Condition:

- Threshold Value
- Mode Applicability
- Basis

Emergency Action Levels are the measurable, observable detailed conditions that must be met in order to classify the event. Classification is not to be made without referencing, comparing and satisfying the Threshold Values specified in the Emergency Action Levels.

A list of definitions is provided as part of this document for terms having specific meaning to the Emergency Action Levels. Site specific definitions are provided for terms with the intent to be used for a particular Initiating Condition/Threshold Value and may not be applicable to other uses of that term at other sites, the Emergency Plan or procedures.

References are also included to documents that were used to develop the EAL Threshold Values.

References to the Emergency Director means the person in Command and Control as defined in the Emergency Plan. Classification of emergencies is a non-delegable responsibility of Command and Control for the onsite facilities with responsibility assigned to the Shift Emergency Director (Control Room Shift Manager) or the Station Emergency Director (TSC). Classification of emergencies remains the responsibility of the applicable onsite facility even after Command and Control is transferred to the Corporate Emergency Director (EOF).

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

Indications used for monitoring and evaluation of plant conditions include the normally used instrumentation, backup or redundant instrumentation, and the use of other parameters that provide information that supports determination if an EAL threshold value has been reached. When an EAL refers to a specific instrument or indication that is determined to be inaccurate or unavailable, then alternate indications shall be used to monitor the specified condition.

During an event that results in changing parameters trending towards an EAL classification, and instrumentation that was available to monitor this parameter becomes unavailable or the parameter goes off scale, the parameter should be assumed to have been exceeded consistent with the trend and the classification made if there are no other direct or indirect means available to determine if the threshold has not been exceeded.

EALs are for unplanned events. A planned evolution involves 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. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL Threshold Value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

When two or more Emergency Action Levels are determined, declaration will be made on the highest classification level for the Unit. When both units are affected, the highest classification for the Station will be used for notification purposes and both units' classification levels will be noted.

3.2 Mode Applicability

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

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 Fission Product Barrier Matrix EALs are applicable only to events that initiate in Hot Shutdown or higher.

If there is a change in 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.

3.3 Emergency Director Judgment

Emergency Director Judgment EALs are provided in the Hazards and Other Condition Affecting Plant Safety section and on the Fission Product Barrier Matrix. Both of the Emergency Director Judgment EALs have specific criteria for when they should be applied.

The Hazards Section Emergency Director Judgment EALs are intended to address unanticipated conditions which are not addressed explicitly by other EALs but warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under specific emergency classifications (UE, Alert, SAE or GE).

The FPB Matrix ED Judgment EALs are intended to include unanticipated conditions, which are not addressed explicitly by any of the other FPB threshold values, but warrant determination because conditions exist that fall under the broader definition for a significant Loss or Potential Loss of the barrier (equal to or greater than the defined FPB threshold values).

3.4 Fission Product Barrier Restoration

Fission Product Barriers (FPBs) are not treated the same as EAL threshold values. Conditions warranting declaration of the loss or potential loss of a Fission Product Barrier may occur resulting in a specific classification. The condition that caused the loss or potential loss declaration could be rectified as the result of Operator action, automatic actions, or designed plant response. Barriers will be considered re-established when there are direct verifiable indications (containment penetration or open valve has been isolated, coolant sample results, etc) that the barrier has been restored and is capable of mitigating future events.

The reestablishment of a fission product barrier does not alter or lower the existing classification. Entry into Termination/Recovery phase is still required for exiting the present classification. However the reestablishment of the barrier should be considered in determining future classifications should plant conditions or events change.

3.5 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 five or more persons violently protesting station operations or activities at the site.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

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

CONTAINMENT CLOSURE: Containment Closure is considered to be as required by procedures.

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

EXTORTION: 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, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

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.

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

NORMAL LEVELS: Normal levels can be considered as 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.

OPERATING MODES:

(1) Power Operations:	Reactor Power > 5%, Keff ≥ 0.99
(2) Startup:	Reactor Power ≤ 5%, Keff ≥ 0.99
(3) Hot Standby:	RCS ≥ 350° F, Keff < 0.99
(4) Hot Shutdown:	200° F < RCS < 350° F, Keff < 0.99
(5) Cold Shutdown:	RCS ≤ 200° F, Keff < 0.99
(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).

Hot Matrix – applies in modes (1), (2), (3), and (4)

Cold Matrix – applies in modes (5), (6), and (D)

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

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: A 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.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) automatic turbine 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 isolated from the Control Room.

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

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

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

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

Emergency Action Level Technical Basis Page Index

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 6 D **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **8.32 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 6 D **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **8.32 E+05 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 6 D **EAL Threshold Values:** 1 VALID reading on any of the following effluent monitors **> 200 times** the high alarm setpoint established by a current radioactive release package for **≥ 15 minutes**. - 0PR001, Liquid Radwaste Effluent Monitor - 0PR002, Gas Decay Tank Effluent Monitor - 0PR010, Station Blowdown Monitor - 1/2 PR001, Containment Purge Effluent Monitor - Discharge Permit specified monitor **OR** 2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 5.53 E+05 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. VALID reading on any of the following effluent monitors **> 2 times** the high alarm setpoint established by a current radioactive release package for **≥ 60 minutes**. - 0PR001, Liquid Radwaste Effluent Monitor - 0PR002, Gas Decay Tank Effluent Monitor - 0PR010, Station Blowdown Monitor - 1/2 PR001, Containment Purge Effluent Monitor - Discharge Permit specified monitor **OR** 2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 2.73 E+05 uCi/sec** for **≥ 60 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Abnormal Rad Levels		**Table R1** **Fuel Handling Incident Radiation Monitors** • Fuel Building Fuel Handling Incident Monitor 0RE-AR055 • Fuel Building Fuel Handling Incident Monitor 0RE-AR056 • Containment Fuel Handling Incident Monitor 1/2 RE-AR011 • Containment Fuel Handling Incident Monitor 1/2 RE-AR012	**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. VALID reading **>1000 mR/hr** on one or more of the radiation monitors in Table R1. **OR** 2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.	**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by: • Refueling Cavity water level **< 23 ft**. above the Reactor Flange (**423 ft**. indicated level). **OR** • Spent Fuel Pool water level **< 23 ft.** above the fuel (**422 ft. 9 in**. indicated level). **OR** • Report of visual observation of an uncontrolled drop in water level in the Fuel Transfer Canal, Refueling Cavity, or Spent Fuel Pool. **AND** b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1. **OR** 2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.
	Table R2 **Areas Requiring Continuous Occupancy** • Main Control Room - 1/2 RE-AR010 • Central Alarm Station - (by survey) • Radwaste Control Room (Aux Bldg 383 ft. el.) - 0RE-AR007	**Table R3** **Areas Requiring Infrequent Access** • Unit 1 and 2 Remote Shutdown Panels (0RE-AR007) • High Radiation Sample Room – HRSS (0RE-AR031) • Containment Air Sample Panel – CASP (by survey) • Fire Hazards Panel (by survey)	**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. VALID radiation monitor or survey readings **>15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions. **OR** 2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.	**RU3** Fuel clad degradation. 1 2 3 4 **EAL Threshold Values:** 1. VALID Gross Failed Fuel Monitor 1/2 RE-PR006 indicating I-135 concentration of **> 5 uCi/cc**. **OR** 2. a. Dose Equivalent I-131specific coolant activity **> 1.0 uCi/gm**. **OR** b. Gross specific coolant activity **> 100 / Ē uCi/gm.**

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

Fission Product Barrier Matrix **Hot Matrix**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
FG1 Loss of ANY Two Barriers AND Loss or Potential Loss of third barrier. 1 2 3 4	**FS1** Loss or Potential Loss of ANY two barriers. 1 2 3 4	**FA1** ANY Loss or ANY Potential Loss of either Fuel Clad or RCS. 1 2 3 4	**FU1** ANY Loss or ANY Potential Loss of Containment. 1 2 3 4

Sub-Category	FC - Fuel Clad		RC – Reactor Coolant System		CT - Containment	
	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
1. CSF Status →	**Core Cooling CSF - RED Path** conditions exist.	1. **Core Cooling CSF - ORANGE Path** conditions exist. **OR** 2. **Heat Sink CSF - RED Path** conditions exist.	None	1. **RCS Integrity CSF - RED Path** conditions exist. **OR** 2. **Heat Sink CSF - RED Path** conditions exist.	None	**Containment CSF - RED Path** conditions exist.
2. RCS Activity →	Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.	None	None	None	None	None
3. Containment Pressure →	None	None	None	None	1. Rapid unexplained drop in Containment pressure following initial pressure rise. **OR** 2. Containment pressure or water level response not consistent with LOCA conditions.	1. Containment pressure **≥ 50 psig** and rising. **OR** 2. Hydrogen concentration in Cont. **≥ 5%** **OR** 3. a. Containment pressure **≥ 20 psig.** **AND** b. Less than one train of Containment Spray operating.
4. CETC Reading →	Average of the ten highest reading core exit thermocouples (CETCs) is **> 1200° F**.	Average of the ten highest reading core exit thermocouples (CETCs) is **> 700° F**.			None	1. a. Average of the ten highest reading core exit thermocouples (CETCs) is **≥ 1200° F** **AND** b. Functional Restoration procedures not effective in **< 15 minutes**. **OR** 2. a. Average of the ten highest reading core exit thermocouples (CETCs) is **≥ 700° F** **AND** b. RVLIS plenum region **= 0%.** **AND** c. Functional Restoration procedures not effective in < **15 minutes.**
5. Reactor Vessel Water Level/RCS Leak Rate →	None	**Core Cooling CSF - ORANGE Path** conditions exist.	RCS leakage > available makeup capacity resulting in loss of subcooling as indicated by CETCs is less than ACCEPTABLE VALUE per Iconic Display or RCS Subcooling Margin Figure 1/2 BwST 2-1.	UNISOLABLE leak exceeding the capacity of one charging pump in the normal charging mode.	None	None
6. S/G Leakage / Rupture →	None	None	Steam Generator Tube Rupture that results in entry into BwEP-3.	None	1. RUPTURED S/G is also FAULTED outside of Containment. **OR** 2. Primary-to-Secondary leakrate **> 10 gpm** with UNISOLABLE steam release from affected S/G to the environment.	None
7. Containment Isolation Valve Status →	None	None	None	None	1. Failure of all isolation valves in any one line to close. **AND** 2. Downstream pathway to the environment exists.	None
8. Containment Rad Monitoring →	Containment radiation monitor reading (AR020(21)) **> Fuel Cladding Loss Threshold**, Table F1.	None	Containment radiation monitor reading (AR020(21)) **> 25 R/hr**.	None	None	Containment radiation (AR020(21)) **> Containment Potential Loss Threshold**, Table F1.
9. ED Judgment →	Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Table F1 – Containment Radiation (AR020(21)) Thresholds

Fuel Cladding - Loss		Containment - Potential Loss	
Time After Shutdown (hrs)	R/hr	Time After Shutdown (hrs)	R/hr
≤ 2	1.95 E+03	≤ 2	4.40 E+03
> 2 to 4	1.70 E+03	> 2 to 4	3.85 E+03
> 4 to 8	1.45 E+03	> 4 to 8	3.35 E+03
> 8 to 16	1.24 E+03	> 8 to 16	2.80 E+03
> 16 to 23	1.09 E+03	> 16 to 23	2.50 E+03
> 23	1.08 E+03	> 23	2.50 E+03

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
Loss of AC Power	**MG1** Prolonged loss of all offsite power and prolonged loss of all onsite AC power to essential busses. 1 2 3 4 **EAL Threshold Values:** 1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2). **AND** 2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses. **AND** 3. a. Restoration of at least one unit ESF bus within 4 hours is **not** likely. **OR** b. **EITHER**: • **Core Cooling CSF - RED Path** conditions exist. • **Core Cooling CSF - ORANGE Path** conditions exist.	**MS1** Loss of all offsite power and loss of all onsite AC power to essential busses. 1 2 3 4 **EAL Threshold Values:** 1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2). **AND** 2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses. **AND** 3. Failure to restore power to at least one unit ESF bus within **15 minutes** from the time of loss of both offsite and onsite AC power.	**MA1** AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout. 1 2 3 4 **EAL Threshold Values:** 1. AC power capability to unit ESF busses reduced to only one of the following power sources for **> 15 minutes**: • Affected unit SAT 142-1(242-1) **OR** 142-2(242-2) • DG 1A(2A) **OR** DG 1B(2B) • Unit crosstie breakers **AND** 2. Any additional single power source failure will result in unit blackout.	**MU1** Loss of all offsite power to essential busses for greater than 15 minutes. 1 2 3 4 5 6 **EAL Threshold Values:** Loss of power to Transformers 142-1(242-1) AND 142-2(242-2) for **> 15 minutes**.
Failure of RPS	**MG3** Failure of the Reactor Protection System to complete an automatic trip and manual trip was NOT successful and there is indication of an extreme challenge to the ability to cool the core. 1 2 **EAL Threshold Values:** 1. Automatic and manual Reactor Trip were not successful from Main Control Board as indicated by: a. Reactor power **≥ 5%** **OR** b. Intermediate Range Start Up Rate is positive **AND** 2. a. **Core Cooling CSF – RED Path** conditions exist. **OR** b. **Heat Sink CSF – RED Path** conditions exist.	**MS3** Failure of the Reactor Protection System to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded and manual trip was NOT successful. 1 2 **EAL Threshold Values:** Automatic and manual Reactor Trip were not successful from Main Control Board as indicated by: a. Reactor power **≥ 5%** **OR** b. Intermediate Range Start Up Rate is positive	**MA3** Failure of the Reactor Protection System to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded. 1 2 3 **EAL Threshold Values:** 1. A Reactor Protection System setpoint was exceeded **AND** 2. A successful automatic Reactor Trip did not occur	**MU3** Inadvertent criticality. 3 4 5 6 **EAL Threshold Values:** An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.
DC Power		**MS4** Loss of all vital DC power. 1 2 3 4 **EAL Threshold Values:** Loss of all vital DC power based on **< 108 VDC** on 125 VDC battery busses 111(211) and 112(212) for **> 15 minutes.**		

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
Heat Sink		**MS5** Complete loss of heat removal capability. 1 2 3 4 **EAL Threshold Values:** 1. **Core Cooling CSF - RED Path** conditions exist. **AND** 2. **Heat Sink CSF - RED Path** conditions exist.		
Annunciators		**MS6** Inability to monitor a SIGNIFICANT TRANSIENT in progress. 1 2 3 4 **EAL Threshold Values:** 1. Loss of most (**approximately 75%)** safety system annunciators (Table M2). **AND** 2. Indications needed to monitor safety functions (Table M3) are unavailable. **AND** 3. SIGNIFICANT TRANSIENT in progress (Table M4). **AND** 4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MA6** UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable. 1 2 3 4 **EAL Threshold Values:** 1. a. UNPLANNED loss of most (**approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** b. UNPLANNED loss of most (**approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes.** **AND** 2. a. SIGNIFICANT TRANSIENT in progress (Table M4). **OR** b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MU6** UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes. 1 2 3 4 **EAL Threshold Values:** 1. UNPLANNED loss of most (**approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** 2. UNPLANNED loss of most (**approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.
		Table M2 - Control Room Panels	**Table M3 - Safety Functions and Related Systems**	**Table M4 - Significant Transients**
		• 1/2 PM01J MCB Gen & Aux Power • 1/2 PM05J MCB Reactor and Chem Volume Control • 1/2 PM06J MCB Eng. Safety Features	• 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) • Fission Product Barriers	• Automatic Turbine Runback **> 25%** thermal reactor power • Electrical load rejection **> 25%** full electrical load • Reactor Trip • Safety Injection Actuation • Thermal power oscillations > 10%

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

<table>
<tr><th></th><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="5">System Malfunction</td></tr>
<tr><td>RCS Leak</td><td></td><td></td><td></td><td>MU7 RCS leakage. 1 2 3 4
<u>EAL Threshold Values:</u>
1. Unidentified or pressure boundary leakage > 10 gpm.
OR
2. Identified leakage > 25 gpm.</td></tr>
<tr><td>Communications</td><td></td><td>
<table>
<tr><th colspan="3">Table M6 – Communications Capability</th></tr>
<tr><th>System</th><th>Onsite</th><th>Offsite</th></tr>
<tr><td>Radios</td><td>X</td><td></td></tr>
<tr><td>Plant page</td><td>X</td><td></td></tr>
<tr><td>Plant Telephone System</td><td>X</td><td></td></tr>
<tr><td>Commercial Telephones</td><td></td><td>X</td></tr>
<tr><td>NARS</td><td></td><td>X</td></tr>
<tr><td>ENS</td><td></td><td>X</td></tr>
<tr><td>HPN</td><td></td><td>X</td></tr>
<tr><td>Cellular phones</td><td></td><td>X</td></tr>
<tr><td>TSO/PJM (Electric Operations)</td><td></td><td>X</td></tr>
<tr><td>Satellite phones</td><td></td><td>X</td></tr>
</table>
</td><td></td><td>MU10 UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5 6
<u>EAL Threshold Values:</u>
1. Loss of all Table M6 Onsite communications capability affecting the ability to perform routine operations.
OR
2. Loss of all Table M6 Offsite communications capability.</td></tr>
<tr><td>T. S. Time</td><td></td><td></td><td></td><td>MU11 Inability to reach required shutdown within Technical Specification limits. 1 2 3 4
<u>EAL Threshold Values:</u>
Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.</td></tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 6 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 6 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 6 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, "Threat Assessment." **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Control Room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per 1/2 BwOA PRI-5, Control Room Inaccessibility procedure in < **15 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 6 D **EAL Threshold Values:** Entry into 1/2 BwOA PRI-5, Control Room Inaccessibility procedure for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • RWSTs • Condensate Storage Tanks • Lake Screen House		**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. Seismic event **> Operating Basis Earthquake (OBE)** as indicated by seismic check 0PA02J. **AND** b. Confirmed by **EITHER:** • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 85 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in the Auxiliary Building as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. Seismic event as indicated by Annunciator 0-38-E5, Accelograph Accel High (0PM01J). **AND** b. Confirmed by **EITHER:** • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 85 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in Auxiliary Building that has the potential to affect safety related equipment needed for the current operating mode.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
	Table H2 **Vital Areas** - Containment - Auxiliary Building - Fuel Handling Building - Main Steam Tunnels - RWSTs - Condensate Storage Tanks - Lake Screen House		**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 6 D **EAL Threshold Values:** 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.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 6 D **EAL Threshold Values:** **Note**: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **8.32 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 6 D **EAL Threshold Values:** **Note**: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **8.32 E+05 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 6 D **EAL Threshold Values:** 1 VALID reading on any of the following effluent monitors **> 200 times** the high alarm setpoint established by a current radioactive release package for **≥ 15 minutes.** • 0PR001, Liquid Radwaste Effluent Monitor • 0PR002, Gas Decay Tank Effluent Monitor • 0PR010, Station Blowdown Monitor • 1/2 PR001, Containment Purge Effluent Monitor • Discharge Permit specified monitor **OR** 2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 5.53 E+05 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. VALID reading on any of the following effluent monitors **> 2 times** the high alarm setpoint established by a current radioactive release package for **≥ 60 minutes**. • 0PR001, Liquid Radwaste Effluent Monitor • 0PR002, Gas Decay Tank Effluent Monitor • 0PR010, Station Blowdown Monitor • 1/2 PR001, Containment Purge Effluent Monitor • Discharge Permit specified monitor **OR** 2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 2.73 E+05 uCi/sec** for **≥ 60 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Abnormal Rad Levels		**Table R1** **Fuel Handling Incident Radiation Monitors** • Fuel Building Fuel Handling Incident Monitor 0RE-AR055 • Fuel Building Fuel Handling Incident Monitor 0RE-AR056 • Containment Fuel Handling Incident Monitor 1/2 RE-AR011 • Containment Fuel Handling Incident Monitor 1/2 RE-AR012	**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1. **OR** 2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.	**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all Irradiated Fuel assemblies remaining covered by water as indicated by: • Refueling Cavity water level **< 23 ft**. above the Reactor Flange (**423 ft**. indicated level). **OR** • Spent Fuel Pool water level **< 23 ft.** above the fuel (**422 ft. 9 in**. indicated level). **OR** • Report of visual observation of an uncontrolled drop in water level in the Fuel Transfer Canal, Refueling Cavity, or Spent Fuel Pool. **AND** b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1. **OR** 2. UNPLANNED VALID Area Radiation Monitor readings rise by a factor of **1000** over NORMAL LEVELS.
	Table R2 **Areas Requiring Continuous Occupancy** • Main Control Room - 1/2 RE-AR010 • Central Alarm Station - (by survey) • Radwaste Control Room (Aux Bldg 383 ft. el.) - 0RE-AR007	**Table R3** **Areas Requiring Infrequent Access** • Unit 1 and 2 Remote Shutdown Panels (0RE-AR007) • High Radiation Sample Room – HRSS (0RE-AR031) • Containment Air Sample Panel – CASP (by survey) • Fire Hazards Panel (by survey)	**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions. **OR** 2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
Loss of AC Power				**MU1** Loss of all offsite power to essential busses for greater than 15 minutes. 1 2 3 4 5 6 **EAL Threshold Values:** Loss of power to Transformers 142-1(242-1) AND 142-2(242-2) for **> 15 minutes**.
			MA2 Loss of all offsite power and loss of all onsite AC power to essential busses. 5 6 D **EAL Threshold Values:** 1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2). **AND** 2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses. **AND** 3. Failure to restore power to at least one unit ESF bus within **15 minutes** from the time of loss of both offsite and onsite AC power.	
RPS				**MU3** Inadvertent criticality. 3 4 5 6 **EAL Threshold Values:** An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.
DC Power				**MU4** UNPLANNED loss of required DC power for greater than 15 minutes. 5 6 **EAL Threshold Values:** 1. UNPLANNED loss of all required vital DC power based on **< 108 VDC** indication on 125 VDC battery busses 111(211) and 112(212). **AND** 2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.
Heat Sink			**MA5** Inability to maintain plant in Cold Shutdown with irradiated fuel in the Reactor Vessel. 5 6 **EAL Threshold Values:** 1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F** for > Table M1 duration. **OR** 2. UNPLANNED Reactor Vessel pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.	**MU5** UNPLANNED loss of decay heat removal capability with irradiated fuel in the Reactor Vessel. 5 6 **EAL Threshold Values:** 1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F**. **OR** 2. Loss of all RCS temperature AND Reactor Vessel level indication for **> 15 minutes**.

Table M1 – RCS Reheat Duration Thresholds

RCS	Containment Closure	Duration
Intact	N/A	60 minutes*
Reduced Inventory (< 397 ft.)	Established	20 minutes*
	Not Established	0 minutes
Not Intact	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.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
RCS Leakage / Inventory	**MG8** Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged with irradiated fuel in the Reactor Vessel. 5 6 **EAL Threshold Values:** 1. Loss of Reactor Vessel inventory per Table M5 indications. **AND** 2. a. RVLIS **≤ 0% Plenum (390 ft. el.)** for **> 30 minutes**. **OR** b. Reactor Vessel level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following: • 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors > **3000 mR/hr** or off-scale high. • Erratic Source Range Monitor indication. **AND** 3. Containment is challenged as indicated by one or more of the following: • Hydrogen concentration in Containment **≥ 5%**. • Containment pressure **≥ 50 psig**. • CONTAINMENT CLOSURE not established.	**MS8** Loss of Reactor Vessel inventory affecting core decay heat removal capability. 5 **EAL Threshold Values:** 1. **Without** CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by RVLIS **≤ 15% Plenum (392.4 ft. el.)**. **OR** b. Reactor Vessel level unknown for **> 30 minutes** with a loss of Reactor Vessel inventory per Table M5 indications. **OR** 2. **With** CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by RVLIS **≤ 0% Plenum (390 ft. el.)**. **OR** b. Reactor Vessel level unknown for **> 30 minutes** with a loss of Reactor Vessel inventory as evidenced by either of the following: • Per Table M5 indications. • Erratic Source Range Monitor indication.	**MA8** Loss of RCS / Reactor Vessel inventory with irradiated fuel in the Reactor Vessel. 5 6 **EAL Threshold Values:** 1. a. Loss of RCS / Reactor Vessel inventory as indicated by RVLIS ≤ **27% Plenum (393 ft. el.)**. **OR** b. Loss of RCS / Reactor Vessel inventory as indicated by LT-046 and LT-049 **< 393 ft. el.** **OR** 2. a. Loss of RCS / Reactor Vessel inventory per Table M5 indications. **AND** b. RCS / Reactor Vessel level unknown for **> 15 minutes**.	**MU8** RCS leakage. 5 **EAL Threshold Values:** 1. Pressurizer level established limit **> 5% Cold Cal** and RCS level **cannot** be restored and maintained **> 5% Cold Cal.** **OR** 2. Pressurizer level established limit **< 5% Cold Cal** and RCS level **cannot** be restored and maintained **>** procedurally established limit.
RCS Leakage / Inventory		**MS9** Loss of Reactor Vessel inventory affecting core decay heat removal capability with irradiated fuel in the Reactor Vessel. 6 **EAL Threshold Values:** 1. **Without** CONTAINMENT CLOSURE established: a. Reactor Vessel Refueling Level Indicators LT-046 and LT-049 **< 393 ft. el.** **OR** b. Reactor Vessel level unknown with indication of core uncovery as evidenced by one or more of the following: • 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high. • Erratic Source Range Monitor indication. **OR** 2. **With** CONTAINMENT CLOSURE established: a. Reactor Vessel Refueling Level Indicators LT-046 and LT-049 **= 392 ft. el.** or off scale low. **OR** b. Reactor Vessel level unknown with indication of core uncovery as evidenced by one or more of the following: • 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high. • Erratic Source Range Monitor indication.	**Table M5 – Indications of RCS Leakage** • Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss	**MU9** UNPLANNED Loss of RCS inventory with irradiated fuel in the Reactor Vessel. 6 **EAL Threshold Values:** 1. UNPLANNED RCS level drop below the Reactor Vessel flange (400 ft.) for **≥ 15 minutes.** **OR** 2. a. Loss of Reactor Vessel inventory per Table M5 indications. **AND** b. Reactor Vessel level unknown.

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

COLD SHUTDOWN / REFUELING MATRIX

System Malfunction

Communications

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Communications		Table M6 - Communications Capability		**MU10** UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5 6 **EAL Threshold Values:** 1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations. **OR** 2. Loss of all Table M6 **Offsite** communications capability.

Table M6 - Communications Capability

System	Onsite	Offsite
Radios	X	
Plant page	X	
Plant Telephone System	X	
Commercial Telephones		X
NARS		X
ENS		X
HPN		X
Cellular Phones		X
TSO/PJM (Electric Operations)		X
Satellite Phones		X

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 6 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 6 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 6 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, "Threat Assessment". **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Control room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per 1/2 BwOA PRI-5, Control Room Inaccessibility procedure in < **15 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 6 D **EAL Threshold Values:** Entry into 1/2 BwOA PRI-5, Control Room Inaccessibility procedure for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • RWSTs • Condensate Storage Tanks • Lake Screen House		**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. Seismic event **> Operating Basis Earthquake (OBE)** as indicated by seismic check 0PA02J. **AND** b. Confirmed by **EITHER:** • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 85 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in Table H2 area or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in the Auxiliary Building as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. Seismic event as indicated by Annunciator 0-38-E5, Accelograph Accel High (0PM01J). **AND** b. Confirmed by **EITHER:** • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 85 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in Auxiliary Building that has the potential to affect safety related equipment needed for the current operating mode.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • RWSTs • Condensate Storage Tanks • Lake Screen House		**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 6 D **EAL Threshold Values:** 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.

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

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RG1

Initiating Condition:

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem 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 Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **8.32 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 1000 mRem TEDE**

      OR

   b. **> 5000 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **>1000 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events that may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in a PAG dose being reached. Assumption and DAPAR inputs are provided in calculation EP-EAL-0601.

The sum of both units' monitors provides the total station release rate.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

Threshold #2 Basis:

The TEDE (1000 mRem) and the CDE Thyroid (5000 mRem) doses are set at the EPA PAG Limits.

The 'site boundary' is defined by an approximately 400-meter (1/4-mile) radius around the plant. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

Basis (cont):

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release has been stopped and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01 Rev 4, AG1
2. EP-AA-112-500 Emergency Environmental Monitoring
3. Exelon DAPAR version 3.0a
4. EP-MW-110-200 Dose Assessment
5. EP-EAL-0601, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Braidwood Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RS1

Initiating Condition:

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem 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 Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **8.32 E+05 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 100 mRem TEDE**

      OR

   b. **> 500 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RS1 (cont)

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in 10% of a PAG dose being reached. Assumption and DAPAR inputs are provided in calculation EP-EAL-601.

The sum of both units' monitors provides the total station release rate.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

Threshold #2 Basis:

The TEDE (100 mRem) and the CDE Thyroid (500 mRem) doses are set at 10% (Ratio 1:5 TEDE to CDE Thyroid) of the EPA PAG Limits.

The 'site boundary' is defined by an approximately 400-meter (1/4-mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

RECOGNITION CATEGORY

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RS1 (cont)

Basis (cont):

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release duration is known and was less than one hour.

  OR

- If is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01 Rev 4, AS1
2. EP-AA-112-500 Emergency Environmental Monitoring
3. Exelon DAPAR version 3.0a
4. EP-MW-110-200 Dose Assessment
5. EP-EAL-0601, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Braidwood Station

RECOGNITION CATEGORY

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer.

Operating Mode Applicability:

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

EAL Threshold Values:

1. VALID reading on any of the following effluent monitors **> 200 times** the high alarm setpoint established by a current radioactive release package for **≥ 15 minutes**.
   - 0PR001, Liquid Radwaste Effluent Monitor
   - 0PR002, Gas Decay Tank Effluent Monitor
   - 0PR010, Station Blowdown Monitor
   - 1/2 PR001, Containment Purge Effluent Monitor
   - Discharge Permit specified monitor

   OR

2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 5.53 E+05 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.

Basis:

UNPLANNED, as used in this context, includes any release for which a radioactive release package 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.

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

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

Basis (cont):

Threshold #1 Basis:

The threshold addresses radioactivity releases (liquid or gaseous) that for whatever reason cause effluent radiation monitor readings to exceed two hundred times the alarm setpoint established by the radioactive release package. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the Offsite Dose Calculation Manual (ODCM) to warn of a release that is not in compliance with the Radiological Effluent Technical Specifications (RETS). Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific release package. 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 flowpath or effluent monitor other than those listed (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the Discharge Permit.

The Liquid Radwaste Effluent monitor high alarm setpoint is typically based on ODCM concentration limits. During periods of release, the high alarm setpoint can be modified based upon tank activity and dilution flow. Detector 0RE-PR001 monitors liquid radwaste effluent from either 30,000-gallon release tank. The release tank discharge valves 0WX353 and 0WX896 close on high radiation.

Threshold #2 Basis:

Braidwood incorporates 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 ODCM. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

This EAL addresses a potential or actual drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. However, the effluent monitor Alert value for gaseous effluents was reduced to a value one half way between the Unusual Event value and the Site Area Emergency value to ensure sequential classifications. Assumptions and inputs for this calculation are provided in EP-EAL-0601. The sum of both units gaseous effluent monitor readings provides a total station release rate. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM, Rev 4. The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition. Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

RECOGNITION CATEGORY

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA1 (cont)

Basis (cont):

Threshold #3 Basis:

Confirmed sample analyses in excess of two hundred times the site ODCM limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by increasing the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mRem/yr basis of the 10 CFR 20 limits for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be approximately 10 mRem/hr. The required release duration was reduced to 15 minutes in recognition of the increased severity.

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

Basis Reference(s):

1. NEI 99-01 Rev 4, AA1
2. Sargent & Lundy calculation ATD-0212, Rev. 0
3. Exelon DAPAR version 3.0a
4. UFSAR Section 11.5.2.3
5. 0BwISR 11.A.3-002, Rev 001 Channel Operation Test of Liquid Radwaste Effluent Radiation Monitor 0PR01J
6. ODCM Section 12.3 Liquid Effluents
7. EP-EAL-0601, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Braidwood Station

RECOGNITION CATEGORY

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer.

Operating Mode Applicability:

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

EAL Threshold Values:

1. VALID reading on any of the following effluent monitors **> 2 times** the high alarm setpoint established by a current radioactive release package for **≥ 60 minutes**.
   - 0PR001, Liquid Radwaste Effluent Monitor
   - 0PR002, Gas Decay Tank Effluent Monitor
   - 0PR010, Station Blowdown Monitor
   - 1/2 PR001, Containment Purge Effluent Monitor
   - Discharge Permit specified monitor

   OR

2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 2.73 E+05 uCi/sec** for **≥ 60 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

Basis:

<u>UNPLANNED</u>, as used in this context, includes any release for which a radioactive release package 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.

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

The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

RECOGNITION CATEGORY

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU1 (cont)

Basis (cont):

Threshold #1 Basis:

The effluent release paths are monitored for radioactivity prior to the flow reaching the point where it would mix with the process flow to the environment. Prior to initiating batch releases, the discharge volume is sampled and analyzed for radioactivity. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate. Radiation monitor alarm setpoints are established to automatically isolate the process flow at the point determined by the discharge permit. These limits are based on the Offsite Dose Calculation Manual ODCM.

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 flowpath or effluent monitor other than those listed (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the Discharge Permit.

The Liquid Radwaste Effluent monitor high alarm setpoint is typically based on ODCM concentration limits. During periods of release, the high alarm setpoint can be modified based upon tank activity and dilution flow. Detector 0RE-PR001 monitors liquid radwaste effluent from either 30,000-gallon release tank. The release tank discharge valves 0WX353 and 0WX896 close on high radiation.

Threshold #2 Basis:

This EAL addresses a potential drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 2, regulatory commitments for an extended period of time. The sum of both gaseous effluent monitor readings provides a total station release rate. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM. Assumptions and calculation inputs are provided in EP-EAL-0601.

The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition. Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

Threshold #3 Basis:

Confirmed sample analyses in excess of two times the site ODCM limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times ODCM for 30 minutes does not exceed this EAL.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

Basis (cont):

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

Basis Reference(s):

1. NEI 99-01 Rev 4, AU1
2. Sargent & Lundy calculation ATD-0212, Rev. 0
3. Exelon DAPAR version 3.0a
4. UFSAR Section 11.5.2.3
5. 0BwSR 11.A.3-002, Rev 001 Channel Operation Test of Liquid Radwaste Effluent Radiation Monitor 0PR01J
6. ODCM Section 12.3 Liquid Effluents
7. EP-EAL-0601, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Braidwood Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA2

Initiating Condition:

Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel.

Operating Mode Applicability:

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

EAL Threshold Values:

1. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1.

Table R1 **Fuel Handling Incident Radiation Monitors**
• Fuel Building Fuel Handling Incident Monitor 0RE-AR055 • Fuel Building Fuel Handling Incident Monitor 0RE-AR056 • Containment Fuel Handling Incident Monitor 1/2 RE-AR011 • Containment Fuel Handling Incident Monitor 1/2 RE-AR012

OR

2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or the Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.

Basis:

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

Threshold #1 Basis:

Uncovering spent fuel represents a substantial degradation of the level of safety of the plant and warrants an Alert classification. Time is available to take corrective actions. 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, no prompt fatalities are predicted and the risk of injury is low. 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 fuel uncovery begins. The value of 1000 mR/hr was conservatively chosen for classification purposes.

Radiation monitor readings are used to provide indication of fuel uncovery and/or fuel damage. High monitor readings associated with the transfer or relocation of a source, stored in or near the pool or readings responding to a planned evolution such as removal of the reactor head or equipment relocation are not classified under this threshold since the reading would not be indicative of fuel uncovery and/or fuel damage.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA2 (cont)

Basis (cont):

Dropping heavy loads onto the spent fuel can cause significant damage to the spent fuel and an Alert is also warranted under these conditions provided that the above radiation monitor threshold readings are reached.

Fuel Building Fuel Handling Incident Monitors 0RE-AR055 and 0RE-AR056 are redundant GM type gamma detectors are mounted on the walls near the edge of the pool to provide reliable and rapid detection of radioactivity released from the pool surface. The monitors alarm locally and in the Main Control Room (5 mR/hr) and initiate control action to route the released activity through the emergency exhaust system. The monitors have an operating range that extends from 0.1 to 1E4 mR/hr.

Containment Fuel Handling Incident Monitors 1/2 RE-AR011 and 1/2 RE-AR012 are redundant GM type gamma detectors that provide reliable and rapid detection of radioactivity released from the water surface. The monitors alarm in the Main Control Room. They alarm at 10 mR/hr above background and isolate containment ventilation (VQ). The monitors have an operating range, which extends from 0.1 to 1E4 mR/hr.

Threshold #2 Basis:

Once Spent Fuel Pool water level drops below the low level alarm setpoint, further drops can be monitored only by visual observation.

Refueling Cavity water level is normally monitored by:

- LT-049 (LI-RY-049), range 392 ft. el. to 426 ft. el.
- LT-047 (LI-RY-047), 413 ft. el. to 426 ft. el.

Loss of inventory from the Refueling Cavity may also be indicated by:

- CNMT DRAIN LEAK DETECT FLOW HIGH alarm (BwAR 1-1-A2, 2-1-A2)
- Abnormal flow on Containment drains flow recorder at 1/2 PM12J
- Floor Drain Sump (1/2 FT-RF008) Flowrate
- RX Cavity Sump (1/2 FT-RF010) Flowrate
- Grid 2 or Grid 4 Containment Area Monitor(s) trends greater than the alert alarm setpoint or increasing.

Without report of the position of spent fuel during transfer between the Reactor Vessel and the Spent Fuel Pool, fuel uncovery cannot be determined directly from the installed water level instrumentation. Visual observation, therefore, provides the only viable mechanism of determining if spent fuel in the spent fuel canal will be uncovered.

This EAL applies to irradiated fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA2 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, AA2
2. UFSAR 11.5.2.2.6, 11.5.2.2.7, 15.7.4, Table 12.3-3
3. Technical Specification Table 3.3-6-1
4. 1/2 BwOA REFUEL-1 Fuel Handling Emergency
5. 1/2 BwOA REFUEL-2 Refueling Cavity or Spent Fuel Pool Level Loss
6. TRM 3.9.A, Refueling Operations, Decay Time
7. BwAR 1-1-A2, 2-1-A2, CNMT DRAIN LEAK DETECT FLOW HIGH alarm

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU2

Initiating Condition:

Unexpected rise in plant radiation.

Operating Mode Applicability:

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

EAL Threshold Values:

1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:

   - Refueling Cavity water level **< 23 ft**. above the Reactor Flange (**423 ft**. indicated level).

     OR

   - Spent Fuel Pool water level **< 23 ft**. above the fuel (**422 ft. 9 in**. indicated level).

     OR

   - Report of visual observation of an uncontrolled drop in water level in the Fuel Transfer Canal, Refueling Cavity, or Spent Fuel Pool.

   AND

   b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1.

Table R1 Fuel Handling Incident Radiation Monitors
• Fuel Building Fuel Handling Incident Monitor 0RE-AR055 • Fuel Building Fuel Handling Incident Monitor 0RE-AR056 • Containment Fuel Handling Incident Monitor 1/2 RE-AR011 • Containment Fuel Handling Incident Monitor 1/2 RE-AR012

OR

2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU2 (cont)

Basis:

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

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

NORMAL LEVELS: Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Threshold #1 Basis:

Refueling Cavity water level is normally monitored by

- LT-049 (LI-RY-049), range 392 ft. el. to 426 ft. el.
- LT-047 (LI-RY-047), 413 ft. el. to 426 ft. el.

Loss of inventory from the Refueling Cavity may also be indicated by:

- Grid 2 or Grid 4 Containment Area Monitor(s) trends greater than the alert alarm setpoint or increasing

Since no remote indication of Spent Fuel Pool water level exists, drops in Spent Fuel Pool water level can normally be detected only through visual observation. Loss of inventory from the Spent Fuel Pool may also be indicated if the Spent Fuel Pool or Transfer Canal leak detection system indicates flow.

The spent fuel transfer canal is normally aligned to either or both of the Refueling Cavity or Spent Fuel Pool when it contains spent fuel as it serves as the transfer path to move the fuel between these locations. Therefore, the level indications available for these two locations will also indicate the level in the spent fuel transfer canal. However, the threshold of “report of visual observation of a rapid drop in water level” is included for the spent fuel transfer canal in the event that it contains spent fuel and is not aligned to the Refueling Cavity or Spent Fuel Pool, and as an additional indication to those level monitors listed for the other locations.

Basis (cont):

Without report of the position of spent fuel during transfer between the Reactor Vessel and the Spent Fuel Pool, fuel uncovery cannot be determined directly from the installed water level instrumentation. Visual observation, therefore, provides the only viable mechanism of determining if spent fuel in the spent fuel canal will be uncovered.

Threshold #2 Basis:

Valid elevated area radiation levels usually have long lead times relative to the potential for radiological release beyond the site boundary, thus impact to public health and safety is very low.

This EAL addresses UNPLANNED rise in radiation levels inside the plant. These radiation levels represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant.

Basis Reference(s):

1. NEI 99-01 Rev 4, AU2
2. Technical Specifications 3.7.14
3. 1/2 BwOA REFUEL-1 Fuel Handling Emergency Unit 1/2
4. 1/2 BwOA REFUEL-2 Refueling Cavity Or Spent Fuel Pool Level Loss Unit 1/2
5. BwAR 1-1-C1 SPENT FUEL PIT LEVEL HIGH LOW
6. 1/2 BwOSR 0.1-6 Unit One(Two) Mode 6 Shiftly and Daily Operating Surveillance
7. BwOP RH-8 Filling the Reactor Cavity for Refueling
8. BwOP RH-9 Pump Down of the Reactor Cavity to the RWSTs
9. BwOP RC-4 Reactor Coolant System Drain
10. BwAR 1-6-C3 REFUELING CAVITY LVL HIGH LOW

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA3

Initiating Condition:

Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown.

Operating Mode Applicability:

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

EAL Threshold Values:

1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions:

Table R2 – Areas Requiring Continuous Occupancy
• Main Control Room – 1/2 RE-AR010 • Central Alarm Station - (by survey) • Radwaste Control Room (Aux Bldg 383 ft. el.) - 0RE-AR007

OR

2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.

Table R3 – Areas Requiring Infrequent Access
• Unit 1 and 2 Remote Shutdown Panels (0RE-AR007) • High Radiation Sample Room – HRSS (0RE-AR031) • Containment Air Sample Panel – CASP (by survey) • Fire Hazards Panel (by survey)

Basis:

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

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

Basis (cont):

Threshold #1 Basis:

This EAL addresses increased radiation levels that impede necessary access to operating stations requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. Areas requiring continuous occupancy include the Main Control Room, the Central Alarm Station (CAS) and the Radwaste Control Room. The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations.

The value of 15 mR/hr is derived from the General Design Criteria (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. A 30-day duration implies an event potentially more significant than an Alert.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EALs may be involved. For example, a dose rate of 15 mR/hr in the Main Control Room may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This EAL is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.).

Threshold #2 Basis:

This EAL addresses increased radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Typically areas requiring infrequent access to maintain plant safety functions include plant VITAL AREAS. Area radiation levels above 2000 mR/hr are indicative of radiation fields that may limit personnel access to equipment, the operation of which may be needed to assure adequate core cooling or shutdown the reactor.

The dose rate threshold selected is based on site administrative limits.

RECOGNITION CATEGORY

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA3 (cont)

Basis (cont):

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved. For example, a dose rate of 2000 mR/hr may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.) or pre-existing radiation areas for which radiological controls already exist. The concern of this threshold is the unanticipated rise in radiation levels that results in unplanned restrictions to areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown.

Basis Reference(s):

1. NEI 99-01 Rev 4, AA3
2. UFSAR Chapter 3.02, UFSAR Table 3.2-1

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU3

Initiating Condition:

Fuel clad degradation.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. VALID Gross Failed Fuel Monitor 1/2 RE-PR006 indicating I-135 concentration **> 5 uCi/cc.**

   OR

2. a. Dose Equivalent I-131 specific coolant activity **> 1.0 uCi/gm**.

      OR

   b. Gross specific coolant activity **> 100 / Ē uCi/gm**.

Basis:

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

Threshold #1 Basis:

The I-135 concentration calculated for a dose equivalent I-131 value of 1 uCi/cc is 0.57 uCi/g. This value is too small to be able to be detected by the Gross Failed Fuel Monitor. Therefore, a monitor value is chosen that is in the detectable range of the monitor and provides reasonable assurance that the 1 uCi/g dose equivalent I-131 value is exceeded based upon the radiation monitor reading.

The modifier "VALID" is appropriate because there are several conditions that may cause the monitor to alarm that are not related to fuel clad degradation and therefore should not result in the declaration of an Unusual Event.

Threshold #2 Basis:

Threshold #2 addresses coolant samples exceeding coolant technical specifications for iodine spike.

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 Reactor Coolant activity unless another cause is known, e.g., Reactor Coolant System chemical decontamination evolution (during shutdown) is ongoing with resulting high activity levels.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU3 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, SU4
2. Technical Specifications 3.4.16
3. 1/2 BwOA PRI-4, High Reactor Coolant Activity Unit 1/2
4. PWR Letdown Rad Monitor Setpoint Calculation for Degraded Fuel Indication

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 Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the General Emergency classification level each barrier is weighted equally.

Basis Reference(s):

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

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 Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Site Area Emergency classification level, each barrier is weighted equally.

Basis Reference(s):

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

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 Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Alert classification level, 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 EAL FS1.

Basis Reference(s):

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

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 Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

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 EAL 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC1 – Loss or Potential Loss

Initiating Condition:

Critical Safety Function Status.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Core Cooling CSF - RED Path conditions exist.

POTENTIAL LOSS

1. **Core Cooling CSF - ORANGE Path** conditions exist.

   OR

2. **Heat Sink CSF - RED Path** conditions exist.

Basis:

Loss Basis:

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.

The Core Cooling Critical Safety Function RED path condition exists when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200° F.

Potential Loss Basis:

The Core Cooling Critical Safety Function ORANGE path condition exists if:

- The average of the ten highest reading core exit thermocouples (CETCs) is reading less than 1200° F but greater than 700° F, and
- RCS subcooling based on CETCs is less than ACCEPTABLE VALUE per Iconic Display or RCS Subcooling Margin Figure 1/2 BwST 2-1.

Any of these conditions indicate subcooling has been lost and that some fuel cladding damage may potentially occur.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC1 – Loss or Potential Loss (cont)

Basis (cont):

The Heat Sink Critical Safety Function (CSF) RED path indicates the heat sink is under extreme challenge and indicates a potential loss of the Fuel Cladding barrier. The Heat Sink Critical Safety Function Red path conditions exist if narrow range levels in all steam generators (S/Gs) are less than or equal to 10% - Unit 1 (31% adverse containment) and 14% - Unit 2 (34% adverse containment) and total feedwater flow to all S/Gs is less than or equal to 500 gpm. If total feed flow is less than 500 gpm due to procedurally directed operator actions then this condition does not apply.

The combination of these two conditions indicates the ultimate heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the Fuel Cladding barrier. This condition also corresponds to RCS barrier potential loss threshold RC1 resulting in at least a Site Area Emergency.

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. 1/2 BwST-2 Core Cooling
3. 1/2 BwFR-C.1 Response to Inadequate Core Cooling
4. 1/2 BwFR-C.2 Response to Degraded Core Cooling
5. 1/2 BwST-3 Heat Sink

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC2 – Loss

Initiating Condition:

Primary Coolant activity level.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.

Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. 300 uCi/gm Dose Equivalent I-131 is well above that expected for iodine spikes and corresponds, generically, to about 2% to 5% fuel cladding damage (0.6% clad failure per S&L calculation BB-ER-02, rev 0). This amount of radioactivity indicates significant clad damage and thus the Fuel Cladding barrier is considered lost.

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. EP-AA-123-1003 Core Damage Assessment Methodology (CDAM) Program Technical Basis
3. S&L calculation BB-ER-02, Rev 0

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC4 – Loss or Potential Loss

Initiating Condition:

Core Exit Thermocouple readings.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Average of the ten highest reading core exit thermocouples (CETCs) is **> 1200° F**.

POTENTIAL LOSS

Average of the ten highest reading core exit thermocouples (CETCs) is **> 700° F**.

Basis:

A failed CETC Channel can lead to indication of the CETC > 700° F until the system removes the failed channel from average. Fission product barrier loss or potential loss is based on VALID CETC readings.

Loss Basis:

The CETC value corresponds to the Core Cooling Critical Safety Function RED path but is evaluated separately from the CSF Status. The elevated temperature corresponds to significant superheating of the coolant and is indicative of a loss of the Fuel Cladding barrier.

Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use (initiation after SI is blocked).

Potential Loss Basis:

The CETC value corresponds to the Core Cooling Critical Safety Function ORANGE path but is evaluated separately from the CSF Status because the CSF evaluation considers the degree of subcooling prior to status determination. The elevated temperature corresponds to a loss of subcooling and is indicative of a Potential Loss of the Fuel Cladding barrier.

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. 1/2 BwST-2 Core Cooling
3. 1/2 BwFR-C.1 Response to Inadequate Core Cooling
4. 1/2 BwFR-C.2 Response to Degraded Core Cooling

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC5 – Potential Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

Core Cooling CSF - ORANGE Path conditions exist.

Basis:

The "Potential Loss" EAL is defined by the Core Cooling CSF - ORANGE path.

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. 1/2 BwFR-C.2 Response to Degraded Core Cooling

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC8 – Loss

Initiating Condition:

Containment radiation monitoring.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Containment radiation monitor reading (AR020(21)) **> Fuel Cladding Loss Threshold**, Table F1.

Table F1 – Containment Radiation (AR020(21)) Thresholds	
Time After Shutdown (hours)	**Fuel Cladding Loss (R/hr)**
≤ 2	1.95 E+03
> 2 to 4	1.70 E+03
> 4 to 8	1.45 E+03
> 8 to 16	1.24 E+03
> 16 to 23	1.09 E+03
> 23	1.08 E+03

Basis:

The containment radiation monitor readings specified in Table F1 provide values that indicate the release of reactor coolant into the containment atmosphere with elevated activity indicative of fuel damage (~2%). The values are a function of time after shutdown and were derived using CDAM with 2% clad damage, no containment sprays in operation, CETC >1200º F and RCS pressure at <1600 psig assuming LOCA depressurized system. The reading is calculated assuming the instantaneous release and dispersal of the above reactor coolant noble gas and iodine inventory into the containment atmosphere.

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations allowed within Technical Specifications (including iodine spiking) and are therefore indicative of fuel damage (approximately 2% - 5% cladding failure). The reading is higher than that specified for the loss of RCS barrier; thus, elevated containment radiation readings at or above the Fuel Cladding barrier loss threshold signify a loss of two fission product barriers.

During at power (including ATWS) conditions the value listed for the “< 2 hours after shutdown” row is used as an indication of fuel damage.

FC8 – Loss (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC9 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.

Basis:

The Emergency Director judgment fuel cladding loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Fuel Cladding barrier. The inability to monitor fuel cladding integrity should also be considered as a factor in judging that the Fuel Cladding barrier may be considered lost or potentially lost.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC1 – Potential Loss

Initiating Condition:

Critical Safety Function Status

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

1. **RCS Integrity CSF - RED Path** conditions exist.

   OR

2. **Heat Sink CSF - RED Path** conditions exist.

Basis:

Threshold #1 Basis

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.

The RCS Integrity Critical Safety Function (CSF) Red path is entered if:

- Temperature drop in any RCS cold leg is greater than or equal to 100° F/hr, and
- Any RCS cold leg temperature/pressure is to the left of Plant Operational Limits Figure 1/2 BwST 4-1 Limit A.

The combination of these two conditions indicates the RCS barrier is under significant challenge and should be considered a potential loss of RCS barrier.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC1 – Potential Loss (cont)

Basis (cont):

Threshold #2 Basis

The Heat Sink Critical Safety Function (CSF) RED path indicates the heat sink is under extreme challenge and indicates a potential loss of the Fuel Cladding barrier. The Heat Sink Critical Safety Function Red path is entered if narrow range levels in all steam generators (S/Gs) are less than or equal to 10% - Unit 1 (31% adverse containment) and 14% - Unit 2 (34% adverse containment) and total feedwater flow to all S/Gs is less than or equal to 500 gpm. If total feed flow is less than 500 gpm due to procedurally directed operator actions then this condition does not apply.

The combination of these two conditions indicates the ultimate heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the RCS barrier. This condition also corresponds to Fuel Cladding barrier potential loss threshold FC1 resulting in at least a Site Area Emergency.

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. 1/2 BwST-4 Integrity
3. 1/2 BwFR-P.1 Response To Imminent Pressurized Thermal Shock Condition
4. 1/2 BwST-3 Heat Sink

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC5 – Loss or Potential Loss

Initiating Condition:

RCS leak rate.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

RCS leakage > available makeup capacity resulting in loss of subcooling as indicated by CETCs is less than ACCEPTABLE VALUE per Iconic Display or RCS Subcooling Margin Figure 1/2 BwST 2-1.

POTENTIAL LOSS

UNISOLABLE leak exceeding the capacity of one charging pump in the normal charging mode.

Basis:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Loss Basis:

This threshold addresses conditions in which 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 Basis:

This threshold is based on the inability to maintain normal liquid inventory within the RCS by normal operation of the Chemical and Volume Control System, which is considered as one centrifugal charging pump discharging to the charging header. The need for a second charging pump would be indicative of a substantial RCS leak. The minimum operability flow rate for each charging pump is 60 gpm.

Normal Charging Lineup refers to the normal charging system flow path through the volume control system including normal and design alternate flow paths, and flow to reactor coolant pump seals.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC5 – Loss or Potential Loss (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. 1/2 BwST-2 Core Cooling
3. 1/2 BwFR-C.1 Response to Inadequate Core Cooling
4. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
5. UFSAR Fig. 6.3-4

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC6 – Loss

Initiating Condition:

S/G Tube Rupture.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Steam Generator Tube Rupture that results in entry into BwEP-3.

Basis:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

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.

To meet this threshold, the leakage must be large enough to cause actuation of ECCS (SI). ECCS (SI) actuation is caused by:

- PZR Low Pressure (≤ 1829 psig)
- Steam Line Low Pressure (< 640 psig)
- Containment High Pressure (> 3.4 psig)
- Manual Safety Injection

This EAL is intended to address the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment Barrier "Loss" EAL CT6 and Fuel Clad Barrier EALs. The "Loss" EAL addresses RUPTURED SG(s) for which the leakage is large enough to cause automatic or manual actuation of ECCS (SI). By itself, this EAL will result in the declaration of an ALERT. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a SITE AREA EMERGENCY per Containment Barrier "Loss" EAL CT6.

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
3. 1/2 BwEP-0 Reactor Trip Or Safety Injection Unit 1/2
4. 1/2 BwEP-3 Steam Generator Tube Rupture

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC8 – Loss

Initiating Condition:

Containment radiation monitoring.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Containment radiation monitor reading (AR020(21)) **> 25 R/hr**.

Basis:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

The containment radiation monitor reading is a value that indicates a significant release of reactor coolant to the containment. A reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the containment atmosphere. Conservative estimates (using Technical Specifications high RCS activity) indicated that the readings from release of the normal RCS inventory would be ~ 25 R/hr. The reading is less than that specified for Fuel Cladding barrier Loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations.

Computer points:

1RE-AR020 – Unit 1 High Range Containment (RA0046)

1RE-AR021 – Unit 1 High Range Containment (RA0047)

2RE-AR020 – Unit 2 High Range Containment (RA0071)

2RE-AR021 – Unit 2 High Range Containment (RA0072)

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC9 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.

Basis:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

The Emergency Director judgment RCS loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the RCS barrier. The inability to monitor RCS integrity should also be considered as a factor in judging that the RCS barrier may be considered lost or potentially lost.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT1 – Potential Loss

Initiating Condition:

Critical Safety Function Status.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

Containment CSF - RED Path conditions exist.

Basis:

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment. Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this EAL is primarily a discriminator between SITE AREA EMERGENCY and GENERAL EMERGENCY representing a potential loss of the third barrier.

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

The Containment Critical Safety Function (CSF) Red path conditions exist if containment pressure is equal to or greater than 50 psig. This pressure is the containment design pressure and is well in excess of that expected from the design basis loss of coolant accident. This threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation (metal-water reaction) or failure to trip in combination with RCS breach. This combination of conditions would be expected to require the declaration of a General Emergency.

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. 1/2 BwST-5 Containment
3. 1/2 BwFR-Z.1 Response to High Containment Pressure

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT3 – Loss or Potential Loss

Initiating Condition:

Containment pressure.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

1. Rapid unexplained drop in Containment pressure following initial pressure rise.

   OR

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

POTENTIAL LOSS

1. Containment pressure **≥ 50 psig** and rising.

   OR

2. Hydrogen concentration in Containment **≥ 5%.**

   OR

3. a. Containment pressure **≥ 20 psig.**

      AND

   b. Less than one train of Containment Spray operating.

Basis:

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

Loss Threshold #1 Basis:

Rapid unexplained loss of pressure (i.e., not attributable to containment spray, cooling or condensation effects) following an initial pressure rise indicates a loss of containment integrity. The referenced analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.8 psig for Unit 1 and 38.4 psig for Unit 2 (experienced during a LOCA).

Loss Threshold #2 Basis:

Containment pressure and sump levels should rise as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure response not consistent with LOCA conditions indicates containment bypass and a loss of containment integrity.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT3 – Loss or Potential Loss (cont)

Basis (cont):

Potential Loss Threshold #1 Basis:

This threshold is the containment design pressure and is well in excess of that expected from the design basis loss of coolant accident. The referenced analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.8 psig for Unit 1 and 38.4 psig for Unit 2 (experienced during a LOCA).

The threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation (metal-water reaction) or failure to trip in combination with RCS breach. This condition would be expected to require the declaration of a General Emergency.

Potential Loss Threshold #2 Basis:

If hydrogen concentration reaches or exceeds 5% in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred. Since this threshold is also indicative of loss of both Fuel Cladding and RCS barriers with the potential loss of the Containment barrier, it therefore will likely warrant declaration of a General Emergency.

Containment hydrogen concentration is indicated on 1/2 HSU-PS345 and 1/2 HSU-PS346 and PPDS.

Potential Loss Threshold #3 Basis:

This threshold represents a potential loss of containment in that the containment depressurization equipment is either lost or performing in a degraded manner.

The Containment Spray System limits post accident conditions to less than the containment design values. The Containment Spray System consists of two separate 100% capacity trains, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping.

During a Design Basis Accident (DBA), a minimum of one containment spray train is required to maintain the containment peak pressure below the design limits.

The containment pressure setpoint (20 psig) is the pressure at which the equipment should have actuated and began performing its function.

CT3 – Loss or Potential Loss (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. UFSAR Section 15.6.5.2.1
3. 1/2 BwST-5 Containment
4. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
5. Technical Specifications B 3.6.6, Containment Spray and Cooling Systems

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT4 – Potential Loss

Initiating Condition:

Core Exit Thermocouple readings.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

1. a. Average of the ten highest reading core exit thermocouples (CETCs) is **≥ 1200° F**.

   AND

   b. Functional Restoration procedures not effective in **< 15 minutes**.

   OR

2. a. Average of the ten highest reading core exit thermocouples (CETCs) is **≥ 700° F**.

   AND

   b. RVLIS plenum region **= 0%.**

   AND

   c. Functional Restoration procedures not effective in **< 15 minutes.**

Basis:

A failed CETC Channel can lead to indication of the CETC ≥ 700° F until the system removes the failed channel from average. Fission product barrier loss or potential loss is based on VALID CETC readings.

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

The conditions in this potential loss EAL 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 Heat Sink criteria in the Fuel and RCS barrier columns, this EAL 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.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT4 – Potential Loss (cont)

Basis (cont)

Potential Loss Threshold #1 Basis:

The Core Cooling Critical Safety Function (CSF) RED path is entered when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200°F. Entry into Core Cooling RED path requires entry into functional restoration procedure 1/2 BwFR-C.1, Response to Inadequate Core Cooling.

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 function restoration procedures to arrest the core melt sequence. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing. Whether or not actions will be effective should be apparent within 15 minutes of entry into restoration procedures. The Emergency Director should make the declaration as soon as it is determined that the procedures have not been, or will not be effective.

Potential Loss Threshold #2 Basis:

Core Cooling Critical Safety Function ORANGE path conditions exist when the average of the ten highest reading core exit thermocouples (CETCs) is reading greater than or equal to 700° F.

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 function restoration procedures to arrest the core melt sequence. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing. Whether or not 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 not been, or will not be effective.

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. 1/2 BwST-2 Core Cooling
3. 1/2 BwFR-C.1 Response to Inadequate Core Cooling
4. 1/2 BwFR-C.2 Response to Degraded Core Cooling

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT6 – Loss

Initiating Condition:

S/G secondary side release with primary to secondary leakage.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

1. RUPTURED S/G is also FAULTED outside of Containment.

   OR

2. Primary-to-Secondary leakrate **> 10 gpm** with UNISOLABLE steam release from affected S/G to the environment.

Basis:

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.

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.

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

Loss Threshold #1 Basis:

This EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. The first threshold addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers. In conjunction with RCS Barrier Loss RC6, this would always result in the declaration of a SITE AREA EMERGENCY.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT6 – Loss (cont)

Basis (cont)

Loss Threshold #2 Basis:

The second threshold addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the affected 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., SGTR with concurrent loss of offsite power and the affected 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 a nonisolable release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs.

A pressure boundary leakage of 10 gpm is also used as the threshold in EAL MU7, RCS leakage.

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. 1/2 BwEP-0 Reactor Trip or Safety Injection Unit 1/2
3. 1/2 BwEP-3 Steam Generator Tube Rupture
4. 1/2 BwOA SEC-8 Steam Generator Tube Leak

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT7 – Loss

Initiating Condition:

Containment Isolation Valve status after Containment Isolation.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

1. Failure of all isolation valves in any one line to close.

   AND

2. Downstream pathway to the environment exists.

Basis:

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

This EAL is intended to address incomplete containment isolation that allows direct release (gaseous or liquid flowpath) to the environment outside of containment (for example into the Auxiliary Bldg, Turbine Bldg or outside atmosphere). It represents a loss of the containment barrier.

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a radiological release pathway to the environment (even if the isolation signal is not VALID). The concern is the UNISOLABLE open pathway to the environment. A failure of the ability to close any open isolation valves in any one line indicates a breach of containment integrity. Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include UNISOLABLE containment atmospheric vent paths as well as UNISOLABLE primary systems (RCS). If the primary system leakage outside containment cannot be isolated, a loss of both the RCS and the Containment barriers results. No leakage threshold is specified since leaks outside containment, particularly under dynamic conditions, are difficult to quantify and may manifest themselves with diverse symptoms.

Symptoms of a primary system discharging outside containment may be indicated via mass balance, lowering RCS inventory without corresponding containment response, or area temperatures and radiation levels outside containment. It is for this reason that Emergency Director judgment should be used in evaluating this criterion. However, it is intended that the magnitude of the primary system leak associated with this EAL be consistent with RCS barrier RC5 Potential Loss of ~100 gpm or greater. Minor release paths such as instrument and sample lines are not considered under this threshold.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT7 – Loss (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
3. UFSAR Fig. 6.3-4

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT8 – Potential Loss

Initiating Condition:

Significant radioactive inventory in Containment.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

Containment radiation (AR020(21)) **> Containment Potential Loss Threshold**, Table F1.

Table F1 – Containment Radiation (AR020(21)) Thresholds

Time After Shutdown (hours)	Containment Potential Loss (R/hr)
≤ 2	4.40 E+03
> 2 to 4	3.85 E+03
> 4 to 8	3.35 E+03
> 8 to 16	2.80 E+03
> 16 to 23	2.50 E+03
> 23	2.50 E+03

Basis:

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

The containment radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Cladding barrier. NUREG-1228 “Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents” states that such readings do not exist when the amount of cladding damage is less than 20%. The values are a function of time after shutdown and were derived using CDAM v.1.1 assuming 20% clad damage, no containment sprays in operation, CETC > 1200º F and RCS pressure at <1600 psig assuming LOCA depressurized system. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a significant failure into the reactor coolant has occurred.

During at power (including ATWS) conditions the value listed for the “≤ 2 hours after shutdown” row is used as an indication of fuel damage.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT8 – Potential Loss (cont)

Basis (cont):

Regardless of whether the Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Containment barrier. The reading is higher than that specified for Fuel Cladding Loss FC8 and RCS Loss RC8. Containment radiation readings at or above the Containment barrier potential loss threshold, therefore, signify a loss of two fission product barriers and potential loss of a third, indicating the need to upgrade the emergency classification to a General Emergency.

Computer points:

1RE-AR020 – Unit 1 High Range Containment (RA0046)

1RE-AR021 – Unit 1 High Range Containment (RA0047)

2RE-AR020 – Unit 2 High Range Containment (RA0071)

2RE-AR021 – Unit 2 High Range Containment (RA0072)

Basis Reference(s):

1. NEI 99-01 Rev 4, Table 5-F-4
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT9 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

<u>LOSS</u>

Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.

<u>POTENTIAL LOSS</u>

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis:

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

The Emergency Director judgment Containment loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Containment barrier. The inability to monitor Containment parameters should also be considered as a factor in judging that the Containment barrier may be considered lost or potentially lost.

Basis Reference(s):

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

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG1

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2).

   AND

2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses.

   AND

3. a. Restoration of at least one unit ESF bus within 4 hours is **not** likely.

   OR

   b. **EITHER**:

   - **Core Cooling CSF - RED Path** conditions exist.
   - **Core Cooling CSF - ORANGE Path** conditions exist.

Basis:

Loss of all AC power to ESF busses compromises the availability of all plant safety systems. Prolonged loss of all AC power may lead to loss of Fuel Cladding, RCS and Containment barriers. The four-hour interval to restore AC power to either unit ESF bus is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, “Station Blackout”.

The likelihood of restoring at least one ESF bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. The ESF busses may be powered from any of the following onsite sources:

- Emergency Diesel Generator 1A(2A) for 4160-V ESF bus 141(241)
- Emergency Diesel Generator 1B(2B) for 4160-V ESF bus 142(242)

Offsite AC power sources feed the ESF busses through the System Auxiliary Transformers 142-1(242-1) and 142-2(242-2). The ESF busses of the affected unit can be powered from the unaffected unit through the crosstie breakers ACB 1414(2414) and ACB 1424(2424). Unit crosstie is considered an adequate source of offsite power when evaluating this EAL.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont):

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

Core Cooling Critical Safety Function (CSF) RED path conditions exist when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200° F. This condition indicates subcooling has been lost and that some fuel cladding damage may potentially occur.

Core Cooling Critical Safety Function (CSF) ORANGE path conditions exist when:

- The average of the ten highest reading core exit thermocouples (CETCs) is reading less than 1200° F but greater than or equal to 700° F, AND
- RCS subcooling based on CETCs is less than ACCEPTABLE VALUE per Iconic Display or RCS Subcooling Margin Figure 1/2 BwST 2-1.

Either of these conditions indicates significant core exit superheating and core uncovery. This is considered a loss of the Fuel Cladding barrier.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG1 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, SG1
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BwOA ELEC-3 Loss Of 4KV ESF Bus
5. 1/2 BwOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BwCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BwCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BwCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BwCA-0.3 Response To Opposite Unit Loss Of All AC Power
10. BwOP AP-37 Unit Two SAT Crosstie To Unit One ESF Bus
11. BwOP AP-38, Unit One SAT Crosstie To Unit Two ESF Bus
12. Safety Evaluations of the Byron Station and Braidwood Station Responses to the Station Blackout (SBO) Rule (TAC NOS. 68522, 68523 AND 68515, 68516)
13. 1/2 BwST-2 Core Cooling
14. 1/2 BwFR-C.1 Response to Inadequate Core Cooling
15. 1/2 BwFR-C.2 Response to Degraded Core Cooling

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS1

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2).

   AND

2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses.

   AND

3. Failure to restore power to at least one unit ESF bus within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power compromises all plant safety systems and represents failures of plant functions required for the protection of the public. The ESF busses may be powered from any of the following onsite sources:

- Emergency Diesel Generator 1A(2A) for 4160-V ESF bus 141(241)
- Emergency Diesel Generator 1B(2B) for 4160-V ESF bus 142(242)

Offsite AC power sources feed the ESF busses through the System Auxiliary Transformers 142-1(242-1) and 142-2(242-2). The ESF busses of the affected unit can be powered from the unaffected unit through the crosstie breakers ACB 1414(2414) and ACB 1424(2424). Unit crosstie is considered an adequate source of offsite power when evaluating this EAL.

Consideration should be given to available loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not available on the energized bus, the bus should not be considered available.

The fifteen-minute interval begins from the time of loss of both onsite and offsite AC power and was selected as a threshold to exclude transient or momentary power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis Reference(s):

1. NEI 99-01 Rev 4, SS1
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BwOA ELEC-3 Loss Of 4KV ESF Bus
5. 1/2 BwOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BwCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BwCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BwCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BwCA-0.3 Response To Opposite Unit Loss Of All AC Power
10. BwOP AP-37 Unit Two SAT Crosstie To Unit One ESF Bus
11. BwOP AP-38, Unit One SAT Crosstie To Unit Two ESF Bus
12. Safety Evaluations of the Byron Station and Braidwood Station Responses to the Station Blackout (SBO) Rule (TAC NOS. 68522, 68523 AND 68515, 68516)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA1

Initiating Condition:

AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. AC power capability to unit ESF busses reduced to only one of the following power sources for **> 15 minutes**:
   - Affected unit SAT 142-1(242-1) **OR** 142-2(242-2)
   - DG 1A(2A) **OR** DG 1B(2B)
   - Unit crosstie breakers

   AND

2. Any additional single power source failure will result in unit blackout.

Basis:

Capability: (pertaining to electrical power supplies) IS equipment that is available to provide and maintain AC power at the required voltage and frequency for the required load.

The reduction of available reliable power sources to a condition in which any additional single failure will result in a Unit Blackout is a substantial degradation in the level of safety of the plant. A Unit Blackout is a loss of AC power to all unit ESF busses. Braidwood blackout coping duration is four hours.

The listed power supplies take into account sources that, if unavailable, establish single-failure vulnerability. This EAL allows for the use of the unit crosstie breaker if they are the only source of power to the affected unit. The Emergency Director must consider the use of the crosstie breaker and the consequent demand on the unaffected unit.

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA1 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, SA5
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BwOA ELEC-3 Loss Of 4KV ESF Bus
5. 1/2 BwOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BwCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BwCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BwCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BwCA-0.3 Response To Opposite Unit Loss Of All AC Power
10. BwOP AP-37 Unit Two SAT Crosstie To Unit One ESF Bus
11. BwOP AP-38, Unit One SAT Crosstie To Unit Two ESF Bus
12. Safety Evaluations of the Byron Station and Braidwood Station Responses to the Station Blackout (SBO) Rule (TAC NOS. 68522, 68523 AND 68515, 68516)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU1

Initiating Condition:

Loss of all offsite power to essential busses for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3, 4, 5, 6

EAL Threshold Values:

Loss of power to Transformers 142-1(242-1) AND 142-2(242-2) for **> 15 minutes**.

Basis:

The Essential busses are the safety-related, ESF busses 141(241) and 142(242). Each Unit 1(Unit 2), 4160-V ESF bus is provided offsite power from the 345-kV system through System Auxiliary Transformer 142-1(242-1) directly to 4160-V ESF bus 141(241).

Each ESF bus for each unit is equipped with an onsite Emergency Diesel Generator; DG 1A(2A) for 4160-V ESF bus 141(241) and DG 1B(2B) for 4160-V ESF bus 142(242). Emergency Diesel Generators for the affected unit should automatically start and be available to carry the essential loads. Balance of plant systems that would assist in plant operations (e.g., condensate pumps, etc.) may be unavailable due to the loss of power.

A loss of offsite AC power reduces the required redundancy and potentially degrades the level of safety of the unit by rendering the station more vulnerable to a complete loss of AC power.

The intent of this EAL is to declare an Unusual Event when offsite power has been lost and the emergency diesel generators have successfully started and energized their respective ESF busses. The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU1 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, SU1 & CU3
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BwOA ELEC-3 Loss Of 4KV ESF Bus
5. 1/2 BwOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BwCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BwCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BwCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BwCA-0.3 Response To Opposite Unit Loss Of All AC Power
10. BwOP AP-37 Unit Two SAT Crosstie To Unit One ESF Bus
11. BwOP AP-38, Unit One SAT Crosstie To Unit Two ESF Bus

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA2

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability:

5, 6, D

EAL Threshold Values:

1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2).

   AND

2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses.

   AND

3. Failure to restore power to at least one unit ESF bus within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power when in Cold Shutdown, Refueling or Defueled mode compromises safety systems required for decay heat removal and represents a substantial degradation of the level of safety of the plant. An Alert declaration (instead of a Site Area Emergency under EAL MS1) is appropriate in these modes because post-shutdown, decay heat energy levels offer more time to restore AC power to heat removal systems than the levels present when the reactor is in Power Operations, Startup, Hot Standby or Hot Shutdown mode. Thus, the threat to the protection of the health and safety of the public is less severe.

The ESF busses may be powered from any of the following onsite sources:

- Emergency Diesel Generator 1A(2A) for 4160-V ESF bus 141(241)
- Emergency Diesel Generator 1B(2B) for 4160-V ESF bus 142(242)

Offsite AC power sources feed the ESF busses through the System Auxiliary Transformers 142-1(242-1) and 142-2(242-2). The ESF busses of the affected unit can be powered from the unaffected unit through the crosstie breakers ACB 1414(2414) and ACB 1424(2424). Unit crosstie is considered an adequate source of offsite power when evaluating this EAL.

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

The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses.

MA2 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, CA3
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BwOA ELEC-3 Loss Of 4KV ESF Bus
5. 1/2 BwOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BwCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BwCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BwCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BwCA-0.3 Response To Opposite Unit Loss Of All AC Power
10. BwOP AP-37 Unit Two SAT Crosstie To Unit One ESF Bus
11. BwOP AP-38, Unit One SAT Crosstie To Unit Two ESF Bus
12. Safety Evaluations of the Byron Station and Braidwood Station Responses to the Station Blackout (SBO) Rule (TAC NOS. 68522, 68523 AND 68515,68516)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG3

Initiating Condition:

Failure of the Reactor Protection System to complete an automatic trip and manual trip was NOT successful and there is indication of an extreme challenge to the ability to cool the core.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. Automatic and manual Reactor Trip were not successful from Main Control Board as indicated by:
    a. Reactor power **≥ 5%**
    OR
    b. Intermediate Range Start Up Rate is positive

    AND

2. a. **Core Cooling CSF – RED Path** conditions exist.
    OR
    b. **Heat Sink CSF – RED Path** conditions exist.

Basis:

Automatic trip and manual trip are not considered successful if action away from the Main Control Board was required to trip the reactor.

This EAL is not applicable if a manual trip is initiated and no RPS setpoints are exceeded.

The reactor power level is the equivalent to the Safety System Design Heat Capacity. Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. Additionally, a continuing temperature rise indicates that this situation could be a precursor for a core melt sequence.

A successful trip has occurred when there is sufficient rod insertion to bring the reactor below Safety System Design Heat Capacity (less than 5%). Subcriticality Critical Safety Function (CSF) RED path is entered based on failure of power range indication to lower below 5% following a reactor trip.

Reactor power levels in the power range are indicated on N-41, 42, 43 and 44. This addresses any manual trip or automatic trip signal followed by a manual trip that fails to shut down the reactor to an extent that the reactor is producing more heat load for which the safety systems were designed.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG3 (cont)

Basis (cont):

A manual trip is any set of actions by the Reactor Operator(s) at the main control board which causes control rods to be rapidly inserted into the core and brings power below that percent power (5%) associated with the ability of the safety systems to remove heat. Automatic and manual trips are not considered successful if action away from the main control board is required to trip the reactor. Note that the operating mode changes to Hot Standby as soon as a successful reactor trip occurs.

Core Cooling Critical Safety Function RED path condition exists when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200° F. This condition indicates subcooling has been lost and that some fuel cladding damage may potentially occur.

The Heat Sink Critical Safety Function Red path condition exists if narrow range levels in all steam generators (S/Gs) are less than or equal to 10% - Unit 1 (31% adverse containment) and 14% - Unit 2 (34% adverse containment) and total feedwater flow to all S/Gs is less than or equal to 500 gpm. If total feed flow is less than 500 gpm due to procedurally directed operator actions then this condition does not apply.

The combination of these conditions (reactor power greater than or equal to 5% and either Core Cooling-RED path or Heat Sink-RED path) indicates the ultimate heat sink function is under extreme challenge. A major consideration is the inability to initially remove heat during the early stages of this sequence.

In the event this challenge occurs at a time when the reactor has not been brought below the power associated with safety system design power (5%), a core melt sequence may exist and rapid degradation of the fuel cladding could begin. To permit maximum offsite intervention time, the General Emergency declaration is therefore appropriate in anticipation of an inevitable General Emergency declaration due to loss and potential loss of fission product barriers.

Basis Reference(s):

1. NEI 99-01 Rev 4, SG2
2. 1/2 BwST-1 Subcriticality
3. 1/2 BwST-2 Core Cooling
4. 1/2 BwST-3 Heat Sink
5. 1/2 BwFR-S.1 Response to Nuclear Power Generation/ATWS
6. 1/2 BwFR-H.1 Response to Loss of Secondary Heat Sink
7. 1/2 BwFR C.1 Response to Inadequate Core Cooling
6\. 1/2 BwOSR 0.1-1,2,3 Unit One(Two) Modes 1, 2, And 3 Shiftly and Daily Operating Surveillance

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS3

Initiating Condition:

Failure of Reactor Protection System to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded and manual trip was NOT successful.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

Automatic and manual Reactor Trip were not successful from Main Control Board as indicated by:

a. Reactor power **≥ 5%**

OR

b. Intermediate Range Start Up Rate is positive

Basis:

Automatic trip and manual trip are not considered successful if action away from the Main Control Board was required to trip the reactor.

This EAL is not applicable if a manual trip is initiated and no RPS setpoints are exceeded.

The reactor power level is the equivalent to the Safety System Design Heat Capacity. Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS.

For the intent of this EAL a successful trip has occurred when there is sufficient rod insertion to bring the reactor below Safety System Design Heat Capacity (less than 5%). Subcriticality Critical Safety Function (CSF) RED path condition is met based on failure of power range indication to lower below 5% following a reactor trip.

This addresses any automatic trip signal followed by a manual trip that fails to shut down the reactor to an extent that the reactor is producing more heat load for which the safety systems were designed. A manual trip is any set of actions by the reactor operator(s) at the main control board which causes control rods to be rapidly inserted into the core and brings power below that percent power (5%) associated with the ability of the safety systems to remove heat. Automatic and manual trips are not considered successful if action away from the main control board is required to trip the reactor.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. Emergency boration is thus required and there is an actual major failure of a system intended for protection of the public.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS3 (cont)

Basis (cont):

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Cladding and RCS barriers and warrants declaration of a Site Area Emergency.

Basis Reference(s):

1. NEI 99-01 Rev 4, SS2
2. 1/2 BwST-1 Subcriticality
3. 1/2 BwFR-S.1 Response to Nuclear Power Generation/ATWS
4. 1/2 BwOSR 0.1-1,2,3 Unit One(Two) Modes 1, 2, And 3 Shiftly and Daily Operating Surveillance

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA3

Initiating Condition:

Failure of the Reactor Protection System to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. A Reactor Protection System setpoint was exceeded.

   AND

2. A successful automatic Reactor Trip did not occur

Basis:

This condition indicates a failure of the automatic reactor protection system to successfully 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 and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue.

In the event that the operator identifies a reactor trip is imminent and successfully initiates a manual reactor trip before the automatic trip setpoint is reached, no declaration is required.

Basis Reference(s):

1. NEI 99-01 Rev 4, SA2
2. 1/2 BwST-1 Subcriticality
3. 1/2 BwFR-S.1 Response to Nuclear Power Generation/ATWS
4. 1/2 BwOSR 0.1-1,2,3 Unit One(Two) Modes 1, 2, And 3 Shiftly and Daily Operating Surveillance

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU3

Initiating Condition:

Inadvertent criticality.

Operating Mode Applicability:

3, 4, 5, 6

EAL Threshold Values:

An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

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

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

This EAL includes criticality events that occur in Cold Shutdown or Refueling modes (NUREG1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events as well as inadvertent criticalities occurring in Hot Shutdown mode. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification.

This condition can be identified using:

- Source Range startup rate channels 1/2 NI-31D (SR STARTUP RATE CH 31) or 1/2 NI-32D (SR STARTUP RATE CH 32)
- Intermediate Range channels 1/2 NI-35B and 1/2 NI-36B
- Nuclear Instrumentation System Source Range/Audio Count Rate Containment Indications
- Post Accident Monitoring System (Gamma Metrics) – Provides measure of flux level from shutdown (0.1 cps) through 200% power level

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU3 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, SU8 & CU8
2. 1BwOS XCB-R1 U0 AND U1 MCR Meter Color Banding
3. 1/2 BwFR-S.1, Response To Nuclear Power Generation/ATWS Unit 1/2
4. Bw TS LCO 3.3.1 RTS Instrumentation
5. 1/2 BwOSR 0.1-4, Unit One(Two) Modes 4 Shiftly And Daily Operating Surveillance
6. 1/2 BwGP 100-2 Plant Startup
7. 1/2 BwGP 100-2T1, 1/2 BwGP 100-2 Flowchart
8. 1/2 BwGP 100-6T4 Defueled To Mode 6 Checklist
9. Regulatory Guide 8.12, Criticality Accident Alarm Systems

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS4

Initiating Condition:

Loss of all vital DC power.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

Loss of all vital DC power based on **< 108 VDC** on 125 VDC battery busses 111(211) and 112(212) for **> 15 minutes**.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of Containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. Battery bus 111(211) or 112(212) are the safety-related, Class 1E 125 VDC power systems.

Each safety-related battery bus, 111(211) or 112(212), has the capacity to continuously supply all the connected normal running loads while maintaining its respective battery in a fully charged condition. Each battery has a guaranteed nominal rating of 2320 ampere-hours at the 8-hour rate to an end voltage of 1.75 volts per cell (or 1.75 VDC/cell x 58 cells = 101.5 VDC).

Each battery was sized based upon supplying the design duty cycle (1- hour overall duration) in the event of a loss of offsite AC power concurrent with a LOCA and a single failure of a diesel generator. If bus voltage drops to 123 VDC, a Control Room annunciator alarms (e.g., BwAR 1-21-E10 for bus 111 and 113). Each battery is equipped with a battery charger that is rated to supply its associated DC loads while fully recharging the battery. Each battery charger is fed from a 480 VAC ESF switchgear bus of the same division. The minimum design voltage limit of each battery is 108 VDC. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate loads.

Basis Reference(s):

1. NEI 99-01 Rev 4, SS3
2. UFSAR 8.3.2.1.1
3. 20E-0-4001 Station One Line Diagram
4. BwAR 1/2-21-E10, 125V DC PNL 111/113(211/213) VOLT LOW
5. 1/2 BwOA ELEC 1 Loss of DC Bus UNIT 1/2
6. BwAR 1/2-22-E10, 125V DC PNL 112/114 (212/214) VOLT LOW

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU4

Initiating Condition:

UNPLANNED loss of required DC power for greater than 15 minutes.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. UNPLANNED loss of all required vital DC power based on **< 108 VDC** indication on 125 VDC battery busses 111(211) and 112(212).

   AND

2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.

Basis:

<u>UNPLANNED</u>: A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

“Unplanned activities” is included in this EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, plants perform maintenance on a bus-related basis during shutdown periods.

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown, refueling or defueled operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. Battery bus 111(211) or 112(212) are the safety-related, Class 1E 125 VDC power systems.

Each safety-related battery bus, 111(211) or 112(212), has the capacity to continuously supply all the connected normal running loads while maintaining its respective battery in a fully charged condition. Each battery has a guaranteed nominal rating of 2320 ampere-hours at the 8-hour rate to an end voltage of 1.75 volts per cell (or 1.75 VDC/cell x 58 cells = 101.5 VDC).

Each battery was sized based upon supplying the design duty cycle (1- hour overall duration) in the event of a loss of offsite AC power concurrent with a LOCA and a single failure of a diesel generator. If bus voltage drops to 123 VDC, a Control Room annunciator alarms (e.g., BwAR 1-21-E10 for bus 111 and 113). Each battery is equipped with a battery charger that is rated to supply its associated DC loads while fully recharging the battery. Each battery charger is fed from a 480 VAC ESF switchgear bus of the same division. The minimum design voltage limit of each battery is 110 VDC. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate loads.

Basis Reference(s):

1. NEI 99-01 Rev 4, CU7
2. UFSAR 8.3.2.1.1
3. 20E-0-4001 Station One Line Diagram
4. BwAR 1/2-21-E10, 125V DC PNL 111/113(211/213) VOLT LOW
5. 1/2 BwOA ELEC 1 Loss of DC Bus UNIT 1/2
6. BwAR 1/2-22-E10, 125V DC PNL 112/114 (212/214) VOLT LOW

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS5

Initiating Condition:

Complete loss of heat removal capability.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. **Core Cooling CSF - RED Path** conditions exist.

   AND

2. **Heat Sink CSF - RED Path** conditions exist.

Basis:

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature.

The Core Cooling Critical Safety Function RED path conditions exist when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200° F. This condition indicates subcooling has been lost and that some fuel cladding damage may potentially occur.

The Heat Sink Critical Safety Function Red path conditions exist if narrow range levels in all steam generators (S/Gs) are less than or equal to 10% - Unit 1 (31% adverse containment) and 14% - Unit 2 (34% adverse containment) and total feedwater flow to all S/Gs is less than or equal to 500 gpm. If total feed flow is less than 500 gpm due to procedurally directed operator actions then this condition does not apply.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted.

Basis Reference(s):

1. NEI 99-01 Rev 4, SS4
2. 1/2 BwST-2 Core Cooling
3. 1/2 BwST-3 Heat Sink
4. 1/2 BwFR C.1, Response to Inadequate Core Cooling
5. 1/2 BwFR H.1, Response to Secondary Heat Sink

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5

Initiating Condition:

Inability to maintain plant in Cold Shutdown with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F** for > Table M1 duration.

Table M1 – RCS Reheat Duration Thresholds

RCS	Containment Closure	Duration
Intact	N/A	60 minutes*
Reduced Inventory (< 397 ft.)	Established	20 minutes*
	Not Established	0 minutes
Not Intact	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. UNPLANNED Reactor Vessel pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.

Basis:

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

CONTAINMENT CLOSURE: Containment Closure is considered to be as required by procedures.

Containment closure status is checked and verified using the Containment Closure Reactor Operator Checklist of procedure 1/2 BwOS XPC-W1.

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 main steam line nozzle plugs, etc.).

Basis (cont):

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as decay heat removal system design and Reactor Vessel water level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200° F).

Threshold #1 Basis:

The first condition in Table M1 addresses complete loss of functions required for core cooling for greater than sixty minutes during Refueling and Cold Shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals, etc.). With containment closure established, a low-pressure barrier to fission product release exists. In this condition, containment status is of less importance than the status of RCS integrity because the RCS is intact and providing a high-pressure barrier to fission product release. The sixty-minute interval should allow sufficient time to restore cooling without a substantial degradation in plant safety. The asterisk highlights the note at the bottom of the table. The note indicates that the first condition is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the sixty-minute interval.

The second condition in Table M1 addresses the complete loss of functions required for core cooling for greater than twenty minutes during Refueling and Cold Shutdown modes when containment closure is established but RCS integrity is not established or Reactor Vessel inventory is reduced. RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals, etc.). RCS inventory is in a reduced condition when water level is three feet below the Reactor Vessel flange (400 ft. el. - 3 ft. = 397 ft. el.).

The allowed twenty-minute interval is 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 barrier to fission product release is established (i.e., containment closure). The asterisk highlights the note at the bottom of the table. The note indicates that the second condition is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the twenty-minute interval.

Basis (cont):

The third condition in Table M1 addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when containment closure and RCS integrity are not established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation. No delay time is allowed for this condition because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above 200° F when the heat removal function is available.

Threshold #2 Basis:

The 10 psig pressure rise due to loss of decay heat removal infers an intact RCS with uncontrolled RCS temperature rise in excess of the Technical Specification cold shutdown limit (200° F) for which MA5 Threshold #1 would permit up to sixty minutes to restore RCS cooling before declaration of an Alert. This EAL therefore covers situations of high decay heat loads, in which the event should be declared without delay.

NRC analyses show that sequences can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Basis Reference(s):

1. NEI 99-01 Rev 4, CA4
2. 1/2 BwOS XPC-W1 Unit One(Two) Containment Penetration Status Weekly Surveillance
3. 1/2 BwOSR 3.4.3.1 Reactor Coolant System Pressure Temperature Limit Surveillance
4. 1/2 BwGP 100-1 Plant Heatup
5. 1/2 BwGP 100-5, Plant Shutdown and Cool Down
6. 1/2 BwGP 100-6, Refueling Outage

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU5

Initiating Condition:

UNPLANNED loss of decay heat removal capability with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F**.

   OR

2. Loss of all RCS temperature AND Reactor Vessel level indication for **> 15 minutes**.

Basis:

<u>UNPLANNED</u>: A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, 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. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the fuel cladding may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel.

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in RCS/Reactor Vessel temperatures depending on the time since shutdown.

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 Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling modes, the second condition of this EAL would result in declaration of an Unusual Event if temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont):

Reactor Vessel water level is normally monitored using the following instruments:

- LT-046, LI-RY-046, Reactor Vessel Refueling Level Indicator
- LT-049, LI-RY-049, Reactor Vessel Refueling Level Indicator
- LT-048, LI-RY-048, Refueling Cavity Water Level
- RVLIS (1LI-RC019, 1LI-RC020)
- LT-047 (LI-RY-047), 413 ft. el. to 426 ft. el.

LT-046 and LT-049 provide Reactor Vessel water level indication in the Control Room during RCS reduced inventory conditions. They are calibrated to indicate level from approximately 392 ft. el. to 402.5 ft. el. The Reactor Vessel flange is at 400 ft. el. LT-048 is calibrated to read refueling cavity water level from 392 ft. el. to the refuel floor level 426 ft. el. LT-047 provides refueling cavity water level in the range of 413 ft. el. to 426 ft. el.

Two independent methods of RCS level indication must be functional and monitored prior to draining below 402 ft. el. If any discrepancies greater than 0.5 ft. between operable level indicators occur, draining is immediately secured until actual level has been verified. Two independent methods of RCS level indication must be continuously monitored when in a Reduced Inventory condition. Two independent RVLIS trains normally remain in service whenever the RCS is intentionally drained and the Reactor Vessel head is in place (except when in the process of removing/installing the head).

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200° F). These include:

- RCS Loop A Cold Leg Wide Range 1/2 TI-413B T0406 1TR-413A
- RCS Loop B Cold Leg Wide Range 1/2 TI-423B T0426 1TR-413B
- RCS Loop C Cold Leg Wide Range 1/2 TI-433B T0446 1TR-433A
- RCS Loop D Cold Leg Wide Range 1/2 TI-443B T0466 1TR-433B
- RCS Loop A Hot Leg Wide Range 1/2 TI-413A T0419 1TR-413A
- RCS Loop B Hot Leg Wide Range 1/2 TI-423A T0439 1TR-413B
- RCS Loop C Hot Leg Wide Range 1/2 TI-433A T0459 1TR-433A
- RCS Loop D Hot Leg Wide Range 1/2 TI-443A T0479 1TR-433B
- 1A RH Pump Discharge Temp. 1/2 TI-612 T0630 1TR-612
- 1B RH Pump Discharge Temp. 1/2 TI-613 T0631 1TE-613

MU5 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, CU4
2. Technical Specifications Table 1.1-1
3. 1/2 BwOSR 0.1-6 Unit One(Two) Mode 6 Shiftly And Daily Operating Surveillance
4. BwOP RH-9 Pump Down of the Reactor Cavity to the RWSTs
5. BwOP RC-4 Reactor Coolant System Drain
6. 1/2 BwOSR 3.3.3.1 Unit One(Two) Accident Monitoring Instrumentation Monthly Channel Checks
7. 1/2 BwOL 3.4.15, LCOAR - Reactor Coolant System Leakage - Leakage Detection Systems
8. 1/2 BwOL 3.4.13, LCOAR - Reactor Coolant System - Operational Leakage
9. 1/2 BwOSR 3.4.3.1 Reactor Coolant System Pressure Temperature Limit Surveillance

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6

Initiating Condition:

Inability to monitor a SIGNIFICANT TRANSIENT in progress.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. Loss of most **(approximately 75%)** safety system annunciators (Table M2).

Table M2 – Control Room Panels
• 1/2 PM01J MCB Gen & Aux Power
• 1/2 PM05J MCB Reactor and Chem Volume Control
• 1/2 PM06J MCB Eng. Safety Features

AND

2. Indications needed to monitor safety functions (Table M3) are unavailable.

Table M3 – Safety Functions and Related Systems
• 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)
• Fission Product Barriers

AND

3. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Automatic Turbine Runback **> 25%** thermal reactor power
• Electrical load rejection **> 25%** full electrical load
• Reactor Trip
• Safety Injection Actuation
• Thermal power oscillations **> 10%**

AND

4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

Basis:

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS and PPDS.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) automatic turbine 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%.

Planned and UNPLANNED actions are not differentiated since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not a factor.

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in 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.

A Site Area Emergency exists if the Control Room staff cannot monitor safety functions needed for protection of the public. Indications needed to monitor critical safety functions necessary for protection of the public must include Control Room indications, computer generated indications and dedicated annunciation capability. The specific parameters should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled and in a coolable geometry, remove heat from the core, and maintain the reactor coolant system and containment intact. These parameters are monitored and controlled in the emergency operating procedures.

Symptoms of a loss of annunciators can be:

- 1/2-4-A7, AN SYS PWR SUP TROUBLE
- 1/2-4-D7, AN SYS GROUND
- 1/2-4-C7, AN SYS ISOL CAB PWR SUP TROUBLE
- 1/2-4-E7, AN SYS FIELD CONTACT PWR TROUBLE

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, SS6
2. Drawing 20E-0-3372B Auxiliary Building Main Control Room Panel El 451'
3. BwAP 300-1A1 At the Controls and Horse-Shoe Areas
4. BwOP AN-1 Plant Annunciator System Startup and Operation
5. UFSAR E.17
6. 1/2 BwST-1 Subcriticality
7. 1/2 BwST-2 Core Cooling
8. 1/2 BwST-3 Heat Sink
9. 1/2 BwST-4 Integrity
10. 1/2 BwST-5 Containment
11. 1/2 BwST-6 Inventory

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 1/2 PM01J MCB Gen & Aux Power
• 1/2 PM05J MCB Reactor and Chem Volume Control
• 1/2 PM06J MCB Eng. Safety Features

OR

b. UNPLANNED loss of most **(approximately 75%)** of indications associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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)
• Fission Product Barriers

AND

2. a. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Automatic Turbine Runback **> 25%** thermal reactor power
• Electrical load rejection **> 25%** full electrical load
• Reactor Trip
• Safety Injection Actuation
• Thermal power oscillations **> 10%**

OR

b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

Basis:

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

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) automatic turbine 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%.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS and PPDS.

For this EAL "most" is approximately 75% of the safety system annunciators or indicators being lost, resulting in 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without Reactor Control, ECCS, Electrical panel, critical safety function, and process/area radiation annunciation or indication equipment. The availability of computer based indication equipment is considered.

Symptoms of a loss of annunciators can be:

- 1/2-4-A7, AN SYS PWR SUP TROUBLE
- 1/2-4-D7, AN SYS GROUND
- 1/2-4-C7, AN SYS ISOL CAB PWR SUP TROUBLE
- 1/2-4-E7, AN SYS FIELD CONTACT PWR TROUBLE

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification. The fifteen-minute interval offers time to recover from transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01 Rev 4, SA4
2. Drawing 20E-0-3372B Auxiliary Building Main Control Room Panel El 451'
3. BwAP 300-1A1 At the Controls and Horse-Shoe Areas
4. BwOP AN-1 Plant Annunciator System Startup and Operation
5. UFSAR E.17

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 1/2 PM01J MCB Gen & Aux Power • 1/2 PM05J MCB Reactor and Chem Volume Control • 1/2 PM06J MCB Eng. Safety Features

OR

2. UNPLANNED loss of most **(approximately 75%)** of indicators associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

Basis:

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

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without Reactor Control, ECCS, Electrical panel, critical safety function, and process/area radiation annunciation or indication equipment. The availability of computer based indication equipment is considered.

Basis (cont):

Symptoms of a loss of annunciators can be:

- 1/2-4-A7, AN SYS PWR SUP TROUBLE
- 1/2-4-C7, AN SYS ISOL CAB PWR SUP TROUBLE
- 1/2-4-D7, AN SYS GROUND
- 1/2-4-E7, ANS SYS FIELD CONTACT PWR TROUBLE

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The fifteen-minute interval offers time to recover from transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01 Rev 4, SU3
2. Drawing 20E-0-3372B Auxiliary Building Main Control Room Panel El 451'
3. BwAP 300-1A1 At the Controls and Horse-Shoe Area
4. BwOP AN-1 Plant Annunciator System Startup and Operation
5. UFSAR E.17

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU7

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

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

   OR

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

Basis:

The conditions of this EAL may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. Positive indications in the Control Room of RCS leakage include one or more of the following:

- PZR pressure decreasing
- PZR level decreasing
- VCT level decreasing
- Charging flow greater than expected
- VCT makeup frequency increased
- RCP seal injection flow(s) abnormal
- PRT pressure, temperature, or level increasing
- Containment pressure, temperature, or humidity increasing
- Containment Recirculation sump level increasing
- Containment floor drain or Reactor cavity sump flow increasing
- Containment radiation levels increasing
- Auxiliary Building radiation levels increasing
- CNMT DRAIN LEAK DETECT FLOW HIGH alarm (1/2-1-1-A2)
- RX VESSEL FLNG LEAKOFF TEMP HIGH alarm (1/2-1-14-E5)

The 10 gpm value for the unidentified leakage and pressure boundary leakage was selected because it is quantifiable with normal Control Room leak detection methods. Station surveillance procedures identify indications to verify leakage from the RCS. A hand calculation or the Process Computer RCS Leakrate Code is used in determining RCS leakage.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU7 (cont)

Basis:

The 25 gpm value for identified leakage is set at a higher value because of the significance of identified leakage in comparison to unidentified or pressure boundary leakage.

No classification under this threshold is made for relief valve operation where the relief valve functions as designed.

Basis Reference(s):

1. NEI 99-01 Rev 4, SU5
2. Technical Specifications 3.4.13 & 3.4.14
3. UFSAR 6.2, 5.24
4. 1/2 BwOSR 3.4.13.1 Unit One(Two) Reactor Coolant System Water Inventory Balance Surveillance
5. BwCP 310-4 Steam Generator - Primary to Secondary Tube Leak Rate
6. 1/2 BwOL 3.4.15 LCOAR - Reactor Coolant System Leakage - Leakage Detection Systems
7. 1/2 BwOL 3.4.13 LCOAR - Reactor Coolant System - Operational Leakage
8. 1/2 BwOA PRI-1 Excessive Primary Leakage Unit 1/2
9. 1/2 BwOSR 0.1-4 Unit One(Two) Modes 4 Shiftly and Daily Operating Surveillance
10. 1/2 BwOS RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
11. 1/2 BwOS XCB-R1 U0 and U1 MCR Meter Color Banding

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8

Initiating Condition:

Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. Loss of Reactor Vessel inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss

AND

2. a. RVLIS **≤ 0% Plenum (390 ft. el.)** for **> 30 minutes**.

OR

b. Reactor Vessel level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following:

- 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high.
- Erratic Source Range Monitor indication.

AND

3. Containment is challenged as indicated by one or more of the following:

- Hydrogen concentration in Containment **≥ 5%.**
- Containment pressure **≥ 50 psig.**
- CONTAINMENT CLOSURE not established.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be as required by procedures.

Containment closure status is checked and verified using the Containment Closure Reactor Operator Checklist of procedure 1/2 BwOS XPC-W1.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8 (cont)

Basis (cont):

The Reactor Vessel Level Indication System (RVLIS) provides the following eight level indicators:

Region	Sensor	Indication	Location
Head Region	1	31%	Near top of Reactor Vessel
	2	0%	Near upper internals support plate
Plenum Region	3	81%	400.7 ft. el. (halfway between upper internals support plate and top of the hot leg)
	4	55%	397.5 ft. el. (Reduced Inventory is 397 ft)
	5	37%	394.2 ft. (top of hot leg)
	6	27%	393 ft. (CL of hot leg)
	7	15%	392.4 ft. (bottom of hot leg)
	8	0%	390 ft. (top of core is 388 ft. 10 in.)

Available RVLIS level indication as close as possible to NEI 99-01 Guidance was chosen for the threshold levels.

Threshold #1 and #2 Basis:

When Reactor Vessel water level drops to RVLIS 0% Plenum (sensor #8), water level has reached 390 ft. el. and core uncovery is about to occur. The top of the core is at 388 ft. 10 in. el. Fuel damage is probable if core uncovery is prolonged and submergence cannot be restored and maintained. Available decay heat will cause boiling and further drop Reactor Vessel water level.

This EAL is 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, (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, etc.) can have a significant affect on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

When Reactor Vessel water level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated containment radiation or unexplained rise in Containment sump levels. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors 1/2 NI-31B and 1/2 NI-32B can be used as a tool for making such determinations.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8 (cont)

Basis (cont):

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The Containment Fuel Handling Incident Radiation Monitors 0RE-AR011 and 0RE-AR012 indication of > 3000 mR/hr. is based on calculation EP-EAL-0501.

- 1/2 RE-AR011 Containment Fuel Handling Incident Monitor - RA0039 for Unit 1/ RA0064 for Unit 2
- 1/2 RE-AR012 Containment Fuel Handling Incident Monitor - RA0040 for Unit 1/ RA0065 for Unit 2

1/2 BwOS RF-1 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting when the Containment Floor Drain Sump annunciator setpoint is exceeded or 1/2 BwOS RF-2 surveillance has failed.

The Containment Floor Drain Sump level indicated range is 0 to 106 in. The Containment Recirculation sump level indicated range is 14 to 150 in. Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment Sump level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area.

Threshold #3 Basis:

Three conditions are associated with the challenge to containment integrity:

- Containment closure is the action taken to secure containment and its assorted structures, systems and components as a functional barrier to fission product release under existing plant conditions. Containment closure is initiated per shift management direction if plant conditions change that could increase the risk of a fission product release as a result of a loss of decay heat removal.
- If hydrogen concentration reaches or exceeds 5% in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside Containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred. Containment hydrogen concentration is indicated on 1/2 HSU-PS345 and 1/2 HSU-PS346.
- The containment design pressure (50 psig) is well in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both RCS and fuel clad barriers in that it is not possible to reach this condition without severe core degradation.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, CG1
2. 1/2 BwOS XPC-W1 Unit One(Two) Containment Penetration Status Weekly Surveillance
3. UFSAR E.17
4. BwOP RH-9 Pump Down of the Reactor Cavity to the RWSTs
5. BwOP RC-4 Reactor Coolant System Drain
6. UFSAR 6.2
7. 1/2 BwOSR 0.1-4 Unit One(Two) Modes 4 Shiftly and Daily Operating Surveillance
8. 1/2 BwOS RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
9. 1/2 BwOS XCB-R1 U0 and U1 MCR Meter Color Banding
10. 1/2 BwGP 100-2 Plant Startup
11. 1/2 BwGP 100-6T4 Defueled to Mode 6 Checklist
12. 1/2 BwOSR 3.3.3.1 Unit One(Two) Accident Monitoring Instrumentation Monthly Channel Checks
13. 1/2 BwFR-C.1, Response to Inadequate Core Cooling Unit 1/2
14. 1/2 BwST-5 Containment
15. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
16. UFSAR stat Section 6.2.5.2.1
17. EP-EAL-0501, Estimation Of Radiation Monitor Readings Indicating Core Uncovery During Refueling

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS8

Initiating Condition:

Loss of Reactor Vessel inventory affecting core decay heat removal capability.

Operating Mode Applicability:

5

EAL Threshold Values:

1. **Without** CONTAINMENT CLOSURE established:
   a. Reactor Vessel inventory as indicated by RVLIS **≤ 15% Plenum (392.4 ft. el.)**

   OR

   b. Reactor Vessel level unknown for **> 30 minutes** with a loss of Reactor Vessel inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss

OR

2. **With** CONTAINMENT CLOSURE established:
   a. Reactor Vessel inventory as indicated by RVLIS **≤ 0% Plenum (390 ft. el.)**

   OR

   b. Reactor Vessel level unknown for **> 30 minutes** with a loss of Reactor Vessel inventory as evidenced by either of the following:
      - Per Table M5 indications.
      - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be as required by procedures.

Containment closure status is checked and verified using the Containment Closure Reactor Operator Checklist of procedure 1/2 BwOS XPC-W1.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS8 (cont)

Basis (cont):

If a low-pressure boundary to fission product release does not exist (i.e., containment closure is not established), the level associated with this threshold is six inches below the bottom inside diameter of the RCS hot leg vessel penetration (i.e., 392.4 ft. el. – 0.5 ft. = 391.9 ft. el.). If containment closure is established, a low-pressure boundary to fission product release exists and water level can drop to the top of active fuel, 388.83 ft. el. before a Site Area Emergency declaration is required. The Reactor Vessel Level Indication System (RVLIS) provides the following eight level indicators:

Region	Sensor	Indication	Location
Head Region	1	31%	Near top of Reactor Vessel
	2	0%	Near upper internals support plate
Plenum Region	3	81%	400.7 ft. el. (halfway between upper internals support plate and top of the hot leg)
	4	55%	397.5 ft. el. (Reduced Inventory is 397 ft)
	5	37%	394.2 ft. (top of hot leg)
	6	27%	393 ft. (CL of hot leg)
	7	15%	392.4 ft. (bottom of hot leg)
	8	0%	390 ft. (top of core is 388 ft. 10 in.)

Available RVLIS level indication as close as possible to NEI 99-01 Guidance was chosen for the threshold levels.

Threshold #1 Basis:

Since there is no direct indication in the Control Room of a level 6 inches below the loop (391.9 ft. el.) RVLIS Sensor # 7 (15% - 392.4 ft. el. bottom of hot leg) was chosen.

If the Reactor Vessel Refueling Level Indicators LT-046 (LI-RY-046) and LT-049 (LI-RY-049) are available they would provide water level indication in the Control Room during RCS reduced inventory conditions and are calibrated to read level to 392 ft. elevation. When readings from these instruments drop off-scale low, the water level threshold with containment closure not established has been reached. RVLIS trend, local indication and perhaps visual observation may also support making this determination.

The thirty-minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS8 (cont)

Basis (cont):

The Containment Floor Drain Sump level indicated range is 0 to 106 in. The Containment Recirculation sump level indicated range is 14 to 150 in. Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment Sump level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area.

Threshold #2 Basis:

Since there is no direct indication in the Control Room of top of active fuel (388 ft. el.) RVLIS Sensor # 8 (390 ft. el.) was chosen. When Reactor Vessel water level drops to RVLIS 0% Plenum (sensor #8), water level has reached 390 ft. el. and core uncovery is about to occur. The inability to restore and maintain water level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Cladding barrier. The closest indication of core uncovery is RVLIS 0% Plenum sensor at 390 ft. el.

When Reactor Vessel water level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated containment radiation or unexplained rise in Containment sump levels. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors 1/2 NI-31B and 1/2 NI-32B can be used as a tool for making such determinations.

The thirty-minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

Basis Reference(s):

1. NEI 99-01 Rev 4, CS1
2. 1/2 BwOS XPC-W1 Unit One (Two) Containment Penetration Status Weekly Surveillance
3. UFSAR E.17
4. BwOP RH-9 Pump Down of the Reactor Cavity to the RWSTs
5. BwOP RC-4 Reactor Coolant System Drain
6. UFSAR 6.2
7. 1/2 BwOSR 0.1-4 Unit One (Two) Modes 4 Shiftly and Daily Operating Surveillance
8. 1/2 BwOS RF-1 Unit One (Two) Containment Floor Drain Monitoring System Non-Routine Surveillance

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS8 (cont)

Basis References (cont):

9. 1/2 BwOS XCB-R1 U0 and U1 MCR Meter Color Banding
10. 1/2 BwOSR 0.1-4, Unit One (Two) Modes 4 Shiftly and Daily Operating Surveillance
11. 1/2 BwGP 100-2 Plant Startup
12. 1/2 BwGP 100-6T4 Defueled to Mode 6 Checklist
13. 1/2 BwOSR 3.3.3.1 Unit One (Two) Accident Monitoring Instrumentation Monthly Channel Checks

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8

Initiating Condition:

Loss of RCS / Reactor Vessel inventory with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. a. Loss of RCS / Reactor Vessel inventory as indicated RVLIS **< 27% Plenum (393 ft. el.)**.

   OR

   b. Loss of RCS / Reactor Vessel inventory as indicated by LT-046 and LT-049 **< 393 ft. el.**

OR

2. a. Loss of RCS / Reactor Vessel inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss

   AND

   b. RCS / Reactor Vessel level unknown for **> 15 minutes.**

Basis:

The threshold value for Reactor Vessel inventory is bottom of RCS Loop Hot Leg (392.4 ft. el.) However because available level indications systems do not provide for an accurate way to positively identify six inches below the loop level, RVLIS sensor # 7 is used in EAL MS8 as the threshold value. Therefore sensor #6 is used for as the threshold value for MA7 to maintain a sequential classification scheme.

When Reactor Vessel water level drops to 393 ft. el., the centerline of the RCS hot leg is uncovered. This EAL serves as a precursor 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 Reactor Vessel water level drop and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

Basis (cont):

In Cold Shutdown mode, the decay heat available to raise Reactor Vessel temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and water level monitoring means are available. In the Refueling mode, the RCS is not intact and water level and inventory are monitored by different means. In the Refueling mode, normal means of water level indication may not be available. Redundant means of water 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.

Reactor Vessel water level is normally monitored using the following instruments:

- LT-046, LI-RY-046, Reactor Vessel Refueling Level Indicator
- LT-049, LI-RY-049, Reactor Vessel Refueling Level Indicator
- LT-048, LI-RY-048, Refueling Cavity Water Level
- RVLIS (1LI-RC019, 1LI-RC020)
- LT-047 (LI-RY-047), 413 ft. el. to 426 ft. el.

LT-046 and LT-049 provide Reactor Vessel water level indication in the Control Room during RCS reduced inventory conditions. They are calibrated to indicate level from approximately 392 ft. el. to 402.5 ft. el. LT-048 is calibrated to read refueling cavity water level from 392 ft. el. to the refuel floor level 426 ft. el. LT-047 provides refueling cavity water level in the range of 413 ft. el. to 426 ft. el.

Two independent methods of RCS level indication must be functional and monitored prior to draining below 402 ft. el. If any discrepancies greater than 0.5 ft. between operable level indicators occur, draining is immediately secured until actual level has been verified. Two independent methods of RCS level indication must be continuously monitored when in a Reduced Inventory condition. Two independent RVLIS trains normally remain in service whenever the RCS is intentionally drained and the Reactor Vessel head is in place (except when in the process of removing/installing the head).

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8 (cont)

Basis (cont):

The Reactor Vessel Level Indication System (RVLIS) provides the following eight level indicators:

Region	Sensor	Indication	Location
Head Region	1	31%	Near top of Reactor Vessel
	2	0%	Near upper internals support plate
Plenum Region	3	81%	400.7 ft. el. (halfway between upper internals support plate and top of the hot leg)
	4	55%	397.5 ft. el. (Reduced Inventory is 397 ft)
	5	37%	394.2 ft. (top of hot leg)
	6	27%	393 ft. (CL of hot leg)
	7	15%	392.4 ft. (bottom of hot leg)
	8	0%	390 ft. (top of core is 388 ft. 10 in.)

Available RVLIS level indication as close as possible to NEI 99-01 Guidance was chosen for the threshold levels.

In the second condition of this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by sump level changes. 1/2 BwOS RF-1 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting when the Containment Floor Drain Sump annunciator setpoint is exceeded or 1/2 BwOS RF-2 surveillance has failed. The Containment Floor Drain Sump level indicated range is 0 to 106 in. The RCDT level indicated range is 0% to 100%.

The Containment Floor Drain Sump level indicated range is 0 to 106 in. The Containment Recirculation sump level indicated range is 14 to 150 in. Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment Sump level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area. The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency duration.

Basis Reference(s):

1. NEI 99-01 Rev 4, CA1 & CA2
2. UFSAR 6.2 & E.17
3. 1/2 BwOSR 0.1-4 Unit One(Two) Modes 4 Shiftly and Daily Operating Surveillance
4. 1/2 BwOS RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8 (cont)

Basis References (cont):

5. 1/2 BwOS XCB-R1 U0 and U1 MCR Meter Color Banding
6. BwOP RH-9 Pump Down of the Reactor Cavity to the RWSTs
7. BwOP RC-4 Reactor Coolant System Drain

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU8

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

5

EAL Threshold Values:

1. Pressurizer level established limit **> 5% Cold Cal** and RCS level **cannot** be restored and maintained **> 5% Cold Cal.**

   OR

2. Pressurizer level established limit **< 5% Cold Cal** and RCS level **cannot** be restored and maintained **>** procedurally established limit.

Basis:

The inability to restore and maintain level after reaching these established limits infers a degradation of the level of safety at the plant.

RCS level in the Cold Shutdown mode is controlled within limits that are established by procedures in effect for the present conditions. The use of Pressurizer level is normally appropriate for the Cold Shutdown mode; however, there are Cold Shutdown mode evolutions that require RCS level to be lowered below the range of the Pressurizer level instrumentation. These evolutions are directed by procedures that require precise control and monitoring of RCS levels that include establishment of low level limits. Examples of such evolutions include draining down to vessel flange level to prepare for reactor head flange bolt de-tensioning, and draining to mid-loop for equipment maintenance. During these evolutions it is appropriate to use the low level limit established by the procedure in effect to determine if RCS leakage is occurring and emergency declaration is required.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU1
2. UFSAR 6.2 & E.17
3. 1/2 BwOSR 0.1-4 Unit One(Two) Modes 4 Shiftly and Daily Operating Surveillance
4. 1/2 BwOS RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
5. 1/2 BwOS XCB-R1 U0 and U1 MCR Meter Color Banding
6. BwOP RH-9 Pump Down of the Reactor Cavity to the RWSTs
7. BwOP RC-4 Reactor Coolant System Drain

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS9

Initiating Condition:

Loss of Reactor Vessel inventory affecting core decay heat removal capability with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

6

EAL Threshold Values:

1. **Without** CONTAINMENT CLOSURE established:

   a. Reactor Vessel Refueling Level Indicators LT-046 and LT-049 **< 393 ft. el.**

   OR

   b. Reactor Vessel level unknown with indication of core uncovery as evidenced by one or more of the following:

   - 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high.
   - Erratic Source Range Monitor indication.

OR

2. **With** CONTAINMENT CLOSURE established:

   a. Reactor Vessel Refueling Level Indicators LT-046 and LT-049 **= 392 ft. el.** or off scale low.

   OR

   b. Reactor Vessel level unknown with indication of core uncovery as evidenced by one or more of the following:

   - 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high.
   - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be as required by procedures.

Containment closure status is checked and verified using the Containment Closure Reactor Operator Checklist of procedure 1/2 BwOS XPC-W1.

Basis (cont):

Threshold #1 and #2 Basis:

Under the refueling conditions specified in this threshold, prolonged loss of the ability to monitor Reactor Vessel water level in conjunction with indirect indications of inventory loss infer a continued drop in water level and loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the vessel.

In the Cold Shutdown mode, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. The heatup and the threat to damaging the fuel cladding thus may be higher for events that occur in the Cold Shutdown mode than for events in the Refueling mode. The elevated RCS heatup rate accelerates boil-off and loss of RCS inventory. When in the Cold Shutdown mode, a Site Area Emergency declaration is therefore associated simply with the decreasing inventory trend rather than indications of actual core uncovery. Note that the heatup threat can be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling. In the Cold Shutdown mode, when Reactor Vessel water level indication is unavailable, the inventory loss must be detected by containment sump level changes or erratic Source Range indication.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to core shine, scattering and radiation bounce off of the solid surfaces in the area will result in readings on the Fuel Handling Incident Radiation Monitors 1/2 RE-AR011 or 1/2 RE-AR012 of > 3000 mR/hr. This threshold radiation value is based on calculations documented in EP-EAL-0501.

- 1/2 RE-AR011 Containment Fuel Handling Incident Monitor - RA0039 for Unit 1/ RA0064 for Unit 2
- 1/2 RE-AR012 Containment Fuel Handling Incident Monitor - RA0040 for Unit 1/ RA0065 for Unit 2

1/2 BwOS RF-1 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting when the Containment Floor Drain Sump annunciator setpoint is exceeded or 1/2 BwOS RF-1 surveillance has failed.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS9 (cont)

Basis (cont):

The Containment Floor Drain Sump level indicated range is 0 to 106 in. The Containment Recirculation sump level indicated range is 14 to 150 in. Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment Sump level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area

Basis Reference(s):

1. NEI 99-01 Rev 4, CS2
2. 1/2 BwOS XPC-W1 Unit One(Two) Containment Penetration Status Weekly Surveillance
3. UFSAR E.17
4. BwOP RH-9 Pump Down of the Reactor Cavity to the RWSTs
5. BwOP RC-4 Reactor Coolant System Drain
6. UFSAR 6.2
7. 1/2 BwOSR 0.1-4 Unit One(Two) Modes 4 Shiftly and Daily Operating Surveillance
8. 1/2 BwOS RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
9. 1/2 BwOS XCB-R1 U0 and U1 MCR Meter Color Banding
10. EP-EAL-0501, Estimation Of Radiation Monitor Readings Indicating Core Uncovery During Refueling

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU9

Initiating Condition:

UNPLANNED loss of RCS inventory with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

6

EAL Threshold Values:

1. UNPLANNED RCS level drop below the Reactor Vessel flange (400 ft.) for **≥ 15 minutes.**

   OR

2. a. Loss of RCS /Reactor Vessel inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss

AND

b. Reactor Vessel level unknown.

Basis:

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

Threshold #1 Basis:

The Reactor Vessel flange is at 400 ft. Reactor Vessel water level is normally monitored using the following instruments:

- LT-046, LI-RY-046, Reactor Vessel Refueling Level Indicator
- LT-049, LI-RY-049, Reactor Vessel Refueling Level Indicator
- LT-048, LI-RY-048, Refueling Cavity Water Level
- LT-047 (LI-RY-047), 413 ft. el. to 426 ft. el.

LT-046 and LT-049 provide Reactor Vessel water level indication in the Control Room during RCS reduced inventory conditions. They are calibrated to indicate level from approximately 392 ft. el. to 402.5 ft. el. LT-048 is calibrated to read refueling cavity water level from 392 ft. el. to the refuel floor level 426 ft. el. LT-047 provides refueling cavity water level in the range of 413 ft. el. to 426 ft. el.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU9 (cont)

Basis (cont):

Two independent methods of RCS level indication must be functional and monitored prior to draining below 402 ft. el. If any discrepancies greater than 0.5 ft. between operable level indicators occur, draining is immediately secured until actual level has been verified. Two independent methods of RCS level indication must be continuously monitored when in a Reduced Inventory condition.

This threshold is applicable only in the Refueling mode and addresses loss of inventory to below the Reactor Vessel flange during refueling operations. Refueling operations that drop water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An Unusual Event is appropriate because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant.

The fifteen-minute interval provides a reasonable time frame to restore level using one or more of the redundant means of refill should be available. If RCS water level cannot be restored in this interval, a more serious condition may exist.

Threshold # 2 Basis:

In the second condition of this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by sump level changes. 1/2 BwOS RF-1 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting when the Containment Floor Drain Sump annunciator setpoint is exceeded or 1/2 BwOS RF-2 surveillance has failed.

The Containment Floor Drain Sump level indicated range is 0 to 106 in. The Containment Recirculation sump level indicated range is 14 to 150 in. Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment Sump level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area MU9 (cont)

Basis Reference(s):

1. NEI 99-01 Rev 4, CU2
2. UFSAR 5.2
3. 1/2 BwOSR 3.4.13.1 Unit One(Two) Reactor Coolant System Water Inventory Balance Surveillance
4. 1/2 BwOL 3.4.15 LCOAR - Reactor Coolant System Leakage - Leakage Detection Systems

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU9 (cont)

Basis (cont):

5. 1/2 BwOA PRI-1 Excessive Primary Leakage Unit 1/2
6. UFSAR 6.2
7. 1/2 BwOSR 0.1-4 Unit One(Two) Modes 6 Shiftly and Daily Operating Surveillance
8. 1/2 BwOS RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
9. BwOP RH-9 Pump Down of the Reactor Cavity to the RWSTs
10. BwOP RC-4 Reactor Coolant System Drain

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU10

Initiating Condition:

UNPLANNED loss of all onsite or offsite communications capabilities.

Operating Mode Applicability:

1, 2, 3, 4, 5, 6

EAL Threshold Values:

1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations.

   OR

2. Loss of all Table M6 **Offsite** communications capability.

Table M6 - Communications Capability

System	Onsite	Offsite
Radios	X	
Plant page	X	
Plant Telephone System	X	
Commercial Telephones		X
NARS		X
ENS		X
HPN		X
Cellular Phones		X
TSO/PJM (Electric Operations)		X
Satellite Phones		X

Basis:

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

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems with offsite authorities or personnel. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

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

Basis Reference(s):

1. NEI 99-01 Rev 4, SU6 & CU6
2. EP-MW-124-1001 Facilities Inventories And Equipment Tests

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU11

Initiating Condition:

Inability to reach required Shutdown within Technical Specification limits.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a prescribed shutdown mode when the Technical Specification configuration cannot be restored. The plant is within its safety envelope when being shut down 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 completion period elapses under Technical Specifications and is not related to how long a condition may have existed.

Basis Reference(s):

1. NEI 99-01 Rev 4, SU2
2. Braidwood Technical Specifications

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1

Initiating Condition:

Security event resulting in loss of physical control of the facility.

Operating Mode Applicability:

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

EAL Threshold Value:

A HOSTILE FORCE has taken control of:

1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1).

Table H1 - Safety Functions and Related Systems
• 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) • Fission Product Barriers

OR

2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).

Basis:

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.

Threshold #1 Basis

This threshold encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions. As a result, equipment control cannot be transferred to and operated from another location.

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

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.

Threshold #2 Basis

This threshold addresses loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely because there is freshly off-loaded fuel in the pool. The condition “freshly off-loaded reactor fuel in pool” equates to fuel off-loaded within the last 120 days in NF-AA-310 Special Nuclear Material And Core Component Movement.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1 (cont)

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HG1
2. Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. 0BwOA Security-1, Security Threat
4. 1/2 BwOA PRI-5 Control Room Inaccessibility
5. SY-AA-101-132, Threat Assessment
6. Station Security Plan – Appendix C
7. NF-AA-310, Special Nuclear Material And Core Component Movement

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1

Initiating Condition:

Site attack.

Operating Mode Applicability:

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

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert ICs (HA1 and HA2) in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA.

Although Nuclear Power Plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for Offsite Response Organizations (ORO) 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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1 (cont)

Basis (cont):

This EAL is intended to address the potential for a very rapid progression of events due to a dedicated attack. It is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs. HOSTILE ACTION identified above encompasses various acts including:

- Air attack (LARGE AIRCRAFT impacting the PROTECTED AREA)
- Land-based attack (HOSTILE FORCE penetrating PROTECTED AREA)
- Waterborne attack (HOSTILE FORCE on water penetrating PROTECTED AREA)
- BOMBs breeching the PROTECTED AREA

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs.

This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for ORO to be notified and to activate in order to be better prepared to respond should protective actions become necessary. If not previously notified by NRC that the LARGE AIRCRAFT impact 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.

LARGE AIRCRAFT is meant to be an 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.

This EAL addresses the immediacy of a threat to impact site VITAL AREAS within a relatively short time. The fact that the site is under serious attack with minimal time available for additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HS4
2. Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. 0BwOA Security-1, Security Threat

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition:

Notification of an airborne attack threat.

Operating Mode Applicability:

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

EAL Threshold Values:

A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LARGE AIRCRAFT is meant to be an 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.

The intent of this EAL is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations (ORO) and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Alert. This EAL is met when a plant receives information regarding a LARGE AIRCRAFT attack threat from NRC and the LARGE AIRCRAFT is less than 30 minutes away from the plant.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from such an attack. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for ORO to be notified and encouraged to activate (if they do not normally) to be better prepared should it be necessary to consider further actions.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA7
2. Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. 0BwOA Security-1, Security Threat

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

Initiating Condition:

Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

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

EAL Threshold Values:

1. A credible site-specific security threat notification as determined per SY-AA-101-132, “Threat Assessment”.

   OR

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

Basis:

Threshold #1 Basis

The intent of this threshold is to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat.

The determination of “credible” is made through use of information found in the Station Security Plan or SY-AA-101-132, “Threat Assessment” procedure.

Threshold #2 Basis

The intent of this threshold is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Unusual Event. This threshold is met when a plant receives information regarding an aircraft threat from NRC. Should the threat involve a LARGE AIRCRAFT (LARGE AIRCRAFT is meant to be an aircraft with the potential for causing significant damage to the plant), then escalation to Alert via HA1 would be appropriate if the LARGE AIRCRAFT is less than 30 minutes away from the plant. The status and size of the plane may be provided by NORAD through the NRC. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NRC Safeguards Advisory 10/6/01
6. 0BwOA Security-1, Security Threat
7. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

Initiating Condition:

Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA.

Operating Mode Applicability:

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

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This EAL is intended to address the potential for a very rapid progression of events due to an attack including:

- Air attack (LARGE AIRCRAFT impacting the OCA)
- Land-based attack (HOSTILE FORCE progressing across licensee property or directing projectiles at the site)
- Waterborne attack (HOSTILE FORCE on water attempting forced entry or directing projectiles at the site)
- BOMBs

This EAL is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2 (cont)

Basis (cont):

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne terrorist attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for Offsite Response Organizations to be notified and to activate in order to be better prepared to respond should protective actions become necessary.

If not previously notified by NRC that the LARGE AIRCRAFT impact 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. LARGE AIRCRAFT is meant to be an 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.

This IC/EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time. The fact that the site is an identified attack candidate with minimal time available for further preparation requires a heightened state of readiness and implementation of protective measures that can be effective (onsite evacuation, dispersal or sheltering) before arrival or impact.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA8
2. Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NRC Safeguards Advisory 10/6/01
6. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
7. 0BwOA Security-1, Security Threat

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS3

Initiating Condition:

Confirmed security event in a plant VITAL AREA.

Operating Mode Applicability:

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

EAL Threshold Value:

Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC (HA3).

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a Station’s security. Only those events that involve Actual or Likely Major failures of plant functions needed for protection of the public need to be considered. The following events would not normally meet this requirement; (e.g., Failure by a Member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or STRIKE ACTIONS).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Station Security Plan.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HS1
2. Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NRC Safeguards Advisory 10/6/01
6. 0BwOA Security-1, Security Threat

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

Initiating Condition:

Confirmed security event in a plant PROTECTED AREA.

Operating Mode Applicability:

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

EAL Threshold Value:

Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Unusual Event.

Multi-unit stations with shared safety functions should further consider how this IC may affect more than one unit and how this may be a factor in escalating the emergency class.

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a station's security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered. The following events would not normally meet this requirement; (e.g., failure by a member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or STRIKE ACTIONS).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA4
2. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events
3. Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
4. SY-AA-101-132, Threat Assessment
5. Station Security Plan – Appendix C
6. NRC Safeguards Advisory 10/6/01
7. 0BwOA Security-1, Security Threat

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU3

Initiating Condition:

Confirmed security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

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

EAL Threshold Value:

Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.

Basis:

Reference is made to Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

This threshold is based on Station Security Plan – Appendix C. 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.

Consideration should be given to the following types of events when evaluating an event against the criteria of the Station Security Plan: CIVIL DISTURBANCE, and STRIKE ACTION.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NRC Safeguards Advisory 10/6/01
6. 0BwOA Security-1, Security Threat

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS4

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 Threshold Values:

1. Control Room evacuation has been initiated.

   AND

2. Control of the plant **cannot** be established per 1/2 BwOA PRI-5, Control Room Inaccessibility procedure in **< 15 minutes**.

Basis:

The 15 minute time period starts when either:

a. Control of the plant is no longer maintained in the Main Control Room

   OR

b. The last Operator has left the Main Control Room.

The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. 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 determination of whether or not control is established outside of the Main Control Room is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the site-specific time for transfer that the licensee has control of the plant. Transfer of control to locations outside the Control Room is considered established when the Shift Manager has determined that the operators are capable of controlling reactivity, core cooling and heat sink functions.

Basis Reference(s):

1. NEI 99-01, Rev 4 HS2
2. 1/2 BwOA PRI-5, Control Room Inaccessibility

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

Initiating Condition:

Control Room evacuation has been initiated.

Operating Mode Applicability:

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

EAL Threshold Values:

Entry into 1/2 BwOA PRI-5, Control Room Inaccessibility procedure for Control Room evacuation.

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency operations centers are necessary. Procedure 1/2 BwOA PRI-5 Control Room Inaccessibility specifies conditions under which Control Room evacuation may be necessary.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA5
2. 1/2 BwOA PRI-5, Control Room Inaccessibility

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5

Initiating Condition:

Natural and destructive phenomena affecting the plant VITAL AREA.

Operating Mode Applicability:

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

EAL Threshold Values:

1. a. Seismic event **> Operating Basis Earthquake (OBE)** as indicated by seismic check 0PA02J.

   AND

   b. Confirmed by **EITHER:**

   - Earthquake felt in plant.
   - National Earthquake Center.

OR

2. Tornado or high winds **> 85 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

OR

3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

OR

4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • RWSTs • Condensate Storage Tanks • Lake Screen House

OR

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

EAL Threshold Values: (cont.)

5. Uncontrolled flooding that results in **EITHER**:
   a. Degraded safety system performance in the Auxiliary Building as indicated in the Control Room.

      OR

   b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.

Basis:

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

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

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

Threshold #1 Basis:

This threshold addresses events that may have resulted in a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this threshold to assess the actual magnitude of the damage.

This threshold is based on seismic ground acceleration in excess of 0.09 g for the UFSAR Operating Basis Earthquake (OBE). Seismic events of this magnitude ~ 4 times greater that the Unusual Event threshold of EAL HU5 and can cause damage to plant safety functions.

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

Threshold #2 Basis:

This threshold addresses events that may have resulted in a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The Alert classification is appropriate if visible damage is observed and relevant plant parameters indicate that the performance of safety systems in these areas has been degraded.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont):

No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

The 85 mph threshold is the UFSAR design basis wind speed. Station Category I structures are designed to withstand wind loads which may exist if wind speeds reach or exceed 85 mph. Wind loads in excess of this magnitude can cause damage to safety functions. The Condensate Storage Tanks are not UFSAR Category I structures, but their safety significance warrants their inclusion in this EAL.

Threshold #3 Basis:

This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes, or impact of projectiles into a Table H2 area. This threshold addresses vehicle crashes that challenge the operability of systems necessary for safe shutdown of the plant. Table H2 areas include Category 1 structures and those Category 2 structures that contain Category 1 Systems and components.

The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Table H2 areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

Threshold #4 Basis:

This threshold covers threats to safety related equipment imposed by missiles generated by failure of the main turbine. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Basis (cont):

Threshold #5 Basis:

This threshold addresses the effect of internal flooding that has resulted in degraded performance of safety systems or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant.

'Uncontrolled' as used in this threshold describes a condition where water is entering an area from an unplanned evolution. This flooding may have been caused by internal events such as component failures, equipment misalignment, and fire suppression system actuation or outage activity mishaps. Water entering an area, which resulted in degraded performance of safety systems within the area due to wetting or submergence, would meet the intent of this threshold. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding.” Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if indications of degraded system performance is available or a shock hazard is known to exist.

The Auxiliary Building has been identified as a potential Internal Flooding Area because it is an area containing systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Basis Reference(s):

1. NEI 99-01, Rev 4 HA1
2. UFSAR Section 2.5.4.9.3
3. 0BwOA-ENV-4, Earthquake
4. Annunciator 0-38-E5 Accelograph Accel High
5. Drawing S-01A Composite Site Plan
6. UFSAR Section 3.2 7
7. UFSAR Section 3.3.1.1
8. UFSAR Section 3.4
9. 0BwOA PRI-8 Auxiliary Building Flooding
10. UFSAR Appendix C
11. 1/2 BwOA TG-7, Main Generator Excessive Hydrogen Leakage

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition:

Natural and destructive phenomena affecting the PROTECTED AREA.

Operating Mode Applicability:

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

EAL Threshold Values:

1. a. Seismic event as indicated by Annunciator 0-38-E5, Accelograph Accel High (0PM01J).

   AND

   b. Confirmed by **EITHER:**

   - Earthquake felt in plant.
   - National Earthquake Center.

   OR

2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 85 mph**, within PROTECTED AREA boundary.

   OR

3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • RWSTs • Condensate Storage Tanks • Lake Screen House

   OR

4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

   OR

5. Uncontrolled flooding in Auxiliary Building that has the potential to affect safety related equipment needed for the current operating mode.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis:

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

Threshold #1 Basis:

This threshold is based on the strong-motion seismograph actuation level which is the sensed earthquake threshold of 0.02 g. Seismic events of this magnitude are ~1/4 of the Alert event threshold (OBE) of EAL HA5 in which it is assumed the earthquake can cause damage to plant safety functions.

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

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. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01 g.”

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

Threshold #2 Basis:

This threshold is based on the assumption that a tornado striking (touching down) or sustained high winds (> 85 mph) within the PROTECTED AREA boundary may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The Protected Area boundary is within the security isolation zone and is defined in the Braidwood Station Security Plan – Appendix C. Verification of a tornado is obtained by direct observation and reporting by station personnel. “Sustained” wind speeds exist for 15 minutes or longer. Wind speed is obtained from meteorological data in the Control Room.

Threshold #3 Basis:

In this context, a "vehicle crash" is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis (cont):

Threshold #4 Basis:

This threshold is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this threshold to classify minor operational leakage.

Threshold #5 Basis:

“Uncontrolled” as used in this threshold describes a condition where water is entering from an unplanned evolution. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if a potential to affect safety related equipment needed for the current operating mode exists.

This threshold addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, fire suppression system actuation or outage activity mishaps. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding.” The Auxiliary Building has been identified as an Internal Flooding Area of concern for the Unusual Event declaration because it is an area having the potential to affect safety related equipment needed for the current operating mode including :

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Basis Reference(s):

1. NEI 99-01, Rev 4 HU1
2. UFSAR Section 2.5.4.9.3
3. 0BwOA-ENV-4 Earthquake
4. Annunciator 0-38-E5 Accelograph Accel High
5. Drawing S-01A Composite Site Plan
6. UFSAR Section 3.2
7. UFSAR Section 3.3.1.1
8. 0BwOA PRI-8 Auxiliary Building Flooding
9. 1/2 BwOA TG-1 Turbine High Vibration, Eccentricity or Differential Expansion
10. 1/2 BwOA TG-7 Main Generator Excessive Hydrogen Leakage
11. BwAR 1/2 PLO1J-1-A2 Hydrogen Pressure high or Low

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6

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 Threshold Values:

1. FIRE or EXPLOSION in any Table H2 area.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • RWSTs • Condensate Storage Tanks • Lake Screen House

AND

2. a. Affected safety system parameter indications show degraded performance.

   OR

   b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.

Basis:

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

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

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6 (cont)

Basis (cont):

The areas listed in Table H2 house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Personnel access to these areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses FIRES and EXPLOSIONS that challenge the operability of systems necessary for safe shutdown of the plant.

The only FIRES and EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment required for safe shutdown. Visual observation of damage infers the ability to approach or enter the affected areas. Lacking the ability to adequately inspect the area for damage, the Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this EAL. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

A steam line break or steam explosion that damages permanent structures or equipment in one of these areas would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA2
2. Drawing S-01A Composite Site Plan
3. UFSAR Section 3.2

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6

Initiating Condition:

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

Operating Mode Applicability:

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

EAL Threshold Values:

1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

OR

2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • RWSTs • Condensate Storage Tanks • Lake Screen House

OR

3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

Basis:

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

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

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6 (cont)

Basis (cont):

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

Thresholds #1 and #2 Basis:

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins with a credible notification that a fire is occurring or indication of a VALID fire detection system alarm. A verified alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm.

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 EAL HA7. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under EAL HA7.

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

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket, etc.). Such fires are excluded from consideration in this threshold since they have no safety consequence.

Threshold #3 Basis:

The only EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment in the PROTECTED AREA.

A steam line break or steam explosion that damages permanent structures or equipment in a PROTECTED AREA would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU2
2. Drawing S-01A Composite Site Plan
3. UFSAR Section 3.2
4. BwAP-1100, Fire Protection Procedures

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA7

Initiating Condition:

Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown.

Operating Mode Applicability:

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

EAL Threshold Values:

1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

   OR

2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • RWSTs • Condensate Storage Tanks • Lake Screen House

Basis:

<u>VITAL AREA:</u> Any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation

<u>IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH):</u> A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

<u>LOWER FLAMMABILITY LIMIT (LFL):</u> The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

Values for LFL for common gases at Braidwood Station are:

- Propane – 2.2% (BOC Gasses MSDS)
- Hydrogen – 4.0% (Air Liquide Safety Data Sheet)
- Acetylene – 2.2% (BOC Gasses MSDS)

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA7 (cont)

Basis (cont)

This EAL is based on toxic, asphyxiant, or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Toxic or flammable gases detected outside of these areas need not be considered for this EAL unless there is a spread of the gasses into one of these areas.

Threshold #1:

Declaration should not be delayed for confirmation from atmospheric testing if it is reasonable to conclude that the IDLH concentrations have been met (e.g., documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards).

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 this EAL. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under this EAL. The first condition is met if measurement of toxic gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a Table H2 area. Non-Toxic Gases which displace oxygen (site examples; Halon or Nitrogen) to a life threatening level due to asphyxiation (oxygen deprivation) should also be considered for this EAL.

An Asphyxiant is a material 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.

Threshold #2:

The second condition is met when the flammable gas concentration in a Table H2 area exceeds the LOWER FLAMMABILITY LIMIT. Flammable gases such as hydrogen and acetylene are routinely used to maintain plant systems (hydrogen – main generator cooling, reactor coolant chemistry control) or repair equipment/components (acetylene - welding). This condition addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a Table H2 area has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage or personnel injury. Once it has been determined that an uncontrolled release of flammable gas is occurring, sampling must be done to determine if the gas concentration exceeds the LOWER FLAMMABILITY LIMIT.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA3
2. Drawing S-01A Composite Site Plan
3. UFSAR Section 3.2 Classification Of Structures, Components, And Systems

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU7

Initiating Condition:

Release of toxic or flammable gases deemed detrimental to normal operation of the plant.

Operating Mode Applicability:

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

EAL Threshold Values:

1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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:

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.

This EAL is based on the existence of uncontrolled releases of toxic, asphyxiant, or flammable gas affecting plant operations or the health of plant personnel. The release may have originated within the PROTECTED AREA boundary, or it may have originated offsite and subsequently drifted inside the PROTECTED AREA boundary. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on NORMAL PLANT OPERATIONS.

It is intended that releases of toxic, asphyxiant, or flammable gases are of sufficient quantity and the release point of such gases is such that safe plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for safe plant operation. The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect safe plant operations or plant personnel safety.

An Asphyxiant is a material 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.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU3
2. Drawing S-01A Composite Site Plan

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG8

Initiating Condition:

Other conditions existing 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 Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels ($>$ 1 Rem TEDE or $>$ 5 Rem CDE Thyroid) outside the site boundary.

Basis Reference(s):

1. NEI 99-01, Rev 4 HG2
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS8

Initiating Condition:

Other conditions existing 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 Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

Basis Reference(s):

1. NEI 99-01, Rev 4 HS3
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA8

Initiating Condition:

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

Operating Mode Applicability:

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

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA6
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU8

Initiating Condition:

Other conditions existing 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 Threshold Values:

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:

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency operating procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU5
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)

Section 4: Emergency Measures

Exelon Nuclear emergency response actions are the same for all nuclear stations and are thus covered by Section E of the Emergency Plan.

4.1 Notification of the Emergency Organization

Standard NARS notifications for the Braidwood Station are made to the State of Illinois Emergency Management Agency (IEMA). At the Braidwood Station, if a General Emergency is the initiating event, the Emergency Director is also responsible for notifying the following local agencies:

- Will County EOC
- Will County Sheriff's Office
- Grundy County EMA
- Grundy County Sheriff's Office
- Kankakee County EOC
- Kankakee County Sheriff's Office

4.2 Assessment Actions

Throughout each emergency situation, continuing assessment will occur. Assessment actions at Braidwood Station may include an evaluation of plant conditions; in-plant, onsite, and initial offsite radiological measurements; and initial estimates of offsite doses. Core damage information is used to refine dose assessments and confirm or extend initial protective action recommendations. Braidwood Station utilizes WCAP-14696-A, Revision 1, (1999) as the basis for the methodology for post-accident core damage assessment. This methodology utilizes real-time plant indications. In addition, Braidwood Station may use samples of plant fluids and atmospheres as inputs to the CDAM (Core Damage Assessment Methodology) program for core damage estimation.

4.3 Protective Actions for the Offsite Public

To aid Control Room personnel during a rapidly developing emergency situation, Figure 4-1, "Protective Action Recommendation (PAR) Determination Flowchart for Braidwood Station" has been developed based on Section J of the Exelon Nuclear Radiological Emergency Plan.

4.3.1 Alert and Notification System (ANS) Sirens

The ANS consists of a permanently installed outdoor notification system within the ten mile radius around the station. The ten mile radius around the station is primarily an agricultural area with a population density below 2000 persons per square mile.

The ANS as installed consists of mechanical and electronic sirens that will cover this entire area with a minimum sound level of 60 db. Additionally, the ANS will cover the heavily populated areas within the ten mile radius around the station with a minimum sound level of 70 db to ensure complete coverage.

4.3.2 Evacuation Time Estimates

The evacuation time estimates were developed per the requirements of NUREG-0654, and to support the Illinois Plan For Radiological Accidents (IPRA) - Braidwood Volume VII. The purpose of the evacuation time estimates is to assess the postulated evacuation times for the Braidwood Station Emergency Planning Zone (EPZ).

The evacuation time estimate data was updated per a study performed by Earth Tech. Inc. documented in their report dated December, 2003 entitled “Evacuation Time Estimates for the Braidwood Station Plume Exposure Pathway Emergency Planning Zone.”

The evacuation times are based on a detailed consideration of the EPZ roadway network and population distribution. The information in Table 4-1 presents representative evacuation times for daytime and nighttime scenarios, for summer and winter seasons, and under various weather conditions for the evacuation of various areas around the Braidwood Station, once a decision has been made to evacuate. The evacuation times noted include notification, mobilization, and travel time. These times are for the general population which include permanent population and special facilities (schools, nursing homes, hospitals, and recreational areas). Table 4-2 provides information on the scenario population distribution (by Subarea) that was used for this study. Table 4-3 provides a representation of the Subarea Locations in relation to the EPZ.

4.4 Protective Actions for Onsite Personnel

Braidwood Station has a siren system to assemble personnel during emergency conditions. Upon hearing a continuous two (2) minute siren, all personnel not having emergency assignments have been instructed to assemble in predesignated assembly areas. Refer to Figure 4-2.

If a site evacuation of non-essential personnel is required by Section J of the Emergency Plan, personnel will be either relocated and monitored at the designated relocation centers or sent home if there is no release or radiological or safety concerns. The designated relocation centers for Braidwood Station are:

- Mazon Relocation Center, Mazon, Illinois
- Dresden Station, Morris, Illinois
- LaSalle County Nuclear Power Station, Marseilles, Illinois

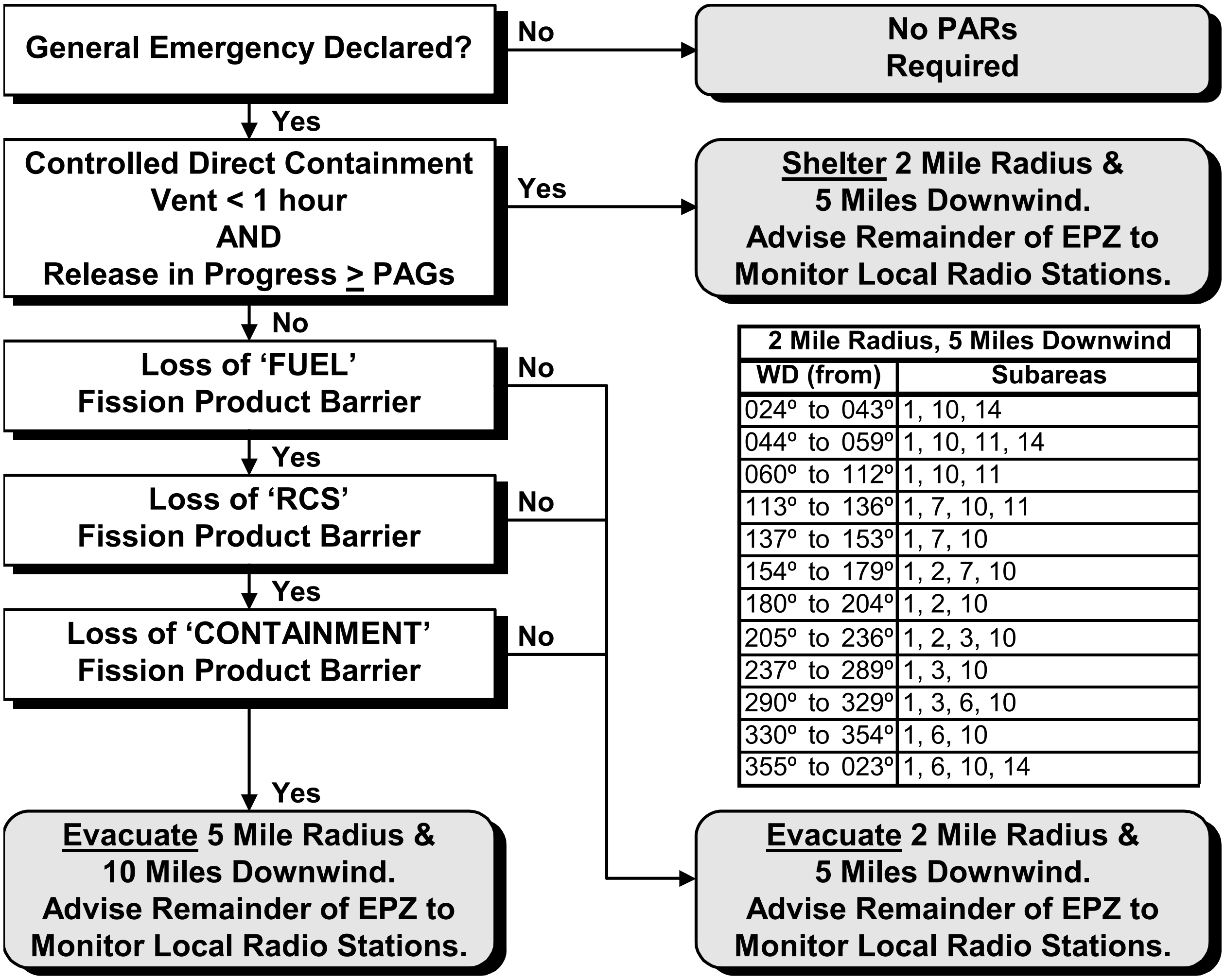
For evacuation routes, refer to EP-AA-113-F-17.

Traffic control for onsite areas will be accomplished by Station personnel, if necessary.

Equipment and personnel would be available at the Mazon Relocation Center, Dresden Station, and LaSalle Station for monitoring and decontamination of evacuated personnel. If major decontamination, follow-up or bioassay samples are necessary, those persons would be sent to Dresden or LaSalle Stations.

Other emergency measures are common to all nuclear stations and are thus discussed in the Emergency Plan.

Figure 4-1: Braidwood Station PAR Determination Flowchart

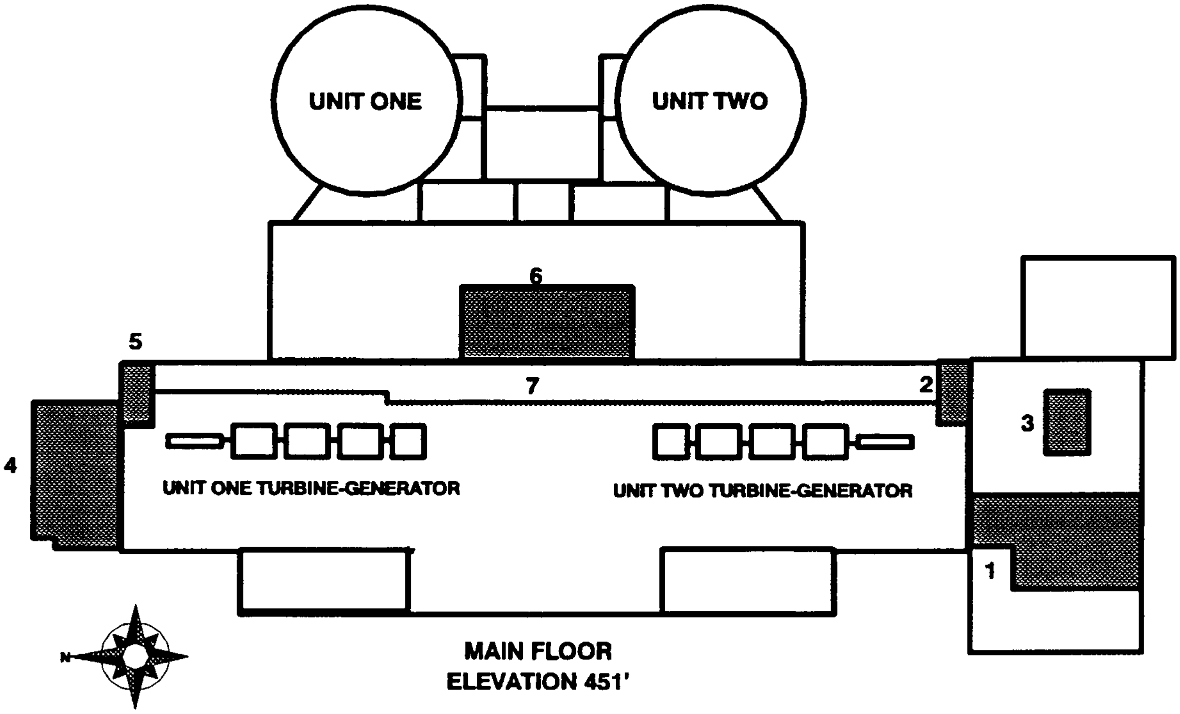


2 Mile Radius, 5 Miles Downwind	
WD (from)	Subareas
024° to 043°	1, 10, 14
044° to 059°	1, 10, 11, 14
060° to 112°	1, 10, 11
113° to 136°	1, 7, 10, 11
137° to 153°	1, 7, 10
154° to 179°	1, 2, 7, 10
180° to 204°	1, 2, 10
205° to 236°	1, 2, 3, 10
237° to 289°	1, 3, 10
290° to 329°	1, 3, 6, 10
330° to 354°	1, 6, 10
355° to 023°	1, 6, 10, 14

5 Mile Radius, 10 Miles Downwind	
WD (from)	Subareas
014° to 027°	1, 2, 3, 6, 7, 10, 11, 14, 15
028° to 056°	1, 2, 3, 6, 7, 10, 11, 14, 15, 16
057° to 073°	1, 2, 3, 6, 7, 10, 11, 12, 14, 15, 16
074° to 083°	1, 2, 3, 6, 7, 10, 11, 12, 14, 16
084° to 093°	1, 2, 3, 6, 7, 10, 11, 12, 14
094° to 104°	1, 2, 3, 6, 7, 8, 10, 11, 12, 14
105° to 121°	1, 2, 3, 4, 6, 7, 8, 10, 11, 12, 14
122° to 149°	1, 2, 3, 4, 6, 7, 8, 10, 11, 14
150° to 163°	1, 2, 3, 4, 6, 7, 10, 11, 14
164° to 181°	1, 2, 3, 4, 5, 6, 7, 10, 11, 14
182° to 210°	1, 2, 3, 5, 6, 7, 10, 11, 14
211° to 242°	1, 2, 3, 5, 6, 7, 10, 11, 13, 14
243° to 274°	1, 2, 3, 6, 7, 10, 11, 13, 14
275° to 307°	1, 2, 3, 6, 7, 9, 10, 11, 13, 14
308° to 013°	1, 2, 3, 6, 7, 9, 10, 11, 14

2 Mile Radius, 5 Miles Downwind	
WD (from)	Subareas
024° to 043°	1, 10, 14
044° to 059°	1, 10, 11, 14
060° to 112°	1, 10, 11
113° to 136°	1, 7, 10, 11
137° to 153°	1, 7, 10
154° to 179°	1, 2, 7, 10
180° to 204°	1, 2, 10
205° to 236°	1, 2, 3, 10
237° to 289°	1, 3, 10
290° to 329°	1, 3, 6, 10
330° to 354°	1, 6, 10
355° to 023°	1, 6, 10, 14

Figure 4-2: Braidwood Onsite Assembly Areas and Emergency Response Facilities



ASSEMBLY AREAS ▨

1. Maintenance Shop El. 401
2. Unit Two Turbine Bldg. Trackway El. 401
3. OSC El. 451 (Access)
4. TSC El. 451 (Access)
5. Unit One Turbine Bldg. Trackway El. 401
6. Control Room El. 451
7. Pathway to TSC

Table 4-1: Braidwood - Summary of General Public Evacuation Time Estimates

	General Population Evacuation Times (minutes) (1)							
	Summer Daytime		**Summer Nighttime**		**Winter Daytime**		**Winter Nighttime**	
PAR Evacuation Zone (2,3,4)	**fair**	**adverse**	**fair**	**adverse**	**fair**	**adverse**	**fair**	**adverse**
2 Mile Radius & 5 Miles Downwind								
WD 024 to 043 [1, 10, 14]	185	215	125	130	180	215	125	130
WD 044 to 059 [1, 10, 11, 14]	185	220	125	130	180	215	125	130
WD 060 to 112 [1, 10, 11]	185	220	125	130	180	215	125	130
WD 113 to 136 [1, 7, 10, 11]	190	220	125	130	180	215	125	130
WD 137 to 153 [1, 7, 10]	190	215	125	130	180	215	125	130
WD 154 to 179 [1, 2, 7, 10]	190	215	125	130	180	215	125	130
WD 180 to 204 [1, 2, 10]	190	215	125	130	180	215	125	130
WD 205 to 236 [1, 2, 3, 10]	190	220	125	130	180	225	125	130
WD 237 to 269 [1, 3, 10]	190	220	125	130	180	225	125	130
WD 270 to 329 [1, 3, 6, 10]	190	220	125	130	180	225	125	130
WD 330 to 354 [1, 6, 10]	190	215	125	130	180	220	125	130
WD 355 to 023 [1, 6, 10, 14]	190	215	125	130	180	220	125	130
5 Mile Radius & 10 Miles Downwind								
WD 014 to 027 [5R, 15]	185	245	130	135	180	225	130	140
WD 028 to 056 [5R, 15, 16]	190	245	130	135	185	225	130	145
WD 057 to 073 [5R, 12, 15, 16]	190	245	130	135	185	225	130	145
WD 074 to 083 [5R, 12, 16]	190	235	130	135	185	225	130	140
WD 084 to 093 [5R, 12]	185	220	130	135	180	225	130	135
WD 094 to 104 [5R, 8, 12]	185	220	130	135	180	225	130	135
WD 105 to 121 [5R, 4, 8, 12]	185	225	130	140	180	225	130	145
WD 122 to 149 [5R, 4, 8]	185	225	130	140	180	225	130	145
WD 150 to 163 [5R, 4]	185	225	130	140	180	225	130	145
WD 164 to 181 [5R, 4, 5]	190	225	130	145	185	225	130	150
WD 182 to 210 [5R, 5]	190	225	130	145	185	225	130	150
WD 211 to 242 [5R, 5, 13]	190	285	130	145	185	225	130	150
WD 243 to 274 [5R, 13]	190	285	130	145	185	225	130	150
WD 275 to 307 [5R, 9, 13]	210	285	130	155	190	235	130	165
WD 308 to 013 [5R, 9]	210	240	130	155	190	235	130	165
Full EPZ	210	285	130	155	190	235	130	165

(1) Times are rounded to the nearest 5 minutes

(2) Subareas in brackets. See Table 4.3 for Subarea locations. PAR evacuation zones per EP-AA-111

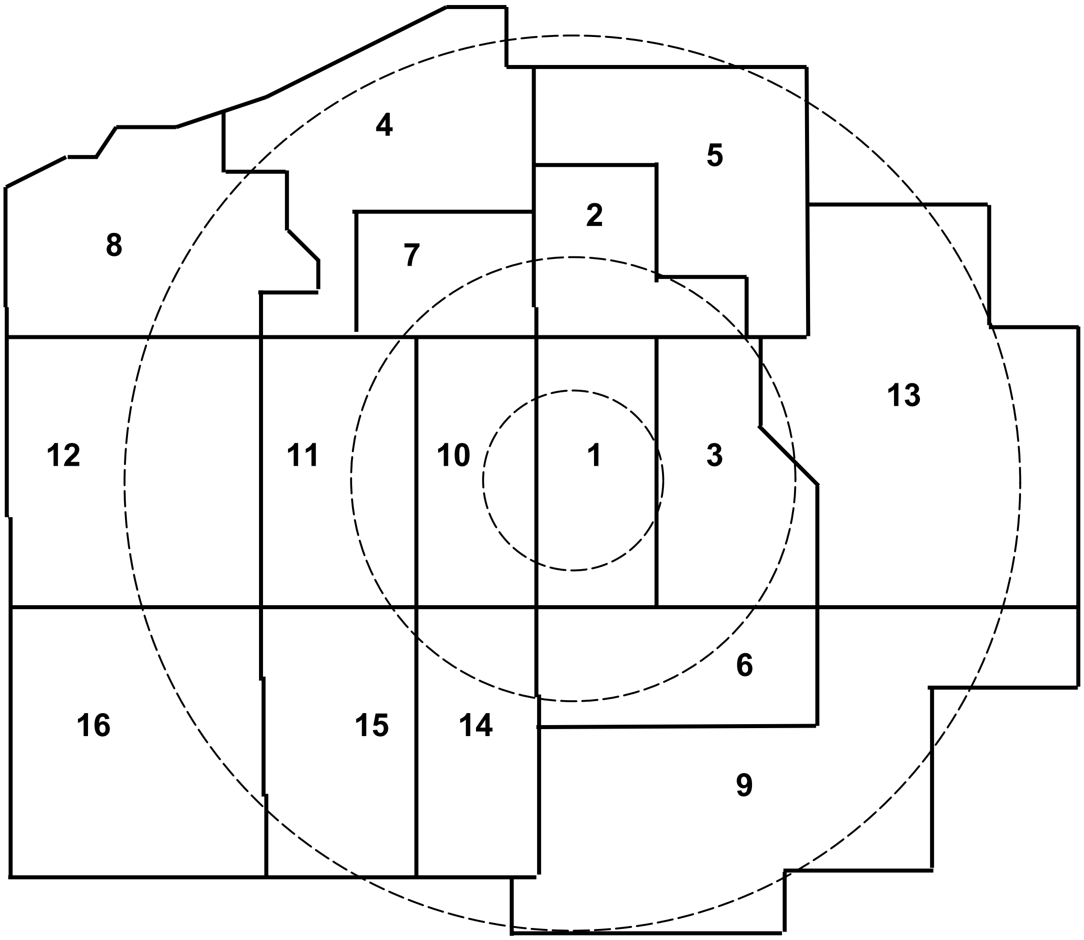
(3) "5R" designates all Subareas within 5-mile radius (Subareas 1, 2, 3, 6, 7, 10, 11, 14)

(4) WD is the direction (in degrees) from which the wind is blowing (00 or 360 represents a wind from north to south)

Table 4-2: Scenario Population Distribution By Subarea

	Summer				Winter			
	Daytime		Nighttime		Daytime		Nighttime	
Subarea	Population	Vehicles	Population	Vehicles	Population	Vehicles	Population	Vehicles
1	14,465	6,240	10,354	3,897	9,520	3,753	6,441	2,525
2	3,668	1,347	3,309	1,176	969	362	904	331
3	2,304	942	1,084	442	788	297	775	289
4	4,751	2,256	3,557	1,302	2,835	1,705	2,021	903
5	7,306	2,966	6,086	2,284	8,107	2,783	5,851	2,205
6	3,787	1,461	1,287	461	895	337	795	297
7	5,101	1,920	4,791	1,760	4,746	1,642	4,019	1,500
8	2,539	945	2,539	945	2,491	929	2,491	929
9	1,449	555	1,424	530	1,640	588	1,400	522
10	7,583	2,858	4,925	1,836	6,821	2,079	4,895	1,826
11	242	90	242	90	242	90	242	90
12	1,470	595	1,430	555	1,831	647	1,412	549
13	15,424	5,864	5,147	1,764	7,517	2,657	4,348	1,535
14	1,671	617	1,258	444	1,093	388	941	352
15	1,590	619	1,571	600	2,082	685	1,565	598
16	404	150	404	150	392	146	392	146
EPZ total	73,753	29,426	49,407	18,237	51,969	19,088	38,492	14,597

Table 4-3: Braidwood Subarea Locations



Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities

Refer to Figure 4-2 for the location of the Braidwood Station Control Room, Technical Support Center (TSC), and Operations Support Center (OSC) within the Station's Protected Area boundary.

5.1.1 Station Control Room

The Braidwood Station Control Room is the initial onsite center of emergency control and is located on the 451' elevation of the Auxiliary Building.

5.1.2 Technical Support Center (TSC)

Braidwood Station has designated a Technical Support Center which exists at the north end of the Turbine Building. The TSC fully meets the requirements of Section H.1.b of the Emergency Plan.

5.1.3 Operational Support Center (OSC)

Braidwood Station has designated a primary Operational Support Center. The Primary OSC is on 451' elevation of the Service Building. The OSC conforms to the requirements of Section H of the Emergency Plan and is the location to which operations support personnel will report during an emergency and from which they will be dispatched for assignments in support of emergency operations.

The backup OSC is the Shift Manager's office on 451' elevation of the Auxiliary Building.

5.2 Assessment Resources

5.2.1 Onsite Meteorological Monitoring Instrumentation

A 320-foot meteorological tower has been erected on the site approximately 1880 feet northeast of the Braidwood U-1 reactor building, the major plant structure closest to the tower. Wind speed and wind direction are measured at 34 feet and 203 feet above grade level. Temperature is measured at 30 feet and temperature difference is determined between the 30-foot and 199-foot levels. A precipitation gauge is utilized to measure rain and snowfall at ground level near the base of the tower.

The onsite meteorological monitoring program is covered in the contract specification and vendor procedures of the meteorological monitoring contractor. These data are used to generate wind roses and to provide estimates of airborne concentrations of gaseous effluents.

5.2.1.1 Instrumentation

The meteorological tower is instrumented with equipment that conforms with the recommendations of Regulatory Guide 1.23 and ANSI/ANS 2.5 (1984). The equipment is placed on booms oriented into the generally prevailing wind at the site.

Equipment signals are brought to an instrument shelter with controlled environmental conditions. The shelter at the base of the tower houses the recording equipment, signal conditions, etc. used to process and re-transmit the data to the end point users.

5.2.1.2 Meteorological Measurement Program During a Disaster

Cooperation between the corporate office and the meteorological contractor assures that a timely restoration of any outage can be made. Emergency field visits to the site are made as quickly as possible after detection of a failure.

Should a disaster of sufficient magnitude occur to destroy the tower structure, a contract is maintained to have a temporary tower erected within 72 hours, weather conditions permitting. Further, the meteorological contractor maintains two levels of sensors (wind speed, wind direction and temperature) in a state of readiness for use on the temporary tower.

Additionally, Exelon Nuclear's existing instrumented towers at other nuclear sites provide a high density measurement network with multiple backup opportunities.

Meteorological data are available to the station Control Room, Technical Support Center and Emergency Operations Facility for use in the Dose Assessment computer model for estimating the environmental impact of unplanned releases of radioactivity from the station.

5.2.2 Onsite Radiation Monitoring Equipment

The onsite radiation monitoring capability includes an installed process, effluent, and area radiation monitoring system; portable survey instrumentation; counting equipment for radiochemical analysis; and a personnel dosimetry program to record integrated exposure. Some onsite equipment is particularly valuable for accident situations and is described in the following subsections.

5.2.2.1 Radiation Monitoring System

Chapters 11 and 12 of the Braidwood UFSAR describe the radiation monitoring system (RMS) in detail. The installed RMS is designed to continuously monitor the containment atmosphere, plant effluents, and various in-plant locations.

The system includes Control Room readouts and recorders for selected parameters that are monitored and an audible Control Room alarm when predetermined setpoints are exceeded. The system can be subdivided into process/effluent instrumentation and an area monitoring system.

- The process/effluent instrumentation consists of pumps, filter samplers, detectors, and associated electronics to determine noble gas, iodine, and particulate concentrations in plant cubicles or liquid and gaseous effluents. Several monitored effluent pathways have control functions that will terminate the release at a predetermined setpoint. These setpoints are premised on compliance with federal regulations.

- The area monitoring system provides information of existing radiation levels in various areas of the plant to ensure safe occupancy. It is equipped with Control Room and local readout and audible alarms to warn personnel of an increased radiation level.

5.2.2.2 Radiological Noble Gas Effluent Monitoring

Two General Atomic Company wide-range gas monitors (WRGMs) are installed on the auxiliary building vent stacks (final release points), one monitor per stack. The monitor has a range for radioactive gas concentration of 1 x 10^{-7} uCi/cc to 1 x 10^{5} uCi/cc. The monitor includes the following: two isokinetic nozzles, one for normal conditions operating at 2 ft3/min. and one for high range conditions operating at 0.06 ft3/min; sampling rack; sample conditioner to filter out large concentrations of radioiodine and particulates; and the wide-range gas detectors assembly, consisting of three radioactive gas detectors, a low-range detector, a mid-range detector, and a high-range detector. Each monitor system has a microprocessor which utilizes digital processing techniques to analyze data and control monitor functions. Control Room readouts include a RM-23 remote display module for all monitored parameters.

Two General Atomic Company detectors are provided for each of the four main steamlines upstream of the safety and relief valves. The range of the monitor is 10^{-1} mR/hr to 10^{4}mRem/hr. The monitors are mounted external to the main steamline piping and corrections made for the loss of low energy gammas.

5.2.2.3 Radioiodine and Particulate Effluent Monitoring

The General Atomic Company wide range gas monitor includes a sampling rack for collection of the auxiliary building vent stack particulate and radioiodine samples. Filter holders and valves are provided to allow grab sample collection for isotopic analyses in the station's counting rooms. The sampling rack is shielded to minimize personnel exposure. The sampling media will be analyzed by a gamma ray spectrometer which utilizes a gamma spectrometer system. In addition, silver zeolite cartridges are available to further reduce the interference of noble gases.

5.2.2.4 High-Range Containment Radiation Monitors

Two high range containment radiation monitors are installed for each operating reactor. The monitors will detect and measure the radiation level within the reactor containment during and following an accident. The range of the monitors is 1 rad/hr to 10^7 Rads/hr.

5.2.2.5 In-plant Iodine Instrumentation

Effective monitoring of increasing iodine levels in buildings under accident conditions will include the use of portable instruments using silver zeolite as a sample media. Braidwood Station has a Post Accident Radionuclide Analysis Portable System (PARAPS) for analyzing samples that cannot be counted and analyzed in the normal Station counting room because of background problems. Auxiliary counting room locations have been identified within the Turbine Building. It is expected that a sample can be obtained, purged, and analyzed for iodine content within a two-hour time frame.

5.2.3 Onsite Process Monitors

An adequate monitoring capability exists to properly assess the plant status for all modes of operation and is described in the Braidwood UFSAR. The operability of the post-accident instrumentation ensures information is available on selected plant parameters to monitor and assess important variables following an accident. Instrumentation is available to monitor the parameters and ranges given in Technical Specifications.

Braidwood Station Emergency Operating Procedures assist personnel in recognizing inadequate core cooling using applicable instrumentation.

5.2.4 Onsite Fire Detection Instrumentation

The fire detection system is designed in accordance with applicable National Fire Protection Association (NFPA) Standards. The System is equipped with electrically supervised ionization smoke and heat detectors to quickly detect any fires and the instrumentation to provide local indication and Control Room annunciation. In addition to the smoke and heat detection systems, each fire protection carbon dioxide, halon, or water system is instrumented to inform the Control Room of its actuation or of system trouble.

In the event that a portion of the fire detection instrumentation is inoperable, fire watches in affected areas may be required.

5.2.5 Facilities and Equipment for Offsite Monitoring

Consult Chapter 11 of the station specific Offsite Dose Calculation Manual (ODCM) for the most current location for fixed continuous air samplers and TLD locations.

Braidwood Station maintains a supply of emergency equipment and supplies for offsite monitoring and sampling. These supplies meet the initial requirements of two (2) environmental field teams. During subsequent phases of an emergency, additional equipment is available from other Exelon Nuclear facilities, vendors and the Corporate Emergency Response Organization.

5.2.6 Site Hydrological Characteristics

The hydrological characteristics of the Braidwood Station vicinity are described in Section 2.4 of the Braidwood UFSAR. The river screen house is the only structure that could be affected by flooding on the Kankakee River. The controlling event for flooding at the site is the probable maximum flood for the cooling pond, resulting in a short-term maximum water surface elevation of 600.6 ft in the immediate plant area.

Although the plant grade elevation is 600 ft, the safety related facilities are situated at elevation 601 ft; 0.4 ft above the estimated maximum water surface elevation.

Low flows in the Kankakee River cannot affect safety related facilities of the plant. In the unlikely event that emergency make-up water requirements cannot be satisfied by surface water withdrawals from the Kankakee River, the Cooling Lake will operate under a closed cycle system. Emergency shut down water is available from the Cooling Lake. Because of the site hydrological characteristics given above, plant operation should not be affected by Kankakee River water level conditions and therefore, hydrological monitors have not been installed.

The Kankakee River was not used for any public water supply within 50 miles downstream of this site prior to 1990. In January, 1990, the City of Wilmington, Illinois began withdrawing water, four miles downstream on the west bed of the Kankakee River. Provisions have been established for weekly samples to be collected and computed for monthly analysis. The City of Joliet, Illinois has submitted a plan to also use the Kankakee River to supply public water. Upon completion of the facility, provisions will be made for weekly sample collection and analysis. In performing dose calculations from liquid releases, the liquid release model has been revised to reflect the change of parameters due to the new public water intake.

5.2 Protective Facilities and Equipment

The principal onsite assembly areas for Braidwood Station are the Machine Shop on the 401-foot elevation of the Service Building and the Turbine Building trackways. These areas are suitable because:

1) They are large open areas suitable for assembling a large number of people in a short time;

2) They can be easily exited if a site evacuation is deemed necessary following an assembly; and

3) They have a low probability of being affected by a serious accident involving the Reactor or primary systems.

The offsite relocation centers for Braidwood Station are discussed in Section 4 of this annex. Both locations are suitable, depending on the emergency condition, with personnel, supplies and communications readily available.

5.3 First Aid and Medical Facilities

Braidwood Station has an inplant first aid/decontamination room on the 426 foot elevation of the auxiliary building near the station laboratory complex. This room is provided with a sink, a shower, and a supply cabinet.

First aid kits, stretchers, sinks, eyewashes and emergency showers have been placed in strategic locations throughout the station.

Medical treatment given to injured persons at the station is of a "first aid" nature. When more professional care is needed, injured persons are transported to a local hospital or clinic. Provena St. Joseph Medical Center in Joliet, Illinois is the designated support hospital for handling contaminated injured personnel. Morris Hospital in Morris, IL is the backup medical facility for evaluation and treatment of persons suffering from traumatic injury, medical illness, or radiation exposure and uptake.

Appendix 1: NUREG-0654 Cross-Reference

Annex Section	**NUREG-0654**
1.0	Part I, Section A
1.1	Part I, Section C
1.2	Part I, Section D
Figure 1-1	Part I, Section D
2.0	Part II, Section A.4
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3.0	Part II, Section D
4.1	Part II, Section E.1 & J.7
4.2	Part II, Section I.2 & 3
4.3	Part II, Section J.10.m
4.3.1	Part II, Section E.6
4.3.2	Part II, Section J.8
4.4	Part II, Section J.1-5
Figure 4-1	Part II, Section J.10.m
Figure 4-2	Part II, Section J.5
4.4	Part II, Section J.2 & 3
Table 4-1	Part II, Section J.8
Table 4-2	Part II, Section J.10.b
5.1	Part II, Section H.1 & G.3
5.2.1	Part II, Section H.5.a & 8
5.2.2	Part II, Section H.5.b & I.2
5.2.3	Part II, Section H.5.c
5.2.4	Part II, Section H.5.d
5.2.5	Part II, Section H.6.b & 7
5.2.6	Part II, Section H.5.a & 6.a
5.3	Part II, Section J.1-5
5.4	Part II, Section L.1 & 2

Appendix 2: Station Letters of Agreement

1. Will County Sheriff – law enforcement.
2. Braidwood Fire Department – fire suppression support.
3. St. Joseph’s Hospital – medical support and treatment.
4. Morris Hospital – backup medical support and treatment.

Attachment 2

EP-AA-1002

"Exelon Nuclear Standardized Radiological Emergency Plan Annex for Byron Station"

Revision 21

EP-AA-1002
Revision 21

EXELON NUCLEAR

RADIOLOGICAL EMERGENCY PLAN ANNEX FOR BYRON STATION

Submitted: Kevin Appel Date: 10/10/07
Midwest Region Emergency Preparedness Manager

Authorized: Jim Meister Date: 10/12/07
Vice President – Operations Support

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APPENDIXES

REVISION HISTORY

Revision 1; September 1984	Revision 4h; November 1995	Revision 12, July 8, 2002
Revision 2; May 1986	Revision 4i; December 1995	Revision 13, October 4, 2002
Revision 3; June 1987	Revision 4k; June 1996	Revision 14, November 15, 2002
Revision 3b; May 1988	Revision 4l; February 1997	Revision 15, May 12, 2003
Revision 3c; May 1989	Revision 4m: January 5, 1998	Revision 16, December 2004
Revision 3d; September 1989	Revision 4n: August 14, 1998	Revision 17, January 2006
Revision 4; January 1991	Revision 4p: October 16, 1998	Revision 18, March 2006
Revision 4a; April 1992	Revision 5: May 13, 1999	Revision 19, September 2006
Revision 4b; November 1992	Revision 6: June 23, 1999	Revision 20, May 2007
Revision 4c; February 1993	Revision 7: January 8, 2001	
Revision 4d; December 1993	Revision 8: October 8, 2001	
Revision 4e; January 1993	Revision 9, October 31, 2001	
Revision 4f; November 1994	Revision 10, November 1, 2001	
Revision 4g; November 1994	Revision 11, January 3, 2002	

Section 1: Introduction

As required in the conditions set forth by the Nuclear Regulatory Commission (NRC) for the operating licenses for the Exelon Nuclear Stations, the management of Exelon recognizes its responsibility and authority to operate and maintain the nuclear power stations in such a manner as to provide for the safety of the general public.

The Exelon Emergency Preparedness Program consists of the Exelon Nuclear Standardized Emergency Plan (E-Plan), Station Annexes, emergency plan implementing procedures, and associated program administrative documents. The Exelon E-Plan outlines the basis for response actions that would be implemented in an emergency. Planning efforts common to all Exelon Nuclear stations are encompassed within the E-Plan.

This document serves as the Byron Station Emergency Plan Annex and contains information and guidance that is unique to the station. This includes Emergency Action Levels (EALs), and facility geography location for a full understanding and representation of the station's emergency response capabilities. The Station Annex is subject to the same review and audit requirements as the Exelon Nuclear Standardized Emergency Plan.

1.1 Facility Description

The Byron Station, Units 1 and 2, are located in Northern Illinois, approximately 3.7 miles south-southwest of the City of Byron in Ogle County. This site is situated near the center of the county in a predominantly agricultural area. At its closest approach, the Rock River is approximately 1.5 miles west of the western site boundary and 2.2 miles west-southwest of the actual plant location. Byron Station occupies approximately 1288 acres of land. The station site is somewhat rectangular in shape, with the plant structures occupying the southeast portion of the site.

Figure 1-1 shows the general location of Byron Station. More specific information on station siting may be found in the Updated Final Safety Analysis Report (UFSAR).

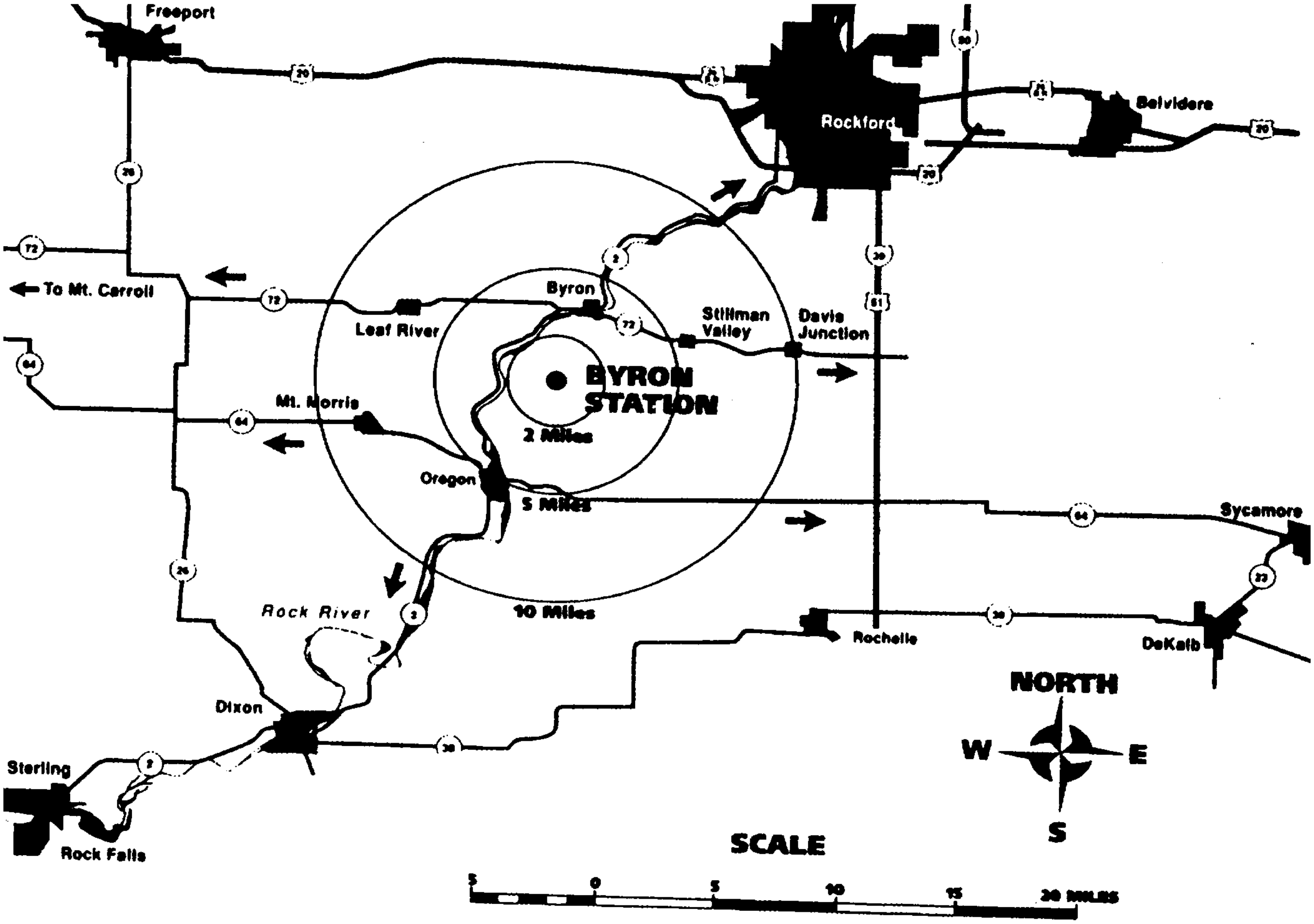
The plant consists of two identical pressurized water reactor (PWR) nuclear steam supply systems (NSSS) and turbine generators furnished by Westinghouse Electric Corporation. Each nuclear steam supply system is designed for a power output of 3586.6 MWt. Cooling for the plant is provided by two natural draft cooling towers for nonessential service cooling water and by mechanical draft cooling towers for essential service cooling water.

1.2 Emergency Planning Zone

The plume exposure Emergency Planning Zone (EPZ) for Byron Station is an area surrounding the station with a radius of about ten miles. (Exact boundaries are determined by the State of Illinois). Refer to Figure 1-1.

The ingestion pathway EPZ for Byron Station is an area surrounding the station with a radius of about 50 miles.

Figure 1-1: Byron Station Location and 10 Mile EPZ



Section 2: Organizational Control of Emergencies

Initial response to any emergency is by the normal plant organization present at the site. This organization includes positions that are onsite 24 hours per day and is described in Section B.1 of the Exelon Nuclear Standardized Emergency Plan.

Once an emergency is declared, the Emergency Response Organization (ERO) is activated as described in the Exelon Nuclear Standardized Emergency Plan.

2.1 Non-Exelon Nuclear Support Groups

Exelon Nuclear has contractual agreements with several companies whose services would be available in the event of a radiological emergency. These agencies and their available services are listed in Appendix 3 of the Exelon Nuclear Standardized Emergency Plan.

Emergency response coordination with governmental agencies and other support organizations is discussed in Section A of the Exelon Nuclear Standardized Emergency Plan.

Agreements exist on file at Byron Station with several support agencies. These agencies and their support roles are listed in Appendix 2, Station Letters of Agreement.

Section 3: Classification of Emergencies

3.1 General

Section D of the Exelon Nuclear Standardized Emergency Plan divides the types of emergencies into four Emergency Classification Levels (ECLs). The first four are the UNUSUAL EVENT, ALERT, SITE AREA EMERGENCY, and GENERAL EMERGENCY. These ECLs are entered by meeting the Emergency Action Level (EAL) Threshold Values provided in this section of the Annex. 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. Depending on the severity of an event, prior to returning to a standard day-to-day organization, a state or phase called RECOVERY may be entered to provide dedicated resources and organization in support of restoration and communication activities following the termination of the emergency.

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 Guideline 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 acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

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

RECOVERY: Recovery can be considered as a phase of the emergency and is entered by meeting emergency termination criteria provided in EP-AA-111 Emergency Classification and Protective Action Recommendations.

An emergency is classified by assessing plant conditions and comparing abnormal conditions to Initiating Conditions and Threshold Values for each Emergency Action Level.

Individuals responsible for the classification of events will refer to the Initiating Condition and Threshold Values on the matrix of the appropriate station Standardized Emergency Plan Annex (this document). This matrix will contain Initiating Conditions, EAL Threshold Values, Mode Applicability Designators, appropriate EAL numbering system, and additional guidance necessary to classify events. It may be provided as a user aid.

The matrix is set up in four Recognition Categories. The first is designated as "R" and relates to Abnormal Radiological Conditions / Abnormal Radiological Effluent Releases. The second is designated as "F" and relates to Fission Product Barrier Degradation. The third is designated as "M" and relates to System Malfunctions. The fourth is designated as "H" and relates to Hazards and Other Conditions.

The matrix is designed to provide an evaluation of the Initiating Conditions from the worst conditions (General Emergencies) on the left to the relatively less severe conditions on the right (Unusual Events). Evaluating conditions from left to right will reduce the possibility that an event will be under classified. All Recognition Categories should be reviewed for applicability prior to classification.

The Initiating Conditions are coded with a two letter and one number code. The first letter is the Recognition Category designator, the second letter is the Classification Level, “U” for (NOTIFICATION OF) UNUSUAL EVENT, “A” for ALERT, “S” for SITE AREA EMERGENCY and “G” for GENERAL EMERGENCY. The EAL number is a sequential number for that Recognition Category series. All Initiating Conditions that are describing the severity of a common condition (series) will have the same number.

The EAL number may then be used to reference a corresponding page(s), which provides the basis information pertaining to the Initiating Condition:

- Threshold Value
- Mode Applicability
- Basis

Emergency Action Levels are the measurable, observable detailed conditions that must be met in order to classify the event. Classification is not to be made without referencing, comparing and satisfying the Threshold Values specified in the Emergency Action Levels.

A list of definitions is provided as part of this document for terms having specific meaning to the Emergency Action Levels. Site specific definitions are provided for terms with the intent to be used for a particular Initiating Condition/Threshold Value and may not be applicable to other uses of that term at other sites, the Emergency Plan or procedures.

References are also included to documents that were used to develop the EAL Threshold Values.

References to the Emergency Director means the person in Command and Control as defined in the Emergency Plan. Classification of emergencies is a non-delegable responsibility of Command and Control for the onsite facilities with responsibility assigned to the Shift Emergency Director (Control Room Shift Manager) or the Station Emergency Director (TSC). Classification of emergencies remains the responsibility of the applicable onsite facility even after Command and Control is transferred to the Corporate Emergency Director (EOF).

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

Indications used for monitoring and evaluation of plant conditions include the normally used instrumentation, backup or redundant instrumentation, and the use of other parameters that provide information that supports determination if an EAL threshold value has been reached. When an EAL refers to a specific instrument or indication that is determined to be inaccurate or unavailable, then alternate indications shall be used to monitor the specified condition.

During an event that results in changing parameters trending towards an EAL classification, and instrumentation that was available to monitor this parameter becomes unavailable or the parameter goes off scale, the parameter should be assumed to have been exceeded consistent with the trend and the classification made if there are no other direct or indirect means available to determine if the threshold has not been exceeded.

EALs are for unplanned events. A planned evolution involves 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. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL Threshold Value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

When two or more Emergency Action Levels are determined, declaration will be made on the highest classification level for the Unit. When both units are affected, the highest classification for the Station will be used for notification purposes and both units' classification levels will be noted.

3.2 Mode Applicability

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

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 Fission Product Barrier Matrix EALs are applicable only to events that initiate in Hot Shutdown or higher.

If there is a change in 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.

3.3 Emergency Director Judgment

Emergency Director Judgment EALs are provided in the Hazards and Other Condition Affecting Plant Safety section and on the Fission Product Barrier Matrix. Both of the Emergency Director Judgment EALs have specific criteria for when they should be applied.

The Hazards Section Emergency Director Judgment EALs are intended to address unanticipated conditions which are not addressed explicitly by other EALs but warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under specific emergency classifications (UE, Alert, SAE or GE).

The FPB Matrix ED Judgment EALs are intended to include unanticipated conditions, which are not addressed explicitly by any of the other FPB threshold values, but warrant determination because conditions exist that fall under the broader definition for a significant Loss or Potential Loss of the barrier (equal to or greater than the defined FPB threshold values).

3.4 Fission Product Barrier Restoration

Fission Product Barriers (FPBs) are not treated the same as EAL threshold values. Conditions warranting declaration of the loss or potential loss of a Fission Product Barrier may occur resulting in a specific classification. The condition that caused the loss or potential loss declaration could be rectified as the result of Operator action, automatic actions, or designed plant response. Barriers will be considered re-established when there are direct verifiable indications (containment penetration or open valve has been isolated, coolant sample results, etc) that the barrier has been restored and is capable of mitigating future events.

The reestablishment of a fission product barrier does not alter or lower the existing classification. Entry into Termination/Recovery phase is still required for exiting the present classification. However the reestablishment of the barrier should be considered in determining future classifications should plant conditions or events change.

3.5 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 five or more persons violently protesting station operations or activities at the site.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

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

CONTAINMENT CLOSURE: Containment Closure is considered to be as required by procedures.

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

EXTORTION: 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, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

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.

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

NORMAL LEVELS: Normal levels can be considered as 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.

OPERATING MODES:

(1) Power Operations:	Reactor Power > 5%, Keff ≥0.99
(2) Startup:	Reactor Power ≤ 5%, Keff ≥ 0.99
(3) Hot Standby:	RCS ≥ 350° F, Keff < 0.99
(4) Hot Shutdown:	200° F < RCS < 350° F, Keff < 0.99
(5) Cold Shutdown:	RCS ≤ 200° F, Keff < 0.99
(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).

Hot Matrix – applies in modes (1), (2), (3), and (4)

Cold Matrix – applies in modes (5), (6), and (D)

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

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: A 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.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) automatic turbine 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 isolated from the Control Room.

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

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

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

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

Emergency Action Level Technical Basis Page Index

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	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** 1 2 3 4 5 6 D Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **3.07 E+07 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** 1 2 3 4 5 6 D Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **3.07 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** 1 2 3 4 5 6 D Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. **EAL Threshold Values:** 1 VALID reading on any of the following effluent monitors **> 200 times** the high alarm setpoint established by a current radioactive release package for **≥ 15 minutes**. • 0PR001, Liquid Radwaste Effluent Monitor • 0PR002, Gas Decay Tank Effluent Monitor • 0PR010, Station Blowdown Monitor • 1/2 PR001, Containment Purge Effluent Monitor • Discharge Permit specified monitor **OR** 2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 1.83 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** 1 2 3 4 5 6 D Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. **EAL Threshold Values:** 1. VALID reading on any of the following effluent monitors **> 2 times** the high alarm setpoint established by a current radioactive release package for **≥ 60 minutes**. • 0PR001, Liquid Radwaste Effluent Monitor • 0PR002, Gas Decay Tank Effluent Monitor • 0PR010, Station Blowdown Monitor • 1/2 PR001, Containment Purge Effluent Monitor • Discharge Permit specified monitor **OR** 2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 5.85 E+05 uCi/sec** for **≥ 60 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Abnormal Rad Levels		**Table R1** **Fuel Handling Incident Radiation Monitors** • Fuel Building Fuel Handling Incident Monitor 0RE-AR055 • Fuel Building Fuel Handling Incident Monitor 0RE-AR056 • Containment Fuel Handling Incident Monitor 1/2 RE-AR011 • Containment Fuel Handling Incident Monitor 1/2 RE-AR012	**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 6 D **<u>EAL Threshold Values:</u>** 1. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1. **OR** 2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.	**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 6 D **<u>EAL Threshold Values:</u>** 1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by: • Refueling Cavity water level **< 23 ft**. above the Reactor Flange (**423 ft**. indicated level). **OR** • Spent Fuel Pool water level **< 23 ft**. above the fuel (**422 ft. 9 in.** indicated level). **OR** • Report of visual observation of an uncontrolled drop in water level in the Fuel Transfer Canal, Refueling Cavity, or Spent Fuel Pool. **AND** b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1. **OR** 2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.
	Table R2 **Areas Requiring Continuous Occupancy** • Main Control Room - 1/2 RE-AR010 • Central Alarm Station - (by survey) • Radwaste Control Room (Aux Bldg 383 ft. el.) - 0RE-AR007	**Table R3** **Areas Requiring Infrequent Access** • Unit 1 and 2 Remote Shutdown Panels (0RE-AR007) • High Radiation Sample Room – HRSS (0RE-AR031) • Containment Air Sample Panel – CASP (by survey) • Fire Hazards Panel (by survey)	**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 6 D **<u>EAL Threshold Values:</u>** 1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions. **OR** 2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.	**RU3** Fuel clad degradation. 1 2 3 4 **<u>EAL Threshold Values:</u>** 1. VALID Gross Failed Fuel Monitor 1/2 RE-PR006 indicating I-135 concentration of **> 5 uCi/cc**. **OR** 2. a. Dose Equivalent I-131specific coolant activity **> 1.0 uCi/gm**. **OR** b. Gross specific coolant activity **> 100 / Ē uCi/gm.**

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

Fission Product Barrier Matrix **Hot Matrix**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
FG1 Loss of ANY Two Barriers AND Loss or Potential Loss of the third barrier. 1 2 3 4	**FS1** Loss or Potential Loss of ANY two barriers. 1 2 3 4	**FA1** ANY Loss or ANY Potential Loss of either Fuel Clad or RCS. 1 2 3 4	**FU1** ANY Loss or ANY Potential Loss of Containment. 1 2 3 4

Sub-Category	FC - Fuel Clad		RC – Reactor Coolant System		CT – Containment	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
1. CSF Status →	**Core Cooling CSF - RED Path** conditions exist.	1. **Core Cooling CSF - ORANGE Path** conditions exist. **OR** 2. **Heat Sink CSF - RED Path** conditions exist.	None	1. **RCS Integrity CSF - RED Path** conditions exist. **OR** 2. **Heat Sink CSF - RED Path** conditions exist.	None	**Containment CSF - RED Path** conditions exist.
2. RCS Activity →	Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.	None	None	None	None	None
3. Containment Pressure →	None	None	None	None	1. Rapid unexplained drop in Containment pressure following initial pressure rise. **OR** 2. Containment pressure or water level response not consistent with LOCA conditions.	1. Containment pressure ≥ **50 psig** and rising. **OR** 2. Hydrogen concentration in Cont. **≥ 5%** **OR** 3. a. Containment pressure ≥ **20 psig**. **AND** b. Less than one train of Containment Spray operating.
4. CETC Reading →	Average of the ten highest reading core exit thermocouples (CETCs) is **> 1200° F**.	Average of the ten highest reading core exit thermocouples (CETCs) is **> 700° F**.			None	1. a. Average of the ten highest reading core exit thermocouples (CETCs) is **≥ 1200° F** **AND** b. Functional Restoration procedures not effective in **< 15 minutes**. **OR** 2. a. Average of the ten highest reading core exit thermocouples (CETCs) is **≥ 700° F** **AND** b. RVLIS **plenum region = 0%.** **AND** c. Functional Restoration procedures not effective in **< 15 minutes.**
5. Reactor Vessel Water Level/RCS Leak Rate →	None	**Core Cooling CSF - ORANGE Path** conditions exist.	RCS leakage > available makeup capacity resulting in loss of Subcooling as indicated by CETCs is less than ACCEPTABLE VALUE per Iconic Display or RCS Subcooling Margin Figure 1/2 BST 2-1.	UNISOLABLE leak exceeding the capacity of one charging pump in the normal charging mode.	None	None
6. S/G Leakage / Rupture →	None	None	Steam Generator Tube Rupture that results in entry into BEP-3.	None	1. RUPTURED S/G is also FAULTED outside of Containment. **OR** 2. Primary-to-Secondary leakrate **> 10 gpm** with UNISOLABLE steam release from affected S/G to the environment.	None
7. Containment Isolation Valve Status →	None	None	None	None	1. Failure of all isolation valves in any one line to close. **AND** 2. Downstream pathway to the environment exits.	None
8. Containment Rad Monitoring →	Containment radiation monitor reading (AR020(21)) **> Fuel Cladding Loss Threshold,** Table F1.	None	Containment Radiation Monitor reading (AR020(21)) **> 25 R/hr**.	None	None	Containment radiation (AR020(21)) **> Containment Potential Loss Threshold**, Table F1.
9. ED Judgment →	Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Table F1 – Containment Radiation (AR020(21)) Thresholds

Fuel Cladding – Loss		Containment - Potential Loss	
Time After Shutdown (hrs)	**R/hr**	**Time After Shutdown (hrs)**	**R/hr**
≤ 2	1.95 E+03	≤ 2	4.40 E+03
> 2 to 4	1.70 E+03	> 2 to 4	3.85 E+03
> 4 to 8	1.45 E+03	> 4 to 8	3.35 E+03
> 8 to 16	1.24 E+03	> 8 to 16	2.80 E+03
> 16 to 23	1.09 E+03	> 16 to 23	2.50 E+03
> 23	1.08 E+03	> 23	2.50 E+03

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
Loss of AC Power	**MG1** Prolonged loss of all offsite power and prolonged loss of all onsite AC power to essential busses. 1 2 3 4 **EAL Threshold Values:** 1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2). **AND** 2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses. **AND** 3. a. Restoration of at least one unit ESF bus within 4 hours is **not** likely. **OR** b. **EITHER**: • **Core Cooling CSF - RED Path** conditions exist. • **Core Cooling CSF - ORANGE Path** conditions exist.	**MS1** Loss of all offsite power and loss of all onsite AC power to essential busses. 1 2 3 4 **EAL Threshold Values:** 1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2). **AND** 2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses. **AND** 3. Failure to restore power to at least one unit ESF bus within **15 minutes** from the time of loss of both offsite and onsite AC power.	**MA1** AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout. 1 2 3 4 **EAL Threshold Values:** 1. AC power capability to unit ESF busses reduced to only one of the following power sources for **> 15 minutes**: • Affected unit SAT 142-1(242-1) **OR** 142-2(242-2) • DG 1A(2A) **OR** DG 1B(2B) • Unit crosstie breakers **AND** 2. Any additional single power source failure will result in unit blackout.	**MU1** Loss of all offsite power to essential busses for greater than 15 minutes. 1 2 3 4 5 6 **EAL Threshold Values:** Loss of power to Transformers 142-1(242-1) and 142-2(242-2) for **> 15 minutes**.
Failure of RPS	**MG3** Failure of the Reactor Protection System to complete an automatic trip and manual trip was NOT successful and there is indication of an extreme challenge to the ability to cool the core. 1 2 **EAL Threshold Values:** 1. Automatic and manual Reactor Trip were not successful from Main Control Board as indicated by: a. Reactor power **≥ 5%** **OR** b. Intermediate Range Start Up Rate is positive **AND** 2. a. **Core Cooling CSF – RED Path** conditions exist. **OR** b. **Heat Sink CSF – RED Path** conditions exist.	**MS3** Failure of the Reactor Protection System to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded and manual trip was NOT successful. 1 2 **EAL Threshold Values:** Automatic and manual Reactor Trip were not successful from Main Control Board as indicated by: a. Reactor power **≥ 5%** **OR** b. Intermediate Range Start Up Rate is positive	**MA3** Failure of the Reactor Protection System to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded. 1 2 3 **EAL Threshold Values:** 1. A Reactor Protection System setpoint was exceeded **AND** 2. A successful automatic Reactor Trip did not occur	**MU3** Inadvertent criticality. 3 4 5 6 **EAL Threshold Values:** An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.
DC Power		**MS4** Loss of all vital DC power. 1 2 3 4 **EAL Threshold Values:** Loss of all vital DC power based on **< 108 VDC** on 125 VDC battery busses 111(211) and 112(212) for **> 15 minutes**.		

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

System Malfunction

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Heat Sink		**MS5** Complete loss of heat removal capability. 1 2 3 4 **EAL Threshold Values:** 1. **Core Cooling CSF - RED Path** conditions exist. **AND** 2. **Heat Sink CSF - RED Path** conditions exist.		
Annunciators		**MS6** Inability to monitor a SIGNIFICANT TRANSIENT in progress. 1 2 3 4 **EAL Threshold Values:** 1. Loss of most **(approximately 75%)** safety system annunciators (Table M2). **AND** 2. Indications needed to monitor safety functions (Table M3) are unavailable. **AND** 3. SIGNIFICANT TRANSIENT in progress (Table M4). **AND** 4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MA6** UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable. 1 2 3 4 **EAL Threshold Values:** 1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes.** **AND** 2. a. SIGNIFICANT TRANSIENT in progress (Table M4). **OR** b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MU6** UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes. 1 2 3 4 **EAL Threshold Values:** 1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** 2. UNPLANNED loss of most **(approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.

Table M2 - Control Room Panels
- 1/2 PM01J MCB Gen & Aux Power - 1/2 PM05J MCB Reactor and Chem Volume Control - 1/2 PM06J MCB Eng. Safety Features

Table M3 - Safety Functions and Related Systems
- 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) - Fission Product Barriers

Table M4 - Significant Transients
- Automatic Turbine Runback **> 25%** thermal reactor power - Electrical load rejection **> 25%** full electrical load - Reactor Trip - Safety Injection Actuation - Thermal power oscillations **> 10%**

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

<table>
<tr><th></th><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="5">**System Malfunction**</td></tr>
<tr><td>**RCS Leak**</td><td></td><td></td><td></td><td>**MU7** RCS leakage. 1 2 3 4
EAL Threshold Values:
1. Unidentified or pressure boundary leakage **> 10 gpm**.
OR
2. Identified leakage **> 25 gpm**.</td></tr>
<tr><td>**Communications**</td><td></td><td>
<table>
<tr><th colspan="3">Table M6 - Communications Capability</th></tr>
<tr><th>System</th><th>Onsite</th><th>Offsite</th></tr>
<tr><td>Radios</td><td>X</td><td></td></tr>
<tr><td>Plant page</td><td>X</td><td></td></tr>
<tr><td>Plant Telephone System</td><td>X</td><td></td></tr>
<tr><td>Commercial Telephones</td><td></td><td>X</td></tr>
<tr><td>NARS</td><td></td><td>X</td></tr>
<tr><td>ENS</td><td></td><td>X</td></tr>
<tr><td>HPN</td><td></td><td>X</td></tr>
<tr><td>Cellular phones</td><td></td><td>X</td></tr>
<tr><td>TSO/PJM (Electric Operations)</td><td></td><td>X</td></tr>
<tr><td>Satellite phones</td><td></td><td>X</td></tr>
</table>
</td><td></td><td>**MU10** UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5 6
EAL Threshold Values:
1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations.
OR
2. Loss of all Table M6 **Offsite** communications capability.</td></tr>
<tr><td>**T. S. Time**</td><td></td><td></td><td></td><td>**MU11** Inability to reach required shutdown within Technical Specification limits. 1 2 3 4
EAL Threshold Values:
Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.</td></tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 6 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 6 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 6 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, “Threat Assessment”. **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Control Room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per 1/2 BOA PRI-5, Control Room Inaccessibility procedure in < **15 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 6 D **EAL Threshold Values:** Entry into 1/2 BOA PRI-5, Control Room Inaccessibility procedure for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • Essential Service Water Cooling Towers • Condensate Storage Tanks • RWSTs		**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. Seismic event **> Operating Basis Earthquake (OBE)** as indicated by seismic check on 0PA02J. **AND** b. Confirmed by **EITHER:** • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 85 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in the Auxiliary Building as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. Seismic event as indicated by Annunciator 0-38-E5, Accelograph Accel High (0PM01J). **AND** b. Confirmed by **EITHER:** • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 85 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in Auxiliary Building that has the potential to affect safety related equipment needed for the current operating mode.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • Essential Service Water Cooling Towers • Condensate Storage Tanks • RWSTs		**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 6 D **EAL Threshold Values:** 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.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 6 D **EAL Threshold Values:** **Note**: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **3.07 E+07 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 6 D **EAL Threshold Values:** **Note**: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **3.07 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 6 D **EAL Threshold Values:** 1 VALID reading on any of the following effluent monitors **> 200 times** the high alarm setpoint established by a current radioactive release package for **≥ 15 minutes**. • 0PR001, Liquid Radwaste Effluent Monitor • 0PR002, Gas Decay Tank Effluent Monitor • 0PR010, Station Blowdown Monitor • 1/2 PR001, Containment Purge Effluent Monitor • Discharge Permit specified monitor **OR** 2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 1.83 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates, in **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. VALID reading on any of the following effluent monitors **> 2 times** the high alarm setpoint established by a current radioactive release package for **≥ 60 minutes**. • 0PR001, Liquid Radwaste Effluent Monitor • 0PR002, Gas Decay Tank Effluent Monitor • 0PR010, Station Blowdown Monitor • 1/2 PR001, Containment Purge Effluent Monitor • Discharge Permit specified monitor **OR** 2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 5.85 E+05 uCi/sec** for **≥ 60 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Abnormal Rad Levels		**Table R1** **Fuel Handling Incident Radiation Monitors** • Fuel Building Fuel Handling Incident Monitor 0RE-AR055 • Fuel Building Fuel Handling Incident Monitor 0RE-AR056 • Containment Fuel Handling Incident Monitor 1/2 RE-AR011 • Containment Fuel Handling Incident Monitor 1/2 RE-AR012	**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. A VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1. **OR** 2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.	**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by: • Refueling Cavity water level **< 23 ft**. above the Reactor Flange (**423 ft**. indicated level). **OR** • Spent Fuel Pool water level **< 23 ft**. above the fuel (**422 ft. 9 in.** indicated level). **OR** • Report of visual observation of an uncontrolled drop in water level in the Fuel Transfer Canal, Refueling Cavity, or Spent Fuel Pool. **AND** b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1. **OR** 2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.
	Table R2 **Areas Requiring Continuous Occupancy** • Main Control Room - 1/2 RE-AR010 • Central Alarm Station - (by survey) • Radwaste Control Room (Aux Bldg 383 ft. el.) - 0RE-AR007	**Table R3** **Areas Requiring Infrequent Access** • Unit 1 and 2 Remote Shutdown Panels (0RE-AR007) • High Radiation Sample Room – HRSS (0RE-AR031) • Containment Air Sample Panel – CASP (by survey) • Fire Hazards Panel (by survey)	**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions. **OR** 2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
Loss of AC Power				**MU1** Loss of all offsite power to essential busses for greater than 15 minutes. 1 2 3 4 5 6 **EAL Threshold Values:** Loss of power to Transformers 142-1(242-1) and 142-2(242-2) for **> 15 minutes**.
			MA2 Loss of all offsite power and loss of all onsite AC power to essential busses. 5 6 D **EAL Threshold Values:** 1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2) **AND** 2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses. **AND** 3. Failure to restore power to at least one unit ESF bus within **15 minutes** from the time of loss of both offsite and onsite AC power.	
RPS				**MU3** Inadvertent criticality. 3 4 5 6 **EAL Threshold Values:** An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.
DC Power				**MU4** UNPLANNED loss of required DC power for greater than 15 Minutes. 5 6 **EAL Threshold Values:** 1. UNPLANNED loss of all required vital DC power based on **< 108 VDC** indication on 125 VDC battery busses 111(211) and 112(212). **AND** 2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.
Heat Sink			**MA5** Inability to maintain plant in Cold Shutdown with irradiated fuel in the Reactor Vessel. 5 6 **EAL Threshold Values:** 1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F** for > Table M1 duration. **OR** 2. UNPLANNED Reactor Vessel pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.	**MU5** UNPLANNED loss of decay heat removal capability with irradiated fuel in the Reactor Vessel. 5 6 **EAL Threshold Values:** 1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F**. **OR** 2. Loss of all RCS temperature AND Reactor Vessel level indication for **> 15 minutes**.

Table M1 – RCS Reheat Duration Thresholds

RCS	Containment Closure	Duration
Intact	N/A	60 minutes*
Reduced Inventory (< 397 ft.)	Established	20 minutes*
	Not Established	0 minutes
Not Intact	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.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
RCS Leakage / Inventory	**MG8** Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged with irradiated fuel in the Reactor Vessel. 5 6 **EAL Threshold Values:** 1. Loss of Reactor Vessel inventory per Table M5 indications. **AND** 2. a. RVLIS **≤ 0% Plenum (390 ft. el.)** for **> 30 minutes**. **OR** b. Reactor Vessel level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following: • 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors > **3000 mR/hr** or off-scale high • Erratic Source Range Monitor indication **AND** 3. CONTAINMENT is challenged as indicated by one or more of the following: • Hydrogen concentration in Containment **≥ 5%** • Containment pressure **≥ 50 psig** • CONTAINMENT CLOSURE not established	**MS8** Loss of Reactor Vessel inventory affecting core decay heat removal capability. 5 **EAL Threshold Values:** 1. **Without** CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by RVLIS **≤ 15% Plenum (392.4 ft. el.)**. **OR** b. Reactor Vessel level unknown for **> 30 minutes** with a loss of Reactor Vessel inventory per Table M5 indications. **OR** 2. **With** CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by RVLIS **≤ 0% Plenum (390 ft. el.)**. **OR** b. Reactor Vessel level unknown for **> 30 minutes** with a loss of Reactor Vessel inventory as evidenced by either of the following: • Per Table M5 indications. • Erratic Source Range Monitor indication.	**MA8** Loss of RCS/Reactor Vessel inventory with irradiated fuel in the Reactor Vessel. 5 6 **EAL Threshold Values:** 1. a. Loss of RCS / Reactor Vessel inventory as indicated by RVLIS **≤ 27% Plenum (393 ft. el.)**. **OR** b. Loss of RCS / Reactor Vessel inventory as indicated by LT-046 and LT-049 **< 393 ft. el.** **OR** 2. a. Loss of RCS / Reactor Vessel inventory per Table M5 indications. **AND** b. RCS / Reactor Vessel level unknown for **> 15 minutes**.	**MU8** RCS leakage. 5 **EAL Threshold Values:** 1. Pressurizer level established limit **> 5% Cold Cal** and RCS level **cannot** be restored and maintained **> 5% Cold Cal.** **OR** 2. Pressurizer level established limit **< 5% Cold Cal** and RCS level **cannot** be restored and maintained **>** procedurally established limit.
RCS Leakage / Inventory		**MS9** Loss of Reactor Vessel inventory affecting core decay heat removal capability with irradiated fuel in the Reactor Vessel. 6 **EAL Threshold Values:** 1. **Without** CONTAINMENT CLOSURE established: a. Reactor Vessel Refueling Level Indicators LT-046 and LT-049 **< 393 ft. el.** **OR** b. Reactor Vessel level unknown with indication of core uncovery as evidenced by one or more of the following: • 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high • Erratic Source Range Monitor indication. **OR** 2. **With** CONTAINMENT CLOSURE established: a. Reactor Vessel Refueling Level Indicators LT-046 and LT-049 **= 392 ft. el.** or off scale low. **OR** b. Reactor Vessel level unknown with indication of core uncovery as evidenced by one or more of the following: • 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high • Erratic Source Range Monitor indication.	**Table M5 – Indications of RCS Leakage** • Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss	**MU9** UNPLANNED Loss of RCS inventory with irradiated fuel in the Reactor Vessel. 6 **EAL Threshold Values:** 1. UNPLANNED RCS level drop below the Reactor Vessel flange (400 ft. el.) for **≥ 15 minutes.** **OR** 2. a. Loss of Reactor Vessel inventory per Table M5 indications. **AND** b. Reactor Vessel level unknown.

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

System Malfunction

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Communications				**MU10** UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5 6 **EAL Threshold Values:** 1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations. **OR** 2. Loss of all Table M6 **Offsite** communications capability.

Table M6 - Communications Capability

System	Onsite	Offsite
Radios	X	
Plant page	X	
Plant Telephone System	X	
Commercial Telephones		X
NARS		X
ENS		X
HPN		X
Cellular Phones		X
TSO/PJM (Electric Operations)		X
Satellite Phones		X

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 6 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 6 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 6 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, "Threat Assessment". **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Control room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per 1/2 BOA PRI-5, Control Room Inaccessibility procedure in < **15 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 6 D **EAL Threshold Values:** Entry into 1/2 BOA PRI-5, Control Room Inaccessibility procedure for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** - Containment - Auxiliary Building - Fuel Handling Building - Main Steam Tunnels - Essential Service Water Cooling Towers - Condensate Storage Tanks - RWSTs		**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. Seismic event **> Operating Basis Earthquake (OBE)** as indicated by seismic check on 0PA02J. **AND** b. Confirmed by **EITHER:** • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 85 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in the Auxiliary Building as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. a. Seismic event as indicated by Annunciator 0-38-E5, Accelograph Accel High (0PM01J). **AND** b. Confirmed by **EITHER:** • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 85 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in Auxiliary Building that has the potential to affect safety related equipment needed for the current operating mode.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm: **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • Essential Service Water Cooling Towers • Condensate Storage Tanks • RWSTs		**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 6 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 6 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 6 D **EAL Threshold Values:** 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.

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

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RG1

Initiating Condition:

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem 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 Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **3.07 E+07 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 1000 mRem TEDE**

      OR

   b. **> 5000 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events that may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in a PAG dose being reached. Assumptions and DAPAR inputs are provided in calculation EP-EAL-0602.

The sum of both units' monitors provides the total station release rate.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

Threshold #2 Basis:

The TEDE (1000 mRem) and the CDE Thyroid (5000 mRem) doses are set at the EPA PAG Limits.

The 'site boundary' is defined by an approximately 800-meter (1/2-mile) radius around the plant. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RG1 (cont.)

Basis (cont.):

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release duration is known, and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev 4 AG1
2. EP-AA-112-500 Emergency Environmental Monitoring
3. Exelon DAPAR version 3.0a
4. EP-MW-110-200 Dose Assessment
5. EP-EAL-0602, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Byron Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RS1

Initiating Condition:

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem 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 Threshold Values:

Note: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) that exceeds or is expected to exceed **3.07 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 100 mRem TEDE**

      OR

   b. **> 500 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RS1 (cont.)

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events that may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in 10% of a PAG dose being reached. Assumptions and DAPAR inputs are provided in calculation EP-EAL-0602.

The sum of both units' monitors provides the total station release rate.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

Threshold #2 Basis:

The TEDE (100 mRem) and the CDE Thyroid (500 mRem) doses are set at 10% (Ratio 1:5 TEDE to CDE Thyroid) of the EPA PAG Limits.

The 'site boundary' is defined by an approximately 800-meter (1/2-mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

Basis (cont.):

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release duration is known and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev 4 AS1
2. EP-AA-112-500 Emergency Environmental Monitoring
3. Exelon DAPAR version 3.0a
4. EP-MW-110-200 Dose Assessment
5. EP-EAL-0602, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Byron Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer.

Operating Mode Applicability:

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

EAL Threshold Values:

1. VALID reading on any of the following effluent monitors **> 200 times** the high alarm setpoint established by a current radioactive release package for **≥ 15 minutes**.
   - 0PR001, Liquid Radwaste Effluent Monitor
   - 0PR002, Gas Decay Tank Effluent Monitor
   - 0PR010, Station Blowdown Monitor
   - 1/2 PR001, Containment Purge Effluent Monitor
   - Discharge Permit specified monitor

   OR

2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 1.83 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.

Basis:

UNPLANNED, as used in this context, includes any release for which a radioactive release package 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.

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

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA1 (cont.)

Basis (cont.):

Threshold #1 Basis:

The threshold addresses radioactivity releases (liquid or gaseous) that for whatever reason cause effluent radiation monitor readings to exceed two hundred times the alarm setpoint established by the radioactive release package. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the Offsite Dose Calculation Manual (ODCM) to warn of a release that is not in compliance with the Radiological Effluent Technical Specifications (RETS). Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific release package. 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 flowpath or effluent monitor other than those listed (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the Discharge Permit.

The Liquid Radwaste Discharge Monitor high alarm setpoint is typically based on ODCM concentration limits. During periods of release, the high alarm setpoint can be modified based upon tank activity and dilution flow. Detector 0RE-PR001 monitors liquid radwaste effluent from either 30,000-gallon release tank. The release tank discharge valves 0WX353 and 0WX896 close on high radiation.

Threshold #2 Basis:

Byron incorporates 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 ODCM. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

This EAL addresses a potential or actual drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. However, the effluent monitor Alert value for gaseous effluents was reduced to a value one half way between the Unusual Event value and the Site Area Emergency value to ensure sequential classifications. Assumptions and inputs for this calculation are provided in EP-EAL-0602. The sum of both units gaseous effluent monitor readings provides a total station release rate. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM, Rev 4. The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA1 (cont.)

Basis (cont.):

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

Threshold #3 Basis:

Confirmed sample analyses in excess of two hundred times the site ODCM limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by increasing the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mRem/yr basis of the 10 CFR 20 limits for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be approximately 10 mRem/hr. The required release duration was reduced to 15 minutes in recognition of the raised severity.

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

Basis Reference(s):

1. NEI 99-01, Rev 4 AA1
2. Sargent & Lundy calculation ATD-0221, Rev. 0
3. Exelon DAPAR version 3.0a
4. UFSAR Section 11.5.2.3
5. 0BISR 11.a.3-002, Channel Operation Test of Liquid Radwaste Effluent Radiation Monitor 0PR01J
6. ODCM, CY-BY-170-301, Section 3.0, Liquid Effluents
7. EP-EAL-0602, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Byron Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer.

Operating Mode Applicability:

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

EAL Threshold Values:

1. VALID reading on any of the following effluent monitors **> 2 times** the high alarm setpoint established by a current radioactive release package for **≥ 60 minutes**.
   - 0PR001, Liquid Radwaste Effluent Monitor
   - 0PR002, Gas Decay Tank Effluent Monitor
   - 0PR010, Station Blowdown Monitor
   - 1/2 PR001, Containment Purge Effluent Monitor
   - Discharge Permit specified monitor

   OR

2. The sum of VALID readings on the Unit 1 and 2 Aux Bldg Vent WRGMs (1/2 RE-PR030) is **> 5.85 E+05 uCi/sec** for **≥ 60 minutes** (as determined from Unit 1 & 2 PF430 or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

Basis:

UNPLANNED, as used in this context, includes any release for which a radioactive release package 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.

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

The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

Basis (cont.):

Threshold #1 Basis:

The effluent release paths are monitored for radioactivity prior to the flow reaching the point where it would mix with the process flow to the environment. Prior to initiating batch releases, the discharge volume is sampled and analyzed for radioactivity. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate. Radiation monitor alarm setpoints are established to automatically isolate the process flow at the point determined by the discharge permit. These limits are based on the Offsite Dose Calculation Manual ODCM.

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 flowpath or effluent monitor other than those listed (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the Discharge Permit.

The Liquid Radwaste Effluent monitor high alarm setpoint is typically based on ODCM concentration limits. During periods of release, the high alarm setpoint can be modified based upon tank activity and dilution flow. Detector 0RE-PR001 monitors liquid radwaste effluent from either 30,000-gallon release tank. The release tank discharge valves 0WX353 and 0WX896 close on high radiation.

Threshold #2 Basis:

This EAL addresses a potential drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 2, regulatory commitments for an extended period of time. The sum of both gaseous effluent monitor readings provides a total station release rate. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM. Assumptions and calculation inputs are provided in EP-EAL-0602.

The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition. Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

Threshold #3 Basis:

Confirmed sample analyses in excess of two times the site ODCM limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times ODCM for 30 minutes does not exceed this EAL.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU1 (cont.)

Basis (cont.):

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

Basis Reference(s):

1. NEI 99-01, Rev 4 AU1
2. Sargent & Lundy calculation ATD-0221, Rev. 0
3. Exelon DAPAR version 3.0a
4. UFSAR Section 11.5.2.3
5. 0BISR 11.a.3-002, Channel Operation Test of Liquid Radwaste Effluent Radiation Monitor 0PR01J
6. ODCM, CY-BY-170-301, Section 3.0, Liquid Effluents
7. EP-EAL-0602, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Byron Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA2

Initiating Condition:

Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel.

Operating Mode Applicability:

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

EAL Threshold Values:

1. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1.

Table R1 Fuel Handling Incident Radiation Monitors
• Fuel Building Fuel Handling Incident Monitor 0RE-AR055
• Fuel Building Fuel Handling Incident Monitor 0RE-AR056
• Containment Fuel Handling Incident Monitor 1/2 RE-AR011
• Containment Fuel Handling Incident Monitor 1/2 RE-AR012

OR

2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.

Basis:

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

Threshold #1 Basis:

Uncovering spent fuel represents a substantial degradation of the level of safety of the plant and warrants an Alert classification. Time is available to take corrective actions. 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, no prompt fatalities are predicted and the risk of injury is low. 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 fuel uncovery begins. The value of 1000 mR/hr was conservatively chosen for classification purposes.

Radiation monitor readings are used to provide indication of fuel uncovery and/or fuel damage. High monitor readings associated with the transfer or relocation of a source, stored in or near the pool or readings responding to a planned evolution such as removal of the reactor head or equipment relocation are not classified under this threshold since the reading would not be indicative of fuel uncovery and/or fuel damage.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA2 (cont.)

Basis (cont.):

Dropping heavy loads onto the spent fuel can cause significant damage to the spent fuel and an Alert is also warranted under these conditions provided that the above radiation monitor threshold readings are reached.

Fuel Building Fuel Handling Incident Monitors 0RE-AR055 and 0RE-AR056 are redundant GM type gamma detectors are mounted on the walls near the edge of the pool to provide reliable and rapid detection of radioactivity released from the pool surface. The monitors alarm locally and in the Main Control Room (5 mR/hr) and initiate control action to route the released activity through the emergency exhaust system. The monitors have an operating range that extends from 0.1 to 1E4 mR/hr.

Containment Fuel Handling Incident Monitors 1/2 RE-AR011 and 1/2 RE-AR012 are redundant GM type gamma detectors that provide reliable and rapid detection of radioactivity released from the water surface. The monitors alarm in the Main Control Room. They alarm at 10 mR/hr above background and isolate containment ventilation (VQ). The monitors have an operating range, which extends from 0.1 to 1E4 mR/hr.

Threshold #2 Basis:

Once Spent Fuel Pool water level drops below the low level alarm setpoint, further drops can be monitored only by visual observation.

Refueling Cavity water level is normally monitored by:

- LT-049 (LI-RY-049), range 392 ft. el. to 426 ft. el.
- LT-047 (LI-RY-047), 413 ft. el. to 426 ft. el.

Loss of inventory from the Refueling Cavity may also be indicated by:

- CNMT DRAIN LEAK DETECT FLOW HIGH alarm (BAR 1-1-A2, 2-1-A2)
- Abnormal flow on Containment drains flow recorder at 1/2 PM12J
- Floor Drain Sump (1/2 FT-RF008) Flowrate
- RX Cavity Sump (1/2 FT-RF010) Flowrate
- Grid 2 or Grid 4 Containment Area Monitor(s) trends greater than the alert alarm setpoint or increasing.

Without report of the position of spent fuel during transfer between the Reactor Vessel and the Spent Fuel Pool, fuel uncovery cannot be determined directly from the installed water level instrumentation. Visual observation, therefore, provides the only viable mechanism of determining if spent fuel in the spent fuel canal will be uncovered.

This EAL applies to irradiated fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA2 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 AA2
2. UFSAR 11.5.2.2.6, 11.5.2.2.7, 15.7.4, Table 12.3-3
3. Technical Specification Table 3.3.6-1
4. 1/2 BOA REFUEL-1 Fuel Handling Emergency Unit 1/2
5. 1/2 BOA REFUEL-2 Refueling Cavity or Spent Fuel Pool Level Loss Unit 1/2
6. TRM 3.9.A, Refueling Operations, Decay Time
7. BAR 1-1-A2, 2-1-A2, CNMT DRAIN LEAK DETECT FLOW HIGH alarm

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU2

Initiating Condition:

Unexpected rise in plant radiation.

Operating Mode Applicability:

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

EAL Threshold Values:

1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:

   - Refueling Cavity water level **< 23 ft**. above the Reactor Flange (**423 ft**. indicated level).

     OR

   - Spent Fuel Pool water level **< 23 ft**. above the fuel (**422 ft. 9 in.** indicated level).

     OR

   - Report of visual observation of an uncontrolled drop in water level in the Fuel Transfer Canal, Refueling Cavity, or Spent Fuel Pool.

   AND

   b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1.

Table R1 **Fuel Handling Incident Radiation Monitors**
• Fuel Building Fuel Handling Incident Monitor 0RE-AR055
• Fuel Building Fuel Handling Incident Monitor 0RE-AR056
• Containment Fuel Handling Incident Monitor 1/2 RE-AR011
• Containment Fuel Handling Incident Monitor 1/2 RE-AR012

OR

2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU2 (cont.)

Basis:

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

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

NORMAL LEVELS: Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Threshold #1 Basis:

Refueling Cavity water level is normally monitored by:

- LT-049 (LI-RY-049), range 392 ft. el. to 426 ft. el.
- LT-047 (LI-RY-047), 413 ft. el. to 426 ft. el.

Loss of inventory from the Refueling Cavity may also be indicated by:

- Grid 2 or Grid 4 Containment Area Monitor(s) trends greater than the alert alarm setpoint or increasing

Since no remote indication of Spent Fuel Pool water level exists, drops in Spent Fuel Pool water level can normally be detected only through visual observation. Loss of inventory from the Spent Fuel Pool may also be indicated if the Spent Fuel Pool or Transfer Canal leak detection system indicates flow.

The spent fuel transfer canal is normally aligned to either or both of the Refueling Cavity or Spent Fuel Pool when it contains spent fuel as it serves as the transfer path to move the fuel between these locations. Therefore, the level indications available for these two locations will also indicate the level in the spent fuel transfer canal. However, the threshold of "report of visual observation of a rapid drop in water level" is included for the spent fuel transfer canal in the event that it contains spent fuel and is not aligned to the Refueling Cavity or Spent Fuel Pool, and as an additional indication to those level monitors listed for the other locations.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU2 (cont.)

Basis (cont.):

Without report of the position of spent fuel during transfer between the Reactor Vessel and the Spent Fuel Pool, fuel uncovery cannot be determined directly from the installed water level instrumentation. Visual observation, therefore, provides the only viable mechanism of determining if spent fuel in the spent fuel canal will be uncovered.

Threshold #2 Basis:

Valid elevated area radiation levels usually have long lead times relative to the potential for radiological release beyond the site boundary, thus impact to public health and safety is very low.

This EAL addresses UNPLANNED rise in radiation levels inside the plant. These radiation levels represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant.

Basis Reference(s):

1. NEI 99-01, Rev 4 AU2
2. Technical Specifications 3.7.14
3. 1/2 BOA REFUEL-1 Fuel Handling Emergency Unit 1/2
4. 1/2 BOA REFUEL-2 Refueling Cavity Or Spent Fuel Pool Level Loss Unit 1/2
5. BAR 1-1-C1 SPENT FUEL PIT LEVEL HIGH LOW
6. 1/2 BOSR 0.1-6 Unit One(Two) Mode 6 Shiftly and Daily Operating Surveillance
7. BOP RH-8 Filling the Refueling Cavity for Refueling
8. BOP RH-9 Pump Down of the Refueling Cavity to the RWST
9. BOP RC-4 Reactor Coolant System Drain
10. BAR 1-6-C3 REFUELING CAVITY LVL HIGH/LOW

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA3

Initiating Condition:

Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown.

Operating Mode Applicability:

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

EAL Threshold Values:

1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions:

Table R2 – Areas Requiring Continuous Occupancy
• Main Control Room - 1/2 RE-AR010 • Central Alarm Station - (by survey) • Radwaste Control Room (Aux Bldg 383 ft. el.) - 0RE-AR007

OR

2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.

Table R3 – Areas Requiring Infrequent Access
• Unit 1 and 2 Remote Shutdown Panels (0RE-AR007) • High Radiation Sample Room – HRSS (0RE-AR031) • Containment Air Sample Panel – CASP (by survey) • Fire Hazards Panel (by survey)

Basis:

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

Basis (cont.):

Threshold #1 Basis:

This EAL addresses increases in radiation levels that impede necessary access to operating stations requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. Areas requiring continuous occupancy include the Main Control Room, the Central Alarm Station (CAS) and the Radwaste Control Room. The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations.

The value of 15 mR/hr is derived from the General Design Criteria (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. A 30 day duration implies an event potentially more significant than an Alert.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the raised radiation levels and determine if any other EALs may be involved. For example, a dose rate of 15 mR/hr in the Main Control Room may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This EAL is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.).

Threshold #2 Basis:

This EAL addresses raised radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Typically areas requiring infrequent access to maintain plant safety functions include plant VITAL AREAS. Area radiation levels above 2000 mR/hr are indicative of radiation fields that may limit personnel access to equipment, the operation of which may be needed to assure adequate core cooling or shutdown the reactor.

The dose rate threshold selected is based on site administrative limits.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RA3 (cont.)

Basis (cont.):

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the raised radiation levels and determine if any other EAL may be involved. For example, a dose rate of 2000 mR/hr may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.) or pre-existing radiation areas for which radiological controls already exist. The concern of this threshold is the unanticipated rise in radiation levels that results in unplanned restrictions to areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown.

Basis Reference(s):

1. NEI 99-01, Rev 4 AA3
2. UFSAR Chapter 3.02, UFSAR Table 3.2-1

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU3

Initiating Condition:

Fuel clad degradation.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. VALID Gross Failed Fuel Monitor 1/2 RE-PR006 indicating I-135 concentration of **> 5 uCi/cc.**

   OR

2. a. Dose Equivalent I-131specific coolant activity **> 1.0 uCi/gm.**

      OR

   b. Gross specific coolant activity **> 100 / Ē uCi/gm.**

Basis:

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

Threshold #1 Basis:

The I-135 concentration calculated for a dose equivalent I-131 value of 1 uCi/cc is 0.57 uCi/g. This value is too small to be able to be detected by the Gross Failed Fuel Monitor. Therefore, a monitor value is chosen that is in the detectable range of the monitor and provides reasonable assurance that the 1 uCi/g dose equivalent I-131 value is exceeded based upon the radiation monitor reading.

The modifier "VALID" is appropriate because there are several conditions that may cause the monitor to alarm that are not related to fuel clad degradation and therefore should not result in the declaration of an Unusual Event.

Threshold #2 Basis:

Threshold #2 addresses coolant samples exceeding coolant technical specifications for iodine spike.

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 Reactor Coolant activity unless another cause is known, e.g., Reactor Coolant System chemical decontamination evolution (during shutdown) is ongoing with resulting high activity levels.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENTS

RU3 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 SU4
2. Technical Specifications 3.4.16
3. 1/2 BOA PRI-4, Abnormal Primary Chemistry Unit 1/2
4. PWR Letdown Rad Monitor Setpoint Calculation for Degraded Fuel Indication

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 Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the General Emergency classification level each barrier is weighted equally.

Basis Reference(s):

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

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 Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Site Area Emergency classification level, each barrier is weighted equally.

Basis Reference(s):

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

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 Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Alert classification level, 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 EAL FS1.

Basis Reference(s):

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

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 Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

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 EAL 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC1 – Loss or Potential Loss

Initiating Condition:

Critical Safety Function Status.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Core Cooling CSF - RED Path conditions exist.

POTENTIAL LOSS

1. **Core Cooling CSF - ORANGE Path** conditions exist.

   OR

2. **Heat Sink CSF - RED Path** conditions exist.

Basis:

Loss Basis:

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.

The Core Cooling Critical Safety Function RED path condition exists when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200° F.

Potential Loss Basis:

The Core Cooling Critical Safety Function ORANGE path condition exists if:

- The average of the ten highest reading core exit thermocouples (CETCs) is reading less than 1200° F but greater than 700° F, and
- RCS subcooling based on CETCs is less than ACCEPTABLE VALUE per Iconic Display or RCS Subcooling Margin Figure 1/2 BST 2-1.

Any of these conditions indicate subcooling has been lost and that some fuel cladding damage may potentially occur.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC1 – Loss or Potential Loss (cont.)

Basis (cont.):

The Heat Sink Critical Safety Function (CSF) RED path indicates the heat sink is under extreme challenge and indicates a potential loss of the Fuel Cladding barrier. The Heat Sink Critical Safety Function Red path conditions exist if narrow range levels in all steam generators (S/Gs) are less than or equal to 10% - Unit 1 (31% adverse containment) and 14% - Unit 2 (34% adverse containment) and total feedwater flow to all S/Gs is less than or equal to 500 gpm. If total feed flow is less than 500 gpm due to procedurally directed operator actions then this condition does not apply.

The combination of these two conditions indicates the ultimate heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the Fuel Cladding barrier. This condition also corresponds to RCS barrier potential loss threshold RC1 resulting in at least a Site Area Emergency.

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. 1/2 BST-2 Core Cooling Unit 1/2
3. 1/2 BFR-C.1 Response to Inadequate Core Cooling Unit 1/2
4 1/2 BFR-C.2 Response to Degraded Core Cooling Unit 1/2
5. 1/2 BST-3 Heat Sink Unit 1/2
6. 1/2 BFR H.1, Response to Loss of Secondary Heat Sink Unit 1/2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC2 – Loss

Initiating Condition:

Primary Coolant activity level.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.

Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. 300 uCi/gm Dose Equivalent I-131 is well above that expected for iodine spikes and corresponds, generically, to about 2% to 5% fuel cladding damage (0.6% clad failure per S&L calculation BB-ER-02, rev 0). This amount of radioactivity indicates significant clad damage and thus the Fuel Cladding barrier is considered lost.

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. EP-AA-123-1003 Core Damage Assessment Methodology (CDAM) Program Technical Basis
3. S&L calculation BB-ER-02, Rev 0

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC4 – Loss or Potential Loss

Initiating Condition:

Core Exit Thermocouple readings.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Average of the ten highest reading core exit thermocouples (CETCs) is **> 1200° F**.

POTENTIAL LOSS

Average of the ten highest reading core exit thermocouples (CETCs) is **> 700° F**.

Basis:

A failed CETC Channel can lead to indication of the CETC > 700° F until the system removes the failed channel from average. Fission product barrier loss or potential loss is based on VALID CETC readings.

Loss Basis:

The CETC value corresponds to the Core Cooling Critical Safety Function RED path but is evaluated separately from the CSF Status. The elevated temperature corresponds to significant superheating of the coolant and is indicative of a loss of the Fuel Cladding barrier.

Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use (initiation after SI is blocked).

Potential Loss Basis:

The CETC value corresponds to the Core Cooling Critical Safety Function ORANGE path but is evaluated separately from the CSF Status because the CSF evaluation considers the degree of subcooling prior to status determination. The elevated temperature corresponds to a loss of subcooling and is indicative of a Potential Loss of the Fuel Cladding barrier.

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. 1/2 BST-2 Core Cooling Unit 1/2
3. 1/2 BFR-C.1 Response to Inadequate Core Cooling Unit 1/2
4. 1/2 BFR-C.2 Response to Degraded Core Cooling Unit 1/2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC5 – Potential Loss

Initiating Condition:

Reactor Vessel water level

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

Core Cooling CSF - ORANGE Path conditions exist.

Basis:

The "Potential Loss" EAL is defined by the Core Cooling CSF - ORANGE path.

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. 1/2 BFR-C.2 Response to Degraded Core Cooling Unit 1/2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC8 – Loss

Initiating Condition:

Containment radiation monitoring.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Containment radiation monitor reading (AR020(21)) **> Fuel Cladding Loss Threshold**, Table F1.

Table F1 – Containment Radiation (AR020(21)) Thresholds	
Time After Shutdown (hours)	**Fuel Cladding Loss (R/hr)**
≤ 2	1.95 E+03
> 2 to 4	1.70 E+03
> 4 to 8	1.45 E+03
> 8 to 16	1.24 E+03
> 16 to 23	1.09 E+03
> 23	1.08 E+03

Basis:

The containment radiation monitor readings specified in Table F1 provide values that indicate the release of reactor coolant into the containment atmosphere with elevated activity indicative of fuel damage (~2%). The values are a function of time after shutdown and were derived using CDAM - with 2% clad damage, no containment sprays, in operation, CETC >1200º F and RCS pressure at <1600 psig assuming LOCA depressurized system. The reading is calculated assuming the instantaneous release and dispersal of the above reactor coolant noble gas and iodine inventory into the containment atmosphere.

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations allowed within Technical Specifications (including iodine spiking) and are therefore indicative of fuel damage (approximately 2% - 5% cladding failure). The reading is higher than that specified for the loss of RCS barrier; thus, elevated containment radiation readings at or above the Fuel Cladding barrier loss threshold signify a loss of two fission product barriers.

During at power (including ATWS) conditions the value listed for the "≤ 2 hours after shutdown" row is used as an indication of fuel damage.

FC8 – Loss (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FC9 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.

Basis:

The Emergency Director judgment fuel cladding loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Fuel Cladding barrier. The inability to monitor fuel cladding integrity should also be considered as a factor in judging that the Fuel Cladding barrier may be considered lost or potentially lost.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC1 – Potential Loss

Initiating Condition:

Critical Safety Function Status.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

1. **RCS Integrity CSF - RED Path** conditions exist.

   OR

2. **Heat Sink CSF - RED Path** conditions exist.

Basis:

Threshold #1 Basis

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.

The RCS Integrity Critical Safety Function (CSF) Red path conditions exist if:

- Temperature drop in any RCS cold leg is greater than or equal to 100° F/hr, and
- Any RCS cold leg temperature/pressure is to the left of Plant Operational Limits Figure 1/2 BST 4-1 Limit A.

The combination of these two conditions indicates the RCS barrier is under significant challenge and should be considered a potential loss of RCS barrier.

Threshold #2 Basis

The Heat Sink Critical Safety Function (CSF) RED path indicates the heat sink is under extreme challenge and indicates a potential loss of the Fuel Cladding barrier. The Heat Sink Critical Safety Function Red path conditions exist if narrow range levels in all steam generators (S/Gs) are less than or equal to 10% - Unit 1 (31% adverse containment) and 14% - Unit 2 (34% adverse containment) and total feedwater flow to all S/Gs is less than or equal to 500 gpm. If total feed flow is less than 500 gpm due to procedurally directed operator actions then this condition does not apply.

The combination of these two conditions indicates the ultimate heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the RCS barrier. This condition also corresponds to Fuel Cladding barrier potential loss threshold FC1 resulting in at least a Site Area Emergency.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC1 – Potential Loss (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. 1/2 BST-4 Integrity Unit 1/2
3. 1/2 BFR-P.1 Response To Imminent Pressurized Thermal Shock Condition Unit 1/2
4. 1/2 BST-3 Heat Sink Unit 1/2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC5 – Loss or Potential Loss

Initiating Condition:

RCS leak rate

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

RCS leakage > available makeup capacity resulting in loss of subcooling as indicated by CETCs is less than ACCEPTABLE VALUE per Iconic Display or RCS Subcooling Margin Figure 1/2 BST 2-1.

POTENTIAL LOSS

UNISOLABLE leak exceeding the capacity of one charging pump in the normal charging mode.

Basis:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Loss Basis:

This threshold addresses conditions in which 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 Basis:

This threshold is based on the inability to maintain normal liquid inventory within the RCS by normal operation of the Chemical and Volume Control System, which is considered as one centrifugal charging pump discharging to the charging header. The need for a second charging pump would be indicative of a substantial RCS leak. The minimum operability flow rate for each charging pump is 60 gpm.

Normal Charging Lineup refers to the normal charging system flow path through the volume control system including normal and design alternate flow paths, and flow to reactor coolant pump seals.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC5 – Loss or Potential Loss (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. 1/2 BST-2 Core Cooling Unit 1/2
3. 1/2 BFR-C.1 Response to Inadequate Core Cooling Unit 1/2
4. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
5. UFSAR Fig. 6.3-4

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC6 – Loss

Initiating Condition:

S/G Tube Rupture.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Steam Generator Tube Rupture that results in entry into BEP-3.

Basis:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

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.

To meet this threshold, the leakage must be large enough to cause actuation of ECCS (SI). ECCS (SI) actuation is caused by:

- PZR Low Pressure (< 1829 psig)
- Steam Line Low Pressure (< 640 psig)
- Containment High Pressure (> 3.4 psig)
- Manual Safety Injection

This EAL is intended to address the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment Barrier "Loss" EAL CT6 and Fuel Clad Barrier EALs. The "Loss" EAL addresses RUPTURED SG(s) for which the leakage is large enough to cause automatic or manual actuation of ECCS (SI). By itself, this EAL will result in the declaration of an ALERT. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a SITE AREA EMERGENCY per Containment Barrier "Loss" EAL CT6.

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
3. 1/2 BEP-0 Reactor Trip Or Safety Injection Unit 1/2
4. 1/2 BEP-3 Steam Generator Tube Rupture Unit 1/2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC8 – Loss

Initiating Condition:

Containment radiation monitoring.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Containment radiation monitor reading (AR020(21)) **> 25 R/hr**.

Basis:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

The containment radiation monitor reading is a value that indicates a significant release of reactor coolant to the containment. A reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the containment atmosphere. Conservative estimates (using Technical Specifications high RCS activity) indicated that the readings from release of the normal RCS inventory would be ~ 25 R/hr. The reading is less than that specified for Fuel Cladding barrier Loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations.

Computer points:

1RE-AR020 – Unit 1 High Range Containment (RA0046)

1RE-AR021 – Unit 1 High Range Containment (RA0047)

2RE-AR020 – Unit 2 High Range Containment (RA0071)

2RE-AR021 – Unit 2 High Range Containment (RA0072)

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RC9 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

<u>LOSS</u>

Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.

<u>POTENTIAL LOSS</u>

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.

Basis:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

The Emergency Director judgment RCS loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the RCS barrier. The inability to monitor RCS integrity should also be considered as a factor in judging that the RCS barrier may be considered lost or potentially lost.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT1 – Potential Loss

Initiating Condition:

Critical Safety Function Status.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

Containment CSF - RED Path conditions exist.

Basis:

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment. Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this EAL is primarily a discriminator between SITE AREA EMERGENCY and GENERAL EMERGENCY representing a potential loss of the third barrier.

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

The Containment Critical Safety Function (CSF) Red path conditions exist if containment pressure is equal to or greater than 50 psig. This pressure is the containment design pressure and is well in excess of that expected from the design basis loss of coolant accident. This threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation (metal-water reaction) or failure to trip in combination with RCS breach. This combination of conditions would be expected to require the declaration of a General Emergency.

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. 1/2 BST-5 Containment Unit 1/2
3. 1/2 BFR-Z.1 Response to High Containment Pressure Unit 1/2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT3 – Loss or Potential Loss

Initiating Condition:

Containment pressure.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

1. Rapid unexplained drop in Containment pressure following initial pressure rise.

   OR

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

POTENTIAL LOSS

1. Containment pressure **≥ 50 psig** and rising.

   OR

2. Hydrogen concentration in Containment **≥ 5%.**

   OR

3. a. Containment pressure **≥ 20 psig.**

      AND

   b. Less than one train of Containment Spray operating.

Basis:

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

Loss Threshold #1 Basis:

Rapid unexplained loss of pressure (i.e., not attributable to containment spray, cooling or condensation effects) following an initial pressure rise indicates a loss of containment integrity. The referenced analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.8 psig for Unit 1 and 38.4 psig for Unit 2 (experienced during a LOCA).

Loss Threshold #2 Basis:

Containment pressure and sump levels should rise as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure response not consistent with LOCA conditions indicates containment bypass and a loss of containment integrity.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT3 – Loss or Potential Loss (cont.)

Basis (cont.):

Potential Loss Threshold #1 Basis:

This threshold is the containment design pressure and is well in excess of that expected from the design basis loss of coolant accident. The referenced analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.8 psig for Unit 1 and 38.4 psig for Unit 2 (experienced during a LOCA).

The threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation (metal-water reaction) or failure to trip in combination with RCS breach. This condition would be expected to require the declaration of a General Emergency.

Potential Loss Threshold #2 Basis:

If hydrogen concentration reaches or exceeds 5% in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred. Since this threshold is also indicative of loss of both Fuel Cladding and RCS barriers with the potential loss of the Containment barrier, it therefore will likely warrant declaration of a General Emergency

Containment hydrogen concentration is indicated on 1/2 HSU-PS345 and 1/2 HSU-PS346 and PPDS.

Potential Loss Threshold #3 Basis:

This threshold represents a potential loss of containment in that the containment depressurization equipment is either lost or performing in a degraded manner.

The Containment Spray System limits post accident conditions to less than the containment design values. The Containment Spray System consists of two separate 100% capacity trains, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping.

During a Design Basis Accident (DBA), a minimum of one containment spray train is required to maintain the containment peak pressure below the design limits.

The containment pressure setpoint (20 psig) is the pressure at which the equipment should have actuated and began performing its function.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT3 – Loss or Potential Loss (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. UFSAR Section 15.6.5.2.1
3. 1/2 BST-5 Containment Unit 1/2
4. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
5. Technical Specifications 3.6.6

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT4 – Potential Loss

Initiating Condition:

Core Exit Thermocouple readings.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

1. a. Average of the ten highest reading core exit thermocouples (CETCs) is **≥ 1200° F**

   AND

   b. Functional Restoration procedures not effective in **< 15 minutes**.

   OR

2. a. Average of the ten highest reading core exit thermocouples (CETCs) is **≥ 700° F**

   AND

   b. RVLIS **plenum region = 0%.**

   AND

   c. Functional Restoration procedures not effective in **< 15 minutes.**

Basis:

A failed CETC Channel can lead to indication of the CETC ≥ 700° F until the system removes the failed channel from average. Fission product barrier loss or potential loss is based on VALID CETC readings.

The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

The conditions in this potential loss EAL 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 Heat Sink criteria in the Fuel and RCS barrier columns, this EAL 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.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT4 – Potential Loss (cont.)

Basis (cont.)

Potential Loss Threshold #1 Basis:

The Core Cooling Critical Safety Function (CSF) RED path is entered when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200°F. Entry into Core Cooling RED path requires entry into functional restoration procedure 1/2 BFR-C.1, Response to Inadequate Core Cooling.

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 function restoration procedures to arrest the core melt sequence. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing. Whether or not actions will be effective should be apparent within 15 minutes of entry into restoration procedures. The Emergency Director should make the declaration as soon as it is determined that the procedures have not been, or will not be effective.

Potential Loss Threshold #2 Basis:

Core Cooling Critical Safety Function ORANGE path conditions exist when the average of the ten highest reading core exit thermocouples (CETCs) is reading greater than or equal to 700°F.

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 function restoration procedures to arrest the core melt sequence. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing. Whether or not 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 not been, or will not be effective.

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. 1/2 BST-2 Core Cooling Unit 1/2
3. 1/2 BFR-C.1 Response to Inadequate Core Cooling Unit 1/2
4. 1/2 BFR-C.2 Response to Degraded Core Cooling Unit 1/2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT6 – Loss

Initiating Condition:

S/G secondary side release with primary to secondary leakage.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

1. RUPTURED S/G is also FAULTED outside of Containment.

   OR

2. Primary-to-Secondary leakrate **> 10 gpm** with UNISOLABLE steam release from affected S/G to the environment.

Basis:

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.

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.

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

Loss Threshold #1 Basis:

This EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. The first threshold addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers. In conjunction with RCS Barrier Loss RC6, this would always result in the declaration of a SITE AREA EMERGENCY.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT6 – Loss (cont.)

Basis (cont.)

Loss Threshold #2 Basis:

The second threshold addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the affected 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., SGTR with concurrent loss of offsite power and the affected 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 a nonisolable release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs.

A pressure boundary leakage of 10 gpm is also used as the threshold in EAL MU7, RCS leakage.

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. 1/2 BEP-0 Reactor Trip Or Safety Injection Unit 1/2
3. 1/2 BEP-3 Steam Generator Tube Rupture Unit 1/2
4. 1/2 BOA SEC-8 Steam Generator Tube Leak Unit 1/2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT7 – Loss

Initiating Condition:

Containment Isolation Valve status after Containment Isolation.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

1. Failure of all isolation valves in any one line to close.

   AND

2. Downstream pathway to the environment exists.

Basis:

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

This EAL is intended to address incomplete containment isolation that allows direct release (gaseous or liquid flowpath) to the environment outside of containment (for example into the Auxiliary Bldg, Turbine Bldg or outside atmosphere). It represents a loss of the containment barrier.

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a radiological release pathway to the environment (even if the isolation signal is not VALID). The concern is the UNISOLABLE open pathway to the environment. A failure of the ability to close any open isolation valves in any one line indicates a breach of containment integrity. Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include UNISOLABLE containment atmospheric vent paths as well as UNISOLABLE primary systems (RCS). If the primary system leakage outside containment cannot be isolated, a loss of both the RCS and the Containment barriers results. No leakage threshold is specified since leaks outside containment, particularly under dynamic conditions, are difficult to quantify and may manifest themselves with diverse symptoms.

Symptoms of a primary system discharging outside containment may be indicated via mass balance, lowering RCS inventory without corresponding containment response, or area temperatures and radiation levels outside containment. It is for this reason that Emergency Director judgment should be used in evaluating this criterion. However, it is intended that the magnitude of the primary system leak associated with this EAL be consistent with RCS barrier RC5 Potential Loss of ~100 gpm or greater. Minor release paths such as instrument and sample lines are not considered under this threshold.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT7 – Loss (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
3. UFSAR Fig. 6.3-4

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT8 – Potential Loss

Initiating Condition:

Significant radioactive inventory in Containment

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

POTENTIAL LOSS

Containment radiation (AR020(21)) **> Containment Potential Loss Threshold**, Table F1.

Table F1 – Containment Radiation (AR020(21)) Thresholds	
Time After Shutdown (hours)	**Containment Potential Loss (R/hr)**
≤ 2	4.40 E+03
> 2 to 4	3.85 E+03
> 4 to 8	3.35 E+03
> 8 to 16	2.80 E+03
> 16 to 23	2.50 E+03
> 23	2.50 E+03

Basis:

The Containment Barrier includes the containment building, its connections up to and including the outboard containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outboard secondary side isolation valve.

The containment radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Cladding barrier. NUREG-1228 “Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents” states that such readings do not exist when the amount of cladding damage is less than 20%. The values are a function of time after shutdown and were derived using CDAM v.1.1 assuming 20% clad damage, no containment sprays in operation, CETC > 1200º F and RCS pressure at <1600 psig assuming LOCA depressurized system. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a significant failure into the reactor coolant has occurred.

During at power (including ATWS) conditions the value listed for the “≤ 2 hours after shutdown” row is used as an indication of fuel damage.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT8 – Potential Loss (cont.)

Basis (cont.):

Regardless of whether the Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Containment barrier. The reading is higher than that specified for Fuel Cladding Loss FC8 and RCS Loss RC8. Containment radiation readings at or above the Containment barrier potential loss threshold, therefore, signify a loss of two fission product barriers and potential loss of a third, indicating the need to upgrade the emergency classification to a General Emergency.

Computer points:

1RE-AR020 – Unit 1 High Range Containment (RA0046)

1RE-AR021 – Unit 1 High Range Containment (RA0047)

2RE-AR020 – Unit 2 High Range Containment (RA0071)

2RE-AR021 – Unit 2 High Range Containment (RA0072)

Basis Reference(s):

1. NEI 99-01, Rev 4 Table 5-F-4
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CT9 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis:

The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

The Emergency Director judgment Containment loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Containment barrier. The inability to monitor Containment parameters should also be considered as a factor in judging that the Containment barrier may be considered lost or potentially lost.

Basis Reference(s):

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

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG1

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2).

   AND

2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses.

   AND

3. a. Restoration of at least one unit ESF bus within 4 hours is **not** likely.

      OR

   b. **EITHER**:
      - **Core Cooling CSF - RED Path** conditions exist.
      - **Core Cooling CSF - ORANGE Path** conditions exist.

Basis:

Loss of all AC power to ESF busses compromises the availability of all plant safety systems. Prolonged loss of all AC power may lead to loss of Fuel Cladding, RCS and Containment barriers. The four-hour interval to restore AC power to either unit ESF bus is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout".

The likelihood of restoring at least one ESF bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. The ESF busses may be powered from any of the following onsite sources:

- Emergency Diesel Generator 1A(2A) for 4160-V ESF bus 141(241)
- Emergency Diesel Generator 1B(2B) for 4160-V ESF bus 142(242)

Offsite AC power sources feed the ESF busses through the System Auxiliary Transformers 142-1(242-1) and 142-2(242-2). The ESF busses of the affected unit can be powered from the unaffected unit through the crosstie breakers ACB 1414(2414) and ACB 1424(2424). Unit crosstie is considered an adequate source of offsite power when evaluating this EAL.

Basis (cont.):

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

Core Cooling Critical Safety Function (CSF) RED path conditions exist when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200° F. This condition indicates subcooling has been lost and that some fuel cladding damage may potentially occur.

Core Cooling Critical Safety Function (CSF) ORANGE path conditions exist when:

- The average of the ten highest reading core exit thermocouples (CETCs) is reading less than 1200° F but greater than or equal to 700° F, AND
- RCS subcooling based on CETCs is less than ACCEPTABLE VALUE per Iconic Display or RCS Subcooling Margin Figure 1/2 BST 2-1.

Either of these conditions indicates significant core exit superheating and core uncovery. This is considered a loss of the Fuel Cladding barrier.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG1 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 SG1
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BOA ELEC-3 Loss Of 4KV ESF Bus Unit 1/2
5. 1/2 BOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BCA-0.3 Response To Opposite Unit Loss Of All AC Power Unit 1/2
10. BOP AP-51, Isolating Unit 1 System AUX Transformer 142-1 & 142-2 While Unit Is At Power
11. BOP AP-52, Restoring Unit 1 System AUX Transformer 142-1 & 142-2 During Power Operation
12. BOP AP-53, Isolating Unit 2 System AUX Transformer 242-1 & 242-2 While Unit Is At Power
13. BOP AP-54, Restoring Unit 2 System AUX Transformer 242-1 & 242-2 During Power Operation
14. Safety Evaluations of the Byron Station and Byron Station Responses to the Station Blackout (SBO) Rule (TAC NOS. 68522, 68523 AND 68515, 68516)
15. 1/2 BST-2 Core Cooling Unit 1/2
16. 1/2 BFR-C.1 Response to Inadequate Core Cooling Unit 1/2
17. 1/2 BFR-C.2 Response to Degraded Core Cooling Unit 1/2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS1

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2).

   AND

2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses.

   AND

3. Failure to restore power to at least one unit ESF bus within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power compromises all plant safety systems and represents failures of plant functions required for the protection of the public. The ESF busses may be powered from any of the following onsite sources:

- Emergency Diesel Generators 1A(2A) for 4160-V ESF bus 141(241)
- Emergency Diesel Generators 1B(2B) for 4160-V ESF bus 142(242)

Offsite AC power sources feed the ESF busses through the System Auxiliary Transformers 142-1(242-1) and 142-2(242-2). The ESF busses of the affected unit can be powered from the unaffected unit through the crosstie breakers ACB 1414(2414) and ACB 1424(2424). Unit crosstie is considered an adequate source of offsite power when evaluating this EAL.

Consideration should be given to available loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not available on the energized bus, the bus should not be considered available.

The fifteen-minute interval begins from the time of loss of both onsite and offsite AC power and was selected as a threshold to exclude transient or momentary power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS1 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 SS1
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BOA ELEC-3 Loss Of 4KV ESF Bus Unit 1/2
5. 1/2 BOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BCA-0.3 Response To Opposite Unit Loss Of All AC Power Unit 1/2
10. BOP AP-51, Isolating Unit 1 System AUX Transformer 142-1 & 142-2 While Unit Is At Power
11. BOP AP-52, Restoring Unit 1 System AUX Transformer 142-1 & 142-2 During Power Operation
12. BOP AP-53, Isolating Unit 2 System AUX Transformer 242-1 & 242-2 While Unit Is At Power
13. BOP AP-54, Restoring Unit 2 System AUX Transformer 242-1 & 242-2 During Power Operation
14. Safety Evaluations of the Byron Station and Byron Station Responses to the Station Blackout (SBO) Rule (TAC NOS. 68522, 68523 AND 68515, 68516)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA1

Initiating Condition:

AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. AC power capability to unit ESF busses reduced to only one of the following power sources for **> 15 minutes**:
   - Affected unit SAT 142-1(242-1) **OR** 142-2(242-2)
   - DG 1A(2A) **OR** DG 1B(2B)
   - Unit crosstie breakers

   AND

2. Any additional single power source failure will result in unit blackout.

Basis:

Capability: (pertaining to electrical power supplies) Is equipment that is available to provide and maintain AC power at the required voltage and frequency for the required load.

The reduction of available reliable power sources to a condition in which any additional single failure will result in a Unit Blackout is a substantial degradation in the level of safety of the plant. A Unit Blackout is a loss of AC power to all unit ESF busses. Byron blackout coping duration is four hours.

The listed power supplies take into account sources that, if unavailable, establish single-failure vulnerability. This EAL allows for the use of the unit crosstie breaker if they are the only source of power to the affected unit. The Emergency Director must consider the use of the crosstie breaker and the consequent demand on the unaffected unit.

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA1 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 SA5
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BOA ELEC-3 Loss Of 4KV ESF Bus Unit 1/2
5. 1/2 BOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BCA-0.3 Response To Opposite Unit Loss Of All AC Power Unit 1/2
10. BOP AP-51, Isolating Unit 1 System AUX Transformer 142-1 & 142-2 While Unit Is At Power
11. BOP AP-52, Restoring Unit 1 System AUX Transformer 142-1 & 142-2 During Power Operation
12. BOP AP-53, Isolating Unit 2 System AUX Transformer 242-1 & 242-2 While Unit Is At Power
13. BOP AP-54, Restoring Unit 2 System AUX Transformer 242-1 & 242-2 During Power Operation
14. Safety Evaluations of the Byron Station and Byron Station Responses to the Station Blackout (SBO) Rule (TAC NOS. 68522, 68523 AND 68515, 68516)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU1

Initiating Condition:

Loss of all offsite power to essential busses for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3, 4, 5, 6

EAL Threshold Values:

Loss of power to Transformers 142-1(242-1) and 142-2(242-2) for **> 15 minutes**.

Basis:

The Essential busses are the safety-related, ESF busses 141(241) and 142(242). Each Unit 1 (Unit 2), each 4160-V ESF bus is provided offsite power from the 345-kV system through System Auxiliary Transformer 142-1(242-1) directly to 4160-V ESF bus 141(241).

Each ESF bus for each unit is equipped with an onsite Emergency Diesel Generator; DG 1A(2A) for 4160-V ESF bus 141(241) and DG 1B(2B) for 4160-V ESF bus 142(242). Emergency Diesel Generators for the affected unit should automatically start and be available to carry the essential loads. Balance of plant systems that would assist in plant operations (e.g., condensate pumps, etc.) may be unavailable due to the loss of power.

A loss of offsite AC power reduces the required redundancy and potentially degrades the level of safety of the unit by rendering the station more vulnerable to a complete loss of AC power.

The intent of this EAL is to declare an Unusual Event when offsite power has been lost and the emergency diesel generators have successfully started and energized their respective ESF busses. The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis Reference(s):

1. NEI 99-01, Rev 4 SU1 & CU3
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BOA ELEC-3 Loss Of 4KV ESF Bus Unit 1/2
5. 1/2 BOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BCA-0.3 Response To Opposite Unit Loss Of All AC Power Unit 1/2
10. BOP AP-51, Isolating Unit 1 System AUX Transformer 142-1 & 142-2 While Unit Is At Power
11. BOP AP-52, Restoring Unit 1 System AUX Transformer 142-1 & 142-2 During Power Operation
12. BOP AP-53, Isolating Unit 2 System AUX Transformer 242-1 & 242-2 While Unit Is At Power
13. BOP AP-54, Restoring Unit 2 System AUX Transformer 242-1 & 242-2 During Power Operation

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA2

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability:

5, 6, D

EAL Threshold Values:

1. Loss of power to Transformers 142-1(242-1) and 142-2(242-2).

   AND

2. Failure of DG 1A(2A) and DG 1B(2B) emergency diesel generators to supply power to unit ESF busses.

   AND

3. Failure to restore power to at least one unit ESF bus within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power when in Cold Shutdown, Refueling or Defueled mode compromises safety systems required for decay heat removal and represents a substantial degradation of the level of safety of the plant. An Alert declaration (instead of a Site Area Emergency under EAL MS1) is appropriate in these modes because post-shutdown, decay heat energy levels offer more time to restore AC power to heat removal systems than the levels present when the reactor is in Power Operations, Startup, Hot Standby or Hot Shutdown mode. Thus, the threat to the protection of the health and safety of the public is less severe.

The ESF busses may be powered from any of the following onsite sources:

- Emergency Diesel Generators 1A(2A) for 4160-V ESF bus 141(241)
- Emergency Diesel Generators 1B(2B) for 4160-V ESF bus 142(242).

Offsite AC power sources feed the ESF busses through the System Auxiliary Transformers 142-1(242-1) and 142-2(242-2). The ESF busses of the affected unit can be powered from the unaffected unit through the crosstie breakers ACB 1414(2414) and ACB 1424(2424). Unit crosstie is considered an adequate source of offsite power when evaluating this EAL.

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

The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA2 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 CA3
2. 20E-0-4001 Station One Line Diagram
3. UFSAR 8.3.1
4. 1/2 BOA ELEC-3 Loss Of 4KV ESF Bus Unit 1/2
5. 1/2 BOA ELEC-4 Loss Of Offsite Power Unit 1/2
6. 1/2 BCA-0.0 Loss Of All AC Power Unit 1/2
7. 1/2 BCA-0.1 Loss Of All AC Power Recovery Without SI Required Unit 1/2
8. 1/2 BCA-0.2 Loss Of All AC Power Recovery With SI Required Unit 1/2
9. 1/2 BCA-0.3 Response To Opposite Unit Loss Of All AC Power Unit 1/2
10. BOP AP-51, Isolating Unit 1 System AUX Transformer 142-1 & 142-2 While Unit Is At Power
11. BOP AP-52, Restoring Unit 1 System AUX Transformer 142-1 & 142-2 During Power Operation
12. BOP AP-53, Isolating Unit 2 System AUX Transformer 242-1 & 242-2 While Unit Is At Power
13. BOP AP-54, Restoring Unit 2 System AUX Transformer 242-1 & 242-2 During Power Operation
14. Safety Evaluations of the Byron Station and Byron Station Responses to the Station Blackout (SBO) Rule (TAC NOS. 68522, 68523 AND 68515, 68516)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG3

Initiating Condition:

Failure of the Reactor Protection System to complete an automatic trip and manual trip was NOT successful and there is indication of an extreme challenge to the ability to cool the core.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. Automatic and manual Reactor Trip were not successful from Main Control Board as indicated by:
   a. Reactor power **≥ 5%**

      OR

   b. Intermediate Range Start Up Rate is positive

   AND

2. a. **Core Cooling CSF – RED Path** conditions exist.

      OR

   b. **Heat Sink CSF – RED Path** conditions exist.

Basis:

Automatic trip and manual trip are not considered successful if action away from the Main Control Board was required to trip the reactor.

This EAL is not applicable if a manual trip is initiated and no RPS setpoints are exceeded.

The reactor power level is the equivalent to the Safety System Design Heat Capacity. Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. Additionally, a continuing temperature rise indicates that this situation could be a precursor for a core melt sequence.

A successful trip has occurred when there is sufficient rod insertion to bring the reactor below Safety System Design Heat Capacity (less than 5%). Subcriticality Critical Safety Function (CSF) RED path is entered based on failure of power range indication to lower below 5% following a reactor trip.

Reactor power levels in the power range are indicated on N-41, 42, 43 and 44. This addresses any manual trip or automatic trip signal followed by a manual trip that fails to shut down the reactor to an extent that the reactor is producing more heat load for which the safety systems were designed.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG3 (cont.)

Basis (cont.):

A manual trip is any set of actions by the Reactor Operator(s) at the main control board which causes control rods to be rapidly inserted into the core and brings power below that percent power (5%) associated with the ability of the safety systems to remove heat. Automatic and manual trips are not considered successful if action away from the main control board is required to trip the reactor. Note that the operating mode changes to Hot Standby as soon as a successful reactor trip occurs.

Core Cooling Critical Safety Function RED path condition exists when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200° F. Entry into CSF Core Cooling RED requires entry into functional restoration procedure 1/2 BFR-C.1, Response to Inadequate Core Cooling. This condition indicates subcooling has been lost and that some fuel cladding damage may potentially occur.

The Heat Sink Critical Safety Function Red path condition exists if narrow range levels in all steam generators (S/Gs) are less than or equal to 10% - Unit 1 (31% adverse containment) and 14% - Unit 2 (34% adverse containment) and total feedwater flow to all S/Gs is less than or equal to 500 gpm. If total feed flow is less than 500 gpm due to procedurally directed operator actions then this condition does not apply.

The combination of these conditions (reactor power greater than or equal to 5% and either Core Cooling-RED path or Heat Sink-RED path) indicates the ultimate heat sink function is under extreme challenge. A major consideration is the inability to initially remove heat during the early stages of this sequence.

In the event this challenge occurs at a time when the reactor has not been brought below the power associated with safety system design power (5%), a core melt sequence may exist and rapid degradation of the fuel cladding could begin. To permit maximum offsite intervention time, the General Emergency declaration is therefore appropriate in anticipation of an inevitable General Emergency declaration due to loss and potential loss of fission product barriers.

Basis Reference(s):

1. NEI 99-01, Rev 4 SG2
2. 1/2 BST-1 Subcriticality Unit 1/2
3. 1/2 BST-2 Core Cooling Unit 1/2
4. 1/2 BST-3 Heat Sink Unit 1/2
5. 1/2 BFR-S.1 Response to Nuclear Power Generation/ATWS Unit 1/2
6. 1/2 BFR H.1 Response to Loss of Secondary Heat Sink Unit 1/2
7. 1/2 BFR C.1 Response to Inadequate Core Cooling Unit 1/2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS3

Initiating Condition:

Failure of Reactor Protection System to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded and manual trip was NOT successful.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

Automatic and manual Reactor Trip were not successful from Main Control Board as indicated by:

a. Reactor power **≥ 5%**

OR

b. Intermediate Range Start Up Rate is positive

Basis:

Automatic trip and manual trip are not considered successful if action away from the Main Control Board was required to trip the reactor.

This EAL is not applicable if a manual trip is initiated and no RPS setpoints are exceeded.

The reactor power level is the equivalent to the Safety System Design Heat Capacity. Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS.

For the intent of this EAL a successful trip has occurred when there is sufficient rod insertion to bring the reactor below Safety System Design Heat Capacity (less than 5%). Subcriticality Critical Safety Function (CSF) RED path condition is met based on failure of power range indication to lower below 5% following a reactor trip.

This addresses any automatic trip signal followed by a manual trip that fails to shut down the reactor to an extent that the reactor is producing more heat load for which the safety systems were designed. A manual trip is any set of actions by the reactor operator(s) at the main control board which causes control rods to be rapidly inserted into the core and brings power below that percent power (5%) associated with the ability of the safety systems to remove heat. Automatic and manual trips are not considered successful if action away from the main control board is required to trip the reactor.

MS3 (cont.)

Basis (cont.):

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. Emergency boration is thus required and there is an actual major failure of a system intended for protection of the public.

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Cladding and RCS barriers and warrants declaration of a Site Area Emergency.

Basis Reference(s):

1. NEI 99-01, Rev 4 SS2
2. 1/2 BST-1 Subcriticality Unit 1/2
3. 1/2 BFR-S.1 Response to Nuclear Power Generation/ATWS Unit 1/2
4. 1/2 BOSR 0.1-1,2,3 Unit One(Two) Mode 1, 2, & 3 Shiftly and Daily Operating Surveillance

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA3

Initiating Condition:

Failure of the Reactor Protection System to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. A Reactor Protection System setpoint was exceeded.

   AND

2. A successful automatic Reactor Trip did not occur

Basis:

This condition indicates a failure of the automatic reactor protection system to successfully 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 and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue.

In the event that the operator identifies a reactor trip is imminent and successfully initiates a manual reactor trip before the automatic trip setpoint is reached, no declaration is required.

Basis Reference(s):

1. NEI 99-01, Rev 4 SA2
2. 1/2 BST-1 Subcriticality Unit 1/2
3. 1/2 BFR-S.1 Response to Nuclear Power Generation/ATWS Unit 1/2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU3

Initiating Condition:

Inadvertent criticality.

Operating Mode Applicability:

3, 4, 5, 6

EAL Threshold Values:

An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

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

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

This EAL includes criticality events that occur in Cold Shutdown or Refueling modes (NUREG1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events as well as inadvertent criticalities occurring in Hot Shutdown mode. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification.

This condition can be identified using:

- Source Range startup rate channels 1/2 NI-31D (SR STARTUP RATE CH 31) or 1/2 NI-32D (SR STARTUP RATE CH 32)
- Intermediate Range channels 1/2 NI-35B and 1/2 NI-36B
- Nuclear Instrumentation System Source Range/Audio Count Rate Containment Indications
- Post Accident Monitoring System (Gamma Metrics) – Provides measure of flux level from shutdown (0.1 cps) through 200% power level

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU3

Basis Reference(s):

1. NEI 99-01, Rev 4 SU8 & CU8
2. 0/1/2 BOSR XZB-R1, Unit 0/1/2 Meter Zone Banding
3. 1/2 BFR-S.1, Response To Nuclear Power Generation/ATWS Unit 1/2
4. Technical Specifications LCO 3.3.1
5. 1/2 BOSR 0.1-4, Unit One(Two) Mode 4 Shiftly And Daily Operating Surveillance
6. 1/2 BGP 100-2 Plant Startup
7. 1/2 BGP 100-2T3 Reactor Startup Flowchart
8. 1/2 BGP 100-6T4 Core Alteration/Fuel Movement Checklist
9. Regulatory Guide 8.12, Criticality Accident Alarm Systems

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS4

Initiating Condition:

Loss of all vital DC power.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

Loss of all vital DC power based on **< 108 VDC** on 125 VDC battery busses 111(211) and 112(212) for **> 15 minutes**.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of Containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. Battery bus 111(211) or 112(212) are the safety-related, Class 1E 125 VDC power systems.

Each safety-related battery bus, 111(211) or 112(212), has the capacity to continuously supply all the connected normal running loads while maintaining its respective battery in a fully charged condition. Each battery has a guaranteed nominal rating of 2320 ampere-hours at the 8-hour rate to an end voltage of 1.75 volts per cell (or 1.75 VDC/cell x 58 cells = 101.5 VDC).

Each battery was sized based upon supplying the design duty cycle (1- hour overall duration) in the event of a loss of offsite AC power concurrent with a LOCA and a single failure of a diesel generator. If bus voltage drops to 123 VDC, a Control Room annunciator alarms (e.g., BAR 1-21-E10 for bus 111 and 113). Each battery is equipped with a battery charger that is rated to supply its associated DC loads while fully recharging the battery. Each battery charger is fed from a 480 VAC ESF switchgear bus of the same division. The minimum design voltage limit of each battery is 108 VDC. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate loads.

Basis Reference(s):

1. NEI 99-01, Rev 4 SS3
2. UFSAR 8.3.2.1.1
3. 6E-0-4001 Station One Line Diagram
4. BAR 1/2-21-E10 125V DC PNL 111/113 (211/213) VOLT LOW
5. BAR 1/2-22-E10 125V DC PNL 112/114 (212/214) VOLT LOW
6. 1/2 BOA ELEC - 1 Loss of DC Bus Unit 1/2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU4

Initiating Condition:

UNPLANNED loss of required DC power for greater than 15 minutes.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. UNPLANNED loss of all required vital DC power based on **< 108 VDC** indication on 125 VDC battery busses 111(211) and 112(212).

   AND

2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.

Basis:

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

“Unplanned activities” is included in this EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, plants perform maintenance on a bus-related basis during shutdown periods.

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown, refueling or defueled operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. Battery bus 111(211) or 112(212) are the safety-related, Class 1E 125 VDC power systems.

Each safety-related battery bus, 111(211) or 112(212), has the capacity to continuously supply all the connected normal running loads while maintaining its respective battery in a fully charged condition. Each battery has a guaranteed nominal rating of 2320 ampere-hours at the 8-hour rate to an end voltage of 1.75 volts per cell (or 1.75 VDC/cell x 58 cells = 101.5 VDC).

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU4 (cont.)

Basis: (cont.)

Each battery was sized based upon supplying the design duty cycle (1- hour overall duration) in the event of a loss of offsite AC power concurrent with a LOCA and a single failure of a diesel generator. If bus voltage drops to 123 VDC, a Control Room annunciator alarms (e.g., BAR 1-21-E10 for bus 111 and 113). Each battery is equipped with a battery charger that is rated to supply its associated DC loads while fully recharging the battery. Each battery charger is fed from a 480 VAC ESF switchgear bus of the same division. The minimum design voltage limit of each battery is 108 VDC. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate loads.

Basis Reference(s):

1. NEI 99-01, Rev 4 CU7
2. UFSAR 8.3.2.1.1
3. 6E-0-4001 Station One Line Diagram
4. 1/2 BOA ELEC -1 Loss of DC Bus Unit 1/2
5. BAR 1/2-21-E10 125V DC PNL 111/113 (211/213) VOLT LOW
6. BAR 1/2-22-E10 125V DC PNL 112/114 (212/214) VOLT LOW

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS5

Initiating Condition:

Complete loss of heat removal capability.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. **Core Cooling CSF - RED Path** conditions exist.

   AND

2. **Heat Sink CSF - RED Path** conditions exist.

Basis:

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature.

The Core Cooling Critical Safety Function RED path conditions exist when the average of the ten highest reading core exit thermocouples (CETCs) is greater than or equal to 1200° F. This condition indicates subcooling has been lost and that some fuel cladding damage may potentially occur.

The Heat Sink Critical Safety Function Red path conditions exist if narrow range levels in all steam generators (S/Gs) are less than or equal to 10% - Unit 1 (31% adverse containment) and 14% - Unit 2 (34% adverse containment) and total feedwater flow to all S/Gs is less than or equal to 500 gpm. If total feed flow is less than 500 gpm due to procedurally directed operator actions then this condition does not apply.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted.

Basis Reference(s):

1. NEI 99-01, Rev 4 SS4
2. 1/2 BST-2 Core Cooling Unit 1/2
3. 1/2 BST-3 Heat Sink Unit 1/2
4. 1/2 BFR C.1, Response to Inadequate Core Cooling Unit 1/2
5. 1/2 BFR H.1, Response to Loss of Secondary Heat Sink Unit 1/2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5

Initiating Condition:

Inability to maintain plant in Cold Shutdown with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F** for > Table M1 duration.

Table M1 – RCS Reheat Duration Thresholds		
RCS	**Containment Closure**	**Duration**
Intact	N/A	60 minutes*
Reduced Inventory (< 397 ft.)	Established	20 minutes*
	Not Established	0 minutes
Not Intact	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. UNPLANNED Reactor Vessel pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.

Basis:

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

CONTAINMENT CLOSURE: Containment Closure is considered to be as required by procedures.

Containment closure status is checked and verified using UNIT 1/2 - CONTAINMENT CLOSURE VERIFICATION CHECKLIST, BOP PC-1T1.

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 main steam line nozzle plugs, etc.).

Basis (cont.):

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as decay heat removal system design and Reactor Vessel water level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200° F).

Threshold #1 Basis:

The first condition in Table M1 addresses complete loss of functions required for core cooling for greater than sixty minutes during Refueling and Cold Shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals, etc.). With containment closure established, a low-pressure barrier to fission product release exists. In this condition, containment status is of less importance than the status of RCS integrity because the RCS is intact and providing a high-pressure barrier to fission product release. The sixty-minute interval should allow sufficient time to restore cooling without a substantial degradation in plant safety. The asterisk highlights the note at the bottom of the table. The note indicates that the first condition is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the sixty-minute interval.

The second condition in Table M1 addresses the complete loss of functions required for core cooling for greater than twenty minutes during Refueling and Cold Shutdown modes when containment closure is established but RCS integrity is not established or Reactor Vessel inventory is reduced. RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals, etc.). RCS inventory is in a reduced condition when water level is three feet below the Reactor Vessel flange (400 ft. el. - 3 ft. = 397 ft. el.).

The allowed twenty-minute interval is 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 barrier to fission product release is established (i.e., containment closure). The asterisk highlights the note at the bottom of the table. The note indicates that the second condition is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the twenty-minute interval.

MA5 (cont.)

Basis (cont.):

The third condition in Table M1 addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when containment closure and RCS integrity are not established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation. No delay time is allowed for this condition because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above 200° F when the heat removal function is available.

Threshold #2 Basis:

The 10 psig pressure rise due to loss of decay heat removal infers an intact RCS with uncontrolled RCS temperature rise in excess of the Technical Specification cold shutdown limit (200° F) for which MA5 Threshold #1 would permit up to sixty minutes to restore RCS cooling before declaration of an Alert. This EAL therefore covers situations of high decay heat loads, in which the event should be declared without delay.

NRC analyses show that sequences can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Basis Reference(s):

1. NEI 99-01, Rev 4 CA4
2. BOP PC-1 Containment Closure Tracking Capability
3. 1/2 BOSR 4.3.1-1 Reactor Coolant System Pressure Temperature Limit Surveillance
4. 1/2 BGP 100-1 Plant Heatup
5. 1/2 BGP 100-5, Plant Shutdown and Cool Down
6. 1/2 BGP 100-6, Unit 1(Unit 2) Refueling Outage

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU5

Initiating Condition:

UNPLANNED loss of decay heat removal capability with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F.**

   OR

2. Loss of all RCS temperature AND Reactor Vessel level indication for **> 15 minutes**.

Basis:

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

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, 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. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the fuel cladding may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel.

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS/Reactor Vessel temperatures depending on the time since shutdown.

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 Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling modes, the second condition of this EAL would result in declaration of an Unusual Event if temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU5 (cont.)

Basis (cont.):

Reactor Vessel water level is normally monitored using the following instruments:

- LT-046, LI-RY-046, Reactor Vessel Refueling Level Indicator
- LT-049, LI-RY-049, Reactor Vessel Refueling Level Indicator
- LT-048, LI-RY-048, Refueling Cavity Water Level
- RVLIS (LI-RC019, LI-RC020)
- LT-047 (LI-RY-047), Refueling Cavity Water Level 413 ft. el. to 426 ft. el.

LT-046 and LT-049 provide Reactor Vessel water level indication in the Control Room during RCS reduced inventory conditions. They are calibrated to indicate level from approximately 392 ft. el. to 402.5 ft. el. The Reactor Vessel flange is at 400 ft. el. LT-048 is calibrated to read refueling cavity water level from 392 ft. el. to the refuel floor level 426 ft. el. LT-047 provides refueling cavity water level in the range of 413 ft. el. to 426 ft. el.

Two independent methods of RCS level indication must be functional and monitored prior to draining below 402 ft. el. If any discrepancies greater than 0.5 ft between operable level indicators occur, draining is immediately secured until actual level has been verified. Two independent methods of RCS level indication must be continuously monitored when in a Reduced Inventory condition. Two independent RVLIS trains normally remain in service whenever the RCS is intentionally drained and the Reactor Vessel head is in place (except when in the process of removing/installing the head).

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200° F). These include:

- RCS Loop A Cold Leg Wide Range 1/2 TI-413B T0406 1TR-413A
- RCS Loop B Cold Leg Wide Range 1/2 TI-423B T0426 1TR-413B
- RCS Loop C Cold Leg Wide Range 1/2 TI-433B T0446 1TR-433A
- RCS Loop D Cold Leg Wide Range 1/2 TI-443B T0466 1TR-433B
- RCS Loop A Hot Leg Wide Range 1/2 TI-413A T0419 1TR-413A
- RCS Loop B Hot Leg Wide Range 1/2 TI-423A T0439 1TR-413B
- RCS Loop C Hot Leg Wide Range 1/2 TI-433A T0459 1TR-433A
- RCS Loop D Hot Leg Wide Range 1/2 TI-443A T0479 1TR-433B
- A RH Pump Discharge Temp. 1/2 TI-612 T0630 1TR-612
- B RH Pump Discharge Temp. 1/2 TI-613 T0631 1TE-613

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU5 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 CU4
2. Technical Specifications Table 1.1-1
3. 1/2 BOSR 0.1-6 Unit One(Two) Mode 6 Shiftly And Daily Operating Surveillance
4. BOP RH-9 Pump Down of the Refueling Cavity to the RWST
5. BOP RC-4 Reactor Coolant System Drain
6. 1/2 BOSR 3.3.1-1 Unit One(Two) Accident Monitoring Instrumentation Monthly Channel Checks
7. 1/2 BOL 4.15, LCOAR - RCS Leakage Detection Instrumentation
8. 1/2 BOL 4.13, LCOAR - RCS Operational Leakage
9. 1/2 BOSR 4.3.1-1 Reactor Coolant System Pressure / Temperature Limit Surveillance

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6

Initiating Condition:

Inability to monitor a SIGNIFICANT TRANSIENT in progress.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. Loss of most **(approximately 75%)** safety system annunciators (Table M2).

Table M2 – Control Room Panels
• 1/2 PM01J MCB Gen & Aux Power
• 1/2 PM05J MCB Reactor and Chem Volume Control
• 1/2 PM06J MCB Eng. Safety Features

AND

2. Indications needed to monitor safety functions (Table M3) are unavailable.

Table M3 – Safety Functions and Related Systems
• 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)
• Fission Product Barriers

AND

3. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Automatic Turbine Runback **> 25%** thermal reactor power
• Electrical load rejection **> 25%** full electrical load
• Reactor Trip
• Safety Injection Actuation
• Thermal power oscillations **> 10%**

AND

4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6 (cont.)

Basis:

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS and PPDS.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) automatic turbine 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%.

Planned and UNPLANNED actions are not differentiated since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not a factor.

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in 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.

A Site Area Emergency exists if the Control Room staff cannot monitor safety functions needed for protection of the public. Indications needed to monitor critical safety functions necessary for protection of the public must include Control Room indications, computer generated indications and dedicated annunciation capability. The specific parameters should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled and in a coolable geometry, remove heat from the core, and maintain the reactor coolant system and containment intact. These parameters are monitored and controlled in the emergency operating procedures.

Symptoms of a loss of annunciators can be:

- 1/2-4-A7, AN SYS PWR SUP TROUBLE
- 1/2-4-D7, AN SYS GROUND
- 1/2-4-E7, AN SYS FIELD CONTACT PWR TROUBLE
- 1/2-4-A6 COMPUTER TROUBLE
- 1/2-B7, SER TROUBLE

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 SS6
2. Drawing 20E-0-3372B Auxiliary Building Main Control Room Panel El 451'
3. BAP 300-1A1 At the Controls and Horse-Shoe Areas
4. BOP AN-1 Annunciator System Startup
5. UFSAR E.17
6. BAR 1/2-4-A7, AN SYS PWR SUP TROUBLE
7. BAR 1/2-4-D7, AN SYS GROUND
8. BAR 1/2-4-E7, AN SYS FIELD CONTACT PWR TROUBLE
9. BAR 1/2-4-A6, COMPUTER TROUBLE
10. BAR 1/2-4-B7, SER TROUBLE
11. 1/2 BOA Elect-7, Loss of Annunciators Unit 1/2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 1/2 PM01J MCB Gen & Aux Power • 1/2 PM05J MCB Reactor and Chem Volume Control • 1/2 PM06J MCB Eng. Safety Features

OR

b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

AND

2. a. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Automatic Turbine Runback **> 25%** thermal reactor power • Electrical load rejection **> 25%** full electrical load • Reactor Trip • Safety Injection Actuation • Thermal power oscillations **> 10%**

OR

b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6 (cont.)

Basis:

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

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) automatic turbine 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%.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS and PPDS.

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without Reactor Control, ECCS, Electrical panel, critical safety function, and process/area radiation annunciation or indication equipment. The availability of computer based indication equipment is considered.

Symptoms of a loss of annunciators can be:

- 1/2-4-A7, AN SYS PWR SUP TROUBLE
- 1/2-4-D7, AN SYS GROUND
- 1/2-4-E7, AN SYS FIELD CONTACT PWR TROUBLE
- 1/2-4-A6, COMPUTER TROUBLE
- 1/2-4-B7, SER TROUBLE

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification. The fifteen-minute interval offers time to recover from transient or momentary power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 SA4
2. Drawing 20E-0-3372B Auxiliary Building Main Control Room Panel El 451'
3. BAP 300-1A1 At the Controls and Horse-Shoe Areas
4. BOP AN-1 Annunciator System Startup
5. UFSAR E.17
6. BAR 1/2-4-A7, AN SYS PWR SUP TROUBLE
7. BAR 1/2-4-D7, AN SYS GROUND
8. BAR 1/2-4-E7, ANS SYS FIELD CONTACT PWR TROUBLE
9. BAR 1/2-4-A6, COMPUTER TROUBLE
10. BAR 1/2-4-B7, SER TROUBLE
11. 1/2 BOA Elect-7, Loss of Annunciators

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 1/2 PM01J MCB Gen & Aux Power
• 1/2 PM05J MCB Reactor and Chem Volume Control
• 1/2 PM06J MCB Eng. Safety Features

OR

2. UNPLANNED loss of most **(approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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)
• Fission Product Barriers

Basis:

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

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without Reactor Control, ECCS, Electrical panel, critical safety function, and process/area radiation annunciation or indication equipment. The availability of computer based indication equipment is considered.

MU6 (cont.)

Basis (cont.):

Symptoms of a loss of annunciators can be:

- 1/2-4-A7, AN SYS PWR SUP TROUBLE
- 1/2-4-D7, AN SYS GROUND
- 1/2-4-E7, AN SYS FIELD CONTACT PWR TROUBLE
- 1/2-4-A6, COMPUTER TROUBLE
- 1/2-4-B7, SER TROUBLE

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The fifteen-minute interval offers time to recover from transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev 4 SU3
2. Drawing 20E-0-3372B Auxiliary Building Main Control Room Panel El 451'
3. BAP 300-1A1 At the Controls Area
4. BOP AN-1 Annunciator System Startup
5. UFSAR E.17
6. BAR 1/2-4-A7, AN SYS PWR SUP TROUBLE
7. BAR 1/2-4-D7, AN SYS GROUND
8. BAR 1/2-4-E7, ANS SYS FIELD CONTACT PWR TROUBLE
9. BAR 1/2-4-A6, COMPUTER TROUBLE
10. BAR 1/2-4-B7, SER TROUBLE
11. 1/2 BOA Elect-7, Loss of Annunciators

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU7

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

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

   OR

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

Basis:

The conditions of this EAL may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. Positive indications in the Control Room of RCS leakage include one or more of the following:

- PZR pressure decreasing
- PZR level decreasing
- VCT level decreasing
- Charging flow greater than expected
- VCT makeup frequency increased
- RCP seal injection flow(s) abnormal
- PRT pressure, temperature, or level increasing
- Containment pressure, temperature, or humidity increasing
- Containment Recirculation sump level increasing
- Containment floor drain or Reactor cavity sump flow increasing
- Containment radiation levels increasing
- Auxiliary Building radiation levels increasing
- CNMT DRAIN LEAK DETECT FLOW HIGH alarm (1/2-1-1-A2)
- RX VESSEL FLNG LEAKOFF TEMP HIGH alarm (1/2-1-14-E5)

The 10 gpm value for the unidentified leakage and pressure boundary leakage was selected because it is quantifiable with normal Control Room leak detection methods. Station surveillance procedures identify indications to verify leakage from the RCS. A hand calculation or the Process Computer RCS Leakrate Code is used in determining RCS leakage.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont.):

The 25 gpm value for identified leakage is set at a higher value because of the significance of identified leakage in comparison to unidentified or pressure boundary leakage.

No classification under this threshold is made for relief valve operation where the relief valve functions as designed.

Basis Reference(s):

1. NEI 99-01, Rev 4 SU5
2. Technical Specifications 3.4.13 & 3.4.14
3. UFSAR. 6.2, 5.2
4. 1/2 BOSR 4.13.1-1 Unit One(Two) Reactor Coolant System Water Inventory Balance 72 Hour Surveillance
5. 1/2 BOL 4.15 LCOAR - RCS Leakage Detection Instrumentation
6. 1/2 BOL 4.13 LCOAR - RCS Operational Leakage
7. 1/2 BOA PRI-1 Excessive Primary Leakage Unit 1/2
8. 1/2 BOSR 0.1-4 Unit One(Two) Mode 4 Shiftly and Daily Operating Surveillance
9. 1/2 BOSR RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
10. 0/1/2 BOSR XZB-R1, Unit 0/1/2 Meter Zone Banding

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8

Initiating Condition:

Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. Loss of Reactor Vessel inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss

AND

2. a. RVLIS ≤ **0% Plenum (390 ft. el.)** for **> 30 minutes**.

   OR

   b. Reactor Vessel level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following:

   - 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high.
   - Erratic Source Range Monitor indication.

AND

3. Containment is challenged as indicated by one or more of the following:
   - Hydrogen concentration in Containment **≥ 5%.**
   - Containment pressure **≥ 50 psig.**
   - CONTAINMENT CLOSURE not established.

Basis:

<u>CONTAINMENT CLOSURE</u>: Containment Closure is considered to be as required by procedures.

Containment closure status is checked and verified using UNIT 1/2 - CONTAINMENT CLOSURE VERIFICATION CHECKLIST, BOP PC-1T1.

MG8 (cont.)

Basis (cont.):

The Reactor Vessel Level Indication System (RVLIS) provides the following eight level indicators:

Region	Sensor	Indication	Location
Head Region	1	31%	Near top of Reactor Vessel
	2	0%	Near upper internals support plate
Plenum Region	3	81%	400.7 ft. el. (halfway between upper internals support plate and top of the hot leg)
	4	55%	397.5 ft. el. (Reduced Inventory is 397 ft)
	5	37%	394.2 ft. (top of hot leg)
	6	27%	393 ft. (CL of hot leg)
	7	15%	392.4 ft. (bottom of hot leg)
	8	0%	390 ft. (top of core is 388 ft. 10 in.)

Available RVLIS level indication as close as possible to NEI 99-01 Guidance was chosen for the threshold levels.

Threshold #1 and #2 Basis:

When Reactor Vessel water level drops to RVLIS 0% Plenum (sensor #8), water level has reached 390 ft. el. and core uncovery is about to occur. The top of the core is at 388 ft 10 in. el. Fuel damage is probable if core uncovery is prolonged and submergence cannot be restored and maintained. Available decay heat will cause boiling and further drop Reactor Vessel water level.

This EAL is 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, (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, etc.) can have a significant affect on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

When Reactor Vessel water level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated containment radiation or unexplained rise in Containment sump levels. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors 1/2 NI-31B and 1/2 NI-32B can be used as a tool for making such determinations.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The Containment Fuel Handling Incident Radiation Monitors 1(2) RE-AR011 or 1(2) RE-AR012 indication of > 3000 mR/hr. is based on calculation EP-EAL-0501.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont.):

- 1/2 RE-AR011 Containment Fuel Handling Incident Monitor - RA0039 for Unit 1/ RA0064 for Unit 2
- 1/2 RE-AR012 Containment Fuel Handling Incident Monitor - RA0040 for Unit 1/ RA0065 for Unit 2

1/2 BOSR RF-1 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting when the Containment Floor Drain Sump annunciator setpoint is exceeded or 1/2 BOSR RF-1 surveillance has failed.

Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment sump level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area.

Threshold #3 Basis:

Three conditions are associated with the challenge to containment integrity:

- Containment closure is the action taken to secure containment and its assorted structures, systems and components as a functional barrier to fission product release under existing plant conditions. Containment closure is initiated per shift management direction if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal.
- If hydrogen concentration reaches or exceeds 5% in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside Containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred. Containment hydrogen concentration is indicated on 1/2 HSU-PS345 and 1/2 HSU-PS346.
- The containment design pressure (50 psig) is well in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both RCS and fuel clad barriers in that it is not possible to reach this condition without severe core degradation.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis Reference(s):

1. NEI 99-01 Rev 4, CG1
2. BOP PC-1, Containment Closure Tracking Capability
3. UFSAR E.17, 6.2, 6.2.5.2.1
4. BOP RH-9 Pump Down of the Refueling Cavity to the RWST
5. BOP RC-4 Reactor Coolant System Drain
6. 1/2 BOSR 0.1-4 Unit One(Two) Mode 4 Shiftly and Daily Operating Surveillance
7. 1/2 BOSR RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
8. 0/1/2 BOSR XZB-R1, Unit 0/1/2 Meter Zone Banding
9. 1/2 BGP 100-2 Plant Startup
10. BGP 100-6T4 Core Alteration / Fuel Movement Checklist
11. 1/2 BOSR 3.3.1-1 Unit One(Two) Accident Monitoring Instrumentation Monthly Channel Checks
12. 1/2 BFR-C.1, Response to Inadequate Core Cooling Unit 1/2
13. 1/2 BST-5 Containment Unit 1/2
14. NES-G-14.02, Calculation No. BYR99-010 / BRW-99-0017-I
15. EP-EAL-0501, Estimation Of Radiation Monitor Readings Indicating Core Uncovery During Refueling

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS8

Initiating Condition:

Loss of Reactor Vessel inventory affecting core decay heat removal capability.

Operating Mode Applicability:

5

EAL Threshold Values:

1. **Without** CONTAINMENT CLOSURE established:
    a. Reactor Vessel inventory as indicated by RVLIS **≤ 15% Plenum (392.4 ft. el.).**

    OR

    b. Reactor Vessel level unknown for **> 30 minutes** with a loss of Reactor Vessel inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss

OR

2. **With** CONTAINMENT CLOSURE established:
    a. Reactor Vessel inventory as indicated by RVLIS **≤ 0% Plenum (390 ft. el.)**

    OR

    b. Reactor Vessel level unknown for **> 30 minutes** with a loss of Reactor Vessel inventory as evidenced by either of the following:
        - Per Table M5 indications.
        - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be as required by procedures.

Containment closure status is checked and verified using UNIT 1/2 - CONTAINMENT CLOSURE VERIFICATION CHECKLIST, BOP PC-1T1.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS8 (cont.)

Basis (cont.):

If a low-pressure boundary to fission product release does not exist (i.e., containment closure is not established), the level associated with this threshold is six inches below the bottom inside diameter of the RCS hot leg vessel penetration (i.e., 392.4 ft. el. – 0.5 ft. = 391.9 ft. el.). If containment closure is established, a low-pressure boundary to fission product release exists and water level can drop to the top of active fuel, 388.83 ft. el. before a Site Area Emergency declaration is required. The Reactor Vessel Level Indication System (RVLIS) provides the following eight level indicators:

Region	Sensor	Indication	Location
Head Region	1	31%	Near top of Reactor Vessel
	2	0%	Near upper internals support plate
Plenum Region	3	81%	400.7 ft. el. (halfway between upper internals support plate and top of the hot leg)
	4	55%	397.5 ft. el. (Reduced Inventory is 397 ft. el.)
	5	37%	394.2 ft. el. (top of hot leg)
	6	27%	393 ft. el. (CL of hot leg)
	7	15%	392.4 ft. el. (bottom of hot leg)
	8	0%	390 ft. el. (top of core is 388 ft. 10 in.)

Available RVLIS level indication as close as possible to NEI 99-01 Guidance was chosen for the threshold levels.

Threshold #1 Basis:

Since there is no direct indication in the Control Room of a level 6 inches below the loop (391.9 ft. el.) RVLIS Sensor # 7 (15% - 392.4 ft. el. bottom of hot leg) was chosen.

If the Reactor Vessel Refueling Level Indicators LT-046 (LI-RY-046) and LT-049 (LI-RY-049) are available they would provide water level indication in the Control Room during RCS reduced inventory conditions and are calibrated to read level to 392 ft. el. When readings from these instruments drop off-scale low, the water level threshold with containment closure not established has been reached. RVLIS trend, local indication and perhaps visual observation may also support making this determination.

The thirty-minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont.):

Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment sump level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area.

Threshold #2 Basis:

Since there is no direct indication in the Control Room of top of active fuel (388 ft. el.) RVLIS Sensor # 8 (390 ft. el.) was chosen. When Reactor Vessel water level drops to RVLIS 0% Plenum (sensor #8), water level has reached 390 ft. el. and core uncovery is about to occur. The inability to restore and maintain water level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Cladding barrier. The closest indication of core uncovery is RVLIS 0% Plenum sensor at 390 ft. el.

When Reactor Vessel water level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated containment radiation or unexplained rise in Containment sump levels. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors 1/2 NI-31B and 1/2 NI-32B can be used as a tool for making such determinations.

The thirty-minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

Basis Reference(s):

1. NEI 99-01, Rev 4 CS1
2. UFSAR E.17, 6.2
3. BOP RH-9 Pump Down of the Refueling Cavity to the RWST
4. BOP RC-4 Reactor Coolant System Drain
5. 1/2 BOSR 0.1-4 Unit One (Two) Mode 4 Shiftly and Daily Operating Surveillance
6. 1/2 BOSR RF-1 Unit One (Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
7. 0/1/2 BOSR XZB-R1, Unit 0/1/2 Meter Zone Banding
8. 1/2 BGP 100-2 Plant Startup
9. BGP 100-6T4 Core Alteration / Fuel Movement Checklist
10. 1/2 BOSR 3.3.1-1 Unit One (Two) Accident Monitoring Instrumentation Monthly Channel Checks

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8

Initiating Condition:

Loss of RCS / Reactor Vessel inventory with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

5, 6

EAL Threshold Values:

1. a. Loss of RCS / Reactor Vessel inventory as indicated by RVLIS **≤ 27% Plenum (393 ft. el.)**.

   OR

   b. Loss of RCS / Reactor Vessel inventory as indicated by LT-046 and LT-049 **< 393 ft. el.**

OR

2. a. Loss of RCS / Reactor Vessel inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss

   AND

   b. RCS / Reactor Vessel level unknown for **> 15 minutes.**

Basis:

The threshold value for Reactor Vessel inventory is bottom of RCS Loop Hot Leg (392.4 ft. el.) However because available level indications systems do not provide for an accurate way to positively identify six inches below the loop level, RVLIS sensor # 7 is used in EAL MS8 as the threshold value. Therefore sensor #6 is used for as the threshold value for MA7 to maintain a sequential classification scheme.

When Reactor Vessel water level drops to 393 ft. el., the centerline of the RCS hot leg is uncovered. This EAL serves as a precursor 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 Reactor Vessel water level drop and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8 (cont.)

Basis (cont.):

In Cold Shutdown mode, the decay heat available to raise Reactor Vessel temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and water level monitoring means are available. In the Refueling mode, the RCS is not intact and water level and inventory are monitored by different means. In the Refueling mode, normal means of water level indication may not be available. Redundant means of water 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.

Reactor Vessel water level is normally monitored using the following instruments:

- LT-046, LI-RY-046, Reactor Vessel Refueling Level Indicator
- LT-049, LI-RY-049, Reactor Vessel Refueling Level Indicator
- LT-048, LI-RY-048, Refueling Cavity Water Level
- RVLIS (LI-RC019, LI-RC020)
- LT-047 (LI-RY-047), Refueling Cavity Water Level 413 ft. el. to 426 ft. el.

LT-046 and LT-049 provide Reactor Vessel water level indication in the Control Room during RCS reduced inventory conditions. They are calibrated to indicate level from approximately 392 ft. el. to 402.5 ft. el. LT-048 is calibrated to read refueling cavity water level from 392 ft. el. to the refuel floor level 426 ft. el. LT-047 provides refueling cavity water level in the range of 413 ft. el. to 426 ft. el.

Two independent methods of RCS level indication must be functional and monitored prior to draining below 402 ft. el. If any discrepancies greater than 0.5 ft. between operable level indicators occur, draining is immediately secured until actual level has been verified. Two independent methods of RCS level indication must be continuously monitored when in a Reduced Inventory condition. Two independent RVLIS trains normally remain in service whenever the RCS is intentionally drained and the Reactor Vessel head is in place (except when in the process of removing/installing the head).

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8 (cont.)

Basis (cont.):

The Reactor Vessel Level Indication System (RVLIS) provides the following eight level indicators:

Region	Sensor	Indication	Location
Head Region	1	31%	Near top of Reactor Vessel
	2	0%	Near upper internals support plate
Plenum Region	3	81%	400.7 ft. el. (halfway between upper internals support plate and top of the hot leg)
	4	55%	397.5 ft. el. (Reduced Inventory is 397 ft)
	5	37%	394.2 ft. (top of hot leg)
	6	27%	393 ft. (CL of hot leg)
	7	15%	392.4 ft. (bottom of hot leg)
	8	0%	390 ft. (top of core is 388 ft. 10 in.)

Available RVLIS level indication as close as possible to NEI 99-01 Guidance was chosen for the threshold levels.

In the second condition of this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by sump level changes. 1/2 BOSR RF-1 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting when the Containment Floor Drain Sump annunciator setpoint is exceeded or 1/2 BOSR RF-1 surveillance has failed. The Containment Floor Drain Sump level indicated range is 0 to 106 in. The RCDT level indicated range is 0% to 100%.

Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment water level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area. The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency duration.

Basis Reference(s):

1. NEI 99-01, Rev 4 CA1 & CA2
2. UFSAR 6.2 & E.17
3. 1/2 BOSR 0.1-4 Unit One(Two) Mode 4 Shiftly and Daily Operating Surveillance
4. 1/2 BOSR RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
5. 0/1/2 BOSR XZB-R1, Unit 0/1/2 Meter Zone Banding

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8 (cont.)

Basis References (cont.):

6. BOP RH-9 Pump Down of the Refueling Cavity to the RWST
7. BOP RC-4 Reactor Coolant System Drain

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU8

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

5

EAL Threshold Values:

1. Pressurizer level established limit **> 5% Cold Cal** and RCS level **cannot** be restored and maintained **> 5% Cold Cal.**

   OR
2. Pressurizer level established limit **< 5% Cold Cal** and RCS level **cannot** be restored and maintained **> procedurally established limit**.

Basis:

The inability to restore and maintain level after reaching these established limits infers a degradation of the level of safety at the plant.

RCS level in the Cold Shutdown mode is controlled within limits that are established by procedures in effect for the present conditions. The use of Pressurizer level is normally appropriate for the Cold Shutdown mode; however, there are Cold Shutdown mode evolutions that require RCS level to be lowered below the range of the Pressurizer level instrumentation. These evolutions are directed by procedures that require precise control and monitoring of RCS levels that include establishment of low level limits. Examples of such evolutions include draining down to vessel flange level to prepare for reactor head flange bolt de-tensioning, and draining to mid-loop for equipment maintenance. During these evolutions it is appropriate to use the low level limit established by the procedure in effect to determine if RCS leakage is occurring and emergency declaration is required.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU1
2. UFSAR 6.2 & E.17
3. 1/2 BOSR 0.1-4 Unit One(Two) Mode 4 Shiftly and Daily Operating Surveillance
4. 1/2 BOS RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
5. 0/1/2 BOSR XZB-R1, Unit 0/1/2 Meter Zone Banding
6. BOP RH-9 Pump Down of the Refueling Cavity to the RWST
7. BOP RC-4 Reactor Coolant System Drain

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS9

Initiating Condition:

Loss of Reactor Vessel inventory affecting core decay heat removal capability with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

6

EAL Threshold Values:

1. **Without** CONTAINMENT CLOSURE established:
    a. Reactor Vessel Refueling Level Indicators LT-046 and LT-049 **< 393 ft. el.**

    OR

    b. Reactor Vessel level unknown with indication of core uncovery as evidenced by one or more of the following:
    - 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high.
    - Erratic Source Range Monitor indication.

OR

2. **With** CONTAINMENT CLOSURE established:
    a. Reactor Vessel Refueling Level Indicators LT-046 and LT-049 **= 392 ft. el.** or off scale low.

    OR

    b. Reactor Vessel level unknown with indication of core uncovery as evidenced by one or more of the following:
    - 1/2 RE-AR011 or 1/2 RE-AR012 Containment Fuel Handling Incident radiation monitors **> 3000 mR/hr** or off-scale high.
    - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be as required by procedures.

Containment closure status is checked and verified using UNIT 1/2 - CONTAINMENT CLOSURE VERIFICATION CHECKLIST, BOP PC-1T1.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont.):

Threshold #1 and #2 Basis:

Under the refueling conditions specified in this threshold, prolonged loss of the ability to monitor Reactor Vessel water level in conjunction with indirect indications of inventory loss infer a continued drop in water level and loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the vessel.

In the Cold Shutdown mode, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. The heatup and the threat to damaging the fuel cladding thus may be higher for events that occur in the Cold Shutdown mode than for events in the Refueling mode. The elevated RCS heatup rate accelerates boil-off and loss of RCS inventory. When in the Cold Shutdown mode, a Site Area Emergency declaration is therefore associated simply with the decreasing inventory trend rather than indications of actual core uncovery. Note that the heatup threat can be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling. In the Cold Shutdown mode, when Reactor Vessel water level indication is unavailable, the inventory loss must be detected by containment sump level changes or erratic Source Range indication.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to core shine, scattering and radiation bounce off of the solid surfaces in the area will result in readings on the Containment Fuel Handling Incident Radiation Monitors 1/2 RE-AR011 or 1/2 RE-AR012 of > 3000 mR/hr. This threshold radiation value is based on calculations documented in EP-EAL-0501.

- 1/2 RE-AR011 Containment Fuel Handling Incident Monitor - RA0039 for Unit 1/ RA0064 for Unit 2
- 1/2 RE-AR012 Containment Fuel Handling Incident Monitor - RA0040 for Unit 1/ RA0065 for Unit 2

1/2 BOSR RF-1 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting when the Containment Floor Drain Sump annunciator setpoint is exceeded or 1/2 BOSR RF-1 surveillance has failed.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont.):

Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment sump level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area

Basis Reference(s):

1. NEI 99-01, Rev 4 CS2
2. 1/2 BOP PC-1 Containment Closure Tracking Capability
3. UFSAR E.17, 6.2
4. BOP RH-9 Pump Down of the Refueling Cavity to the RWST
5. BOP RC-4 Reactor Coolant System Drain
6. 1/2 BOSR 0.1-4 Unit One(Two) Mode 4 Shiftly and Daily Operating Surveillance
7. 1/2 BOSR RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
8. 0/1/2 BOSR XZB-R1, Unit 0/1/2 Meter Zone Banding
9. EP-EAL-0501, Estimation Of Radiation Monitor Readings Indicating Core Uncovery During Refueling

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU9

Initiating Condition:

UNPLANNED loss of RCS inventory with irradiated fuel in the Reactor Vessel.

Operating Mode Applicability:

6

EAL Threshold Values:

1. UNPLANNED RCS level drop below the Reactor Vessel flange (400 ft. el.) for **≥ 15 minutes**.

   OR

2. a. Loss of Reactor Vessel inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained Containment Sump level rise • Unexplained Auxiliary Bldg. Sump level rise • Unexplained Tank level rise • Unexplained rise in RCS makeup • Observation of leakage or inventory loss

   AND

   b. Reactor Vessel level unknown.

Basis:

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

Threshold #1 Basis:

The Reactor Vessel flange is at 400 ft. el. Reactor Vessel water level is normally monitored using the following instruments:

- LT-046, LI-RY-046, Reactor Vessel Refueling Level Indicator
- LT-049, LI-RY-049, Reactor Vessel Refueling Level Indicator
- LT-048, LI-RY-048, Refueling Cavity Water Level
- LT-047 (LI-RY-047), Refueling Cavity Water Level 413 ft. el. to 426 ft. el.

LT-046 and LT-049 provide Reactor Vessel water level indication in the Control Room during RCS reduced inventory conditions. They are calibrated to indicate level from approximately 392 ft. el. to 402.5 ft. el. LT-048 is calibrated to read refueling cavity water level from 392 ft. el. to the refuel floor level 426 ft. el. LT-047 provides refueling cavity water level in the range of 413 ft. el. to 426 ft. el.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont.):

Two independent methods of RCS level indication must be functional and monitored prior to draining below 402 ft. el. If any discrepancies greater than 0.5 ft. between operable level indicators occur, draining is immediately secured until actual level has been verified. Two independent methods of RCS level indication must be continuously monitored when in a Reduced Inventory condition.

This threshold is applicable only in the Refueling mode and addresses loss of inventory to below the Reactor Vessel flange during refueling operations. Refueling operations that drop water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An Unusual Event is appropriate because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant.

The fifteen-minute interval provides a reasonable time frame to restore level using one or more of the redundant means of refill should be available. If RCS water level cannot be restored in this interval, a more serious condition may exist.

Threshold # 2 Basis:

In the second condition of this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by sump level changes. 1/2 BOSR RF-1 provides instructions for calculating primary system leak rate by water inventory balances for off normal events and for operations troubleshooting when the Containment Floor Drain Sump annunciator setpoint is exceeded or 1/2 BOSR RF-1 surveillance has failed.

Containment Sump Flow recorder on 1/2 PM12J displays Containment Floor Drain Sump Flow (0-15 gpm) and Containment Equipment Flow (0-15 gpm) with alarms at 1 gpm and 6 gpm, respectively. Containment level 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. Sump level rise in the Auxiliary Building could also be caused by a leak of Reactor Coolant System water into the area

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU9 (cont.)

Basis Reference(s):

1. NEI 99-01, Rev 4 CU2
2. UFSAR 5.2, 6.2
3. 1/2 BOL 4.15 LCOAR - RCS Leakage Detection Instrumentation
4. 1/2 BOA PRI-1 Excessive Primary Plant Leakage Unit 1/2
5. 1/2 BOSR 0.1-6 Unit One(Two) Mode 6 Shiftly and Daily Operating Surveillance
6. 1/2 BOSR RF-1 Unit One(Two) Containment Floor Drain Monitoring System Non-Routine Surveillance
7. BOP RH-9 Pump Down of the Refueling Cavity to the RWST
8. BOP RC-4 Reactor Coolant System Drain

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU10

Initiating Condition:

UNPLANNED loss of all onsite or offsite communications capabilities.

Operating Mode Applicability:

1, 2, 3, 4, 5, 6

EAL Threshold Values:

1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations.

   OR

2. Loss of all Table M6 **Offsite** communications capability.

Table M6 - Communications Capability		
System	**Onsite**	**Offsite**
Radios	X	
Plant page	X	
Plant Telephone System	X	
Commercial Telephones		X
NARS		X
ENS		X
HPN		X
Cellular Phones		X
TSO/PJM (Electric Operations)		X
Satellite Phones		X

Basis:

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

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems with offsite authorities or personnel. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

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

Basis Reference(s):

1. NEI 99-01, Rev 4 SU6 & CU6
2. EP-MW-124-1001 Facilities Inventories And Equipment Tests

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU11

Initiating Condition:

Inability to reach required shutdown within Technical Specification limits.

Operating Mode Applicability:

1, 2, 3, 4

EAL Threshold Values:

Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a prescribed shutdown mode when the Technical Specification configuration cannot be restored. The plant is within its safety envelope when being shut down 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 completion period elapses under Technical Specifications and is not related to how long a condition may have existed.

Basis Reference(s):

1. NEI 99-01, Rev 4 SU2
2. Byron Technical Specifications

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1

Initiating Condition:

Security event resulting in loss of physical control of the facility.

Operating Mode Applicability:

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

EAL Threshold Value:

A HOSTILE FORCE has taken control of:

1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1).

Table H1 - Safety Functions and Related Systems
• 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) • Fission Product Barriers

OR

2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).

Basis:

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.

Threshold #1 Basis

This threshold encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions. As a result, equipment control cannot be transferred to and operated from another location.

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

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.

Threshold #2 Basis

This threshold addresses loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely because there is freshly off-loaded fuel in the pool. The condition “freshly off-loaded reactor fuel in pool” equates to fuel off-loaded within the last 120 days in NF-AA-310 Special Nuclear Material And Core Component Movement.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1 (cont.)

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HG1
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. 0BOA Security-1, Security Threat Unit 0
4. 1/2 BOA PRI-5 Control Room Inaccessibility Unit 1/2
5. SY-AA-101-132, Threat Assessment
6. Station Security Plan – Appendix C
7. NF-AA-310, Special Nuclear Material And Core Component Movement

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1

Initiating Condition:

Site attack.

Operating Mode Applicability:

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

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert ICs (HA1 and HA2) in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA.

Although Nuclear Power Plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for Offsite Response Organizations (ORO) 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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1 (cont.)

Basis (cont.):

This EAL is intended to address the potential for a very rapid progression of events due to a dedicated attack. It is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs. HOSTILE ACTION identified above encompasses various acts including:

- Air attack (LARGE AIRCRAFT impacting the PROTECTED AREA)
- Land-based attack (HOSTILE FORCE penetrating PROTECTED AREA)
- Waterborne attack (HOSTILE FORCE on water penetrating PROTECTED AREA)
- BOMBs breeching the PROTECTED AREA.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs.

This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for ORO to be notified and to activate in order to be better prepared to respond should protective actions become necessary. If not previously notified by NRC that the LARGE AIRCRAFT impact 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.

LARGE AIRCRAFT is meant to be an 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.

This EAL addresses the immediacy of a threat to impact site VITAL AREAS within a relatively short time. The fact that the site is under serious attack with minimal time available for additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HS4
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. 0BOA Security-1, Security Threat Unit 0

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition:

Notification of an airborne attack threat.

Operating Mode Applicability:

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

EAL Threshold Values:

A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LARGE AIRCRAFT is meant to be an 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.

The intent of this EAL is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations (ORO) and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Alert. This EAL is met when a plant receives information regarding a LARGE AIRCRAFT attack threat from NRC and the LARGE AIRCRAFT is less than 30 minutes away from the plant.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from such an attack. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for ORO to be notified and encouraged to activate (if they do not normally) to be better prepared should it be necessary to consider further actions.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA7
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. 0BOA Security-1, Security Threat Unit 0

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

Initiating Condition:

Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

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

EAL Threshold Values:

1. A credible site-specific security threat notification as determined per SY-AA-101-132, “Threat Assessment”.

   OR

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

Basis:

Threshold #1 Basis

The intent of this threshold is to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat.

The determination of “credible” is made through use of information found in the Station Security Plan or SY-AA-101-132, “Threat Assessment” procedure.

Threshold #2 Basis

The intent of this threshold is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Unusual Event. This threshold is met when a plant receives information regarding an aircraft threat from NRC. Should the threat involve a LARGE AIRCRAFT (LARGE AIRCRAFT is meant to be an aircraft with the potential for causing significant damage to the plant), then escalation to Alert via HA1 would be appropriate if the LARGE AIRCRAFT is less than 30 minutes away from the plant. The status and size of the plane may be provided by NORAD through the NRC. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NRC Safeguards Advisory 10/6/01
6. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
7. 0BOA Security-1, Security Threat Unit 0

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

Initiating Condition:

Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA.

Operating Mode Applicability:

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

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This EAL is intended to address the potential for a very rapid progression of events due to an attack including:

- Air attack (LARGE AIRCRAFT impacting the OCA)
- Land-based attack (HOSTILE FORCE progressing across licensee property or directing projectiles at the site)
- Waterborne attack (HOSTILE FORCE on water attempting forced entry or directing projectiles at the site)
- BOMBs

This EAL is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2 (cont.)

Basis (cont.):

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne terrorist attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for Offsite Response Organizations to be notified and to activate in order to be better prepared to respond should protective actions become necessary.

If not previously notified by NRC that the LARGE AIRCRAFT impact 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. LARGE AIRCRAFT is meant to be an 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.

This IC/EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time. The fact that the site is an identified attack candidate with minimal time available for further preparation requires a heightened state of readiness and implementation of protective measures that can be effective (onsite evacuation, dispersal or sheltering) before arrival or impact.

Basis Reference(s):

1 NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA8
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NRC Safeguards Advisory 10/6/01
6. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
7. 0BOA Security-1, Security Threat Unit 0

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS3

Initiating Condition:

Confirmed security event in a plant VITAL AREA.

Operating Mode Applicability:

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

EAL Threshold Value:

Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC (HA3).

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a Station's security. Only those events that involve Actual or Likely Major failures of plant functions needed for protection of the public need to be considered. The following events would not normally meet this requirement; (e.g., Failure by a Member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or STRIKE ACTIONS).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Station Security Plan.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HS1
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NRC Safeguards Advisory 10/6/01
6. 0BOA Security-1, Security Threat Unit 0

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

Initiating Condition:

Confirmed security event in a plant PROTECTED AREA.

Operating Mode Applicability:

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

EAL Threshold Value:

Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Unusual Event.

Multi-unit stations with shared safety functions should further consider how this IC may affect more than one unit and how this may be a factor in escalating the emergency class.

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a station’s security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered. The following events would not normally meet this requirement; (e.g., failure by a member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or STRIKE ACTIONS).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA4
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NRC Safeguards Advisory 10/6/01
6. 0BOA Security-1, Security Threat Unit 0

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU3

Initiating Condition:

Confirmed security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

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

EAL Threshold Value:

Notification by the Security Force of a security event as determined from Station Security Plan - Appendix C.

Basis:

Reference is made to Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

This threshold is based on Station Security Plan – Appendix C. 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.

Consideration should be given to the following types of events when evaluating an event against the criteria of the Station Security Plan: CIVIL DISTURBANCE, and STRIKE ACTION.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NRC Safeguards Advisory 10/6/01
6. 0BOA Security-1, Security Threat Unit 0

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS4

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 Threshold Values:

1. Control Room evacuation has been initiated.

   AND

2. Control of the plant **cannot** be established per 1/2 BOA PRI-5 Control Room Inaccessibility procedure in **< 15 minutes**.

Basis:

The 15 minute time period starts when either:

a. Control of the plant is no longer maintained in the Main Control Room.

   OR

b. The last Operator has left the Main Control Room.

The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. 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 determination of whether or not control is established outside of the Main Control Room is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the site-specific time for transfer that the licensee has control of the plant. Transfer of control to locations outside the Control Room is considered established when the Shift Manager has determined that the operators are capable of controlling reactivity, core cooling and heat sink functions.

Basis Reference(s):

1. NEI 99-01, Rev 4 HS2
2. 1/2 BOA PRI-5, Control Room Inaccessibility Unit 1/2

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

Initiating Condition:

Control Room evacuation has been initiated.

Operating Mode Applicability:

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

EAL Threshold Values:

Entry into 1/2 BOA PRI-5, Control Room Inaccessibility procedure for Control Room evacuation.

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency operations centers are necessary. Procedure 1/2 BOA PRI-5 Control Room Inaccessibility specifies conditions under which Control Room evacuation may be necessary.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA5
2. 1/2 BOA PRI-5, Control Room Inaccessibility Unit 1/2

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5

Initiating Condition:

Natural and destructive phenomena affecting the plant VITAL AREA.

Operating Mode Applicability:

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

EAL Threshold Values:

1. a. Seismic event **> Operating Basis Earthquake (OBE)** as indicated by seismic check on 0PA02J.

   AND

   b. Confirmed by **EITHER:**

   - Earthquake felt in plant.
   - National Earthquake Center.

   OR

2. Tornado or high winds **> 85 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

   OR

3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

   OR

4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • Essential Service Water Cooling Towers • Condensate Storage Tanks • RWSTs

OR

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5 (cont.)

EAL Threshold Values: (cont.)

5. Uncontrolled flooding that results in **EITHER**:
   a. Degraded safety system performance in the Auxiliary Building as indicated in the Control Room.

   OR

   b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.

Basis:

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

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

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

Threshold #1 Basis:

This threshold addresses events that may have resulted in a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this threshold to assess the actual magnitude of the damage.

This threshold is based on seismic ground acceleration in excess of 0.09 g for the UFSAR Operating Basis Earthquake (OBE). Seismic events of this magnitude ~ 4 times greater that the Unusual Event threshold of EAL HU5 can cause damage to plant safety functions.

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

Threshold #2 Basis:

This threshold addresses events that may have resulted in a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The Alert classification is appropriate if visible damage is observed and relevant plant parameters indicate that the performance of safety systems in these areas has been degraded.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5 (cont.)

Basis (cont.):

No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

The 85 mph threshold is the UFSAR design basis wind speed. Station Category I structures are designed to withstand wind loads which may exist if sustained wind speeds reach or exceed 85 mph. Wind loads in excess of this magnitude can cause damage to safety functions. The Condensate Storage Tanks are not UFSAR Category I structures, but their safety significance warrants their inclusion in this EAL.

Threshold #3 Basis:

This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes, or impact of projectiles into a Table H2 area. This threshold addresses vehicle crashes that challenge the operability of systems necessary for safe shutdown of the plant. Table H2 areas include Category 1 structures and those Category 2 structures that contain Category 1 Systems and components.

The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Table H2 areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

Threshold #4 Basis:

This threshold covers threats to safety related equipment imposed by missiles generated by failure of the main turbine. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.

Threshold #5 Basis:

This threshold addresses the effect of internal flooding that has resulted in degraded performance of safety systems or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5 (cont.)

Basis (cont.):

“Uncontrolled” as used in this threshold describes a condition where water is entering an area from an unplanned evolution. This flooding may have been caused by internal events such as component failures, equipment misalignment, and fire suppression system actuation or outage activity mishaps. Water entering an area, which resulted in degraded performance of safety systems within the area due to wetting or submergence, would meet the intent of this threshold. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding'. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if indications of degraded system performance is available or a shock hazard is known to exist.

The Auxiliary Building has been identified as a potential Internal Flooding Area because it is an area containing systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Basis Reference(s):

1. NEI 99-01, Rev 4 HA1
2. UFSAR Section 2.5.4.9.3, 3.02, 3.3.1.1, and 3.04, Appendix C
3. 0BOA-ENV-4, Earthquake Unit 0
4. Annunciator 0-38-E5 Accelograph Accel High
5. Drawing S-01A Composite Site Plan
6. 1/2 BOA TG-7, Main Generator Excessive Hydrogen Leakage Unit 1/2

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition:

Natural and destructive phenomena affecting the PROTECTED AREA.

Operating Mode Applicability:

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

EAL Threshold Values:

1. a. Seismic event as indicated by Annunciator 0-38-E5, Accelograph Accel High (0PM01J).

   AND

   b. Confirmed by **EITHER:**

   - Earthquake felt in plant.
   - National Earthquake Center.

2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 85 mph**, within PROTECTED AREA boundary.

   OR

3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • Essential Service Water Cooling Towers • Condensate Storage Tanks • RWSTs

   OR

4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

   OR

5. Uncontrolled flooding in Auxiliary Building that has the potential to affect safety related equipment needed for the current operating mode.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont.)

Basis:

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

Threshold #1 Basis:

This threshold is based on the strong-motion seismograph actuation level which is the sensed earthquake threshold of 0.02 g. Seismic events of this magnitude are ~1/4 of the Alert event threshold (OBE) of EAL HA5 in which it is assumed the earthquake can cause damage to plant safety functions.

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

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. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01 g.”

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

Threshold #2 Basis:

This threshold is based on the assumption that a tornado striking (touching down) or sustained high winds (> 85 mph) within the PROTECTED AREA boundary may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The Protected Area boundary is within the security isolation zone and is defined in the Byron Station Security Plan – Appendix C. Verification of a tornado is obtained by direct observation and reporting by station personnel. “Sustained” wind speeds exist for 15 minutes or longer. Wind speed is obtained from meteorological data in the Control Room.

Threshold #3 Basis:

In this context, a "vehicle crash" is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont.)

Basis (cont.):

Threshold #4 Basis:

This threshold is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this threshold to classify minor operational leakage.

Threshold #5 Basis:

“Uncontrolled” as used in this threshold describes a condition where water is entering an area from an unplanned evolution. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if a potential to affect safety related equipment needed for the current operating mode exists.

This threshold addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, fire suppression system actuation or outage activity mishaps. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding.” The Auxiliary Building has been identified as an Internal Flooding Area of concern for the Unusual Event declaration because it is an area having the potential to affect safety related equipment needed for the current operating mode including:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Basis Reference(s):

1. NEI 99-01, Rev 4 HU1
2. UFSAR Section 2.5.4.9.3, 3.02, 3.3.1.1
3. 0BOA-ENV-4 Earthquake Unit 0
4. Annunciator 0-38-E5 Accelograph Accel High
5. Drawing S-01A Composite Site Plan
6. 1/2 BOA TG-1 Turbine High Vibration, Eccentricity or Differential Expansion Unit 1/2
7. 1/2 BOA TG-7 Main Generator Excessive Hydrogen Leakage Unit 1/2
8. BAR 1/2 PL01J-1-A2 Hydrogen Pressure High or Low

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6

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 Threshold Values:

1. FIRE or EXPLOSION in any Table H2 area.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • Essential Service Water Cooling Towers • Condensate Storage Tanks • RWSTs

AND

2. a. Affected safety system parameter indications show degraded performance.

   OR

   b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.

Basis:

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

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

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6 (cont.)

Basis (cont.):

The areas listed in Table H2 house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Personnel access to these areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses FIRES and EXPLOSIONS that challenge the operability of systems necessary for safe shutdown of the plant.

The only FIRES and EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment required for safe shutdown. Visual observation of damage infers the ability to approach or enter the affected areas. Lacking the ability to adequately inspect the area for damage, the Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this EAL. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

A steam line break or steam explosion that damages permanent structures or equipment in one of these areas would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA2
2. Drawing S-01A Composite Site Plan
3. UFSAR Section 3.02

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6

Initiating Condition:

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

Operating Mode Applicability:

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

EAL Threshold Values:

1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

   OR

2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • Essential Service Water Cooling Towers • Condensate Storage Tanks • RWSTs

   OR

3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

Basis:

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

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

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6 (cont.)

Basis (cont.):

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

Thresholds #1 and #2 Basis:

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins with a credible notification that a fire is occurring or indication of a VALID fire detection system alarm. A verified alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm.

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 EAL HA7. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under EAL HA7.

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

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket, etc.). Such fires are excluded from consideration in this threshold since they have no safety consequence.

Threshold #3 Basis:

The only EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment in the PROTECTED AREA.

A steam line break or steam explosion that damages permanent structures or equipment in a PROTECTED AREA would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU2
2. Drawing S-01A Composite Site Plan
3. UFSAR Section 3.02
4. BAP-1100, Fire Protection Procedure Series

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA7

Initiating Condition:

Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown.

Operating Mode Applicability:

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

EAL Threshold Values:

1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

   OR

2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Main Steam Tunnels • Essential Service Water Cooling Towers • Condensate Storage Tanks • RWSTs

Basis:

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

Values for LFL for common gases at Byron Station are:

- Propane – 2.2% (BOC Gasses MSDS)
- Hydrogen – 4.0% (Air Liquide Safety Data Sheet)
- Acetylene – 2.2% (BOC Gasses MSDS)

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Basis (cont.)

This EAL is based on toxic, asphyxiant, or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Toxic or flammable gases detected outside of these areas need not be considered for this EAL unless there is a spread of the gasses into one of these areas.

Threshold #1:

Declaration should not be delayed for confirmation from atmospheric testing if it is reasonable to conclude that the IDLH concentrations have been met (e.g., documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards).

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 this EAL. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under this EAL. The first condition is met if measurement of toxic gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a Table H2 area. Non-Toxic Gases which displace oxygen (site examples; Halon or Nitrogen) to a life threatening level due to asphyxiation (oxygen deprivation) should also be considered for this EAL.

An Asphyxiant is a material 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.

Threshold #2:

The second condition is met when the flammable gas concentration in a Table H2 area exceeds the LOWER FLAMMABILITY LIMIT. Flammable gases such as hydrogen and acetylene are routinely used to maintain plant systems (hydrogen – main generator cooling, reactor coolant chemistry control) or repair equipment/components (acetylene - welding). This condition addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a Table H2 area has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage or personnel injury. Once it has been determined that an uncontrolled release of flammable gas is occurring, sampling must be done to determine if the gas concentration exceeds the LOWER FLAMMABILITY LIMIT.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA3
2. Drawing S-01A Composite Site Plan
3. UFSAR Section 3.02

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU7

Initiating Condition:

Release of toxic or flammable gases deemed detrimental to normal operation of the plant.

Operating Mode Applicability:

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

EAL Threshold Values:

1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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:

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.

This EAL is based on the existence of uncontrolled releases of toxic, asphyxiant, or flammable gas affecting plant operations or the health of plant personnel. The release may have originated within the PROTECTED AREA boundary, or it may have originated offsite and subsequently drifted inside the PROTECTED AREA boundary. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on NORMAL PLANT OPERATIONS.

It is intended that releases of toxic, asphyxiant, or flammable gases are of sufficient quantity and the release point of such gases is such that safe plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for safe plant operation. The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect safe plant operations or plant personnel safety.

An Asphyxiant is a material 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.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU3
2. Drawing S-01A Composite Site Plan

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG8

Initiating Condition:

Other conditions existing 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 Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels (> 1 Rem TEDE or > 5 Rem CDE Thyroid) outside the site boundary.

Basis Reference(s):

1. NEI 99-01, Rev 4 HG2
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS8

Initiating Condition:

Other conditions existing 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 Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

Basis Reference(s):

1. NEI 99-01, Rev 4 HS3
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA8

Initiating Condition:

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

Operating Mode Applicability:

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

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA6
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU8

Initiating Condition:

Other conditions existing 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 Threshold Values:

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:

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency operating procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU5
2. NEI Industry White Paper - Enhancements to Emergency Preparedness Programs For Hostile Action May 2005 (Revised November 18, 2005)

Section 4: Emergency Measures

Exelon Nuclear emergency response actions are the same for all nuclear stations and are thus covered by Section E of the Emergency Plan.

4.1 Notification of the Emergency Organization

Standard NARS notifications for the Byron Station are made to the State of Illinois Emergency Management Agency (IEMA). If a General Emergency is the initiating event, the Emergency Director is also responsible for notifying the following offsite agencies:

- Ogle County Sheriff

4.2 Assessment Actions

Throughout each emergency situation, continuing assessment will occur. Assessment actions at Byron Station may include an evaluation of plant conditions; inplant, onsite, and initial offsite radiological measurements; and initial estimates of offsite doses. Core damage information is used to refine dose assessments and confirm or extend initial protective action recommendations. Byron Station utilizes WCAP-14696-A, Revision 1, (1999) as the basis for the methodology for post-accident core damage assessment. This methodology utilizes real-time plant indications. In addition, Byron Station may use samples of plant fluids and atmospheres as inputs to the CDAM (Core Damage Assessment Methodology) program for core damage estimation.

4.3 Protective Actions for the Offsite Public

Protective actions concerning the public within the 10 mile EPZ involve prompt notification, evacuation and sheltering. Prompt notification involves the use of the permanently installed outdoor notification sirens located within the EPZ.

To aid Control Room personnel during a rapidly developing emergency situation, Figure 4-1 "Byron Station (PAR) Determination Flowchart" has been developed based on Section J.10.m of the Exelon Nuclear Standardized Emergency Plan.

4.3.1 Alert and Notification System (ANS) Sirens

The ANS consists of a permanently installed outdoor notification system within a ten mile radius around the station. The ten mile radius around the station is primarily an agricultural area with a population density below 2000 persons per square mile. The ANS as installed consists of mechanical and electronic sirens that will cover this entire area with a minimum sound level of 60 db. Additionally, the ANS will cover the heavily populated areas within the ten mile radius around the station with a minimum sound level of 70 db to ensure complete coverage.

Once the public has tuned to designated radio stations in an emergency, detailed instructional messages will be given to the public. State and local procedures provide for these messages.

4.3.2 Evacuation Time Estimates

The evacuation time estimates were developed per the requirements of NUREG-0654, and to support the Illinois Plan For Radiological Accidents (IPRA) - Byron Station Volume VI. The purpose of the evacuation time estimates is to assess the postulated evacuation times for the Byron Station Emergency Planning Zone (EPZ).

The evacuation time estimate data was updated per a study performed by Earth Tech. Inc. documented in their report dated December, 2003 entitled “Evacuation Time Estimates for the Byron Station Plume Exposure Pathway Emergency Planning Zone.”

The evacuation times are based on a detailed consideration of the EPZ roadway network and population distribution. The information in Table 4-1 presents representative evacuation times for daytime and nighttime scenarios, for summer and winter seasons, and under various weather conditions for the evacuation of various areas around the Byron Station, once a decision has been made to evacuate. The evacuation times noted include notification, mobilization, and travel time. These times are for the general population which include permanent population and special facilities (schools, nursing homes, hospitals, and recreational areas). Table 4-2 provides information on the scenario population distribution (by Subarea) that was used for this study. Table 4-3 provides a representation of the Subarea Locations in relation to the EPZ.

4.4 Protective Actions for Onsite Personnel

Byron Station has a siren system to warn personnel of emergency conditions. Upon hearing a continuous two (2) minute siren, all personnel not having emergency assignments have been instructed to assemble in predesignated assembly areas. Refer to Figure 4-2. Station ERO personnel report to the Technical Support Center and Control Room personnel report to the Main Control Room. Radiation Protection, Chemistry and Operations personnel not assigned to the Main Control Room report to the Operational Support Center.

If a site evacuation of non-essential personnel is required by Section J of the Emergency Plan, personnel will be either relocated and monitored at the relocation centers or sent home if there is no release or radiological/safety concerns. The designated relocation centers for Byron Station are:

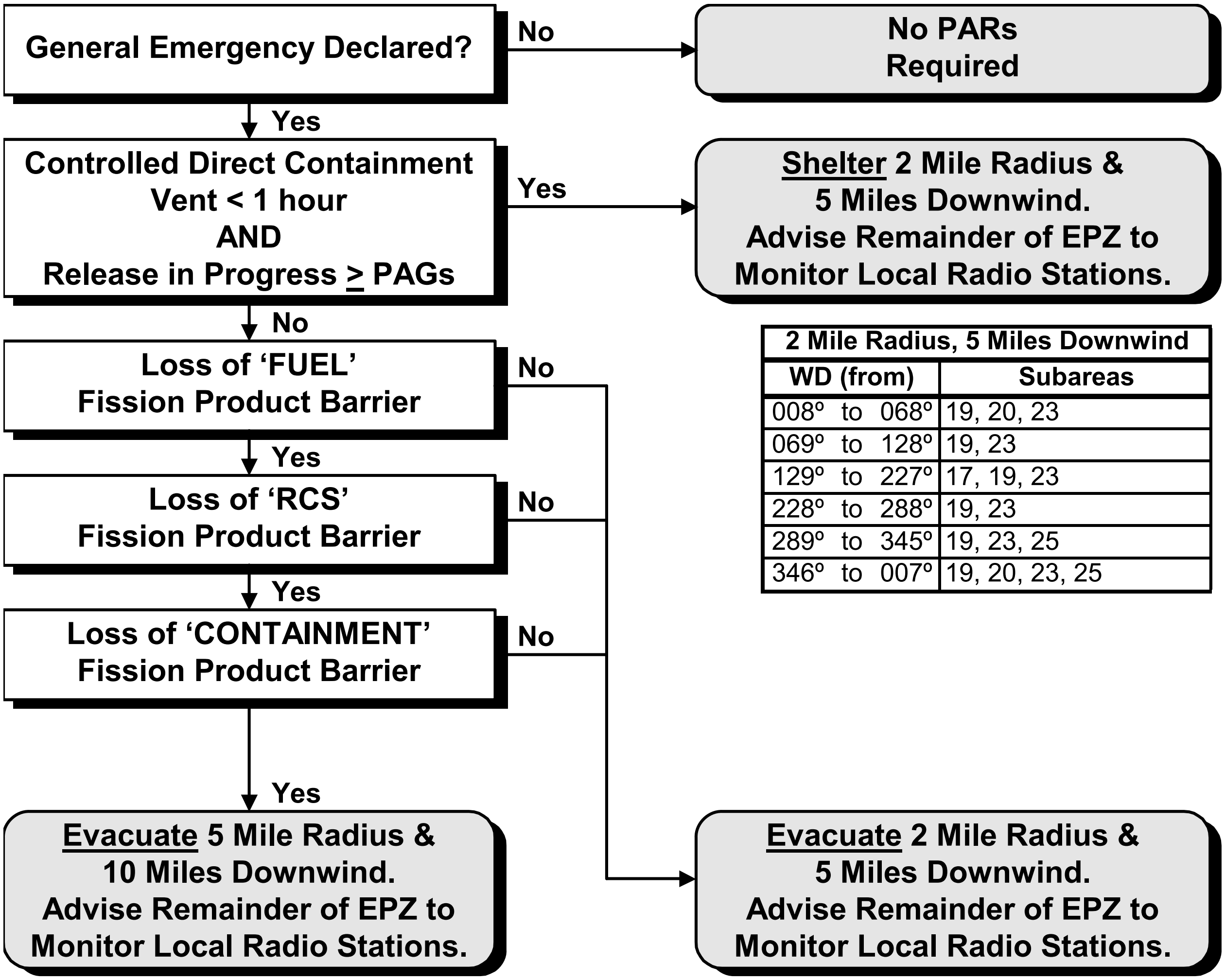
- Morrison Relocation Center, Morrison, Illinois
- Quad Cities Station, Cordova, Illinois

For evacuation routes, refer to EP-AA-113-F-18.

Traffic control for onsite areas will be handled by Byron Station personnel, if necessary.

Equipment and personnel would be available at the Morrison Relocation Center and Quad Cities Station for monitoring and decontamination of evacuated personnel. If major decontamination, follow-up, or bioassay samples are necessary, those persons would be sent to Quad Cities Station.

Figure 4-1: Byron Station PAR Determination Flowchart

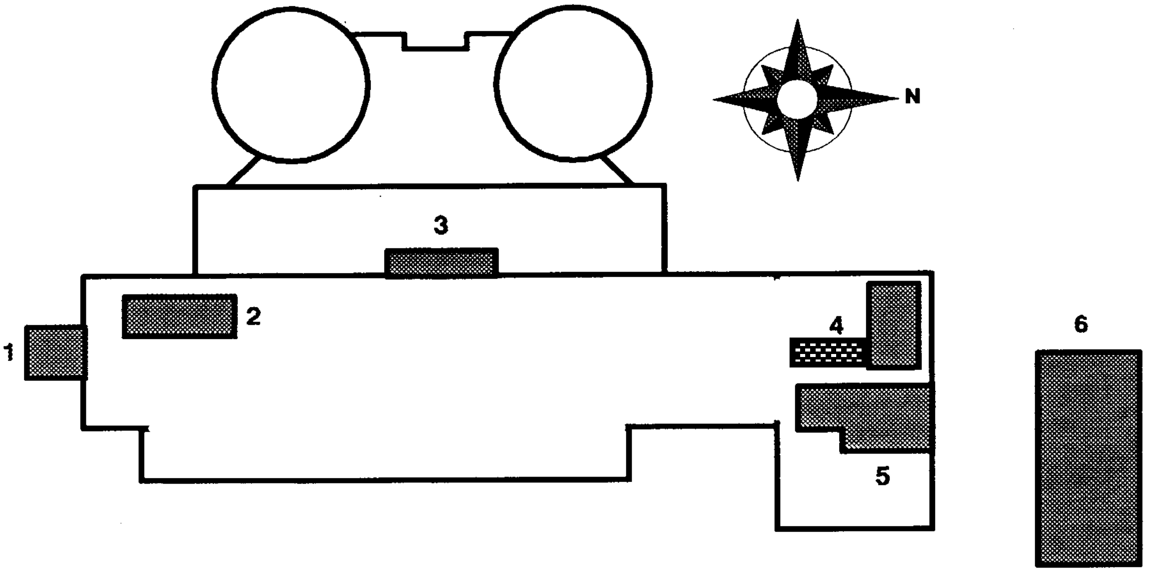


2 Mile Radius, 5 Miles Downwind	
WD (from)	**Subareas**
008° to 068°	19, 20, 23
069° to 128°	19, 23
129° to 227°	17, 19, 23
228° to 288°	19, 23
289° to 345°	19, 23, 25
346° to 007°	19, 20, 23, 25

5 Mile Radius, 10 Miles Downwind	
WD (from)	**Subareas**
022° to 051°	14, 17, 19, 20, 23, 25
052° to 082°	12, 14, 17, 19, 20, 23, 25
083° to 100°	12, 17, 19, 20, 23, 25
101° to 125°	10, 12, 17, 19, 20, 23, 25
126° to 139°	10, 17, 19, 20, 23, 25
140° to 171°	10, 17, 19, 20, 23, 25, 40
172° to 209°	17, 19, 20, 23, 25, 40
210° to 220°	17, 19, 20, 23, 25, 39, 40
221° to 248°	17, 19, 20, 23, 25, 27, 39
249° to 264°	17, 19, 20, 23, 25, 27
265° to 285°	17, 19, 20, 23, 25, 27, 28
286° to 326°	17, 19, 20, 23, 25, 28
327° to 021°	17, 19, 20, 23, 25

2 Mile Radius, 5 Miles Downwind	
WD (from)	**Subareas**
008° to 068°	19, 20, 23
069° to 128°	19, 23
129° to 227°	17, 19, 23
228° to 288°	19, 23
289° to 345°	19, 23, 25
346° to 007°	19, 20, 23, 25

Figure 4-2: Byron Station Assembly Areas and Onsite Emergency Response Facilities



ASSEMBLY AREAS ▨

1. TECHNICAL SUPPORT CENTER (TSC) EL 451'
2. UNIT ONE TURBINE BLDG. TRACKWAY EL 401'
3. CONTROL ROOM EL 451'
4. OPERATIONAL SUPPORT CENTER (OSC) EL 451'
5. MAINTENANCE SHOP EL 401'
6. SECURITY GATEHOUSE EL 401'

Table 4-1: Byron - Summary of General Public Evacuation Time Estimates (1)

	Summer				Winter			
	Daytime		Nighttime		Daytime		Nighttime	
PAR Evacuation Zone (2, 3, 4)	**Fair**	**Adverse**	**Fair**	**Adverse**	**Fair**	**Adverse**	**Fair**	**Adverse**
2 mile radius & 5 miles downwind								
WD 008 to 068 [19, 20, 23]	180	220	130	145	180	225	130	145
WD 069 to 128 [19, 23]	170	210	125	135	170	210	125	135
WD 129 to 227 [17, 19, 23]	180	220	130	145	180	225	130	145
WD 228 to 288 [19, 23]	170	210	125	135	170	210	125	135
WD 289 to 345 [19, 23, 25]	170	210	125	135	170	210	125	135
WD 346 to 007 [19, 20, 23, 25]	180	220	130	145	180	225	125	145
5 mile radius & 10 miles downwind								
WD 022 to 051 [5R, 14]	185	225	135	155	185	230	135	160
WD 052 to 082 [5R, 12, 14]	185	225	135	155	185	230	135	160
WD 083 to 100 [5R, 12]	185	225	135	155	185	230	135	160
WD 101 to 125 [5R, 10, 12]	185	225	135	155	185	230	135	160
WD 126 to 139 [5R, 10]	185	225	135	150	185	230	135	155
WD 140 to 171 [5R, 10, 40]	185	225	135	150	185	230	135	155
WD 172 to 209 [5R, 40]	185	220	130	145	185	225	130	150
WD 210 to 220 [5R, 39, 40]	185	225	135	150	185	230	135	155
WD 221 to 248 [5R, 27, 39]	185	225	135	150	185	230	135	155
WD 249 to 264 [5R, 27]	185	220	130	145	185	225	130	145
WD 265 to 285 [5R, 27, 28]	185	220	130	145	185	225	130	145
WD 286 to 326 [5R, 28]	185	220	130	145	185	225	130	145
WD 327 to 021 [5R]	185	220	130	145	185	225	130	145
Entire 10-mile EPZ	185	225	135	155	185	230	135	160

(1) Times are rounded to the nearest 5 minutes

(2) Subareas in brackets. See Table 4.3 for Subarea locations. PAR evacuation zones per EP-AA-111

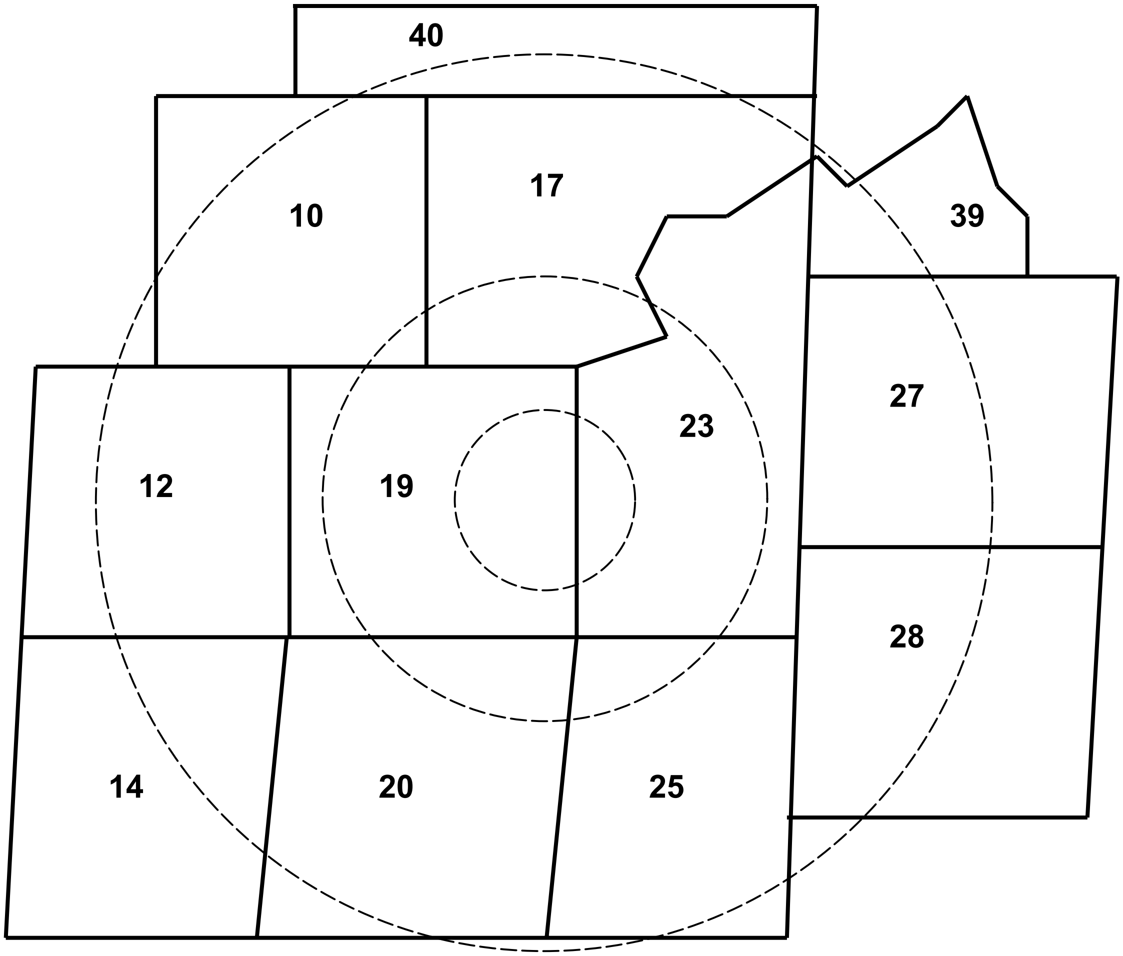
(3) "5R" designates all Subareas within 5-mile radius (Subareas 17, 19, 20, 23, 25)

(4) WD is the direction (in degrees) from which the wind is blowing (00 or 360 represents a wind from north to south)

Table 4-2: Byron Scenario Population Distribution By Subarea

	Summer				Winter			
	Daytime		Nighttime		Daytime		Nighttime	
Subarea	Population	Vehicles	Population	Vehicles	Population	Vehicles	Population	Vehicles
10	1,260	481	1,260	481	1,453	505	1,260	481
12	5,403	2,701	4,518	1,876	6,075	2,737	4,518	1,876
14	3,590	1,302	2,876	973	1,237	529	1,095	429
17	8,226	3,356	7,406	2,836	8,950	3,155	6,362	2,524
19	4,417	2,174	3,197	1,258	3,108	1,665	2,192	901
20	7,915	3,558	6,555	2,653	8,831	3,432	6,217	2,470
23	7,030	2,792	4,412	1,641	5,510	1,715	3,804	1,432
25	1,129	434	1,054	404	1,069	420	979	374
27	1,697	664	1,671	638	1,697	664	1,671	638
28	1,099	385	1,094	380	878	293	709	271
39	616	235	616	235	616	235	616	235
40	508	194	508	194	508	194	508	194
EPZ total	42,890	18,274	35,167	13,567	39,932	15,542	29,931	11,823

Table 4-3: Byron Subarea Locations



Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities

Refer to Figure 4-2 for the location of the Byron Station Control Room, Technical Support Center (TSC), and Operations Support Center (OSC) within the Station's Protected Area boundary.

5.1.1 Station Control Room

The Byron Station Control Room is the initial onsite center of emergency control and is located on the 451' elevation of the Auxiliary Building.

5.1.2 Technical Support Center (TSC)

Byron Station has designated a TSC which exists at the south end of the Turbine Building. The TSC fully meets the requirements of Section H.1.b of the Exelon Nuclear Standardized Emergency Plan.

5.1.3 Operational Support Center (OSC)

Byron Station has designated a primary Operational Support Center. The primary OSC is the Response Center and Meeting Room #1 on the 451' elevation of the Service Building. The OSC conforms to the requirements of Section H.1.c of the Emergency Plan and is the location to which operations support personnel will report during an emergency and from which they will be dispatched for assignments in support of emergency operations.

Assessment Resources

5.2.1 Onsite Meteorological Monitoring Instrumentation

A 250-foot meteorological tower has been erected on the site approximately 3400 feet southwest of the Byron Station reactor building, the major plant structure closest to the tower.

Wind speed, wind direction and temperature are measured at 30 feet and 250 feet above grade level. Temperature difference is determined between the 30-foot and 250-foot levels. A precipitation gauge is utilized to measure rain and snowfall at ground level near the base of the tower.

The onsite meteorological monitoring program is covered in the contract specification and vendor procedures of the meteorological monitoring contractor. These data are used to generate wind roses and to provide estimates of airborne concentrations of gaseous effluents.

5.2.1.1 Instrumentation

The meteorological tower is instrumented with equipment that conforms with the recommendations of Regulatory Guide 1.23 and ANSI/ANS 2.5 (1984). The equipment is placed on booms oriented into the generally prevailing wind at the site. Equipment signals are brought to an instrument shack with controlled environmental conditions. The shack at the base of the tower houses the recording equipment, signal conditioners, etc., used to process and re-transmit the data to the end point users.

5.2.1.2 Meteorological Measurement Program During a Disaster

Cooperation between the corporate office and the meteorological contract assures that a timely restoration of any outage can be made. Emergency field visits to the site are made as quickly as possible after detection of a failure.

Should a disaster of sufficient magnitude occur to destroy the tower structure, a contract is maintained to have a temporary tower erected within 72 hours, weather conditions permitting. Further, the meteorological contractor maintains two levels of sensors (wind speed, wind direction and temperature) in a state of readiness for use on the temporary tower.

Additionally, Exelon Nuclear's existing instrumented towers at other nuclear sites provide a high density measurement network with multiple backup opportunities.

Meteorological data are available to the station Control Room, Technical Support Center, and Emergency Operations Facility for use in the Dose Projection Computer Model for estimating the environmental impact of unplanned releases of radioactivity from the station.

5.2.2 Onsite Radiation Monitoring Equipment

The onsite radiation monitoring capability includes an installed process, effluent, and area radiation monitoring system; post accident sampling capability; portable survey instrumentation; counting equipment for radiochemical analysis; and a personnel dosimetry program to record integrated exposure. Some onsite equipment is particularly valuable for accident situations and is described in the following subsections.

5.2.2.1 Radiation Monitoring System

Chapters 11 and 12 of the Byron UFSAR describe the radiation monitoring system (RMS) in detail. The installed RMS is designed to continuously monitor the containment atmosphere, plant effluents, and various inplant locations.

The system includes Control Room readouts and recorders for selected parameters that are monitored and an audible Control Room alarm when predetermined setpoints are exceeded. The system can be subdivided into process/effluent instrumentation and an area monitoring system.

- The process/effluent instrumentation consists of pumps, filter samplers, detectors, and associated electronics to determine noble gas, iodine, and particulate concentrations in plant cubicles or liquid and gaseous effluents. Several monitored effluent pathways have control functions which will terminate the release at a predetermined setpoint. These setpoints are premised on compliance with federal regulations.
- The area monitoring system provides information of existing radiation levels in various areas of the plant to ensure safe occupancy. It is equipped with Control Room and local readout and audible alarms to warn personnel of an increased radiation level.

Some onsite equipment is particularly valuable for accident situations and is described in the following sections.

5.2.2.2 Radiological Noble Gas Effluent Monitoring

Two General Atomic Company wide-range gas monitors (WRGM) are installed for sampling the auxiliary building vent stacks which are the final release points for gaseous effluents. The monitors have a range of 1 x 10-7 uCi/cc to 1 x 105 uCi/cc. Each monitor includes the following:

One isokinetic nozzle located in the vent stack, a sample conditioning skid to filter out radioiodine and particulate activity, the wide range gas detector assembly including three gas detectors of the low, mid, and high ranges, two sample pumps (high flow used in the low range mode and low flow used for the mid/high range), and an auxiliary pump skid which boosts flow when using the mid/high range of the WRGM.

The system also includes a microprocessor which utilizes digital processing techniques to analyze data and control monitor functions. Readouts are available in the control room with associated audible alarms. Two General Atomic Company RD-12 detectors are provided for each of the four main steamlines upstream of the safety and relief valves. The range of the monitor is 10^{-1} mR/hr to 10^{4} mR/hr. The monitors will be mounted external to the main steamline piping and corrections made for the loss of low energy gammas.

5.2.2.3 Radioiodine and Particulate Effluent Monitoring

The General Atomic Company wide range gas monitor includes a sampling rack for collection of the auxiliary building vent stack particulate and radioiodine samples. Filter holders and valves are provided to allow grab sample collection for isotopic analyses in the station's counting rooms. The sampling rack is shielded to minimize personnel exposure. The sampling media will be analyzed by a gamma ray spectrometer which utilizes a gamma spectrometer system detector. In addition, silver zeolite cartridges are available to further reduce the interference of noble gases.

5.2.2.4 High-Range Containment Radiation Monitors

Two high range containment radiation monitors are installed for each operating reactor. The monitors will detect and measure the radiation level within the reactor containment during and following an accident. The range of the monitors is 1 rad/hr to 10^7 Rads/hr.

5.2.2.5 In-plant Iodine Instrumentation

Effective monitoring of increasing iodine levels in buildings under accident conditions will include the use of portable instruments using silver zeolite as a sample media. Auxiliary counting room locations have been identified within the Turbine Building. It is expected that a sample can be obtained and analyzed for iodine content within a two-hour time frame.

5.2.3 Onsite Process Monitors

An adequate monitoring capability exists to properly assess the plant status for all modes of operation and is described in the Byron UFSAR. The operability of the post-accident instrumentation ensures information is available on selected plant parameters to monitor and assess important variables following an accident. Instrumentation is available to monitor the parameters and ranges given in Technical Specifications. Byron Station Emergency Operating Procedures aid personnel in recognizing inadequate core cooling using applicable instrumentation.

5.2.4 Onsite Fire Detection Instrumentation

The fire detection system is designed in accordance with applicable National Fire Protection Association (NFPA) Standards. The system is equipped with electrically supervised ionization smoke and heat detectors to quickly detect any fires and the instrumentation to provide local indication and Control Room annunciation. In addition to the smoke and heat detection systems, each fire protection carbon dioxide, halon, or water system is instrumented to inform the Control Room of its actuation or of system trouble. In the event that a portion of the fire detection instrumentation is inoperable, fire watches in affected areas may be required.

5.2.5 Facilities and Equipment for Offsite Monitoring

Consult Chapter 11 of the station specific Offsite Dose Calculation Manual (ODCM) for the most current location for fixed continuous air samplers and TLD locations. Byron Station maintains a supply of emergency equipment and supplies for offsite monitoring and sampling by environmental field teams.

5.2.6 Site Hydrological Characteristics

The hydrological characteristics of the Byron Station vicinity are described in the Byron UFSAR. The river screen house is the only structure that could be affected by flooding on the Rock River and is designed for a combined event flood, where a combined event flood is defined as a flood on the Rock River having a 1×10^{-6} annual probability of being exceeded at a 90% confidence level. All other Byron Station structures are 161 feet or more above the Probable Maximum Flood level of the Rock River.

The minimum design operating level of the essential service water makeup pumps is 3.8 feet lower than the water level for the 1-day 100 year low flow drought condition. In the unlikely event that emergency make-up water requirements cannot be satisfied by surface water withdrawals from the Rock River, groundwater wells will serve for makeup to the essential service water cooling towers.

Because of the site hydrological characteristics given above, plant operation should not be affected by Rock River water level conditions and therefore, hydrological monitors have not been installed. The Rock River is not used for any public water supply. There are no recorded plans for any future public water supply usage from the Rock River. The nearest surface water users downstream from Byron Station are on the Mississippi River over 115 miles away. This allows for sufficient mixing that makes permanently installed hydrological monitors unnecessary. In performing dose calculations from liquid releases, Byron Station uses a historical average river flow value, Fw, as a parameter in the liquid release model.

Protective Facilities and Equipment

The principal onsite assembly areas for Byron Station are the Machine Shop on the 401-foot elevation of the Service Building and the Unit #1 Turbine Building track-way. These areas are suitable because:

1. They are large open areas suitable for assembling a large number of people in a short time;
2. They can be easily exited if a site evacuation is deemed necessary following an assembly; and
3. They have a low probability of being affected by a serious accident involving the Reactor and it’s primary systems.

The offsite relocation centers for Byron Station are discussed in Section 4 of this annex. These locations are suitable, depending on the emergency condition. These locations are owned by Exelon; thus, personnel, supplies, and communications are readily available.

First Aid and Medical Facilities

Byron Station has an inplant first aid/decontamination room on the 426-foot elevation of the auxiliary building near the station laboratory complex. This room is provided with a sink, a shower, and a supply cabinet. First aid kits, stretchers, sinks, eyewashes, and emergency showers have been placed in strategic locations throughout the station.

Medical treatment given to injured persons at the station is of a "first aid" nature. When more professional care is needed, injured persons are transported to a local hospital or clinic. Rockford Memorial Hospital in Rockford, Illinois is the designated support hospital for handling contaminated injured persons. Provena St. Joseph Medical Center in Joliet, Illinois is the backup medical facility for evaluation and treatment of persons suffering from traumatic injury, medical illness, or radiation exposure and uptake.

Appendix 1: NUREG-0654 Cross-Reference

Annex Section	NUREG-0654
1.0	Part I, Section A
1.1	Part I, Section C
1.2	Part I, Section D
Figure 1-1	Part I, Section D
2.0	Part II, Section A.4
2.1	Part II, Section A.3
3.0	Part II, Section D
4.1	Part II, Section E.1 & J.7
4.2	Part II, Section I.2 & 3
4.3	Part II, Section J.10.m
4.3.1	Part II, Section E.6
4.3.2	Part II, Section J.8
4.4	Part II, Section J.1-5
Figure 4-1	Part II, Section J.10.m
Table 4-2	Part II, Section J.8 & 10.b
Figure 4-2	Part II, Section J.5
4.4	Part II, Section J.2 & 3
5.1	Part II, Section H.1 & G.3
5.2.1	Part II, Section H.5.a & 8
5.2.2	Part II, Section H.5.b & I.2
5.2.3	Part II, Section H.5.c
5.2.4	Part II, Section H.5.d
5.2.5	Part II, Section H.6.b & 7
5.2.6	Part II, Section H.5.a & 6.a
5.3	Part II, Section J.1-5
5.4	Part II, Section L.1 & 2

Appendix 2: Station Letters of Agreement

1. Byron Fire Protection District and Rescue – fire protection.
2. Rockford Memorial Hospital – medical services.
3. The Ogle County Sheriff's Office – law enforcement.

Attachment 3

EP-AA-1003

"Exelon Nuclear Standardized Radiological Emergency Plan Annex for Clinton Station"

Revision 11

EP-AA-1003
Revision 11

EXELON NUCLEAR

RADIOLOGICAL EMERGENCY PLAN ANNEX FOR CLINTON STATION

Submitted: Kevin Appel Date: 10/10/07
Midwest Region Emergency Preparedness Manager

Authorized: Jim Meister Date: 10/12/07
Vice President – Operations Support

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APPENDIXES

REVISION HISTORY

Revision 0; March 2002	Revision 9; November 2006	
Revision 1; May 2002	Revision 10, April 2007	
Revision 2; August 2002		
Revision 3; May 2003		
Revision 4; August 2003		
Revision 5; January 2004		
Revision 6, December 2004		
Revision 7, May 2005		
Revision 8; January 2006		

Section 1: Introduction

As required in the conditions set forth by the Nuclear Regulatory Commission (NRC) for the operating licenses for the Exelon Nuclear Stations, the management of Exelon recognizes its responsibility and authority to operate and maintain the nuclear power stations in such a manner as to provide for the safety of the general public.

The Exelon Emergency Preparedness Program consists of the Exelon Nuclear Standardized Emergency Plan (Emergency Plan) Station Annexes, emergency plan implementing procedures, and associated program administrative documents. The Emergency Plan outlines the basis for response actions that would be implemented in an emergency. Planning efforts common to all Exelon Nuclear stations are encompassed within the Emergency Plan.

This document serves as the Clinton Station Annex and contains information and guidance that is unique to the station. This includes Emergency Action Levels (EALs), and facility geography and location for a full understanding and representation of the station’s emergency response capabilities. The Station Annex is subject to the same review and audit requirements as the Emergency Plan.

1.1 Facility Description

The Clinton Station, is located in approximately 6 miles east of Clinton, Illinois, in DeWitt County in Central Illinois. The Clinton Station is operated by AmerGen Energy Company, LLC.

The location can be defined by placing the station in the approximate center of a triangle formed by Bloomington, 22 miles to the north, Decatur, 22 miles to the south, and Champaign, 30 miles to the east. The reactor containment, the focal point for the Station, is located approximately 3 miles northeast of the confluence of the Salt Creek North Fork and the Salt Creek.

The site encompasses about 14,000 acres. This includes the Station of about 150 acres and a man-made, irregular U-shaped cooling reservoir of about 4,895 acres, known as Clinton Lake.

The surrounding area is mostly rural with no major population centers (greater than 25,000 people) or industrial complexes within a 10 mile radius of the Station. Recreational facilities are also limited in the area with Clinton Lake offering the largest variety.

The Clinton Station is a Boiling Water Reactor (BWR), the unit is rated at 3473 MWt. The rated electric output of the unit is 1062 MWe; from the General Electric (GE) turbine generator. The Nuclear Steam System Supplier (NSSS) was GE (Nuclear Energy Division). The entire plant, except for the NSSS, was designed by Sargent & Lundy (S&L) Engineers.

The containment system designed by Sargent & Lundy employs the drywell/pressure suppression features of the BWR-MARK III containment concept. The containment is a right cylindrical, reinforced concrete, steel-lined pressure vessel with a hemispherical dome.

The power generation complex includes several adjacent buildings, including an Auxiliary Building, Control Building (housing the Main Control Room), the Fuel Building, the Turbine Building, Diesel Generator and HVAC Building, the Radwaste Building, and the Service Building. Other buildings such as the gatehouse, circulating water screenhouse, makeup water pump house, warehouses, etc., are also located in the general plant area.

The Circulating Water Screen House located on the Clinton Lake, provides makeup water for the Clinton Station.

The ultimate heat sink for emergency core cooling is a submerged pond and intake flume of 590 acre-feet capacity that underlies the cooling lake and the natural grade of the site.

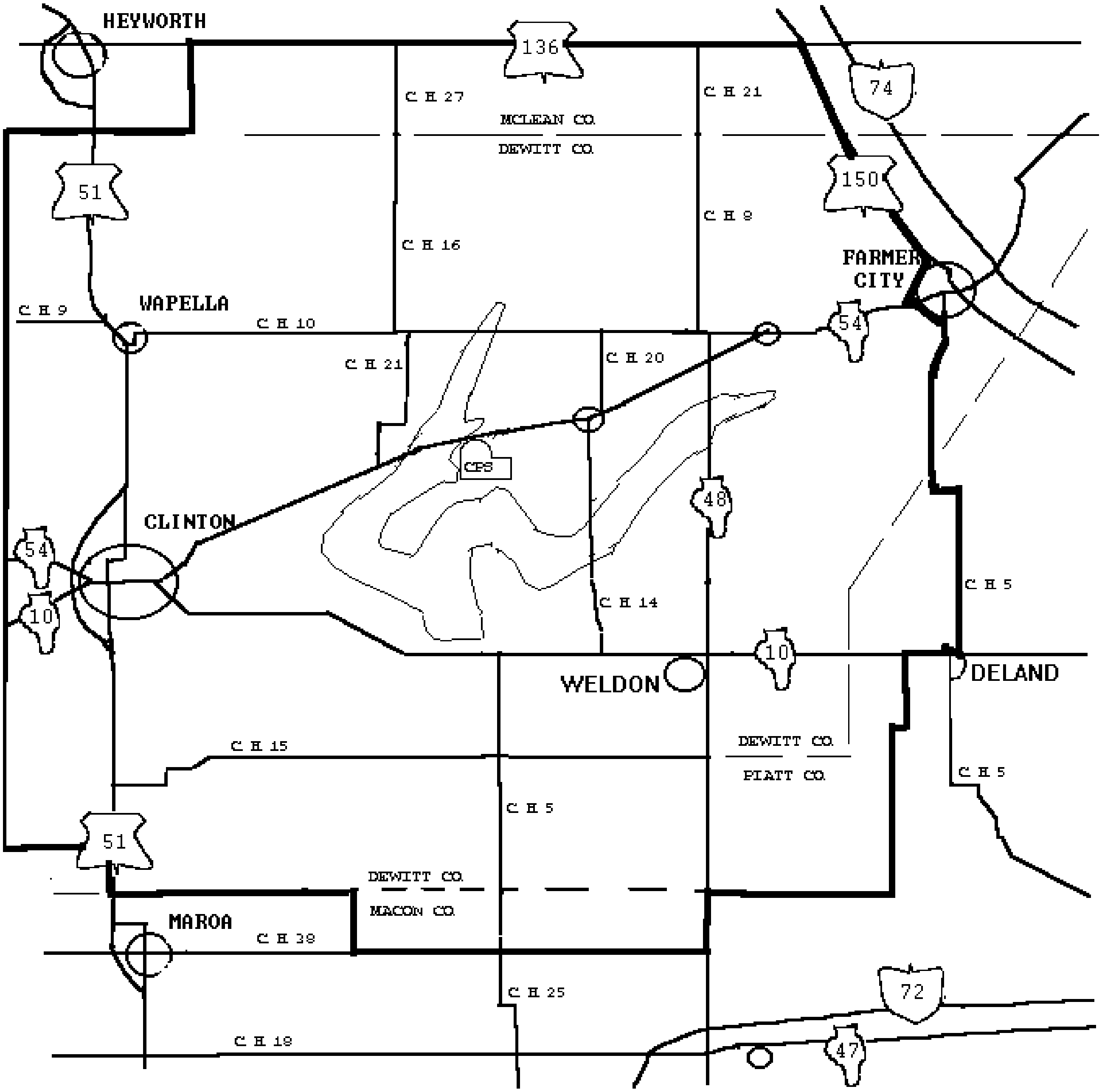
The Clinton Station utilizes a single vent stack of approximately 200 feet in height for the release of all gaseous waste. For more specific site location information, refer to the Station USAR.

1.2 Emergency Planning Zones

The plume exposure Emergency Planning Zone (EPZ) for Clinton Station shall be an area surrounding the Station with a radius of about ten miles (exact boundaries are determined by the State of Illinois). Refer to Figure 1-1.

The ingestion pathway EPZ for Clinton Station shall be an area surrounding the Station with a radius of about 50 miles.

Figure 1-1: Clinton Station Location and 10 Mile EPZ



Section 2: Organizational Control of Emergencies

2.1 On-Shift Emergency Response Organization Assignments

Initial response to any emergency is by the normal plant organization present at the site. This organization includes positions that are onsite 24 hours per day and is described in Table B-1 below.

Table B-1: Minimum Staffing Requirements for the On-Shift Clinton Station ERO

Functional Area	Major Tasks	Emergency Positions	Minimum Shift Size
1. Plant Operations and Assessment of Operational Aspects	Control Room Staff	Shift Manager	1
		Shift Supervisor	1
		Nuclear Station Operator	2
		Non-Licensed Operator	2
2. Emergency Direction and Control	Command and Control	Shift Emergency Director (CR)	1(a)
3. Notification & Communication	Emergency Communications	Plant Shift Personnel	1
4. Radiological Assessment	Offsite Dose Assessment	Station Personnel	1(a)
	In-plant Surveys	RP Personnel	2
	Chemistry	Chemistry Personnel	1
5. Plant System Engineering, Repair, and Corrective Actions	Technical Support	STA or Incident Assessor (CR)	1
	Repair and Corrective Actions	MM/Operations Shift Personnel(OSC	1
		Electrical/I&C Maintenance (OSC)	1
6. In-Plant Protective Actions	Radiation Protection	RP/Operations Shift Personnel	2
7. Fire Fighting	--	Fire Brigade	5(a)(c)
8. First Aid and Rescue Operations	--	Plant Personnel	2(a)
9. Site Access Control and Personnel Accountability	Security & Accountability	Security Team Personnel	(d)
		TOTAL:	**15**

(a) May be provided by personnel assigned other functions.

(c) Fire Brigade per USAR/Technical Specifications, as applicable.

(d) Per Security Plan.

2.2 Incident Assessor

Clinton Station has the option of using an Incident Assessor in these cases where the STA qualification is held by others such as the Shift Manager. Upon declaration of an emergency, the Incident Assessor fulfills the role of the on-shift technical advisor and reports to the Shift Emergency Director (Shift Manager). The Incident Assessor shall function as an advisor to the Shift Manager on matters of safety and act as an on-shift technical advisor, and, if qualified, the Nuclear Engineer. The Incident Assessor is an ERO position that can be filled by an individual who is qualified as the Shift Technical Advisor or Incident Assessor.

As an advisor to the Shift Manager, the Incident Assessor shall have no authority to direct the activities of the shift during an emergency. The Incident Assessor shall be available for briefing individuals who are preparing to assume command authority. The Incident Assessor is required to be present in all modes. The Incident Assessor will be present within the Owner Controlled area when filling the Emergency Plan function.

2.3 Emergency Response Organization Positional Responsibilities

Once an emergency is declared, the Emergency Response Organization (ERO) is activated as described in the Exelon Nuclear Radiological Emergency Plan.

NOTE: Credit is provided towards the Exelon Standard 60-minute minimum staffing requirements for those on-shift personnel already filling the response functions. (three RP positions, one Electrical/I&C Maintenance position and one Mechanical Maintenance position).

2.4 Non-Exelon Nuclear Support Groups

Exelon Nuclear has contractual agreements with several companies whose services would be available in the event of a radiological emergency. These agencies and their available services are listed in Appendix 3 of the Exelon Nuclear Radiological Emergency Plan.

Emergency response coordination with governmental agencies and other support organizations is discussed in Section A of the Exelon Nuclear Radiological Emergency Plan.

Agreements exist on file at Clinton Station with several support agencies. These agencies and their support roles are listed in Appendix 2, Station Letters of Agreement.

Section 3: Classification of Emergencies

3.1 General

Section D of the Exelon Nuclear Standardized Emergency Plan divides the types of emergencies into four Emergency Classification Levels (ECLs). The first four are the UNUSUAL EVENT, ALERT, SITE AREA EMERGENCY, and GENERAL EMERGENCY. These ECLs are entered by meeting the Emergency Action Level (EAL) Threshold Values provided in this section of the Annex. 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. Depending on the severity of an event, prior to returning to a standard day-to-day organization, a state or phase called RECOVERY may be entered to provide dedicated resources and organization in support of restoration and communication activities following the termination of the emergency.

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 Guideline 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 acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

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

RECOVERY: Recovery can be considered as a phase of the emergency and is entered by meeting emergency termination criteria provided in EP-AA-111 Emergency Classification and Protective Action Recommendations.

An emergency is classified by assessing plant conditions and comparing abnormal conditions to Initiating Conditions and Threshold Values for each Emergency Action Level.

Individuals responsible for the classification of events will refer to the Initiating Condition and Threshold Values on the matrix of the appropriate station Standardized Emergency Plan Annex (this document). This matrix will contain Initiating Conditions, EAL Threshold Values, Mode Applicability Designators, appropriate EAL numbering system, and additional guidance necessary to classify events. It may be provided as a user aid.

The matrix is set up in four Recognition Categories. The first is designated as "R" and relates to Abnormal Radiological Conditions / Abnormal Radiological Effluent Releases. The second is designated as "F" and relates to Fission Product Barrier Degradation. The third is designated as "M" and relates to System Malfunctions. The fourth is designated as "H" and relates to Hazards and Other Conditions.

The matrix is designed to provide an evaluation of the Initiating Conditions from the worst conditions (General Emergencies) on the left to the relatively less severe conditions on the right (Unusual Events). Evaluating conditions from left to right will reduce the possibility that an event will be under classified. All Recognition Categories should be reviewed for applicability prior to classification.

The Initiating Conditions are coded with a two letter and one number code. The first letter is the Recognition Category designator, the second letter is the Classification Level, “U” for (NOTIFICATION OF) UNUSUAL EVENT, “A” for ALERT, “S” for SITE AREA EMERGENCY and “G” for GENERAL EMERGENCY. The EAL number is a sequential number for that Recognition Category series. All Initiating Conditions that are describing the severity of a common condition (series) will have the same number.

The EAL number may then be used to reference a corresponding page(s), which provides the basis information pertaining to the Initiating Condition:

- Threshold Value
- Mode Applicability
- Basis

Emergency Action Levels are the measurable, observable detailed conditions that must be met in order to classify the event. Classification is not to be made without referencing, comparing and satisfying the Threshold Values specified in the Emergency Action Levels.

A list of definitions is provided as part of this document for terms having specific meaning to the Emergency Action Levels. Site specific definitions are provided for terms with the intent to be used for a particular Initiating Condition/Threshold Value and may not be applicable to other uses of that term at other sites, the Emergency Plan or procedures.

References are also included to documents that were used to develop the EAL Threshold Values.

References to the Emergency Director means the person in Command and Control as defined in the Emergency Plan. Classification of emergencies is a non-delegable responsibility of Command and Control for the onsite facilities with responsibility assigned to the Shift Emergency Director (Control Room Shift Manager) or the Station Emergency Director (TSC). Classification of emergencies remains the responsibility of the applicable onsite facility even after Command and Control is transferred to the Corporate Emergency Director (EOF).

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

Indications used for monitoring and evaluation of plant conditions include the normally used instrumentation, backup or redundant instrumentation, and the use of other parameters that provide information that supports determination if an EAL threshold value has been reached. When an EAL refers to a specific instrument or indication that is determined to be inaccurate or unavailable, then alternate indications shall be used to monitor the specified condition.

During an event that results in changing parameters trending towards an EAL classification, and instrumentation that was available to monitor this parameter becomes unavailable or the parameter goes off scale, the parameter should be assumed to have been exceeded consistent with the trend and the classification made if there are no other direct or indirect means available to determine if the threshold has not been exceeded.

EALs are for unplanned events. A planned evolution involves 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. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL Threshold Value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

When two or more Emergency Action Levels are determined, declaration will be made on the highest classification level for the Unit. When both units are affected, the highest classification for the Station will be used for notification purposes and both units’ classification levels will be noted.

3.2 Mode Applicability

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

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 Fission Product Barrier Matrix EALs are applicable only to events that initiate in Hot Shutdown or higher.

If there is a change in 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.

3.3 Emergency Director Judgment

Emergency Director Judgment EALs are provided in the Hazards and Other Condition Affecting Plant Safety section and on the Fission Product Barrier Matrix. Both of the Emergency Director Judgment EALs have specific criteria for when they should be applied.

The Hazards Section Emergency Director Judgment EALs are intended to address unanticipated conditions which are not addressed explicitly by other EALs but warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under specific emergency classifications (UE, Alert, SAE or GE).

The FPB Matrix ED Judgment EALs are intended to include unanticipated conditions, which are not addressed explicitly by any of the other FPB threshold values, but warrant determination because conditions exist that fall under the broader definition for a significant Loss or Potential Loss of the barrier (equal to or greater than the defined FPB threshold values).

3.4 Fission Product Barrier Restoration

Fission Product Barriers (FPBs) are not treated the same as EAL threshold values. Conditions warranting declaration of the loss or potential loss of a Fission Product Barrier may occur resulting in a specific classification. The condition that caused the loss or potential loss declaration could be rectified as the result of Operator action, automatic actions, or designed plant response. Barriers will be considered re-established when there are direct verifiable indications (containment penetration or open valve has been isolated, coolant sample results, etc) that the barrier has been restored and is capable of mitigating future events.

The reestablishment of a fission product barrier does not alter or lower the existing classification. Entry into Termination/Recovery phase is still required for exiting the present classification. However the reestablishment of the barrier should be considered in determining future classifications should plant conditions or events change.

3.5 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 five or more persons violently protesting station operations or activities at the site.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

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

CONTAINMENT CLOSURE: Containment Closure is considered to Containment as defined by Technical Specifications.

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

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

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute 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, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

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.

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

NORMAL LEVELS: Normal levels can be considered as 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.

OPERATING MODES	REACTOR MODE SWITCH POSITION	TEMP
(1) Power Operation:	Run	N/A
(2) Startup:	Refuel (a) or Startup/Hot Standby	N/A
(3) Hot Shutdown (a):	Shutdown	> 200° F
(4) Cold Shutdown (a):	Shutdown	≤ 200° F
(5) Refueling (b):	Shutdown or Refuel	N/A
(D) Defueled:	All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).	

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Hot Matrix – applies in modes (1), (2), and (3)

Cold Matrix – applies in modes (4), (5), and (D)

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

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

SABOTAGE: A 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.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations >10% Reactor Power change.

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 isolated from the Control Room.

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

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

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

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

Emergency Action Level Technical Basis Page Index

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the HVAC and SGTS Radiation Monitors that exceeds or is expected to exceed **4.50E+07 uCi/sec** for **≥ 15 minutes** (as found on Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the HVAC and SGTS Radiation Monitors that exceeds or is expected to exceed **4.50E+06 uCi/sec** for **≥ 15 minutes** (As found on Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1 VALID reading on any effluent monitor **> 200 times** the high alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**. **OR** 2. The sum of VALID readings on the HVAC and SGTS Radiation Monitors is **> 2.75E+06 uCi/sec** for **≥ 15 minutes** (as found on Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on any effluent monitor **> 2 times** the high alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**. **OR** 2. The sum of VALID readings on the HVAC and SGTS Radiation Monitors is **> 9.93E+05 uCi/sec** for **≥ 60 minutes** (as found on Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates in excess of **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

<table>
<tr><th></th><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="5">Abnormal Rad Levels / Radiological Effluent</td></tr>
<tr><td rowspan="2">Abnormal Radiation Levels</td><td></td><td>
<table>
<tr><th>Table R1
Fuel Handling Incident Radiation Monitors</th></tr>
<tr><td>• VF (1PR006A-D)
• CCP monitors (1PR042A-D)
• Main VR (1PR001A-D)
• Fuel Transfer VR (1PR008A-D)</td></tr>
</table>
</td><td>
RA2 Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 D

<u>EAL Threshold Values:</u>
1. VALID reading > 1000 mR/hr on one or more of the radiation monitors in Table R1.
OR
2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool, Upper Containment Fuel Storage Pool, Inclined Fuel Transfer System, or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.
</td><td>
RU2 Unexpected rise in plant radiation. 1 2 3 4 5 D

<u>EAL Threshold Values:</u>
1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool, Upper Containment Fuel Storage Pool, Inclined Fuel Transfer System, or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:
• Refueling Cavity water level < 22 ft. 8 in. above the Reactor Vessel Flange.
OR
• Spent Fuel Pool or Upper Containment Fuel Storage Pool water level < 23 ft.
OR
• Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool, Upper Containment Fuel Storage Pool, Inclined Fuel Transfer System or Fuel Transfer Canal.
AND
b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1.
OR
2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of 1000 over NORMAL LEVELS.
</td></tr>
<tr><td>
<table>
<tr><th>Table R2
Areas Requiring Continuous Occupancy</th></tr>
<tr><td>• Main Control Room (1RIX-AR035)
• Central Alarm Station (by survey)
• Radwaste Control Room (by survey)
• Remote Shutdown Panel (by survey)</td></tr>
</table>
</td><td>
<table>
<tr><th>Table R3
Areas Requiring Infrequent Access</th></tr>
<tr><td>• Containment
• Auxiliary Building
• Fuel Building
• Control Building
• Diesel Generator & HVAC Building
• Screenhouse</td></tr>
</table>
</td><td>
RA3 Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 D

<u>EAL Threshold Values:</u>
1. VALID radiation monitor or survey readings >15 mR/hr in areas requiring continuous occupancy (Table R2) to maintain plant safety functions.
OR
2. VALID radiation monitor or survey readings > 2000 mR/hr in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.
</td><td>
RU3 Fuel clad degradation. 1 2 3

<u>EAL Threshold Values:</u>
1. Offgas system isolation due to VALID Offgas post-treatment radiation monitor signal with charcoal adsorbers in service.
OR
2. Specific coolant activity > 4.0 uCi/gm Dose Equivalent I-131.
</td></tr>
</table>

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

Fission Product Barrier Matrix

Hot Matrix

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
FG1 Loss of ANY two barriers AND Loss or Potential Loss of the third barrier. $\boxed{1}\boxed{2}\boxed{3}$	**FS1** Loss or Potential Loss of ANY two barriers. $\boxed{1}\boxed{2}\boxed{3}$	**FA1** ANY Loss or ANY Potential Loss of either Fuel Clad or RCS. $\boxed{1}\boxed{2}\boxed{3}$	**FU1** ANY Loss or ANY Potential Loss of Containment. $\boxed{1}\boxed{2}\boxed{3}$

Sub-Category	FC - Fuel Clad		RC - Reactor Coolant System		CT - Containment	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
1. RCS Activity →	Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.	None	None	None	None	None
2. RPV Water Level →	RPV level **< -187 in.**	RPV level **< -162 in.** (TAF).	RPV level **< -162 in.** (TAF).	None	None	Plant conditions indicate Primary Containment flooding is required (entry into SAGs)
3. Drywell Pressure →	None	None	1. Drywell Pressure **> 1.68 psig.** **AND** 2. Drywell Pressure rise due to RCS leakage.	None	1. Rapid unexplained drop in containment pressure following initial pressure rise. **OR** 2. Containment pressure response not consistent with LOCA conditions.	1. Containment pressure **≥ 15 psig** and rising. **OR** 2. a. Drywell H2 concentration **≥ 9%.** **OR** b. Containment H2 concentration **≥ SAG-2, Deflagration Limit.**
4. RCS Leakrate →	None	None	1. UNISOLABLE Main Steam Line (MSL) break as indicated by the failure of both MSIVs in ANY one line to close. **AND** 2. a. High MSL Flow AND High Aux. Bldg. Steam Tunnel Temperature/Turbine Building Temperature. **OR** b. Direct report of steam release.	1. RCS leakage **> 50 gpm** inside the drywell or containment. **OR** 2. UNISOLABLE primary system leakage outside Primary Containment as indicated by Secondary Containment area temperatures or radiation levels **> EOP-8, Maximum Normal** operating levels.		
5. Hi Drywell / Cnmt Rad →	Drywell/Cnmt radiation monitor reading **> Fuel Cladding Loss Threshold**, Table F1.	None	1. a. Drywell Radiation monitor reading **> 100 R/hr** **OR** b. Containment Radiation monitor reading **> 33 R/hr** **AND** 2. Indications of RCS leakage into the Drywell.	None	None	Drywell/Cnmt. radiation monitor reading **> Containment Potential Loss Threshold**, Table F2.
6. Breach/Bypass →			None	None	1. a. Failure of all isolation valves in any one line to close. **AND** b. Downstream pathway to the environment exists. **OR** 2. Intentional venting/purging of Primary Containment per EOPs or SAGs due to accident conditions. **OR** 3. UNISOLABLE primary system leakage outside primary containment as indicated by Secondary Containment area temperatures or radiation levels **> EOP-8, Maximum Safe** operating levels.	None
7. ED Judgment.→	Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

Table F2 Containment Potential Loss
Drywell / Containment Radiation Thresholds

Time After Shutdown (hrs)	Drywell Rad (R/hr)	Containment Rad (R/hr)
≤ 2	5.90 E+02	9.70 E+01
> 2 to 4	4.40 E+02	7.00 E+01
> 4 to 8	2.90 E+02	4.60 E+01
> 8 to 16	1.40 E+02	2.20 E+01
> 16 to 23	5.90 E+01	9.50 E+00
> 23	5.10 E+01	8.00 E+00

Table F1 Fuel Cladding Loss
Drywell / Containment Radiation Thresholds

Time After Shutdown (hrs)	Drywell Rad (R/hr)	Cnmt Rad (R/hr)
≤ 2	2.60 E+02	4.13 E+01
> 2 to 4	1.95 E+02	3.10 E+01
> 4 to 8	1.30 E+02	2.06 E+01
> 8 to 16	6.00 E+01	9.53 E+00
> 16 to 23	2.60 E+01	4.13 E+00
> 23	2.20 E+01	3.49 E+00

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

System Malfunction

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
AC Electrical Distribution	**MG1** Prolonged loss of all offsite power and prolonged loss of all onsite AC power to Division I and Division II essential busses. 1 2 3 **<u>EAL Threshold Values:</u>** 1. Loss of power to Emergency Reserve Auxiliary Transformer (ERAT) and Reserve Auxiliary Transformer (RAT). **AND** 2. Failure of DG 1A and DG 1B emergency diesel generators to supply power to vital busses 1A1 and 1B1. **AND** 3. a. Restoration of at least one vital bus (excluding Division III) within **4 hours** is <u>not</u> likely **OR** b. RPV level **<u>cannot</u>** be determined to be **> -162 in.** (TAF)	**MS1** Loss of all offsite power and loss of all onsite AC power to Division I and Division II essential busses. 1 2 3 **<u>EAL Threshold Values:</u>** 1. Loss of power to Emergency Reserve Auxiliary Transformer (ERAT) and Reserve Auxiliary Transformer (RAT). **AND** 2. Failure of DG1A and DG1B emergency diesel generators to supply power to vital busses 1A1 and 1B1. **AND** 3. Failure to restore power to at least one vital bus (excluding Division III) within **15 minutes** from the time of loss of both offsite and onsite AC power.	**MA1** AC power capability to Division I and Division II essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout. 1 2 3 **<u>EAL Threshold Values:</u>** 1. AC power capability to vital busses (1A1, 1B1) reduced to only one of the following power sources for **> 15 minutes**: • Emergency Reserve Auxiliary transformer (ERAT) • Reserve Auxiliary Transformer (RAT) • One Emergency Diesel Generator: DG 1A or DG 1B **AND** 2. Any additional single power source failure will result in station blackout.	**MU1** Loss of all offsite power to Division I and Division II essential busses for greater than 15 minutes. 1 2 3 4 5 **<u>EAL Threshold Values:</u>** Loss of power to Emergency Reserve Auxiliary Transformer (ERAT) **AND** Reserve Auxiliary Transformer (RAT) for **> 15 minutes**.
RPS / Inadvertent Criticality	**MG3** Failure of the Reactor Protection System to complete an automatic scram and manual scram was NOT successful and there is indication of an extreme challenge to the ability to cool the core. 1 2 **<u>EAL Threshold Values:</u>** 1. Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**: a. Reactor power remains **> 5% APRM**. **OR** b. Suppression pool temperature **> Boron Injection Temperature (EOP 1A, Fig. G) AND** boron injection required for reactivity control. **AND** 2. a. RPV level cannot be restored and maintained **> -187 in.** **OR** b. Heat Capacity Limit (EOP-6, Fig. P) exceeded.	**MS3** Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded and manual scram was NOT successful. 1 2 **<u>EAL Threshold Values:</u>** Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**: 1. Reactor power remains **> 5% APRM**. **OR** 2. Suppression pool temperature **> Boron Injection Temperature (EOP-1A, Fig. G) AND** boron injection required for reactivity control.	**MA3** Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded. 1 2 **<u>EAL Threshold Values:</u>** 1. A Reactor Protection System setpoint was exceeded. **AND** 2. Automatic scram did not reduce reactor power **< IRM Range 6**.	**MU3** Inadvertent criticality. 3 4 5 **<u>EAL Threshold Values:</u>** An UNPLANNED extended positive period observed on nuclear instrumentation.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
DC Power		**MS4** Loss of all vital DC power. 1 2 3 **EAL Threshold Values:** Loss of all vital DC power based on **< 108 VDC** on 125 VDC battery busses 1A and 1B for **> 15 minutes**.		
Heat		**MS5** Complete loss of heat removal capability. 1 2 3 **EAL Threshold Values:** Heat Capacity Temperature Limit (EOP-6, Fig. P) exceeded.		
Annunciators		**MS6** Inability to monitor a SIGNIFICANT TRANSIENT in progress. 1 2 3 **EAL Threshold Values:** 1. Loss of most **(approximately 75%)** safety system annunciators (Table M2). **AND** 2. Indications needed to monitor safety functions (Table M3) are unavailable. **AND** 3. SIGNIFICANT TRANSIENT in progress (Table M4). **AND** 4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MA6** UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) A SIGNIFICANT TRANSIENT in progress, Or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable. 1 2 3 **EAL Threshold Values:** 1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes.** **AND** 2. a. SIGNIFICANT TRANSIENT in progress (Table M4). **OR** b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MU6** UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes. 1 2 3 **EAL Threshold Values:** 1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** 2. UNPLANNED loss of most **(approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.
RCS Leakage				**MU7** RCS leakage. 1 2 3 **EAL Threshold Values:** 1. Unidentified or pressure boundary leakage **> 10 gpm**. **OR** 2. Identified leakage **> 25 gpm**.

Table M2 - Control Room Panels	Table M3 – Safety Functions and Related Systems	Table M4 - Significant Transients
• 1H13-P601 • 1H13-P877 • 1H13-P680 • 1H13-P801 (VC & VG)	• 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) • Fission Product Barriers	• Turbine trip • Reactor Scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10 %** Reactor Power change

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

<table>
<tr><th></th><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="5">System Malfunction</td></tr>
<tr><td>Communications</td><td></td><td></td><td>
<table>
<tr><th colspan="3">Table M6 - Communications Capability</th></tr>
<tr><th>System</th><th>Onsite</th><th>Offsite</th></tr>
<tr><td>Plant Radio System</td><td>X</td><td></td></tr>
<tr><td>Plant Paging System</td><td>X</td><td></td></tr>
<tr><td>Sound Power Phones</td><td>X</td><td></td></tr>
<tr><td>In-Plant Telephones</td><td>X</td><td></td></tr>
<tr><td>PCS Phones</td><td>X</td><td>X</td></tr>
<tr><td>All Telephone Lines (commercial and microwave)</td><td></td><td>X</td></tr>
<tr><td>NARS</td><td></td><td>X</td></tr>
<tr><td>ENS</td><td></td><td>X</td></tr>
<tr><td>Satellite Phones</td><td></td><td>X</td></tr>
<tr><td>HPN</td><td></td><td>X</td></tr>
<tr><td>Cellular Phones</td><td></td><td>X</td></tr>
</table>
</td><td>MU10 UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5
<u>EAL Threshold Values:</u>
1. Loss of all Table M6 Onsite communications capability affecting the ability to perform routine operations.
OR
2. Loss of all Table M6 Offsite communications capability.</td></tr>
<tr><td>T. S. Time</td><td></td><td></td><td></td><td>MU11 Inability to reach required shutdown within Technical Specification limits. 1 2 3
<u>EAL Threshold Values:</u>
Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.</td></tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the Site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, “Threat Assessment”. **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 D **EAL Threshold Values:** 1. Control Room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per CPS 4003.01, in < **15 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 D **EAL Threshold Values:** Entry into CPS 4003.01 for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse	**Table H3** **Internal Flooding Areas** • Auxiliary Building • Fuel Building • Diesel Building • Control Building • Screenhouse • Turbine Building • Radwaste Building	**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by seismic instrumentation **> 0.10 g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 85 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in any Table H3 area as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment. **OR** 6. High lake level **> 697 ft**.	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1 a. Seismic event as indicated by seismic instrumentation **> 0.02g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained **(> 15 minutes)** high winds **> 85 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode. **OR** 6. High lake level **> 696 ft**.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm: **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse		**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** 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.

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

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	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the HVAC and SGTS Radiation Monitors that exceeds or is expected to exceed **4.50E+07 uCi/sec** for **≥ 15 minutes** (as found on Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the HVAC and SGTS Radiation Monitors that exceeds or is expected to exceed **4.50E+06 uCi/sec** for **≥ 15 minutes** (As found on Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1 VALID reading on any effluent monitor is **> 200 times** the high alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**. **OR** 2. The sum of VALID readings on the HVAC and SGTS Radiation Monitors is **> 2.75E+06 uCi/sec** for **≥ 15 minutes** (as found on Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on any effluent monitor is **> 2 times** the high alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**. **OR** 2. The sum of VALID readings on the HVAC and SGTS Radiation Monitors is **> 9.93E+05 uCi/sec** for **≥ 60 minutes** (as found on by Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates in excess of **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

<table>
<tr><th></th><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="5">Abnormal Rad Levels / Radiological Effluent</td></tr>
<tr>
<td rowspan="2">Abnormal Radiation Levels</td>
<td></td>
<td>
<table>
<tr><th>Table R1
Fuel Handling Incident Radiation Monitors</th></tr>
<tr><td>• VF (1PR006A-D)
• CCP monitors (1PR042A-D)
• Main VR (1PR001A-D)
• Fuel Transfer VR (1PR008A-D)</td></tr>
</table>
</td>
<td>RA2 Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 D

<u>EAL Threshold Values:</u>

1. VALID reading > 1000 mR/hr on one or more of the radiation monitors in Table R1.
OR
2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool, Upper Containment Fuel Storage Pool, Inclined Fuel Transfer System or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.</td>
<td rowspan="2">RU2 Unexpected rise in plant radiation. 1 2 3 4 5 D

<u>EAL Threshold Values:</u>

1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool, Upper Containment Fuel Storage Pool, Inclined Fuel Transfer System or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:
• Refueling Cavity water level < 22 ft. 8 in. above the Reactor Vessel Flange.
OR
• Spent Fuel Pool or Upper Containment Fuel Storage Pool water level < 23 ft.
OR
• Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool, Upper Containment Fuel Storage Pool, Inclined Fuel Transfer System or Fuel Transfer Canal.
AND
b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1.
OR
2. UNPLANNED VALID Area Radiation Monitor readings rise by a factor of 1000 over NORMAL LEVELS.</td>
</tr>
<tr>
<td>
<table>
<tr><th>Table R2
Areas Requiring Continuous Occupancy</th></tr>
<tr><td>• Main Control Room (1RIX-AR035)
• Central Alarm Station (by survey)
• Radwaste Control Room (by survey)
• Remote Shutdown Panel (by survey)</td></tr>
</table>
</td>
<td>
<table>
<tr><th>Table R3
Areas Requiring Infrequent Access</th></tr>
<tr><td>• Containment
• Auxiliary Building
• Fuel Building
• Control Building
• Diesel Generator & HVAC Building
• Screenhouse</td></tr>
</table>
</td>
<td>RA3 Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 D

<u>EAL Threshold Values:</u>

1. VALID radiation monitor or survey readings >15 mR/hr in areas requiring continuous occupancy (Table R2) to maintain plant safety functions.
OR
2. VALID radiation monitor or survey readings > 2000 mR/hr in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.</td>
</tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
AC Electrical Distribution				**MU1** Loss of all offsite power to Division I and Division II essential busses for greater than 15 minutes. 1 2 3 4 5 **EAL Threshold Values:** Loss of power to Emergency Reserve Auxiliary Transformer (ERAT) **AND** Reserve Auxiliary Transformer (RAT) for **> 15 minutes**.
			MA2 Loss of all offsite power and loss of all onsite AC power to Division I and Division II essential busses. 4 5 D **EAL Threshold Values:** 1. Loss of power to Emergency Reserve Auxiliary Transformer (ERAT) and Reserve Auxiliary Transformer (RAT). **AND** 2. Failure of DG 1A and DG 1B emergency diesel generators to supply power to vital busses 1A1 and 1B1. **AND** 3. Failure to restore power to at least one vital bus (excluding Division III) within **15 minutes** from the time of loss of both offsite and onsite AC power.	
RPS				**MU3** Inadvertent criticality. 3 4 5 **EAL Threshold Values:** An UNPLANNED extended positive period observed on nuclear instrumentation.
DC Power				**MU4** UNPLANNED loss of required DC power for greater than 15 minutes. 4 5 **EAL Threshold Values:** 1. UNPLANNED Loss of all required vital DC Power based on **< 108 VDC** indications on 125 VDC battery busses 1A and 1B. **AND** 2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.
Heat Sink			**MA5** Inability to maintain plant in cold shutdown with irradiated fuel in the RPV. 4 5 **EAL Threshold Values:** 1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F** for > Table M1 duration. **OR** 2. UNPLANNED RPV pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.	**MU5** UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV. 4 5 **EAL Threshold Values:** 1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F.** **OR** 2. Loss of all RCS temperature **AND** RPV level indication for **> 15 minutes**.

Table M1 – RCS Reheat Duration Thresholds

RCS	Secondary Containment Closure	Duration
Intact	N/A	60 minutes*
Not Intact	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.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
RCS Leakage / Inventory	**MG8** Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged with irradiated fuel in the RPV. 4 5 **EAL Threshold Values:** 1. Loss of RPV inventory per Table M5 indications. **AND** 2. a. RPV level **< - 162 in**. (TAF) for **> 30 min**. **OR** b. RPV level unknown with Indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following: • Containment High Range Monitors 1RIX-CM061 **OR** 1RIX-CM062 **> 3 R/hr** or off-scale high • Erratic Source Range Monitor indication **AND** 3. Containment is challenged as indicated by one or more of the following: • Drywell Hydrogen concentration ≥ **9%.** • Containment hydrogen concentration ≥ **Fig. R, Deflagration Limit.** • Containment pressure ≥ **15 psig.** • Primary and Secondary CONTAINMENT CLOSURE not established. • Any Secondary Containment radiation reading **> EOP-8 Table U, Maximum Safe** operating level.	**MS8** Loss of RCS/RPV inventory affecting core decay heat removal capability. 4 **EAL Threshold Values:** 1. **Without** Primary or Secondary CONTAINMENT CLOSURE established: a. RPV level **< -151 in.** **OR** b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory per Table M5 indications. **OR** 2. **With** Primary or Secondary CONTAINMENT CLOSURE established a. RPV level **< -162 in.** (TAF) **OR** b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory as evidenced by either of the following: • Per Table M5 indications. • Erratic Source Range Monitor indication.	**MA8** Loss of RCS/RPV inventory with irradiated fuel in the RPV. 4 5 **EAL Threshold Values:** 1. Loss of RCS/RPV inventory as indicated by RPV level **< -145.5 in.** **OR** 2. a. Loss of RPV inventory per Table M5 indications. **AND** b. RCS/RPV level unknown for **> 15 minutes.**	**MU8** RCS leakage. 4 **EAL Threshold Values:** RPV level **cannot** be restored and maintained **> Level 3 (8.9 in.).**
RCS Leakage / Inventory		**MS9** Loss of RPV inventory affecting core decay heat removal capability with irradiated fuel in the RPV. 5 **EAL Threshold Values:** 1. **Without** Secondary CONTAINMENT CLOSURE established: a. RPV level **< -151in.** **OR** b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following: • Containment High Range Monitors 1RIX-CM061 **OR** 1RIX-CM062 **> 3 R/hr** or off-scale high. • Erratic Source Range Monitor indication. **OR** 2. **With** Secondary CONTAINMENT CLOSURE established: a. RPV level **< -162 in** (TAF). **OR** b. RPV level unknown with Indication of core uncovery as evidenced by one or more of the following: • Containment High Range Monitors 1RIX-CM061 **OR** 1RIX-CM062 **> 3 R/hr** or off-scale high. • Erratic Source Range Monitor indication.	**Table M5 – Indications of RCS Leakage** • Unexplained floor or equipment sump level rise • Unexplained Suppression Pool level rise • Unexplained vessel make up rate rise • Observation of leakage	**MU9** UNPLANNED loss of RCS inventory with irradiated fuel in the RPV. 5 **EAL Threshold Values:** 1. UNPLANNED RPV level **< 204 in**. Shutdown Range (RPV flange) for **≥ 15 minutes**. **OR** 2. a. Loss of RPV inventory per Table M5 indications. **AND** b. RPV level unknown.

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

<table>
<tr><th></th><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="5">System Malfunction</td></tr>
<tr><td>Communications</td><td></td><td></td><td>
<table>
<tr><th colspan="3">Table M6 - Communications Capability</th></tr>
<tr><th>System</th><th>Onsite</th><th>Offsite</th></tr>
<tr><td>Plant Radio System</td><td>X</td><td></td></tr>
<tr><td>Plant Paging System</td><td>X</td><td></td></tr>
<tr><td>Sound Power Phones</td><td>X</td><td></td></tr>
<tr><td>In-Plant Telephones</td><td>X</td><td></td></tr>
<tr><td>PCS Phones</td><td>X</td><td>X</td></tr>
<tr><td>All Telephone Lines (commercial and microwave)</td><td></td><td>X</td></tr>
<tr><td>NARS</td><td></td><td>X</td></tr>
<tr><td>ENS</td><td></td><td>X</td></tr>
<tr><td>Satellite Phones</td><td></td><td>X</td></tr>
<tr><td>HPN</td><td></td><td>X</td></tr>
<tr><td>Cellular Phones</td><td></td><td>X</td></tr>
</table>
</td><td>
MU10 UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5

<u>EAL Threshold Values:</u>

1. Loss of all Table M6 Onsite communications capability affecting the ability to perform routine operations.
OR
2. Loss of all Table M6 Offsite communications capability.
</td></tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security Event resulting in loss of physical control of the facility. 1 2 3 4 5 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the Site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, "Threat Assessment". **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 D **EAL Threshold Values:** 1. Control room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per CPS 4003.01 in < **15 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 D **EAL Threshold Values:** Entry into CPS 4003.01 for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse	**Table H3** **Internal Flooding Areas** • Auxiliary Building • Fuel Building • Diesel Building • Control Building • Screenhouse • Turbine Building • Radwaste Building	**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by seismic instrumentation **> 0.10 g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 85 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding in that results in **EITHER**: a. Degraded safety system performance in any Table H3 area as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment. **OR** 6. High lake level **> 697 ft**.	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1 a. Seismic event as indicated by seismic instrumentation **> 0.02g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained **(> 15 minutes)** high winds **> 85 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode. **OR** 6. High lake level **> 696 ft**.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas	**Table H2** **Vital Areas** • Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse		**HA7** Release of toxic or flammable Gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 D **EAL Threshold Values:** 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.

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

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RG1

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the HVAC and SGTS Radiation Monitors that exceeds or is expected to exceed **4.50 E+07 uCi/sec** for **≥ 15 minutes** (as found on Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 1000 mRem TEDE**

      OR

   b. **> 5000 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in a PAG dose being reached.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RG1 (cont)

Basis (cont):

Threshold #2 Basis:

The TEDE (1000 mRem) and the CDE Thyroid (5000 mRem) doses are set at the EPA PAG Limits.

The “site boundary” is defined by an approximately 800-meter (1/2-mile) radius around the plant. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release has been stopped and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AG1
2. EP-AA-112-500, Emergency Environmental Monitoring
3. Exelon DAPAR version 3.0a
4. EP-MW-110-200 Dose Assessment
5. ODCM Section 6.3.1, Gaseous Effluents and Total Dose
6. CPS 3315.03, Radiation Monitoring (AR/PR)
7. CPS 4979.01, Abnormal Release of Airborne Radioactivity
8. EP-EAL-0603, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Clinton Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RS1

Initiating Condition:

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Note: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the HVAC and SGTS Radiation Monitors that exceeds or is expected to exceed **4.50 E+06 uCi/sec** for **≥ 15 minutes** (as found on Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 100 mRem TEDE**

      OR

   b. **> 500 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events that may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in 10% of a PAG dose being reached. Assumptions and DAPAR inputs are provided in EP-EAL-065.

The same value is used for ground level and elevated release points. An elevated release may not affect offsite areas as close to plant as ground level release; however, use of ground level values provides conservative estimates for exposure (cloud shine) to an overhead plume (EPA-400, section 5.6.1).

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RS1 (cont)

Basis (cont):

Threshold #2 Basis:

The TEDE (100 mRem) and the CDE Thyroid (500 mRem) doses are set at 10% of the EPA PAG Limits.

The “site boundary” is defined by an approximately 800-meter (1/2-mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release has been stopped and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AS1
2. EP-AA-112-500, Emergency Environmental Monitoring
3. EP-MW-110-200 Dose Assessment
4. Exelon DAPAR version 3.0a
5. ODCM Section 6.3.1, Gaseous Effluents
6. CPS 3315.03, Radiation Monitoring (AR/PR)
7. CPS 4979.01, Abnormal Release of Airborne Radioactivity
8. EP-EAL-0603, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Clinton Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading on any effluent monitor **> 200 times** the high alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**.

   OR

2. The sum of VALID readings on the HVAC and SGTS radiation monitors is **> 2.75 E+06 uCi/sec** for **≥ 15 minutes** (as found on Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.

Basis:

UNPLANNED: As used in this context, 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.

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

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

Basis (cont):

Threshold #1 Basis:

The threshold addresses radioactivity releases (liquid or gaseous) that for whatever reason cause effluent radiation monitor readings to exceed two hundred times the alarm setpoint established by the radioactive discharge permit. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the Offsite Dose Calculation Manual (ODCM) to warn of a release that is not in compliance with the Radiological Effluent Technical Specifications (RETS). Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

An elevated monitor reading while the effluent flow path is isolated is NOT considered to be a VALID reading.

The liquid radwaste discharge radiation monitors (0RIX-PR040) measure the radioactivity concentration of the liquid radwaste discharge into the plant service water (WS) discharge header. Monitor 0RIX-PR040 is only operable when conducting a radioactive discharge, which is rarely done at the Clinton Station. The sample extraction point is located upstream of the discharge isolation valve prior to the connection to the WS system. The sample point is located sufficiently upstream of the isolation valve to ensure that the monitor response will isolate radioactivity before passing the valve.

The High Alarm closes the radwaste discharge to WS isolation valve (1WE036), regardless of whether WS or CW is providing dilution flow. The High Alarm is set sufficiently high to minimize nuisance alarms due to natural radioactive radionuclides, but low enough to provide for a several decade buffer between the initial alarm and 10CFR50 release to the public limits. The alarm provides ~ one decade of buffer between the action point and any potential ODCM limits.

Threshold #2 Basis:

CPS incorporates 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.

RA1 (cont)

Basis (cont):

This EAL addresses a potential or actual drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. However, the effluent monitor Alert value for gaseous effluents was reduced to a value one half way between the Unusual Event value and the Site Area Emergency value to ensure sequential classifications. The sum of both gaseous effluent monitor readings provides a total station release rate. Assumptions and calculation inputs are provided in EP-EAL-0603. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM, Rev 21. The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

Threshold #3 Basis:

Confirmed sample analyses in excess of two hundred times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by increasing the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mRem/yr basis of the 10 CFR 20 limits for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be approximately 10 mRem/hr. The required release duration was reduced to 15 minutes in recognition of the increased severity.

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

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

Basis Reference(s):

1. NEI 99-01, Rev. 4 AA1
2. ODCM Section 6.3.1, Gaseous Effluents
3. ODCM Section 6.3.2, Liquid Effluents
4. CPS 3315.03, Radiation Monitoring (AR/PR)
5. CPS 4979.01, Abnormal Release of Airborne Radioactivity
6. CPS 4979.05, Abnormal Release of Radioactive Liquids
7. USAR Section 11.5.2.2.6, Liquid Radwaste Discharge Radiation Monitor
8. USAR Figure 2.1-7, CPS Restricted Area
9. EP-EAL-0603, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Clinton Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading on any effluent monitor **> 2 times** the high alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**.

   OR

2. The sum of VALID readings on the HVAC and SGTS radiation monitors is > **9.93 E+05 uCi/sec** for **≥ 60 minutes** (as found on Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates in excess of **2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

Basis:

UNPLANNED, As used in this context, 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.

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

The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU1 (cont)

Basis (cont):

Threshold #1 Basis:

The effluent release paths are monitored for radioactivity prior to the flow reaching the point where it would mix with the process flow to the environment. Prior to initiating batch releases, the discharge volume is sampled and analyzed for radioactivity. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate. Radiation monitor alarm setpoints are established to automatically isolate the process flow at the point determined by the discharge permit. These limits are based on the Offsite Dose Calculation Manual ODCM.

An elevated monitor reading while the effluent flow path is isolated is NOT considered to be a VALID reading."

The liquid radwaste discharge radiation monitors (0RIX-PR040) measure the radioactivity concentration of the liquid radwaste discharge into the plant service water (WS) discharge header. Monitor 0RIX-PR040 is only operable when conducting a radioactive discharge, which is rarely done at the Clinton Station. The sample extraction point is located upstream of the discharge isolation valve prior to the connection to the WS system. The sample point is located sufficiently upstream of the isolation valve to ensure that the monitor response will isolate radioactivity before passing the valve.

The High Alarm closes the radwaste discharge to WS isolation valve (1WE036), regardless of whether WS or CW is providing dilution flow. The High Alarm is set sufficiently high to minimize nuisance alarms due to natural radioactive radionuclides, but low enough to provide for a several decade buffer between the initial alarm and 10 CFR 50 release to the public limits. The alarm provides ~ one decade of buffer between the action point and any potential ODCM limits.

Threshold #2 Basis:

This EAL addresses a potential drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 2, regulatory commitments for an extended period of time. The sum of both gaseous effluent monitor readings provides a total station release rate. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM.

The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition. Assumptions and inputs for this calculation are provided in EP-EAL-0603.

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

Basis (cont):

Threshold #3 Basis:

Confirmed sample analyses in excess of two times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times ODCM for 30 minutes does not exceed this EAL. Grab samples are used to determine release concentrations or rates to confirm meter readings or when the effluent monitors are not in service. Maximum instantaneous release rate limits are calculated in accordance with the ODCM. These are indicated on discharge permits.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AU1
2. ODCM Section 6.3.1, Gaseous Effluents
3. ODCM Section 6.3.2, Liquid Effluents
4. CPS 3315.03, Radiation Monitoring (AR/PR)
5. CPS 4979.01, Abnormal Release of Airborne Radioactivity
6. CPS 4979.05, Abnormal Release of Radioactive Liquids
7. USAR Section 11.5.2.2.6, Liquid Radwaste Discharge Radiation Monitor
8. EP-EAL-0603, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Clinton Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA2

Initiating Condition:

Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1.

Table R1 - Fuel Handling Incident Radiation Monitors
• VF (1PR006A-D) • CCP monitors (1PR042A-D) • Main VR (1PR001A-D) • Fuel Transfer VR (1PR008A-D)

OR

2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool, Upper Containment Fuel Storage Pool, Inclined Fuel Transfer System or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.

Basis:

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

Threshold #1 Basis:

Uncovering spent fuel represents a substantial degradation of the level of safety of the plant and warrants an Alert classification. Time is available to take corrective actions. 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, no prompt fatalities are predicted and the risk of injury is low. 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 fuel uncovery begins. The value of 1000 mR/hr was conservatively chosen for classification purposes.

Radiation monitor readings are used to provide indication of fuel uncovery and/or fuel damage. High monitor readings associated with the transfer or relocation of a source, stored in or near the pool or readings responding to a planned evolution such as removal of the reactor head or equipment relocation are not classified under this threshold since the reading would not be indicative of fuel uncovery and/or fuel damage.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA2 (cont)

Basis (cont):

Dropping heavy loads onto the spent fuel can cause significant damage to the spent fuel and an Alert is also warranted under these conditions provided that the above radiation monitor threshold readings are reached.

Threshold #2 Basis:

Once Spent Fuel Pool water level drops below the low level alarm setpoint (8 in. below normal level), further drops can be monitored only by visual observation.

When the RPV head is removed and cavity flooded, there are no remote level indications available. Fuel that becomes uncovered while suspended from the refuel grapple may not be indicated. Without report of the vertical position of the grapple, fuel uncovery cannot be determined. Visual observation, therefore, provides the only viable mechanism of determining if spent fuel will be uncovered.

This EAL applies to irradiated fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AA2
2. USAR 1.2.2.4.11.3
3. Technical Specifications 3.7.7
4. CPS 4011.02, Spent Fuel Pool Abnormal Water Level Drop
5. CPS 8117.01, Reactor Pressure Vessel Disassembly

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU2

Initiating Condition:

Unexpected rise in plant radiation.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool, Upper Containment Fuel Storage Pool, Inclined Fuel Transfer System or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by **EITHER**:

- Refueling Cavity water level **< 22 ft. 8 in.** above the Reactor Vessel Flange.

  OR

- Spent Fuel Pool or Upper Containment Fuel Storage Pool water level **< 23 ft**.

  OR

- Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool, Upper Containment Fuel Storage Pool, Inclined Fuel Transfer System or Fuel Transfer Canal.

AND

b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1.

Table R1 - Fuel Handling Incident Radiation Monitors
• VF (1PR006A-D) • CCP monitors (1PR042A-D) • Main VR (1PR001A-D) • Fuel Transfer VR (1PR008A-D)

OR

2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU2 (cont)

Basis:

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

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

NORMAL LEVELS: Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Threshold #1 Basis:

Technical Specifications require the Spent Fuel Pool water level be maintained at least 19 ft. over the top of the irradiated fuel assemblies seated in the pool racks.

Since no remote indication of Spent Fuel Pool water level exists, drops in Spent Fuel Pool water level can normally be detected only through visual observation.

In preparation for refueling operations, the Refueling Cavity and the separator storage pool are flooded and water level is increased to the elevation of the upper containment pool water level. The upper containment pools and spent fuel storage pools are in different buildings on different elevations and are connected by the IFTS (Inclined Fuel Transfer System). Normal upper containment pool water level is 827 ft 3 in. el. Technical Specifications require Refueling Cavity water level be maintained at least 22 ft above the top of the RPV flange (804 ft 4-1/16 in. + 22 ft) or 826 ft 4-1/16 in. el. The threshold value corresponds to 471.5 in. above instrument zero. Plant procedure CPS 8117.01, Reactor Pressure Vessel Disassembly provides alternate level monitoring capabilities when the normal level instrumentation is unavailable for the desired level range or the head vent piping is removed. In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed.

Threshold #2 Basis:

Valid elevated area radiation levels usually have long lead times relative to the potential for radiological release beyond the site boundary, thus impact to public health and safety is very low.

This EAL addresses unplanned increases in radiation levels inside the plant. These radiation levels represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU2 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 AU2
2. RP-AA-203, Exposure Control and Authorization
3. Technical Specifications 3.7.7
4. CPS 4011.02, Spent Fuel Pool Abnormal Water Level Drop
5. USAR Figure 3.8-31
6. USAR Table 7.1-13
7. USAR 9.1.4.2.10
8. CPS 8117.01, Reactor Pressure Vessel Disassembly

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA3

Initiating Condition:

Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions:

Table R2 – Areas Requiring Continuous Occupancy
• Main Control Room (1RIX-AR035) • Central Alarm Station (by survey) • Radwaste Control Room (by survey) • Remote Shutdown Panel (by survey)

OR

2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.

Table R3 – Areas Requiring Infrequent Access
• Containment • Auxiliary Building • Fuel Building • Control Building • Diesel Generator & HVAC Building • Screenhouse

Basis:

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

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

Basis (cont):

Threshold #1 Basis:

This EAL addresses increased radiation levels that impede necessary access to operating stations requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. Areas requiring continuous occupancy include the Main Control Room, the central alarm station (CAS), TSC (when staffed) and the Radwaste Control Room. The security alarm station is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations.

The value of 15 mR/hr is derived from the General Design Criteria (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. A 30 day duration implies an event potentially more significant than an Alert.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EALs may be involved. For example, a dose rate of 15 mR/hr in the Main Control Room may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This EAL is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.).

Threshold #2 Basis:

This EAL addresses increased radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Typically areas requiring infrequent access to maintain plant safety functions include plant VITAL AREAS. Area radiation levels above 2000 mR/hr are indicative of radiation fields which may limit personnel access to equipment, the operation of which may be needed to assure adequate core cooling or shutdown the reactor.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA3 (cont)

Basis (cont):

The dose rate threshold selected is based on site administrative limits.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved. For example, a dose rate of 2000 mR/hr may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation rises due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.) or pre-existing radiation areas for which radiological controls already exist. The concern of this threshold is the unanticipated rise in radiation levels that results in unplanned restrictions to areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AA3
2. USAR Table 12.3-2
3. USAR Appendix F, Fire Protection Safe Shutdown Analysis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU3

Initiating Condition:

Fuel clad degradation.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Offgas system isolation due to VALID Offgas post-treatment radiation monitor signal with charcoal adsorbers in service.

   OR

2. Specific coolant activity **> 4.0 uCi/gm** Dose Equivalent I-131.

Basis:

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

Threshold #1 Basis:

During unit operation, the steam jet air ejectors (SJAEs) remove all non-condensable gases from the main condenser including air in-leakage and disassociated products originating in the reactor and exhausts them to the charcoal adsorbers. A rise in offgas activity could therefore indicate damage to the fuel cladding, a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

The gas from the main condenser normally includes relatively low levels of radioactivity. If radioactivity of the gas reaches the OFF GAS HIGH-HIGH RADIATION annunciator setpoint the Offgas system isolates (i.e. vent stack valve auto closes).

The modifier "VALID" is appropriate because there are several conditions that may cause the monitor to alarm that are not related to fuel clad degradation and therefore should not result in the declaration of an Unusual Event.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU3 (cont)

Basis (cont):

Threshold #2 Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding coolant Technical Specifications for iodine spiking. The specific iodine activity ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) accident is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits.

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 Reactor Coolant activity unless another cause is known, e.g., Reactor Coolant System chemical decontamination evolution (during shutdown) is ongoing with resulting high activity levels.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU4 & CU5
2. Technical Specifications 3.4.8
3. USAR 3.7.5
4. CPS 3215.01, Off-Gas (OG)
5. CPS 5140.46, AR/PR Annunciator - Off Gas Post-Treat PRM #1 - 1RIX-PR035
6. CPS 5140.47, AR/PR Annunciator - Off Gas Post-Treat PRM #2 - 1RIX-PR041
7. CPS 4004.02, Loss of Vacuum

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FG1

Initiating Condition:

Loss of ANY two barriers AND Loss or Potential Loss of the third barrier.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the General Emergency classification level each barrier is weighted equally.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FS1

Initiating Condition:

Loss or Potential Loss of ANY two barriers.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Site Area Emergency classification level, each barrier is weighted equally.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FA1

Initiating Condition:

ANY Loss or ANY Potential Loss of either Fuel Clad or RCS.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

At the Alert classification level, 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FU1

Initiating Condition:

ANY Loss or ANY Potential Loss of Containment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Unlike the Fuel Cladding and RCS barriers, the loss of either of which results in an Alert under EAL 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC1 – Loss

Initiating Condition:

Primary coolant activity level

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.

Basis:

Loss Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

300 uCi/gm Dose Equivalent I-131 is well above that expected for iodine spikes and corresponds, generically, to about 2% to 5% fuel cladding damage. When reactor coolant activity reaches this level, significant clad damage has occurred and thus the Fuel Cladding barrier is considered lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. CPS 4010.01, Reactor Coolant High Activity

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC2 – Loss or Potential Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

RPV level **< -187 in.**

POTENTIAL LOSS

RPV level **< -162 in**. (TAF)

Basis:

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAG program.

Loss Basis:

Core submergence is the preferred method of core cooling and as such, the failure to re-establish RPV level above the top of active fuel for an extended period of time could lead to significant fuel damage. RPV level < -187 in. (Minimum Steam Cooling RPV Water Level (MSCRWL)) could be indicative of a large break Loss Of Coolant Accident (LOCA) or a small LOCA with the inability of emergency core cooling systems to reflood the RPV.

Potential Loss Basis:

An RPV level reading of -162 in. indicates RPV level is at the top of active fuel (TAF). When RPV level is above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is at or below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC2 – Loss or Potential Loss (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. 4401.01, EOP-1 RPV Control
3. 4404.01, EOP-1 ATWS RPV Control
4. 4403.01, EOP-2 RPV Flooding
5. Clinton Power Station Severe Accident Guidelines Technical Bases
6. STA/IA Guide/1005.09M002
7. Clinton Power Station Emergency Operating Procedures Technical Bases
8. 4000.01, Abnormal RPV Water Level

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC5 – Loss

Initiating Condition:

Drywell/Cnmt radiation monitoring.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Drywell/Cnmt radiation monitor reading **> Fuel Cladding Loss Threshold**, Table F1

Table F1 – Fuel Cladding Loss Drywell/Cnmt Radiation Thresholds

Time After Shutdown (hours)	Drywell Rad R/hr	Cnmt Rad R/hr
≤ 2	2.60 E+02	4.13 E+01
> 2 to 4	1.95 E+02	3.10 E+01
> 4 to 8	1.30 E+02	2.06 E+01
> 8 to 16	6.00 E+01	9.53 E+00
> 16 to 23	2.60 E+01	4.13 E+00
> 23	2.20 E+01	3.49 E+00

Basis:

The drywell/containment radiation monitor readings specified in Table F1 provide values that indicates the release of reactor coolant into the drywell with elevated activity indicative of fuel damage (~2%). The values are a function of time after shutdown and were derived using Core Damage Assessment Methodology (CDAM) with 2% clad damage, no containment sprays in operation and a LOCA depressurized system. The reading is calculated assuming the instantaneous release and dispersal of the above reactor coolant noble gas and iodine inventory into the drywell atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations allowed within Technical Specifications (including iodine spiking) and are therefore indicative of fuel damage (approximately 2% - 5% cladding failure).

During at power (including ATWS) conditions the value listed for the "< 2 hours after shutdown" row is used as an indication of fuel damage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.

Basis:

The Emergency Director judgment fuel cladding loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Fuel Cladding barrier. The inability to monitor fuel cladding integrity should also be considered as a factor in judging that the Fuel Cladding barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC2 – Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

RPV level **< -162 in.** (TAF)

Basis:

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAG program.

Loss Basis:

RPV level reading of -162 in. indicates RPV level is at the top of active fuel (TAF). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment barriers, and initiation of all ECCS. If RPV level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of any unplanned loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a loss of the RCS barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. 4401.01, EOP-1 RPV Control
3. Clinton Power Station Emergency Operating Procedures Technical Bases, Section
4. 4001.01, Reactor Coolant Leakage

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC3 – Loss

Initiating Condition:

Drywell pressure.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. Drywell pressure **> 1.68 psig.**

   AND

2. Drywell pressure rise due to RCS leakage.

Basis:

The drywell pressure value is the ECCS initiation setpoint and the drywell high pressure scram setpoint and is therefore indicative of a Loss of Coolant Accident (LOCA) event that requires ECCS response. The drywell high pressure scram setpoint is an entry condition to EOP-1, RPV Control, and EOP-6, Primary Containment Control. Normal drywell pressure control functions (e.g., operation of drywell mixers, and drywell cooling) are specified in EOP 6.

In the Clinton design basis, drywell pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the drywell for which normal pressure control systems are inadequate or incapable of reversing the increasing pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to vent the drywell.

The second threshold focuses the fission product barrier loss threshold on a failure of the RCS instead of the non-LOCA malfunctions that may adversely affect drywell pressure. Therefore:

- Drywell pressure greater than 1.68 psig with corollary indications (drywell temperature) should therefore be considered a loss of RCS.
- Loss of drywell cooling that results in greater than 1.68 psig should not be considered a loss of RCS.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. 4401.01, EOP-1 RPV Control
3. 4402.01, EOP-6 Primary Containment Control
4. Clinton Power Station EOP Technical Bases
5. Technical Specifications Table 3.3.1.1-1
6. Technical Specifications Table 3.3.5.1-1
7. 4001.01, Reactor Coolant Leakage

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC4 – Loss or Potential Loss

Initiating Condition:

RCS leak rate.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. UNISOLABLE Main Steam Line (MSL) break as indicated by the failure of both MSIVs in ANY one line to close.

   AND

2. a. High MSL Flow AND High Aux. Bldg. Steam Tunnel Temperature/Turbine Building Temperature.

      OR

   b. Direct report of steam release.

POTENTIAL LOSS

1. RCS leakage **> 50 gpm** inside the drywell or containment.

   OR

2. UNISOLABLE primary system leakage outside Primary Containment as indicated by Secondary Containment area temperatures or radiation levels **> EOP-8 Maximum Normal** operating levels.

Basis:

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Loss Basis:

High Steam Flow and High Steam Tunnel Temperature Annunciators are both indications of a Main Steam Line Break. Both of these parameters will cause a signal for closure of the MSIVs. Should both valves in any one line fail to isolate, this event would be considered a Loss of the RCS.

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside primary containment exists when flow is not prevented by downstream isolations. In the case of a failure of both Main Steam Isolation Valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required.

Direct report of steam release is meant to provide an alternate means of classification if the Hi Steam Flow Annunciator or the Hi Steam Tunnel Temperature Annunciator fails to operate and the observation of conditions indicates a Main Steam Line Break in the judgment of the Emergency Director. This is not meant to cause a declaration based on leaks such as valve packing leaks where the consequences offsite would be negligible.

RC4 – Loss or Potential Loss (cont)

Basis (cont):

Potential Loss Threshold #1 Basis:

The potential loss of RCS based on leakage is set at a level indicative of a small breach of the RCS but which is well within the makeup capability of normal and emergency high-pressure systems. Core uncovery is not a significant concern for a 50 gpm leak; however, break propagation leading to significantly larger loss of inventory is possible.

The primary detection methods for small unidentified leaks within the drywell include monitoring of drywell floor drain sump flowrate, and drywell cooler condensate flow rate rises. These variables are continuously indicated and/or recorded in the control room. The secondary detection methods, (i.e., the monitoring of pressure and temperature of the drywell atmosphere and airborne gaseous and particulate radioactivity increases) are used to detect gross unidentified leakage. High drywell pressure will alarm and trip the isolation logic which will result in closure of selected containment isolation valves. Inventory loss events, such as a stuck open SRV, should not be considered when referring to “RCS leakage” because they are not indications of a break, which could propagate.

Potential Loss Threshold #2 Basis:

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of an unisolable primary system leakage outside the primary containment. For radiation levels, the specified maximum normal values are four to five time the routine survey results. The area temperature is defined to be the alarm setpoint. This is the maximum normal operating value and signifies the onset of abnormal system operation. When parameters reach these levels, equipment failure or miss-operation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-8, Secondary Containment Control. Tables F2 and F3 represent EOP-8 Tables T and U.

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the Reactor Enclosure since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the Reactor Enclosure, an unexpected rise in Feedwater flowrate, or unexpected Main Turbine Control Valve closure) may indicate that a primary system is discharging into the Reactor Building.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC4 – Loss or Potential Loss (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. M05-1002, Main steam
3. USAR 5.2.5
4. USAR Tables 5.2-9a and 5.2-9b
5. 9043.06, Drywell Floor Drain Sump Flow Test 00PS404
6. 9443.01, Drywell Equipment Drain Sump Flow E31-N578 Channel Cal 01PS274
7. 4406.01, EOP-8 Secondary Containment Control
8. Clinton Power Station Emergency Operating Procedures Technical Bases, Section 10
9. USAR Figure 6.2-132

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC5 – Loss

Initiating Condition:

Drywell/Cont Radiation Monitoring

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. a. Drywell Radiation monitor reading **> 100 R/hr**

   OR

   b. Containment Radiation monitor reading **> 33 R/hr**

   AND

2. Indications of RCS leakage into the Drywell.

Basis:

The drywell / containment radiation monitor readings are values that indicates a significant release of reactor coolant to the drywell. A reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the drywell atmosphere. Conservative estimates (high RCS uCi/cc) indicated that the Drywell readings from release of the normal RCS inventory would be ~ 100 R/hr. Due to their location, the containment monitors would indicate a third of the indication on the Drywell monitors which is ~ 33 R/hr. These readings are less than that specified for Fuel Cladding barrier Loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations.

Indication of a RCS leak into the drywell is added to qualify the radiation monitor indication to avoid declaring the loss of RCS barrier for situations where the radiation rise is not due to primary a system leak. For situations that involve failure of the Fuel Clad barrier alone, radiation monitor readings would rise due to shine and potentially giving a false indication of a loss of the RCS barrier. Therefore this EAL contains a qualifier to preclude over classification of the event if only fuel clad barrier failed.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Calc. EP-EAL-0611

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.

Basis:

The Emergency Director judgment RCS loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the RCS barrier. The inability to monitor RCS integrity should also be considered as a factor in judging that the RCS barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT2 – Potential Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

POTENTIAL LOSS

Plant conditions indicate that Primary Containment flooding is required (entry into SAGs).

Basis:

Potential Loss Basis:

When primary containment flooding is required, all symptom-based EOPs are exited and the Severe Accident Guideline flowcharts (SAGs) are entered in order to restore and maintain cooling to the core and any core debris. Since it may not be possible to recover the core inside the RPV, flooding the primary containment to the elevation of the top of active fuel in the containment may be required.

The EOP conditions requiring primary containment flooding represent imminent core melt sequences that, if not corrected, could lead to RPV failure and increased potential for containment failure.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Clinton Power Station Severe Accident Guidelines Technical Bases
3. STA/IA Guide/1005.09M002

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT3 – Loss or Potential Loss

Initiating Condition:

Containment pressure.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. Rapid unexplained drop in containment pressure following initial pressure rise.

   OR

2. Containment pressure response not consistent with LOCA conditions.

POTENTIAL LOSS

1. Containment pressure **≥ 15 psig** and rising.

   OR

2. a. Drywell hydrogen concentration **≥ 9%.**

      OR

   b. Containment hydrogen concentration **≥ SAG-2, Deflagration Limit.**

Basis:

Loss Threshold #1 Basis:

Rapid unexplained loss of containment pressure (i.e., not attributable to containment sprays or condensation effects) following an initial pressure rise indicates a loss of containment integrity.

Loss Threshold #2 Basis:

Rapid unexplained loss of pressure (i.e., not attributable to containment sprays, or condensation effects) following an initial pressure rise indicates a loss of containment integrity. Containment pressure should rise as a result of mass and energy release into the containment from a LOCA. Thus, containment pressure response not consistent with LOCA conditions indicates a loss of containment integrity. This indicator relies on operator recognition of an unexpected response for the condition and therefore does not include a specific pressure value or trend. Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. For example, blowdown mass flowrate may be only 60-80% of the analyzed rate. The unexpected response is important because it is the indicator for a containment bypass condition.

A breach of the drywell causing an abnormal containment pressure rise does not constitute a loss of containment.

CT3 – Loss or Potential Loss (cont)

Basis (cont):

Potential Loss Threshold #1 Basis:

When the Primary Containment design pressure is challenged, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded. This condition, if compounded by further plant degradation may challenge primary containment integrity and is, therefore, an appropriate threshold for potential loss of the Primary Containment barrier.

A Containment pressure of 15 psig is based on the containment/drywell design pressure. If the containment design pressure is exceeded this represents a challenge to the containment structure because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a potential loss of the containment barrier even if a breach has NOT occurred.

Potential Loss Threshold #2 Basis:

Explosive mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of SAG-2indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to primary containment integrity. The Deflagration Limit is the highest containment hydrogen concentration at which a deflagration will not generate pressures in excess of the structural capability of the containment. 9% hydrogen concentration in the drywell is the Drywell Hydrogen Deflagration Overpressure Limit. This limit is the lesser of:

- The highest drywell hydrogen concentration at which the peak differential pressure between the drywell and containment resulting from a deflagration will not exceed the peak differential pressure resulting from a main steam line or recirculation line break in the drywell.
- The highest drywell hydrogen concentration at which the rate of change of drywell pressure resulting from a deflagration will not exceed the maximum rate of change resulting from a main steam line or recirculation line break in the drywell.

The specified values for this potential loss threshold are readily recognizable because the Deflagration Limit and 9% drywell hydrogen concentration are well above the containment/drywell hydrogen monitor alarm setpoint (0.5% hydrogen) and the Primary Containment Control EOP entry condition. The containment Deflagration Limit requires intentional containment venting, which is defined to be a barrier loss under Primary Containment CT6.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS
CT3 – Loss or Potential Loss (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. USAR 6.2.1.1.3
3. USAR Table 1.3-4
4. CPS 4402.01, EOP-6 Primary Containment Control
5. SAG2
6. Clinton Power Station Emergency Operating Procedures Technical Bases, Sections 9 and 12

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT5 – Potential Loss

Initiating Condition:

Significant radioactive inventory in Containment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

POTENTIAL LOSS

Drywell/Cnmt radiation monitor reading **> Containment Potential Loss Threshold**, Table F2.

Table F2 – Containment Potential Loss Drywell/ Containment Radiation Thresholds

Time After Shutdown (hours)	Drywell Rad (R/hr)	Containment Rad (R/hr)
≤ 2	5.90 E+02	9.70 E+01
> 2 to 4	4.40 E+02	7.00 E+01
> 4 to 8	2.90 E+02	4.60 E+01
> 8 to 16	1.40 E+02	2.20 E+01
> 16 to 23	5.90 E+01	9.50 E+00
> 23	5.10 E+01	8.00 E+00

Basis:

The drywell/containment radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the Fuel Cladding barrier. NUREG-1228 “Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents” states that such readings do not exist when the amount of cladding damage is less than 20%. The values are a function of time after shutdown and were derived using Core Damage Assessment Methodology (CDAM) assuming 20% clad damage, no containment sprays in operation and a LOCA depressurized system. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a significant failure into the reactor coolant has occurred.

During at power (including ATWS) conditions the value listed for the “< 2 hours after shutdown” row is used as an indication of fuel damage.

Regardless of whether the Containment barrier itself is challenged, this amount of activity in the Containment/Drywell could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Containment barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT6 – Loss

Initiating Condition:

Containment isolation failure or bypass.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

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

   AND

   b. Downstream pathway to the environment exists.

   OR

2. Intentional venting/purging of Primary Containment per EOPs or SAGs due to accident conditions.

   OR

3. UNISOLABLE primary system leakage outside primary containment as indicated by Secondary Containment area temperatures or radiation levels **> EOP-8, Maximum Safe** operating levels.

Basis:

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Threshold #1 Basis:

This threshold addresses failure of open isolation devices that should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway to the environment. The concern is the unisolable open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of primary containment integrity.

Failure of containment isolation valves to isolate with a downstream pathway to the environment is only a concern during an accident. If this condition exists during normal Power Operation, a Technical Specification Action Statement will address it. However, during accident conditions, this will represent a breach of Primary Containment.

Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include unisolable Main Steam Line Isolation Valves, RCIC steam line breaks, unisolable RWCU system breaks, and unisolable containment atmosphere vent paths. Minor release paths such as instrument and sample lines are not considered under this threshold.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT6 – Loss (cont)

Basis (cont):

Examples of "downstream pathway to the Environment" could be through Turbine/Condenser, or direct release to the Turbine Enclosure or Reactor Enclosure.

The breach is NOT isolable from the Control Room if an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation from the Control Room should be made prior to the accident classification. If Operator actions from the Control Room are successful, then this IC is not applicable. Credit is NOT given for Operator actions taken in-plant (outside the Control Room) to isolate the leak.

This EAL is intended to cover containment isolation failures allowing a direct flow path to the environment such as failure of both MSIVs to close with open valves downstream to the turbine or to the condenser, even if these systems are not breached.

Threshold #2 Basis:

Intentional venting of the primary containment to the secondary containment and/or the environment per the EOPs/SAG due to accident conditions is considered a loss of the Primary Containment barrier.

Threshold #3 Basis:

The presence of elevated general area temperatures and/or area radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. Temperatures and radiation levels beyond their maximum safe operating temperatures are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This EAL threshold addresses problematic discharges outside primary containment that may not originate from a high-energy line break.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. CPS 4402.01, EOP-6 Primary Containment Control
3. SAG-2
4. Clinton Power Station Emergency Operating Procedures Technical Bases, Sections 8 and 9
5. 4406.01, EOP-8 Secondary Containment Control
6. Clinton Power Station Emergency Operating Procedures Technical Bases, Section 10
7. USAR Figure 6.2-132

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis:

The Emergency Director judgment Containment loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Containment barrier. The inability to monitor Containment parameters should also be considered as a factor in judging that the Containment barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG1

Initiating Condition:

Prolonged loss of all offsite power and prolonged loss of all onsite AC power to Division I and Division II essential busses.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of power to Emergency Reserve Auxiliary Transformer (ERAT) and Reserve Auxiliary Transformer (RAT).

   AND

2. Failure of DG 1A and DG 1B emergency diesel generators to supply power to vital busses 1A1 and 1B1.

   AND

3. a. Restoration of at least one vital bus (excluding Division III) within **4 hours** is **not** likely.

      OR

   b. RPV level **cannot** be determined to be **> -162 in**. (TAF)

Basis:

Loss of all AC power to vital busses compromises the availability of all plant safety systems. Prolonged loss of all AC power may lead to loss of Fuel Cladding, RCS and Primary Containment barriers. The four-hour interval to restore AC power to either unit vital bus is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout.”

The likelihood of restoring at least one vital bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

The 138-kV offsite power system connects the station to the Illinois Power Company Bloomington and Clinton Route 54 Substations. This system provides power to the Emergency Reserve Auxiliary Transformer (ERAT), which in turn supplies power to the vital busses. The 345-kV offsite power system connects the station to the Illinois Power Company grid at Brokaw, Rising-Oreana, and Latham-Oreana Substations. All three lines terminate at the station switchyard ring bus which feeds the Reserve Auxiliary Transformer (RAT), which in turn supplies power to the three vital busses.

Onsite AC power sources include the UATs powered from the main generator and emergency diesel generators DG 1A and DG 1B.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

Basis (cont):

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAG program.

A reading of -162 in. below instrument zero indicates RPV level is at the top of active fuel (TAF). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops below TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SG1
2. USAR Figure 8.3-1
3. USAR Section 8.1.5
4. USAR Section 8.3.1
5. CPS 4200.01, Loss of AC
6. Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Station Blackout, 10 CFR 50.63 Illinois Power Company, et al Clinton Power Station, Unit 1 Docket No. 50-461
7. CPS 4401.01, EOP-1 RPV Control

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS1

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to Division I and Division II essential busses.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of power to Emergency Reserve Auxiliary Transformer (ERAT) and Reserve Auxiliary Transformer (RAT).

   AND

2. Failure of DG 1A and DG 1B emergency diesel generators to supply power to vital busses 1A1 and 1B1.

   AND

3. Failure to restore power to at least one vital bus (excluding Division III) within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power compromises all plant safety systems and represents failures of plant functions required for the protection of the public. The 138-kV offsite power system connects the station to the Illinois Power Company Bloomington and Clinton Route 54 Substations. This system provides power to the Emergency Reserve Auxiliary Transformer (ERAT), which in turn supplies power to the vital busses. The 345-kV offsite power system connects the station to the Illinois Power Company grid at Brokaw, Rising-Oreana, and Latham-Oreana Substations. All three lines terminate at the station switchyard ring bus, which feeds the Reserve Auxiliary Transformer (RAT), which in turn, supplies power to the three vital busses.

The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when BOTH onsite and offsite power is lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS1
2. USAR Figure 8.3-1
3. USAR Section 8.1.5
4. USAR Section 8.3.1
5. CPS 4200.01, Loss of AC
6. Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Station Blackout, 10 CFR 50.63 Illinois Power Company, et al Clinton Power Station, Unit 1 Docket No. 50-461

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA1

Initiating Condition:

AC power capability to Division I and Division II essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. AC power capability to vital busses (1A1, 1B1) reduced to only one of the following power sources for **> 15 minutes**:
    - Emergency Reserve Auxiliary transformer (ERAT).
    - Reserve Auxiliary Transformer (RAT).
    - One Emergency Diesel Generator: DG 1A or DG 1B.

    AND

2. Any additional single power source failure will result in station blackout.

Basis:

Capability: (pertaining to electrical power supplies) is equipment that is available to provide and maintain AC power at the required voltage and frequency for the required load.

The reduction of available reliable power sources to a condition in which any additional single failure will result in a Unit Blackout is a substantial degradation in the level of safety of the plant. A Unit Blackout is a loss of AC power to all unit vital busses. Clinton blackout coping duration is four hours.

The listed power supplies take into account sources that, if unavailable, establish single-failure vulnerability.

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA1 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA5
2. USAR Figure 8.3-1
3. USAR Section 8.1.5
4. USAR Section 8.3.1
5. CPS 4200.01, Loss of AC
6. Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Station Blackout, 10 CFR 50.63 Illinois Power Company, et al Clinton Power Station, Unit 1 Docket No. 50-461

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU1

Initiating Condition:

Loss of all offsite power to Division I and Division II essential busses for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3, 4, 5

EAL Threshold Values:

Loss of power to Emergency Reserve Auxiliary Transformer (ERAT) AND Reserve Auxiliary Transformer (RAT) for **> 15 minutes**.

Basis:

The emergency busses are the safety-related, VITAL busses 1A1 (Division 1) and 1B1 (Division 2). The 138-kV and the 345-kV offsite power systems supply the 1A1 and 1B1 busses. This system provides power to the Emergency Reserve Auxiliary Transformer (ERAT), which in turn supplies power to the vital busses. The 345-kV offsite power system connects the station to the Illinois Power Company grid at Brokaw, Rising-Oreana, and Latham-Oreana Substations. All three lines terminate at the station switchyard ring bus which feeds the Reserve Auxiliary Transformer (RAT), which in turn supplies power to the vital busses.

Loss of all offsite power causes emergency diesel generators to automatically start and be available to carry the essential loads. Balance of plant systems that would assist in plant operations (e.g., condensate pumps, etc.) may be unavailable due to the loss of power.

A loss of offsite AC power reduces the 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.

The intent of this EAL is to declare an Unusual Event when offsite power has been lost and the emergency diesel generators have successfully started and energized the vital busses. The fifteen-minute interval was selected as a threshold to exclude transient power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU1 & CU3
2. USAR Figure 8.3-1
3. USAR Section 8.1.5
4. USAR Section 8.3.1
5. CPS 4200.01, Loss of AC

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA2

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to Division I and Division II essential busses.

Operating Mode Applicability:

4, 5, D

EAL Threshold Values:

1. Loss of power to Emergency Reserve Auxiliary Transformer (ERAT) and Reserve Auxiliary Transformer (RAT)

   AND

2. Failure of DG 1A and DG 1B emergency diesel generators to supply power to vital busses 1A1 and 1B1.

   AND

3. Failure to restore power to at least one vital bus (excluding Division III) within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power when in Cold Shutdown, Refueling or Defueled modes compromises safety systems required for decay heat removal and represents a substantial degradation of the level of safety of the plant. An Alert declaration (instead of a Site Area Emergency under EAL MS1) is appropriate in these modes because post-shutdown, decay heat energy levels offer more time to restore AC power to heat removal systems than the levels present when the reactor is in Power Operation, Startup or Hot Shutdown mode. Thus, the threat to the protection of the health and safety of the public is less severe.

The 138-kV offsite power system connects the station to the Illinois Power Company Bloomington and Clinton Route 54 Substations. This system provides power to the Emergency Reserve Auxiliary Transformer (ERAT), which in turn supplies power to the vital busses. The 345-kV offsite power system connects the station to the Illinois Power Company grid at Brokaw, Rising-Oreana, and Latham-Oreana Substations. All three lines terminate at the station switchyard ring bus which feeds the Reserve Auxiliary Transformer (RAT), which in turn supplies power to the three vital busses.

Consideration should be given to available loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not available on the energized bus, the bus should not be considered available.

The fifteen-minute interval begins from the time of loss of both onsite and offsite AC power and was selected as a threshold to exclude transient or momentary power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA2 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA3
2. USAR Figure 8.3-1
3. USAR Section 8.1.5
4. USAR Section 8.3.1
5. CPS 4200.01, Loss of AC
6. Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Station Blackout, 10 CFR 50.63 Illinois Power Company, et al Clinton Power Station, Unit1 Docket No. 50-461

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG3

Initiating Condition:

Failure of the Reactor Protection System to complete an automatic scram and manual scram was NOT successful and there is indication of an extreme challenge to the ability to cool the core.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**:
   a. Reactor power remains **> 5 % APRM.**

      OR

   b. Suppression pool temperature **> Boron Injection Temperature (EOP-1A, Fig. G) AND** boron injection required for reactivity control.

   AND

2. a. RPV level cannot be restored and maintained **> -187 in.**

      OR

   b. Heat Capacity Limit (EOP-6, Fig. P) exceeded.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed The reactor power threshold (5%) is approximately equal to the APRM downscale trip setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, Suppression Pool temperature trend) can be used to determine if reactor power is greater than 5% power.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG3 (cont)

Basis (cont):

Classification at the General Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

The Suppression Pool water temperature criterion (110° F) is the Boron Injection Initiation Temperature (BIT). The BIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the Suppression Pool heats the suppression pool to the Heat Capacity Temperature Limit (HCTL). If Suppression Pool temperature exceeds the BIT, reactor power is heating the Suppression Pool and the suppression pool cooling may be inadequate or incapable of performing its design function.

The second condition of this EAL indicates either:

- An extreme challenge to the ability to cool the core as indicated when RPV level cannot be held above -187 in. or unknown. The specified water level is the Minimum Steam Cooling RPV Water Level (MSCRWL). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500° F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core cooling could be a precursor of a core melt sequence.
  An extreme challenge to the primary containment as indicated when heat cannot be removed from the primary containment resulting in elevated suppression pool temperature. The Heat Capacity Limit is the highest suppression pool temperature from which a blowdown will not increase containment temperature to the design limit (185°F). The Heat Capacity Limit is a function of RPV pressure and suppression pool temperature and level and is a measure of the maximum heat load which the primary containment can withstand. Plant parameters in excess of the Heat Capacity Limit could be a precursor of primary containment failure. The Heat Capacity Limit is given in Fig. P of EOP-6, Primary Containment Control.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SG2
2. CPS 4100.01, Reactor scram
3. CPS 4401.01, EOP-1 RPV Control
4. CPS 4404.01, EOP-1A ATWS RPV Control
5. CPS 3304.02, Rod Control and Information System (RC&IS)
6. CPS 4402.01, EOP-6 Primary Containment Control
7. Clinton Power Station Emergency Operating Procedures Technical Bases, Sections 4, 5, 8 and 12

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS3

Initiating Condition:

Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded and manual scram was NOT successful.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**:

1. Reactor power remains **> 5 % APRM.**

   OR

2. Suppression pool temperature **> Boron Injection Temperature (EOP-1A, Fig. G) AND** boron injection required for reactivity control.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed The reactor power threshold (5%) is approximately equal to the APRM downscale trip setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, Suppression Pool temperature trend) can be used to determine if reactor power is greater than 5% power.

The Suppression Pool water temperature criterion is the Boron Injection Initiation Temperature (BIT). The BIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the Suppression Pool heats the suppression pool to the Heat Capacity Temperature Limit (HCTL). If Suppression Pool temperature exceeds the BIT, reactor power is heating the Suppression Pool and the suppression pool cooling may be inadequate or incapable of performing its design function.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS3 (cont)

Basis (cont):

Classification at the Site Area Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS2
2. CPS 4100.01, Reactor scram
3. CPS 4401.01, EOP-1 RPV Control
4. CPS 4404.01, EOP-1A ATWS RPV Control
5. CPS 3304.02, Rod Control and Information System (RC&IS)
6. CPS 4402.01, EOP-6 Primary Containment Control
7. Clinton Power Station Emergency Operating Procedures Technical Bases, Sections 4, 5, 8 and 12

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA3

Initiating Condition:

Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. A reactor protection system setpoint was exceeded.

   AND

2. Automatic scram did not reduce reactor power to **< IRM Range 6**.

Basis:

This condition indicates a failure of the automatic reactor protection system to successfully scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. Site-specific indication of reactor shutdown is included as the criteria of whether the scram was successful when required. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue.

A successful scram has occurred when there is sufficient rod insertion to bring the reactor subcritical (< IRM Range 6).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function.

MA3 (cont)

Basis (cont):

The second condition of this EAL indicates a failure of the automatic RPS scram function to rapidly insert a sufficient number of control rods to achieve reactor shutdown. The CRD system backup scram valves and the Alternate Rod Insertion (ARI) system provide automatic, alternate methods of completing the scram function. These backups, however, insert control rods at a much slower rate than the automatic RPS scram function. For the purpose of emergency classification at the Alert level, reactor shutdown achieved by automatic backup scram valve operation and ARI initiation does not constitute a successful RPS automatic scram.

Following any automatic RPS scram signal EOP-1A, ATWS RPV Control, prescribes insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Alert

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded, or first-out annunciators), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA2
2. Technical Specifications Table 3.3.1.1-1
3. CPS 4100.01, Reactor scram
4. CPS 4401.01, EOP-1 RPV Control
5. CPS 4404.01, EOP-1A ATWS RPV Control
6. CPS 3304.02, Rod Control and Information System (RC&IS)

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU3

Initiating Condition:

Inadvertent criticality.

Operating Mode Applicability:

3, 4, 5

EAL Threshold Values:

An UNPLANNED extended positive period observed on nuclear instrumentation.

Basis:

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

The term “extended” is used in order to allow exclusion of expected short-term positive periods from planned fuel bundle or control rod movements during core alteration. These short-term positive periods are the result of the rise in neutron population due to subcritical multiplication.

This EAL includes criticality events that occur in Cold Shutdown or Refueling modes (NUREG1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events as well as inadvertent criticalities occurring in Hot Shutdown mode. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification.

This condition can be identified using:

- Any of the four SRM channels (A-D) that display log count rate indication
- Period indication
- Short period annunciation
- White light display on Panel 1H13-P680

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU8 & CU8
2. Technical Specifications Table 3.3.1.2-1
3. USAR Table 7.1-13
4. USAR 7.7.1.22
5. USAR Table 7.7-2
6. CPS 3306.01, Source/Intermediate Range Monitors (SRM/IRM)

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS4

Initiating Condition:

Loss of all Vital DC power.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Loss of All Vital DC power based on **< 108 VDC** on 125 VDC battery busses 1A and 1B for **> 15 minutes**.

Basis:

Loss of all vital DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses. Failure of distribution busses such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. The DC power system consists of the following subsystems:

- Two non-Class 1E, 125-VDC subsystems that supply power to the generator air side seal oil pump, turbine DC emergency bearing oil pump, and other non-safety related 125 VDC loads.
- Four Class 1E, 125 VDC power subsystems that supply power to safety-related loads, two of which, are addressed by this EAL. Battery busses 1A and 1B are two electrically and physically separated divisions (Divisions 1 and 2, respectively) and power the bulk of the loads necessary for safe shutdown and core cooling. There are no divisional interconnections or bus crossties. CPS 4201.01, Loss of DC Power, gives a summary of Division 1 and 2 loads.

The Class 1E system design allows for the single failure or loss of any redundant DC subsystem during simultaneous accident and loss of offsite power conditions without adversely affecting safe shutdown of the plant. Only Division 1 and Division 2 125 VDC subsystems are required to be considered for safe shutdown analysis of the plant.

The ampere-hour capacity of each battery is adequate to supply expected essential loads for a period of 4 hours following station trip and loss of all AC power coincident with a design-basis accident without battery terminal voltage falling below 84% (105 VDC). The minimum design voltage limit of each battery is 108 VDC. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate loads.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS4 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS3
2. USAR 8.3.2
3. USAR Figure 8.3-7
4. USAR Table 8.3-5
5. USAR 8.3.2.1.1
6. CPS 4201.01, Loss of DC Power
7. Technical Specifications B3.8.4

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU4

Initiating Condition:

UNPLANNED loss of required DC power for greater than 15 minutes.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. UNPLANNED Loss of all required vital DC power based on **< 108 VDC** indication on 125 VDC battery busses 1A and 1B.

   AND

2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.

Basis:

<u>UNPLANNED</u>: A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

“Unplanned activities” is included in this EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, plants perform maintenance on a bus-related basis during shutdown periods.

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown, refueling or defueled modes. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses. Failure of distribution busses such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. The DC power system of consists of the following subsystems:

- Two non-Class 1E, 125-Vdc subsystems that supply power to the generator air side seal oil pump, turbine DC emergency bearing oil pump, and other non-safety related 125 VDC loads.
- Four Class 1E, 125 VDC power subsystems that supply power to safety-related loads, two of which, are addressed by this EAL. Battery busses 1A and 1B are two electrically and physically separated divisions (Divisions 1 and 2, respectively) and power the bulk of the loads necessary for safe shutdown and core cooling. There are no divisional interconnections or bus crossties. CPS 4201.01, Loss of DC Power, gives a summary of Division 1 and 2 loads.

The Class 1E system design allows for the single failure or loss of any redundant DC subsystem during simultaneous accident and loss of offsite power conditions without adversely affecting safe shutdown of the plant. Only Division 1 and Division 2 125 VDC subsystems are required to be considered for safe shutdown analysis of the plant.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU4 (cont)

Basis (cont):

The ampere-hour capacity of each battery is adequate to supply expected essential loads for a period of 4 hours following station trip and loss of all AC power coincident with a design-basis accident without battery terminal voltage falling below 84% (105 VDC).

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU7
2. USAR 8.3.2
3. USAR Figure 8.3-7
4. USAR Table 8.3-5
5. USAR 8.3.2.1.1
6. CPS 4201.01, Loss of DC Power
7. Technical Specifications B3.8.4

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS5

Initiating Condition:

Complete loss of heat removal capability.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Heat Capacity Temperature Limit (EOP-6, Fig. P) exceeded.

Basis:

Plant parameters associated with the Heat Capacity Limit (EOP-6 Fig. P) are RPV pressure, suppression pool level and suppression pool temperature. The Heat Capacity Limit is a function of RPV pressure and suppression pool temperature and is a measure of the maximum heat load that the containment can withstand.

The Heat Capacity Limit threshold (limit) for performing an Emergency Depressurization is designed to ensure that prompt actions (Emergency Depressurization) are taken such that the containment design limit (185^{o} F) is not exceeded. However, if the Emergency Depressurization is not successful (i.e., depressurize to < 50 psig) or performed at the Heat Capacity threshold (limit), then the containment design limit (185^{o} F) may be exceeded.

Exceeding the Heat Capacity Limit signals the loss of functions required to maintain hot shutdown. If compounded by further plant degradation, the event may challenge containment design temperature.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS4
2. CPS 4404.01, EOP-6, Primary Containment Control

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA5

Initiating Condition:

Inability to maintain plant in cold shutdown with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F** for > Table M1 duration.

Table M1 – RCS Reheat Duration Thresholds		
RCS	**Secondary Containment Closure**	**Duration**
Intact	N/A	60 minutes*
Not Intact	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. UNPLANNED RPV pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.

Basis:

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

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

Containment closure status is checked and verified using the applicable sections of CPS 3002.01, Heatup and Pressurization: Non-Nuclear Heat Up and CPS 3002.01C003, MODE 3 Checklist.

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 main steam line nozzle plugs, etc.)

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA5 (cont)

Basis (cont):

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as decay heat removal system design and RPV level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200° F). These include:

- Recirculation loop suction temperatures: 1B33-R604, Pts 1&2 (P614)
- Reactor vessel metal temperatures
- RWCU Inlet from Rx: G33DA011, G33NA011
- RHR HX Inlet Temperatures: 1E12-R601, Pts 1&2 (P601)
- DCS screens: 1B/C, 4B/E/C/G/J, 3G, EP Display 2

Threshold #1 Basis:

The first condition in Table M1 addresses complete loss of functions required for core cooling for greater than sixty minutes during Refueling and Cold Shutdown modes when RCS integrity is established. As in the second and third thresholds, RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., reactor head vents closed, reactor head on, no freeze seals or main steam line nozzle plugs, etc.). With containment closure established, a low-pressure barrier to fission product release exists. In this condition, containment status is of less importance than the status of RCS integrity because the RCS is intact and providing a high-pressure barrier to fission product release. The sixty-minute interval should allow sufficient time to restore cooling without a substantial degradation in plant safety. The asterisk highlights the note at the bottom of the table. The note indicates that the first condition is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the sixty-minute interval.

The second threshold in Table M1 addresses the complete loss of functions required for core cooling for greater than twenty minutes during Refueling and Cold Shutdown modes when containment closure is established but RCS integrity is not established or RPV inventory is reduced. RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., reactor head vents closed, reactor head on, no freeze seals or main steam line nozzle plugs, etc.).

Basis (cont):

The allowed twenty-minute interval is 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 barrier to fission product release is established (i.e., containment closure). The asterisk highlights the note at the bottom of the table. The note indicates that the second threshold is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the twenty-minute interval.

The third threshold in Table M1 addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when containment closure and RCS integrity are not established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., reactor head vents closed, reactor head on, no freeze seals or main steam line nozzle plugs, etc.). No delay time is allowed for this threshold because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary unplanned excursion above 200°F when the heat removal function is available.

Threshold #2 Basis:

The 10 psig pressure rise due to loss of decay heat removal infers an intact RCS with uncontrolled RPV temperature rise in excess of the Technical Specification cold shutdown limit (200° F) for which MA5 Threshold #1 would permit up to sixty minutes to restore RCS cooling before declaration of an Alert. This EAL therefore covers situations due to high decay heat loads, event should be declared without delay.

NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA5 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA4
2. Technical Specifications 3.6.1.1
3. Technical Specifications 3.6.4.1
4. OU-AA-103, Shutdown Safety Management Program
5. CPS 3002.01, Heatup and Pressurization
6. CPS 3002.01C003, MODE 3 Checklist
7. CPS 4006.01, Loss of Shutdown Cooling
8. CPS 9000.06, Reactor Coolant and Vessel Metal/Pressure/Temperature Limit Logs
9. CPS 9433.13, ECCS Reactor Steam Dome Pressure B21-N097A(B) Channel 6 Calibration

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU5

Initiating Condition:

UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F**.

   OR

2. Loss of all RCS temperature **AND** RPV level indication for **> 15 minutes**.

Basis:

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

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, 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. In Cold Shutdown, the decay heat available to increase RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the fuel cladding may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. 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 Reactor Vessel level so that escalation to the Alert under EAL MA5 will occur if required.

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in RCS/Reactor Vessel temperatures depending on the time since shutdown.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU5 (cont)

Basis (cont):

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 Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling modes, the second condition of this EAL would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication.

Reactor Vessel water level is normally monitored using the following instruments:

- Narrow Range (+60 in. to +5 in.)
- Wide Range (+60 in. to -160 in.) – Used for RPV level directed EOP/SAG actions whenever they are available. The bottom of Wide Range (-160 in. variable leg tap) is considered equivalent to the top of active fuel (-162 in.).
- Fuel Zone (-112 in. to -310 in.) – Used only for RPV level directed EOP/SAG actions when no other RPV level instruments are available.
- Upset Range (+180 in. to +0 in.) – Preferred use is when level is above the Narrow Range with the RPV pressurized and used to maintain RPV level below the main steam lines.
- Shutdown Range (+400 in. to +7 in.) – Preferred use is when the reactor is in cold shutdown and it is desired to maintain level above Narrow Range.

Detail A of EOP 1, RPV Control, and CPS 4411.07, RPV Level Instrumentation, provide guidance concerning when an instrument may be used for RPV level indication when EOPs are entered.

During shutdown conditions, the Shutdown Range and Upset Range are the primary instruments for monitoring RPV level as the RPV is flooded in preparation for vessel head removal and refueling operations. Plant procedure CPS 8117.01, Reactor Pressure Vessel Disassembly provides alternate level monitoring capabilities when the normal level instrumentation is unavailable for the desired level range or the head vent piping is removed. In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed.

Basis (cont):

Several instruments and computer points are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (200° F), such as:

- Recirculation loop suction temperatures: 1B33-R604, Pts 1&2 (P614)
- Reactor vessel metal temperatures
- RWCU Inlet from Rx: G33DA011, G33NA011
- RHR HX Inlet Temperatures: 1E12-R601, Pts 1&2 (P601)
- DCS screens: 1B/C, 4B/E/C/G/J, 3G, EP Display 2

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU4
2. Technical Specifications Table 1.1-1
3. CPS 4401.01, EOP-1 RPV Control
4. Clinton Power Station Emergency Operating Procedures Technical Bases, Section 12
5. CPS 4411.07, RPV Level Instrumentation
6. CPS 8117.01, Reactor Pressure Vessel Disassembly
7. CPS 3002.01, Heatup and Pressurization
8. CPS 9000.06, Reactor Coolant and Vessel Metal/Pressure/Temperature Limit Logs

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS6

Initiating Condition:

Inability to monitor a SIGNIFICANT TRANSIENT in progress.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of most **(approximately 75%)** safety system annunciators (Table M2).

Table M2 – Control Room Panels
• 1H13-P601 • 1H13-P877 • 1H13-P680 • 1H13-P801 (VC & VG)

AND

2. Indications needed to monitor safety functions (Table M3) are unavailable.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

AND

3. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Turbine trip • Reactor Scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10 %** Reactor Power change

AND

4. COMPENSATORY NON-ALARMING INDICATIONS (Computer Points) are unavailable.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS6 (cont)

Basis:

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25%, Reactor Power change or (5) thermal power oscillations >10% Reactor Power change.

Planned and unplanned actions are not differentiated since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not a factor.

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in 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.

A Site Area Emergency exists if the Control Room staff cannot monitor safety functions needed for protection of the public. 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 parameters should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled and in a coolable geometry, remove heat from the core, and maintain the reactor coolant system and containment intact. These parameters are monitored and controlled in the symptom-based emergency operating procedures (EOPs).

Symptoms of a loss of annunciators can be:

- ALARM POTENTIAL FAILURE or ANNUNCIATOR DC POWER FAILURE alarms on one or more panels
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

Station procedures provide instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS6 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS6
2. CPS 4401.01, EOP-1 RPV Control
3. CPS 4402.01, EOP-6 Primary Containment Control
4. USAR 7.7.1.26
5. CPS 3512.01, Display Control system (DCS/CX) & Performance Monitoring System
6. SPDS-DD-102, Safety Parameter Display System

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) A SIGNIFICANT TRANSIENT in progress, Or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 1H13-P601 • 1H13-P877 • 1H13-P680 • 1H13-P801 (VC & VG)

OR

b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

AND

2. a. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Turbine trip • Reactor Scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10 %** Reactor Power change

OR

b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA6 (cont)

Basis:

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

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations > 10% Reactor Power change

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in a 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the Reactor Control, ECCS, and Electrical panel annunciation or indication equipment. The availability of computer based indication equipment is considered.

Symptoms of a loss of annunciators can be:

- ALARM POTENTIAL FAILURE or ANNUNCIATOR DC POWER FAILURE alarms on one or more panels
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor
- Station procedures provide instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The 15 interval offers time to recover from transient or momentary power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA6 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA4
2. USAR 7.7.1.26
3. CPS 3512.01, Display Control system (DCS/CX) & Performance Monitoring System
4. SPDS-DD-102, Safety Parameter Display System

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 1H13-P601 • 1H13-P877 • 1H13-P680 • 1H13-P801 (VC & VG)

OR

2. UNPLANNED loss of most **(approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

Basis:

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

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the Reactor Control, ECCS, and Electrical panel annunciation or indication equipment. The availability of computer based indication equipment is considered.

MU6 (cont)

Basis (cont):

Symptoms of a loss of annunciators can be:

- ALARM POTENTIAL FAILURE or ANNUNCIATOR DC POWER FAILURE alarms on one or more panels
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

Station procedures provide instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The fifteen-minute interval offers time to recover from transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU3
2. USAR 7.7.1.26
3. CPS 3512.01, Display Control system (DCS/CX) & Performance Monitoring System
4. SPDS-DD-102, Safety Parameter Display System

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU7

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

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

   OR

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

Basis:

The conditions of this EAL threshold may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. Various indications may be used to identify or verify potential leakage from the RCS. They include monitoring: drywell pressure and temperature, airborne radioactivity rises, and the drain lines to the drywell equipment drain sumps from various potential leakage sources (i.e., containment pool seal drain flow, reactor recirculation pump seal drain flow, valve stem leakoff drain line temperatures, and reactor vessel head seal drain line pressure). Station procedures provide direction for determining RCS leakage.

The 10 gpm value for unidentified leakage was selected because it is observable with normal Control Room measurement of sump pumpout rates. It is consistent with the Technical Specification threshold for leaks beyond which increased risk of crack propagation exists.

The 25 gpm value for identified leakage is set at a higher value because of the significance of identified leakage in comparison to unidentified or pressure boundary leakage.

No classification under this threshold is made for relief valve operation or leakage.

Both threshold values are observable on Control Room instrumentation and do not require a mass balance calculation.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU7 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU5
2. USAR 5.2.5.1
3. CPS 9043.06, Drywell Floor Drain Sump Flow Test 00PS404
4. CPS 9443.01, Drywell Equipment Drain Sump Flow E31-N578 Channel Cal 01PS274
5. CPS 4401.01, EOP-1 RPV Control
6. CPS 4001.01, Reactor Coolant Leakage
7. Clinton Power Station Emergency Operating Procedures Technical Bases, Section 12
8. ITS 3.4.5

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG8

Initiating Condition:

Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Suppression Pool level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

AND

2. a. RPV level **< -162 in.** (TAF) for **> 30 minutes**.

   OR

   b. RPV level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following:

   - Containment High Range Monitors 1RIX-CM061 **OR** 1RIX-CM062 **> 3 R/hr** or off-scale high.
   - Erratic Source Range Monitor Indication.

AND

3. Containment is challenged as indicated by one or more of the following:
   - Drywell Hydrogen concentration ≥ **9%**.
   - Containment hydrogen concentration ≥ **Fig. R, Deflagration Limit.**
   - Containment pressure ≥ **15 psig.**
   - Primary and Secondary CONTAINMENT CLOSURE not established.
   - Any Secondary Containment radiation reading **> EOP-8 Table U Maximum Safe** operating level.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

Basis (cont):

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAG program.

Threshold #1 and #2 Basis:

This EAL represents the inability to restore and maintain RPV level to above the top of active fuel, -162 in. (TAF). Fuel damage is probable if core uncovery is prolonged and submergence cannot be restored and maintained. Available decay heat will cause boiling and further drop RPV level.

This EAL is 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, (e.g., decay heat removal system design, etc.) can have a significant affect on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

When RPV level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated drywell/containment radiation or unexplained rise in drywell floor or equipment drain sump pumpout rate. Detail A of EOP 1, RPV Control, and CPS 4411.07, RPV Level Instrumentation, provide guidance on determining if RPV level can be monitored. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors (SRM Channels A-D) can be used as a tool for making such determinations.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The Containment High Range Monitors 1RIX-CM061 OR 1RIX-CM062 > 3 R/hr or off-scale high indication is based on calculation EP-EAL-0501.

Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG8 (cont)

Basis (cont):

Threshold #3 Basis:

Four conditions are associated with the challenge to containment integrity:

- When hydrogen concentrations in drywell or containment reach or exceed the deflagration limits, imminent loss of the Primary Containment barrier exists. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred.
- The containment design pressure (15 psig) is well in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation.
- Containment Closure provides a barrier to the release of radioactivity to the environment. When this barrier is not established with prolonged core uncovery, the health and safety of the public may be threatened.
- The secondary containment area radiation level is the EOP-8 Maximum Safe Operating level. The Maximum Safe Operating radiation level is based on the highest radiation level at which neither equipment necessary for the safe shutdown of the plant will fail nor personnel access necessary for the safe shutdown of the plant will be precluded. The maximum safe operating radiation level is 25 Rem/hr for areas to which access is required and 400 Rem/hr for areas to which access is not required.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CG1
2. CPS 4401.01, EOP-1 RPV Control
3. CPS 4402.01, EOP-6 Primary Containment Control
4. Clinton Power Station Emergency Operating Procedures Technical Bases, Sections 9, 10 and 12
5. CPS 4406.01, EOP-8 Secondary Containment Control
6. USAR Figure 6.2-132
7. USAR Table 1.3-4
8. Technical Specifications 3.6.1.1
9. Technical Specifications 3.6.4.1
10. OU-AA-103, Shutdown Safety Management Program
11. Clinton Power Station Emergency Operating Procedures Technical Bases
12. CPS 4411.07, RPV Level Instrumentation
13. CPS 3306.01, Source/Intermediate Range Monitors (SRM/IRM)

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS8

Initiating Condition:

Loss of RCS/RPV inventory affecting core decay heat removal capability

Operating Mode Applicability:

4

EAL Threshold Values:

1. **Without** Primary or Secondary CONTAINMENT CLOSURE established:

    a. RPV level **< - 151 in.**

    OR

    b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Suppression Pool level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

OR

2. **With** Primary or Secondary CONTAINMENT CLOSURE established:

    a. RPV level **< - 162 in**. (TAF).

    OR

    b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory as evidenced by either of the following:

    - Per Table M5 indications.
    - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS8 (cont)

Basis (cont):

Threshold #1 Basis:

Under the conditions specified by this threshold, continued drop in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV. If a low-pressure boundary to fission product release does not exist (i.e., containment closure is not established), the RPV level associated with this threshold is six inches below the low-pressure ECCS actuation setpoint (i.e., -145.5 in. - 6 in. = -151.5 in.). If containment closure is established, a low-pressure boundary to fission product release exists and RPV level can drop to the top of active fuel, -162 in. (TAF), before a Site Area Emergency declaration is required. 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 drop and potential core uncovery.

In Cold Shutdown, the decay heat available to raise RPV temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding 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.

Threshold #2 Basis:

This threshold is 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, (e.g., decay heat removal system design, etc.) can have a significant impact on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

When RPV level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated drywell/containment radiation or unexplained rise in drywell floor or equipment drain sump pumpout rate. Detail A of EOP 1, RPV Control, and CPS 4411.07, RPV Level Instrumentation, provide guidance on determining if RPV level can be monitored. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors (SRM Channels A-D) can be used as a tool for making such determinations.

The thirty-minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CS1
2. Technical Specifications Table 3.3.5.1-1
3. ORM Attachment 2-7, Table 5
4. CPS 4401.01, EOP-1 RPV Control
5. Clinton Power Station Emergency Operating Procedures Technical Bases, Section 12
6. CPS 4411.07, RPV Level Instrumentation
7. USAR 5.2.5.1
8. CPS 9043.06, Drywell Floor Drain Sump Flow Test 00PS404
9. CPS 9443.01, Drywell Equipment Drain Sump Flow E31-N578 Channel Cal 01PS274
10. CPS 3306.01, Source/Intermediate Range Monitors (SRM/IRM)
11. Technical Specifications 3.6.1.1
12. Technical Specifications 3.6.4.1
13. OU-AA-103, Shutdown Safety Management Program
14. CPS 3002.01, Heatup and Pressurization

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA8

Initiating Condition:

Loss of RCS/RPV inventory with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. Loss of RCS/RPV inventory as indicated by RPV level **< - 145.5 in**.

   OR

2. a. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise
• Unexplained Suppression Pool level rise
• Unexplained vessel make up rate rise
• Observation of leakage or inventory loss

   AND

   b. RCS/RPV level unknown for **> 15 minutes.**

Basis:

This EAL serves as a precursor 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 drop and potential core uncovery. The low-pressure ECCS actuation setpoint is -145.5 in. below RPV instrument zero. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAG program.

In Cold Shutdown mode, the decay heat available to raise RPV temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding 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.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA8 (cont)

Basis (cont):

In Cold Shutdown mode, the RCS will normally be intact and standard RPV inventory and RPV level monitoring means are available. In the Refueling mode, the RCS is not intact and RPV level and inventory are monitored by different means. In the Refueling mode, normal means of RPV level indication may not be available.

Redundant means of RPV level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

RPV level is normally monitored using the following instruments:

- Narrow Range (+60 in. to +5 in.)
- Wide Range (+60 in. to -160 in.) – Used for RPV level directed EOP/SAG actions whenever they are available. The bottom of Wide Range (-160 in. variable leg tap) is considered equivalent to the top of active fuel (-162 in.).
- Fuel Zone (-112 in. to -310 in.) – Used only for RPV level directed EOP/SAG actions when no other RPV level instruments are available.
- Upset Range (+180 in. to +0 in.) – Preferred use is when level is above the Narrow Range with the RPV pressurized and used to maintain RPV level below the main steam lines.
- Shutdown Range (+400 in. to +7 in.) – Preferred use is when the reactor is in cold shutdown and it is desired to maintain level above Narrow Range.

In the second condition of this EAL, all RPV level indication would be unavailable. Detail A of EOP 1, RPV Control, and CPS 4411.07, RPV Level Instrumentation, provide guidance on determining if RPV level can be monitored. RPV inventory loss, therefore, must be detected by alternate means (i.e., drywell floor and equipment drain sump pumpout rates). Sump pumpout rate rises must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency duration.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA8 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA1 & CA2
2. Technical Specifications Table 3.3.5.1-1
3. ORM Attachment 2-7, Table 5
4. CPS 4401.01, EOP-1 RPV Control
5. CPS Emergency Operating Procedures Technical Bases, Section 12
6. CPS 4411.07, RPV Level Instrumentation
7. USAR 5.2.5.1
8. CPS 9043.06, Drywell Floor Drain Sump Flow Test 00PS404
9. CPS 9443.01, Drywell Equipment Drain Sump Flow E31-N578 Channel Cal

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU8

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

4

EAL Threshold Values:

RPV level **cannot** be restored and maintained **> Level 3 (8.9 in.).**

Basis:

The inability to restore and maintain level after reaching the RPS low level scram setpoint infers a degradation of the level of safety at the plant.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU1
2. Technical Specifications Table 3.3.5.1-1
3. ORM Attachment 2-7, Table 5
4. CPS 4100.01, Reactor SCRAM.
4. CPS 4401.01, EOP-1 RPV Control
5. CPS Emergency Operating Procedures Technical Bases, Section 12
6. CPS 4411.07, RPV Level Instrumentation
7. USAR 5.2.5.1
8. CPS 9043.06, Drywell Floor Drain Sump Flow Test 00PS404
9. CPS 9443.01, Drywell Equipment Drain Sump Flow E31-N578 Channel Cal

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS9

Initiating Condition:

Loss of RPV inventory affecting core decay heat removal capability with irradiated fuel in the RPV.

Operating Mode Applicability:

5

EAL Threshold Values:

1. **Without** Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< - 151 in**.

   OR

   b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
      - Containment High Range Monitors 1RIX-CM061 **OR** 1RIX-CM062 **> 3 R/hr** or off-scale high.
      - Erratic Source Range Monitor indication.

OR

2. **With** Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< - 162 in** (TAF).

   OR

   b. RPV level unknown with Indication of core uncovery as evidenced by one or more of the following:
      - Containment High Range Monitors 1RIX-CM061 **OR** 1RIX-CM062 **> 3 R/hr** or off-scale high.
      - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

Threshold #1 and #2 Basis:

Under the refueling conditions specified in this threshold, prolonged loss of the ability to monitor RPV level in conjunction with indirect indications of inventory loss infer a continued drop in RPV level and loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS9 (cont)

Basis (cont):

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAG program.

In the refueling mode, when RPV level indication is unavailable, the inventory loss must be detected by drywell floor and equipment drain sump pumpout rates or erratic Source Range Monitor indication. Detail A of EOP 1, RPV Control, and CPS 4411.07, RPV Level Instrumentation, provide guidance on determining if RPV level can be monitored. Sump pumpout rate rises must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The Containment High Range Monitors 1RIX-CM061 OR 1RIX-CM062 > 3 R/hr or off-scale high indication is based on calculation EP-EAL-0501.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CS2
2. Technical Specifications Table 3.3.5.1-1
3. CPS 4401.01, EOP-1 RPV Control
4. Clinton Power Station Emergency Operating Procedures Technical Bases, Section 12
5. CPS 4411.07, RPV Level Instrumentation
6. USAR 5.2.5.1
7. CPS 9043.06, Drywell Floor Drain Sump Flow Test 00PS404
8. CPS 9443.01, Drywell Equipment Drain Sump Flow E31-N578 Channel Cal
9. CPS 3306.01, Source/Intermediate Range Monitors (SRM/IRM)
10. Technical Specifications 3.6.1.1
11. Technical Specifications 3.6.4.1
12. OU-AA-103, Shutdown Safety Management Program
13. CPS 3002.01, Heatup and Pressurization

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU9

Initiating Condition:

UNPLANNED loss of RCS inventory with irradiated fuel in the RPV.

Operating Mode Applicability:

5

EAL Threshold Values:

1. UNPLANNED RPV level **< 204 in.** Shutdown Range (RPV flange) for **≥ 15 minutes**.

OR

2. a. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Suppression Pool level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

AND

b. RPV level unknown.

Basis:

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

Threshold #1 Basis:

The RPV flange is at 804 ft 4-1/16 in. el. or 203.5 in. above instrument zero. RPV level at this plant elevation is normally indicated by the Shutdown Range instrument (+7 in. to +400 in.). Plant procedures provide alternate level monitoring capabilities when the normal level instrumentation is unavailable for the desired level range or the head vent piping is removed. In order to expand the indicating range . In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed.

This threshold is applicable only in the Refueling mode and addresses loss of inventory to below the RPV flange during refueling operations. Refueling operations that drop RPV level below the RPV flange are carefully planned and procedurally controlled. An Unusual Event is appropriate because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant.

The fifteen-minute interval provides a reasonable time frame to restore level using one or more of the redundant means of refill that should be available. If RPV level cannot be restored in this interval, a more serious condition may exist.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU9 (cont)

Basis (cont):

Threshold #2 Basis:

In the second condition of this threshold, all RPV level indication would be unavailable. Detail A of EOP 1, RPV Control, and CPS 4411.07, RPV Level Instrumentation, provide guidance concerning when an instrument may be used for RPV level indication when EOPs are entered. RPV inventory loss, therefore, must be detected by alternate means (i.e., drywell floor and equipment drain sump pumpout rates). Sump pumpout rate rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RCS leakage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU2
2. CPS 4401.01, EOP-1 RPV Control
3. Clinton Power Station Emergency Operating Procedures Technical Bases, Section12
4. CPS 4411.07, RPV Level Instrumentation
5. CPS 8117.01, Reactor Pressure Vessel Disassembly
6. Technical Specifications Table 3.3.5.1-1
7. USAR Figure 3.8-31
8. USAR Table 7.1-13

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU10

Initiating Condition:

UNPLANNED loss of all onsite or offsite communications capabilities.

Operating Mode Applicability:

1, 2, 3, 4, 5

EAL Threshold Values:

1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations.

   OR

2. Loss of all Table M6 **Offsite** communications capability.

Table M6 - Communications Capability		
System	**Onsite**	**Offsite**
Plant Radio System	X	
Plant Paging System	X	
Sound Power Phones	X	
In-Plant Telephones	X	
PCS phones	X	X
All Telephone Lines (commercial and microwave)		X
NARS		X
ENS		X
Satellite Phones		X
HPN		X
Cellular Phones		X

Basis:

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

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems with offsite authorities or personnel. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

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

MU10 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU6 & CU6
2. EP-MW-124-1001, Facilities Inventories and Equipment Tests
3. UFSAR Section 9.5.2

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU11

Initiating Condition:

Inability to reach required shutdown within Technical Specification limits.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a prescribed shutdown mode when the Technical Specification configuration cannot be restored. The plant is within its safety envelope when being shut down 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 completion period elapses under Technical Specifications and is not related to how long a condition may have existed.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU2
2. Clinton Technical Specifications

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1

Initiating Condition:

Security event resulting in loss of physical control of the facility.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

A HOSTILE FORCE has taken control of:

1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1).

Table H1 - Safety Functions and Related Systems
• 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) • Fission Product Barriers

OR

2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).

Basis:

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.

Threshold #1 Basis

This threshold encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions. As a result, equipment control cannot be transferred to and operated from another location.

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

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.

Threshold #2 Basis

This threshold addresses loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely because there is freshly off-loaded fuel in the pool. The condition “freshly off-loaded reactor fuel in pool” equates to fuel off-loaded within the last 120 days in NF-AA-310 Special Nuclear Material and Core Component Movement.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HG1
2. CPS 4003.01, Remote Shutdown
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. NF-AA-310, Special Nuclear Material And Core Component Movement

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1

Initiating Condition:

Site attack.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A notification from the Site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert ICs (HA1 and HA2) in that a hostile force has progressed from the OWNER CONTROLLED AREA to the Protected Area.

Although Nuclear Power Plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for Offsite Response Organizations (ORO) 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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1 (cont)

Basis (cont):

This EAL is intended to address the potential for a very rapid progression of events due to a dedicated attack. It is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs. HOSTILE ACTION identified above encompasses various acts including:

- Air attack (LARGE AIRCRAFT impacting the protected area)
- Land-based attack (HOSTILE FORCE penetrating protected area)
- Waterborne attack (HOSTILE FORCE on water penetrating protected area)
- BOMBs breeching the protected area

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs.

This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for ORO to be notified and to activate in order to be better prepared to respond should protective actions become necessary. If not previously notified by NRC that the LARGE AIRCRAFT impact 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.

LARGE AIRCRAFT is meant to be an 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.

This EAL addresses the immediacy of a threat to impact site vital areas within a relatively short time. The fact that the site is under serious attack with minimal time available for additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HS4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition:

Notification of an airborne attack threat.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LARGE AIRCRAFT is meant to be an 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.

The intent of this EAL is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations (ORO) and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Alert. This EAL is met when a plant receives information regarding a LARGE AIRCRAFT attack threat from NRC and the LARGE AIRCRAFT is less than 30 minutes away from the plant.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from such an attack. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for OROs to be notified and encouraged to activate (if they do not normally) to be better prepared should it be necessary to consider further actions.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA7
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

Initiating Condition:

Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. A credible site-specific security threat notification as determined per SY-AA-101-132, “Threat Assessment.”

   OR

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

Basis:

Threshold #1 Basis

The intent of this threshold is to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat.

The determination of “credible” is made through use of information found in the Station Security Plan or SY-AA-101-132, “Threat Assessment” procedure.

Threshold #2 Basis

The intent of this threshold is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Unusual Event. This threshold is met when a plant receives information regarding an aircraft threat from NRC. Should the threat involve a LARGE AIRCRAFT (LARGE AIRCRAFT is meant to be an aircraft with the potential for causing significant damage to the plant), then escalation to Alert via HA1 would be appropriate if the LARGE AIRCRAFT is less than 30 minutes away from the plant. The status and size of the plane may be provided by NORAD through the NRC. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

Initiating Condition:

Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This EAL is intended to address the potential for a very rapid progression of events due to an attack including:

- Air attack (LARGE AIRCRAFT impacting the OCA)
- Land-based attack (HOSTILE FORCE progressing across licensee property or directing projectiles at the site)
- Waterborne attack (HOSTILE FORCE on water attempting forced entry or directing projectiles at the site)
- BOMBs

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Basis (cont):

This EAL is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne terrorist attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for Offsite Response Organizations to be notified and to activate in order to be better prepared to respond should protective actions become necessary.

If not previously notified by NRC that the LARGE AIRCRAFT impact 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. LARGE AIRCRAFT is meant to be an 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.

This IC/EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time. The fact that the site is an identified attack candidate with minimal time available for further preparation requires a heightened state of readiness and implementation of protective measures that can be effective (onsite evacuation, dispersal or sheltering) before arrival or impact.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA8
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS3

Initiating Condition:

Confirmed security event in a plant VITAL AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC (HA3).

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a Station’s security. Only those events that involve Actual or Likely Major failures of plant functions needed for protection of the public need to be considered. The following events would not normally meet this requirement; (e.g., Failure by a Member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Station Security Plan.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HS1
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

Initiating Condition:

Confirmed security event in a Plant PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Unusual Event.

Multi-unit stations with shared safety functions should further consider how this IC may affect more than one unit and how this may be a factor in escalating the emergency class.

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a station’s security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered. The following events would not normally meet this requirement; (e.g., failure by a member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU3

Initiating Condition:

Confirmed security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.

Basis:

Reference is made to Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

This threshold is based on Station Security Plan – Appendix C. 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.

Consideration should be given to the following types of events when evaluating an event against the criteria of the Station Security Plan: CIVIL DISTURBANCE, and STRIKE ACTION.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS4

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Control Room evacuation has been initiated.

   AND

2. Control of the plant **cannot** be established per CPS 4003.01 in **< 15 minutes**.

Basis:

The 15 minute time period starts when either:

a. Control of the plant is no longer maintained in the Main Control Room

   OR

b. The last Operator has left the Main Control Room.

The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. 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 determination of whether or not control is established outside of the Main Control Room is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the site-specific time for transfer that the licensee has control of the plant. Transfer of control to locations outside the Control Room is considered established when the Shift Manager has determined that the operators are capable of controlling reactivity, core cooling and heat sink functions.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HS2
2. CPS 4003.01, Remote Shutdown

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

Initiating Condition:

Control Room evacuation has been initiated.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Entry into CPS 4003.01 for Control Room evacuation.

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency operations centers are necessary. Procedure CPS 4003.01 Control Room Evacuation specifies conditions under which Control Room evacuation may be necessary.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA5
2. CPS 4003.01 Control Room Evacuation

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5

Initiating Condition:

Natural and destructive phenomena affecting the plant VITAL AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by seismic instrumentation **> 0.10 g**.

   AND

   b. Confirmed by **EITHER**:

   - Earthquake felt in plant.
   - National Earthquake Center.

   OR

2. Tornado or high winds **> 85 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

   OR

3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

   OR

4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of a Table H2 area.

   OR

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5 (cont)

EAL Threshold Value(s) (cont):

5. Uncontrolled flooding that results in **EITHER**:

   a. Degraded safety system performance in any Table H3 area as indicated in the Control Room.

Table H3 – Internal Flooding Areas
• Auxiliary Building • Fuel Building • Diesel Building • Control Building • Screenhouse • Turbine Building • Radwaste Building

   OR

   b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.

OR

6. High lake level **> 697 ft**.

Basis:

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

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

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

Threshold #1 Basis:

This threshold addresses events that may have resulted in a Table H2 being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this threshold to assess the actual magnitude of the damage.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Basis (cont):

This threshold is based on seismic ground acceleration in excess of 0.1 g for the USAR Operating Basis Earthquake (OBE). Seismic events of this magnitude are a factor of five greater that the Unusual Event threshold of EAL HU5 and can cause damage to plant safety functions.

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

Threshold #2 Basis:

This threshold addresses events that may have resulted in a VITAL AREA (Table H2) being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The Alert classification is appropriate if visible damage is observed and relevant plant parameters indicate that the performance of safety systems in these areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

The 85 mph threshold is the USAR design basis wind speed and is indicated on PPDS, computer point, or meter readings. Station structures are designed to withstand wind loads which may exist if sustained wind speeds reach or exceed 85 mph. Wind loads in excess of this magnitude can cause damage to safety functions. Sustained winds present a more severe loading on the buildings than a gust.

Wind speed measurement is provided by an instrument located on the station meteorological tower. The meteorological tower at Clinton Power Station is located outside of the PROTECTED AREA but within the OWNER CONTROLLED AREA; therefore, meteorological data provided is considered valid for applicability to the PROTECTED AREA classification thresholds.

Threshold #3 Basis:

This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes, or impact of projectiles into a Table H2 area. This threshold addresses vehicle crashes that challenge the operability of systems necessary for safe shutdown.

The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Table H2 areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Basis (cont):

Threshold #4 Basis:

This threshold covers threats to safety related equipment imposed by missiles generated by failure of the main turbine. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.

Threshold #5 Basis:

This threshold addresses the effect of internal flooding that has resulted in degraded performance of safety systems or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant.

“'Uncontrolled” as used in this threshold describes a condition where water is entering an area from an unplanned evolution. This flooding may have been caused by internal events such as component failures, equipment misalignment, and fire suppression system actuation or outage activity mishaps. Water entering an area, which resulted in degraded performance of safety systems within the area due to wetting or submergence, would meet the intent of this threshold. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding'. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if indications of degraded system performance is available or a shock hazard is known to exist.

The Internal Flooding Areas listed in Table H3 include areas containing systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Threshold #6 Basis:

Lake level rising above 697 ft el. exceeds the 100-year flood elevation. Prior to reaching this level, electrical equipment in the intake structure is deenergized and an orderly plant shutdown conducted. If not already shutdown prior to this elevation, a rapid shutdown is ordered and electrical equipment in the screen house deenergized.

A dam failure may result in failure of the 345-KV Rising and Latham power lines, threatening the availability of this offsite AC power supply.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA1
2. USAR 2.5.2, 3.3.1.1, 3.7, D3.6.4 and Appendix F
3. CPS 4301.01, Earthquake
4. CPS 4303.02, Abnormal Lake Level
5. CPS 4304.01, Flooding
6. Sargent & Lundy Report SL-4576 “Internal Flooding - Safe Shutdown Analysis and INPO SOER No. 85-5 Comparison Evaluation Report,” Clinton Power Station, January 31, 1990

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition:

Natural and destructive phenomena affecting the PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1 a. Seismic event as indicated by seismic instrumentation **> 0.02g**.

AND

b. Confirmed by **EITHER**:

- Earthquake felt in plant.
- National Earthquake Center.

OR

2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 85 mph**, within PROTECTED AREA boundary.

OR

3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse

OR

4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

OR

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

EAL Threshold Values (cont):

5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode.

Table H3 – Internal Flooding Areas
• Auxiliary Building • Fuel Building • Diesel Building • Control Building • Screenhouse • Turbine Building • Radwaste Building

OR

6. High lake level **> 696 ft**.

Basis:

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

Threshold #1 Basis:

This threshold is based on the Seismic Central Recording Unit actuation level which is the sensed earthquake threshold of 0.02 g. Seismic events of this magnitude are 1/5 of the Alert Event threshold of EAL HA5 in which it is assumed the earthquake can cause damage to plant safety functions.

The method of detection relies on the agreement of the shift operators on duty in the Control Room that the suspected ground motion is a “felt earthquake” as well as the actuation of the CPS seismic instrumentation. Consensus of the Control Room operators with respect to ground motion helps avoid unnecessary classification if the seismic switches inadvertently trip or detect vibrations not related to an earthquake. 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. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01 g.”

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis (cont):

Threshold #2 Basis:

This EAL is based on the assumption that a tornado striking (touching down) or design force winds (> 85 mph) within the Protected Area boundary may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The Protected Area boundary is within the security isolation zone and is defined in USAR Figure 2.1-7 Sheet 2, Nuisance Fence and Perimeter Boundary Fence. Verification of a tornado is obtained by direct observation and reporting by station personnel. “Sustained” wind speeds exist for 15 minutes or longer. Wind speed is obtained from meteorological data in the Control Room.

Threshold #3 Basis:

In this context, a "vehicle crash" is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

Threshold #4 Basis:

This threshold is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this threshold to classify minor operational leakage.

Threshold #5 Basis:

“Uncontrolled” as used in this threshold describes a condition where water is entering the area from an unplanned evolution. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if a potential to affect safety related equipment needed for the current operating mode exists.

This threshold addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, fire suppression system actuation or outage activity mishaps. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding.”

The Internal Flooding Areas listed in Table H3 include areas containing systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis (cont):

Threshold #6 Basis:

A lake level of 696 ft el. is one foot below the 100-year flood elevation and represents a decision point; based on rate of lake level rise, and expected level crest, to shutdown the plant in order to assure safe plant shutdown (MODE 3) prior to lake level reaching 697 ft.

A dam failure may result in failure of the 345-KV Rising and Latham power lines, threatening the availability of this offsite AC power supply. If there is potential for dam failure, the dam tender should be performing inspections per the CPS Main Dam - EAP Section 2.4 criteria. In addition, the following actions may be required:

- Ensure the Brokow line is available.
- Restore all DGs to available status.
- Shift safety-related busses to the 138-KV source.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU1
2. USAR 2.5.2
3. USAR 3.7
4. CPS 4301.01, Earthquake
5. USAR 3.3.1.1
6. USAR Appendix F
7. USAR Figure 2.1-7 Sheet 2
8. CPS 4303.02, Abnormal Lake Level
9. USAR D3.6.4
10. CPS 4304.01, Flooding
11. Sargent & Lundy Report SL-4576 “Internal Flooding - Safe Shutdown Analysis and INPO SOER No. 85-5 Comparison Evaluation Report,” Clinton Power Station, January 31, 1990

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6

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, D

EAL Threshold Values:

1. FIRE or EXPLOSION in any Table H2 area.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse

AND

2. a. Affected safety system parameter indications show degraded performance.

   OR

   b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.

Basis:

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.

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

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6 (cont)

Basis (cont):

The areas listed in Table H2 house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Personnel access to these areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses fires and EXPLOSIONS that challenge the operability of systems necessary for safe shutdown of the plant.

The only fires and EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment required for safe shutdown. Visual observation of damage infers the ability to approach or enter the affected areas. Lacking the ability to adequately inspect the area for damage, the Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this EAL. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

A steam line break or steam EXPLOSION that damages permanent structures or equipment in one of these areas would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA2
2. USAR Appendix F

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

OR

2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse

OR

3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

Basis:

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

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.

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Basis (cont):

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

Thresholds #1 and #2 Basis:

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins with a credible notification that a fire is occurring or indication of a valid fire detection system alarm. A verified alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm.

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 EAL HA7, Release of Toxic or Flammable Gases. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under EAL HA7.

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

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket, etc.). Such fires are excluded from consideration in this threshold since they have no safety consequence.

Threshold #3 Basis:

The only EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment in the PROTECTED AREA.

A steam line break or steam EXPLOSIONS that damages permanent structures or equipment in a PROTECTED AREA would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU2
2. USAR Figure 1.2-3, Principal Station Structure
3. USAR Appendix F

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA7

Initiating Condition:

Release of toxic, or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).

Table H2 – Vital Areas
• Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse

Basis:

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

Values for LFL for common gases at Clinton Station are:

- Propane – 2.2% (BOC Gasses MSDS)
- Hydrogen – 4% (Air Liquide Safety Data Sheet)
- Acetylene – 2.2% (BOC Gasses MSDS)

Basis (cont)

This EAL is based on toxic, asphyxiant, or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Toxic or flammable gases detected outside of these areas need not be considered for this EAL unless there is a spread of the gasses into one of these areas.

Threshold #1:

Declaration should not be delayed for confirmation from atmospheric testing if it is reasonable to conclude that the IDLH concentrations have been met (e.g., documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards).

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 this EAL. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under this EAL.

The first condition is met if measurement of toxic gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a Table H2 area. Non-Toxic Gases which displace oxygen (site examples; Halon or Nitrogen) to a life threatening level due to asphyxiation (oxygen deprivation) should also be considered for this EAL.

An Asphyxiant is a material 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.

Threshold #2:

The second condition is met when the flammable gas concentration in a Table H2 area exceeds the lower flammability limit. Flammable gases such as hydrogen and acetylene are routinely used to maintain plant systems (hydrogen – main generator cooling, reactor coolant chemistry control) or repair equipment/components (acetylene - welding). This condition addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a Table H2 area has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage or personnel injury. Once it has been determined that an uncontrolled release of flammable gas is occurring, sampling must be done to determine if the gas concentration exceeds the lower flammability limit.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA3
2. USAR Appendix F

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU7

Initiating Condition:

Release of toxic or flammable gases deemed detrimental to normal operation of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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:

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.

This EAL is based on the existence of uncontrolled releases of toxic, asphyxiant, or flammable gas affecting plant operations or the health of plant personnel. The release may have originated within the Protected Area boundary, or it may have originated offsite and subsequently drifted inside the Protected Area boundary. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on normal plant operations.

It is intended that releases of toxic, asphyxiant, or flammable gases are of sufficient quantity and the release point of such gases is such that safe plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for safe plant operation. The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect safe plant operations or plant personnel safety.

An Asphyxiant is a material 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.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU3
2. USAR Appendix F

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels (≥ 1 Rem TEDE or ≥ 5 Rem CDE Thyroid) outside the site boundary.

Basis Reference(s):

1. NEI 99-01, Rev 4 HG2
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or 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 that exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

Basis Reference(s):

1. NEI 99-01, Rev 4 HS3
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA6
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

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:

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency operating procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU5
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)

Section 4: Emergency Measures

4.1 Notification of the Emergency Organization

Standard NARS notifications for the Clinton Station are made to the State of Illinois Emergency Management Agency (IEMA). If a General Emergency is the initiating event, the Emergency Director is also responsible for notifying the following local agencies:

- DeWitt County Sheriff /ESDA

4.2 Assessment Actions

Throughout each emergency situation, continuing assessment will occur. Assessment actions at Clinton Station may include an evaluation of plant conditions; inplant, onsite, and initial offsite radiological measurements; and initial estimates of offsite doses.

Core damage information is used to refine dose assessments and confirm or extend initial protective action recommendations. Clinton Station utilizes NEDC-33045P-A, Revision 0, (2001) as the basis for the methodology for post-accident core damage assessment. This methodology utilizes real-time plant indications. In addition Clinton Station may use samples of plant fluids and atmospheres as inputs to the CDAM (Core Damage Assessment Methodology) program for core damage estimation.

4.3 Protective Actions for the Offsite Public

Protective actions concerning the public within the 10 mile EPZ involve prompt notification, evacuation and sheltering. Prompt notification involves primarily the use of the permanently installed outdoor notification sirens located within the EPZ.

To aid Main Control Room personnel during a rapidly developing emergency situation, Figure 4-1, "Protective Action Recommendation (PAR) Determination Flowchart for Clinton Station" has been developed based on Section J.10 of the Exelon Nuclear Radiological Emergency Plan.

4.3.1 Alert and Notification System (ANS) Sirens

The alert and notification system consists of a permanently installed outdoor notification system within the zero (0) to ten (10) mile radius around the station. The zero (0) to ten (10) mile radius around the station is primarily an agricultural area with a population density below 2000 persons per square mile. The alert and notification system as installed consists of mechanical and electronic sirens that will cover this entire area with a minimum sound level of 60 db. Additionally, the prompt notification system will cover the heavily populated areas within the zero (0) to ten (10) mile radius around the station with a minimum sound level of 70 db to ensure complete coverage.

Once the public has tuned to designated radio stations in an emergency, detailed instructional messages will be given to the public. State and local procedures provide for these messages.

4.3.2 Evacuation Time Estimates

The evacuation time estimates were developed per the requirements of NUREG-0654, and to support the Illinois Plan For Radiological Accidents (IPRA) - Clinton Station Volume VIII. The purpose of the evacuation time estimates is to assess the postulated evacuation times for the Clinton Station Emergency Planning Zone (EPZ) for the 10-Mile Plume Exposure Pathway.

The evacuation time estimate data was updated per a study performed by Earth Tech. Inc. documented in their report dated June, 2005 entitled “Update of Evacuation Time Estimates for the Plume Exposure Pathway Emergency Planning Zone for Clinton Nuclear Generating Station.” The ETE study was revised and reissued in January, 2006 with minor changes to several of the evacuation times.

An updated set of evacuation time estimates (ETEs) for Clinton Generating Station has been developed, using population data from the 2000 Census. The assumptions and analysis procedures for the present study closely followed the approach of the previous (1993) study . A full ETE update study was not judged to be necessary for Clinton Station. Comparison of the 1990 and 2000 Census data showed relatively small changes in population within the Emergency Planning Zone (EPZ), and no significant changes to the roadway network have occurred since 1993 . This “partial” update study was therefore performed using data from the 1993 study to characterize the roadway network and the populations for special and transient facilities; only the population numbers for permanent and seasonal residents have been updated.

The evacuation times are based on a detailed consideration of the EPZ roadway network and population distribution. The information in Table 4-1 presents representative evacuation times for daytime and nighttime scenarios under various weather conditions for the evacuation of various areas around the Clinton Station, once a decision has been made to evacuate. The evacuation times noted include notification, mobilization, and travel time. These times are for the general population which include permanent population and special facilities (schools, nursing homes, hospitals, and recreational areas).

Table 4-2 provides information on the scenario population distribution that was used for this study. Figure 4-2 provides a representation of the Subarea Locations in relation to the EPZ.

4.4 Protective Actions for Onsite Personnel

Clinton Station has a plant alarm system to warn personnel of emergency conditions. Upon hearing a continuous two (2) minute alarm, or receiving notification by other means of communication, all personnel not having emergency assignments have been instructed to assemble in a predesignated assembly area. The onsite assembly areas (Figure 4-3) are located in the:

- Outage Control Center (OCC), and
- 762' elevation of the Radwaste Building.

Accountability of site personnel is accomplished by the Station Security force.

If a site evacuation of non-essential personnel is required, personnel will be released to their homes or relocated and monitored at one of the following designated relocation centers for CPS are:

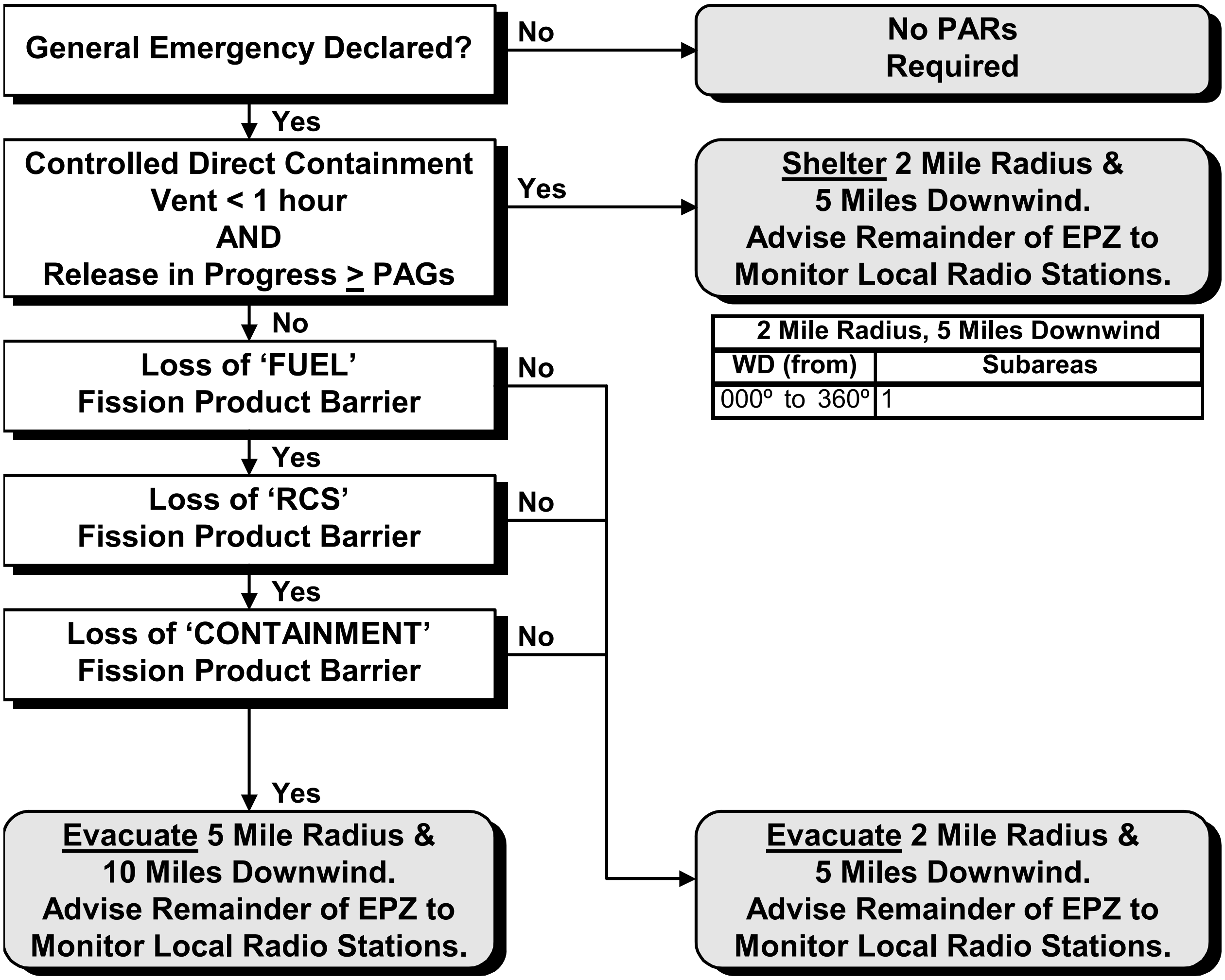
- ISU Horton Field House, Normal, Illinois
- Monticello High School, Monticello, Illinois
- Richland Community College, Decatur, Illinois

For evacuation routes, refer to EP-AA-113-F-22.

Traffic control for onsite areas will be the responsibility of the Station Security force. When a site evacuation is imminent, the Station Security force will post guards as necessary to assist in the evacuation.

Equipment and personnel would be available at all three locations for monitoring and decontamination of evacuated personnel. If major decontamination, follow-up or bioassay samples are necessary, those persons would be sent to either the Dresden or Braidwood Stations.

Figure 4-1: Clinton Station PAR Determination Flowchart



2 Mile Radius, 5 Miles Downwind	
WD (from)	**Subareas**
000° to 360°	1

5 Mile Radius, 10 Miles Downwind	
WD (from)	**Subareas**
021° to 048°	1, 5, 6
049° to 066°	1, 6
067° to 090°	1, 6, 7
091° to 094°	1, 7
095° to 132°	1, 7, 8
133° to 157°	1, 2, 8
158° to 196°	1, 2
197° to 228°	1, 2, 3
229° to 251°	1, 3
252° to 281°	1, 3, 4
282° to 308°	1, 4
309° to 338°	1, 4, 5
339° to 020°	1, 5

2 Mile Radius, 5 Miles Downwind	
WD (from)	**Subareas**
000° to 360°	1

Figure 4-2: Clinton Subarea Locations

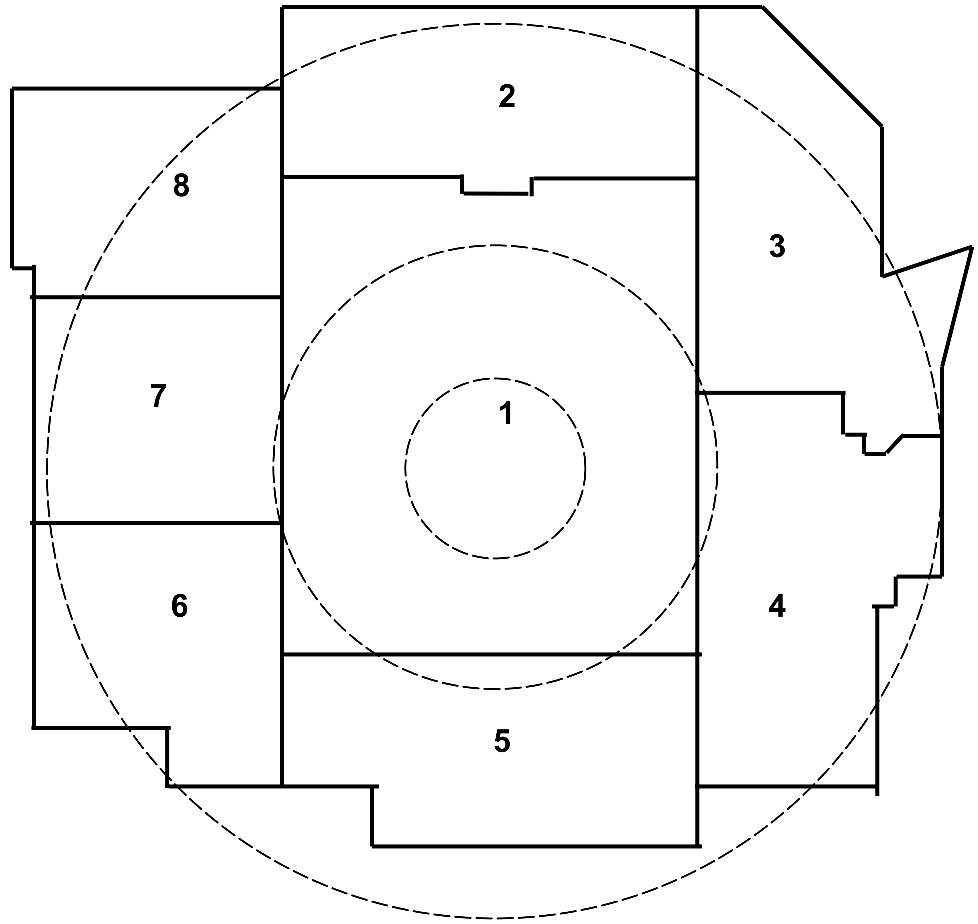


Figure 4-3: Onsite Assembly Areas

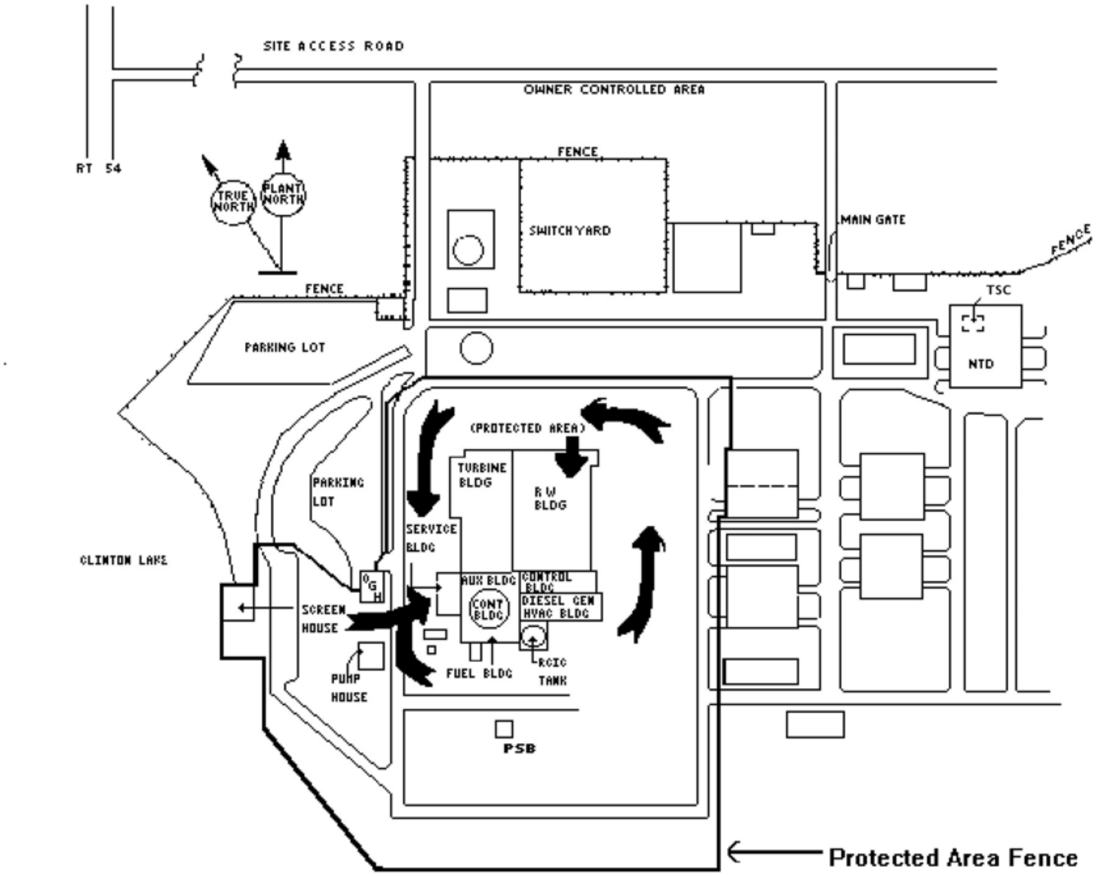


Table 4-1: Evacuation Time Estimates

General Population Evacuation Times (minutes)

PAR Evacuation Zone		**Summer Day**		**Summer Weekend**		**Winter Day**		**Winter Night**	
Wind Direction	**Sub-areas**	Normal weather	Adverse weather	Normal weather	Adverse weather	Normal weather	Adverse weather	Normal weather	Adverse weather
2 mile Radius, 5 Miles Downwind									
All	1	180	180	170	180	180	190	170	170
5 mile Radius, 10 miles Downwind									
021 to 048	1, 5, 6	180	190	180	190	180	200	170	180
049 to 066	1, 6	180	190	180	190	180	200	170	180
067 to 090	1, 6, 7	195	220	185	220	185	230	185	200
091 to 094	1, 7	195	220	185	220	185	230	185	200
095 to 132	1, 7, 8	195	220	185	220	185	230	185	200
133 to 157	1, 2, 8	180	190	170	180	180	200	170	170
158 to 196	1, 2	180	190	170	180	180	200	170	170
197 to 228	1, 2, 3	190	190	170	180	185	200	170	180
229 to 251	1, 3	190	190	170	180	185	200	170	180
252 to 281	1, 3, 4	190	200	180	190	185	200	170	180
282 to 308	1, 4	190	200	180	190	185	200	170	180
309 to 338	1, 4, 5	190	200	180	190	185	200	170	180
339 to 020	1, 5	180	190	180	190	180	200	170	170
Entire EPZ		195	220	185	220	185	230	185	200

Table 4-2: Clinton Scenario Population Distribution By Subarea

	Winter Day		Winter Night		Summer Day		Summer Weekend	
Subarea	Population	Vehicles	Population	Vehicles	Population	Vehicles	Population	Vehicles
1	3,790	1,961	2,015	930	8,459	3,512	12,044	4,171
2	211	86	211	86	211	86	211	86
3	627	245	477	195	527	213	527	213
4	460	121	174	71	178	75	174	71
5	181	74	181	74	181	74	181	74
6	1,307	513	1,032	422	2,597	916	5,837	1,979
7	11,997	4,811	8,871	3,821	9,922	4,568	8,930	3,687
8	1,280	456	1,004	411	1,618	553	1,034	421
EPZ	19,853	8,268	13,965	6,012	23,693	9,998	28,938	10,703

Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities

5.1.1 Station Main Control Room

The Main Control Room is the initial onsite center of emergency control and is located on the 800 foot elevation of the Control Building.

5.1.2 Technical Support Center (TSC)

Clinton Station has a designated TSC on the first floor of the Nuclear Training Building on the east side of the site. Standard air sampling equipment is used to monitor air-borne radioactivity levels in the TSC. The TSC fully meets the requirements of Section H.1.b of the E-Plan.

5.1.3 Operational Support Center (OSC)

A designated Operational Support Center (OSC) is located in the Outage Control Center (OCC) in the Service Building. The OSC conforms to the requirements of Section H.1.c of the Exelon Nuclear Radiological Emergency Plan and is the location which operations support personnel will report during an emergency and from which they will be dispatched for assignments in support of emergency operations.

In the event that the OSC has to be abandoned, OSC personnel and functions shall be relocated to the Machine Shop on the 737' elevation of the Radwaste Building.

5.2 Assessment Resources

5.2.1 Onsite Meteorological Monitoring Instrumentation

Clinton Station shall maintain meteorological instrumentation to ensure that sufficient meteorological data is available. This system shall provide measurements and calculations for the following parameters:

- Wind direction and speed at the 10-meter and 60-meter levels
- Standard deviation of wind direction fluctuations at all measured levels
- Vertical temperature difference for at least one layer (50 meters)
- Ambient temperature (10 meters and 60 meters)
- Dew point temperature (10 meters)
- Pasquill stability class used for diffusion estimates

A tower 199 feet high with two levels of instrumentation has been erected with the base at 735 feet above mean sea level. There are no trees, tall obstructions or significant topographical features in the immediate vicinity of the tower.

The tower is instrumented at the 33 foot (10 meter) and 198 foot (60 meter) levels. Heating and ventilation are thermostatically controlled in the equipment building located at the base of the tower to provide a controlled environment for the signal translating equipment.

Meteorological parameters measured are transmitted to the 781-foot level of the control building via a dedicated telephone line. There the signals are received and converted to electrical signals, and fed individually to a microprocessor and chart recorders. The microprocessor is part of the CPS Process Radiation Monitoring System. This system calculates 10-minute averages of the meteorological parameters, and stores hourly averages on floppy disk. The meteorological system shall be equipped with instrumentation and operated by procedures to maximize the availability of meteorological data.

An emergency generator with auto-transfer switch has been installed to supply electric service upon loss of the distribution circuit. Surge suppressors have been installed on the power circuits at the tower and the equipment building. Recorders shall be located in the Main Control Room. Digital information shall be available through CRT output in the Main Control Room, TSC, and EOF. A magnetic tape or other electronic medium shall be in use to archive the data. The capability shall exist for retrieving meteorological data for input to the dose assessment model should any component of the data acquisition system fail.

A backup meteorological tower is located at the CPS site and is instrumented at the 33 foot (10 meter) level. The meteorological parameters measured are wind speed and wind direction. Sigma Theta is calculated from the changes in wind direction. All three of these variables are available in the Main Control Room envelope and are read from a line printer. The average period may be changed by the operator as necessary.

Administrative arrangements have been made with the National Weather Service (NWS) office in Lincoln, Illinois, to provide CPS with meteorological measurements and forecast information on a 24-hour basis, if requested. This letter of agreement is maintained on file. These measurements are representative of CPS meteorology due to the homogeneity of the local terrain. Thus, the NWS data will serve as a back-up to the meteorological data measurements.

5.2.2 Onsite Radiation Monitoring Equipment

Clinton Station shall maintain various radiological monitoring systems that will monitor processes, areas and effluents. The constant air monitors (CAM) shall provide ambient air monitoring for detecting airborne particulate radiation, iodine, and noble gases in Station areas or cubicles. Area radiation monitoring (ARM) instruments provide a local visual and audible alarm if their high radiation set points are exceeded. The process radiation monitors (PRM) provide monitoring of Station HVAC exhaust, standby gas treatment, pre- and post-treatment air ejector off-gas, Station service water, shutdown service water, and liquid radwaste discharge effluents.

These radiological monitoring systems ensure that sufficient radiological data are available for estimating the danger to personnel and the public as a result of an incident or abnormal occurrence. Further, an alarm and/or automatic action is initiated when the setpoint of the equipment is exceeded. Portable survey instruments are identified in Table 12.5.2 of the USAR and are available for in-plant and offsite monitoring.

A computer network located in the Main Control Room provides an operator interface with select field units of the RMS. The Central Server of the computer network polls select field units and provides the radiation/radioactivity levels, alarm status, and monitor status to other computers within this network.

Hardwired input from the Accident Range system stack monitors provide radiological control alarm status information to the SPDS display for purposes of concise monitoring of this critical safety function.

5.2.2.1 Area Radiation Monitoring System (AR System)

These are three types of area radiation monitors (ARM) in the AR system:

1) Analog ARM
2) Fixed digital ARM
3) Portable digital ARM

5.2.2.2 Analog Area Radiation Monitors

There are analog ARMs on each of the fuel handling platforms and on the containment polar crane. Each monitor has a single GM detector. These monitors are independent from the rest of the RMS and are provided for the operators' safety. There are also associated interlocks on the lifting mechanisms on the fuel handling platforms.

5.2.2.3 Fixed Digital Area Radiation Monitor

There are numerous fixed ARMs throughout the Station utilizing GM detectors. ARMs have a range of 10^{-1} to $2.2 \times 10^{+3}$ mR/hr. The microprocessor associated with each ARM is designed to accept input from a second, high range detector with a range from to 10 to 10^{+4} R/hr. These additional, high range detectors can be added, as necessary. Select micro-processors communicate directly with the Main Control Room RMS Unit. Local indication and annunciation are also provided on each individual microprocessor. All ARMs have integral battery power backup which can provide eight hours of operation.

5.2.2.4 Portable Digital Area Radiation Monitor

The portable digital ARMs are identical to the fixed digital ARMs. An ARM can be used as a stand alone monitor as long as a 120 VAC power supply is available. The portable ARMs can be tied into the communication network and communicate directly with the CR RMS Computer Network.

5.2.2.5 Process Radiation Monitoring System (PR System)

Certain Station processes are monitored to detect radiation/radioactivity in excess of acceptable limits. The PR system consists of 4 types of monitors:

a) off-line liquid sampling monitors

b) off-line gas sampling monitors

c) constant air monitors (CAM)

d) ventilation duct monitors

5.2.2.6 Off-line Liquid Sampling Monitors

The off-line liquid sampling monitors monitor process lines which are either direct release paths to the environment or are used to detect inter-system leakage. The detector is a sodium iodide gamma scintillator. The monitors draw a sample of the process liquid, measure the radioactivity, and normally return the sample to the process stream. The monitors provide local and remote indication of radioactivity levels in the process streams and provide alarms when predetermined levels are exceeded.

5.2.2.7 Off-line Gas Sampling Monitor

The off-line gas sampling monitor functions in the same manner as a liquid monitor except the sampled media is a gas from the process stream. The monitor utilizes a GM detector. The pretreatment air ejector off-gas monitor is an example of a monitor in the PR system that falls into this category. This monitor monitors the air ejector off-gas system downstream of the air ejectors and prior to the charcoal adsorbers.

5.2.2.8 Constant Air Monitors (CAM)

There are two types of CAMs,

a) Fixed digital CAM

b) Portable digital CAM

CAMs are provided to monitor the ambient air surrounding the monitor or to monitor gases in a ventilation duct or process stream. Each CAM contains a fixed particulate filter, iodine collection charcoal cartridge and the associated equipment to draw and maintain a constant sample flow. Each CAM, with the exception of the standby gas treatment system (SGTS) PRM and the common station heating, ventilation, air conditioning (HVAC) exhaust PRM, and Post Treat Off Gas PRM contain the following detectors:

a) Three detectors for measuring airborne radioactivity:
   1) Particulate: beta-scintillation detector
   2) Iodine: sodium iodide gamma-scintillation detector, gain stabilized
   3) Noble gas: beta-scintillation detector

b) Two detectors measuring background radiation for subsequent subtraction for the appropriate channels:
   1) Gamma (external): GM tube detector
   2) Alpha (naturally occurring Rn and Th): alpha - scintillation detector

The SGTS, HVAC, and Post Treat Off Gas PRM contains an additional noble gas channel.

The ventilation air discharge from various buildings is continuously monitored for radioactivity in the air. The two gaseous discharge paths are the common station HVAC and SGTS stack. The monitoring of these effluents provides a record of the gross radioactivity discharged through these paths into the environs. Post treat monitors provide a record of gross radioactivity downstream of the charcoal bed.

The portable digital CAMs are identical to the fixed digital CAMs with the exception of a strip chart recorder and communication capability.

5.2.2.9 Ventilation Duct Monitors

With the exception of Main Control Room Air Intake PRM, the ventilation duct monitors monitor gross gamma radioactivity in the ventilation system. Each ventilation duct monitor consists of four directionally shielded GM tubes oriented such that they monitor the radiation level inside the ducts. Each GM tube has its own microprocessor. An alarm by one channel in each division initiates an isolation signal if radiation levels exceed a predetermined value.

The Main Control Room Air Intake PRM has two unshielded GM detectors on each of the air intakes. With the above exception, the Main Control Room Air Intake PRMs are identical to the remainder of the ventilation duct monitors. The Main Control Room habitability is discussed in the USAR.

5.2.2.10 Main Steam Line Radiation Monitors (MSLRM)

The main steam lines located in the steam tunnel (downstream of outer isolation valves) between the nuclear reactor and the main turbine are monitored continuously for gamma radiation for the purpose of detecting increased radiation levels caused by gross fuel failures This system is separate from the PR system.

5.2.2.11 High-Range Containment Radiation Monitoring (Post-Accident Primary Containment Atmosphere and Gross Gamma Monitoring)

The purpose of the containment atmosphere and gross gamma monitoring system is to provide the signals necessary to indicate and alarm high hydrogen concentration, high oxygen concentration, or high gross gamma radiation in the drywell following a loss-of-coolant-accident (LOCA).

The containment atmosphere monitoring subsystem monitors hydrogen and oxygen in the drywell resulting from radiolytic and chemical phenomena associated with an accident condition. The gross gamma monitoring subsystem, consisting of two high range (1 R/hr to 10^8 R/hr) containment radiation detectors, monitors gamma radiation resulting from the gross release of fission products from the reactor fuel. Each subsystem has two redundant channels of instrumentation that are physically separated and electrically independent. Each channel provides a local measurement and transmits the signals to the Main Control Room where a permanent record is made on seismically qualified recorders.

5.2.2.12 Station Survey and Counting Equipment

The Station Counting Room contains Germanium gamma spectrometer systems and gas-flow proportional counters for alpha and beta analysis. An alternate power supply for the Counting Room is provided from an essential power bus. An alternate (HPGe) spectrometer system is available on the Auxiliary Building 768 elevation, should the Station Counting Room become unavailable for use. Additionally, during emergency situations, samples may be taken to an alternate facility.

5.2.3 Onsite Process Monitors

There are many methods available to Control Room personnel to monitor critical reactor and Station parameters. These parameters, regardless where they are monitored, can be used by Control Room personnel to assess abnormal Station conditions and, based on these indications and their trained judgment, declare and classify emergencies as conditions dictate. A description of the process monitors used to initiate emergency conditions is found below:

a) Vessel Pressure - Reactor vessel pressure is monitored and indicated in four ranges:

   1. For monitoring vessel pressure during normal operating conditions

      Range: 0-1200 psig

   2. Narrow range monitoring of reactor pressure during power operations

      Range: 850-1050 psig

   3. Wide range monitoring of reactor pressure during pressure transients

      Range: 0-1500 psig

   4. Narrow range monitoring of reactor pressure during accident/transient conditions.

      Range: 0-300 psig

b) Vessel Temperature - The reactor vessel temperature is measured in four areas:

   1. Vessel Bottom Head
   2. Vessel Head Flange
   3. Bottom Head Drain
   4. Shell Flange

c) Reactor Water Level - There are five ranges available to measure reactor water level. The five types of reactor water level instrumentation are described below:

1. Narrow Range
    - (a) Range: 0" to +60"
    - (b) Used for feedwater control level inputs and is most precise indication of normal water level.
    - (c) Calibrated to read correctly at normal operating temperature and pressure.
2. Wide Range
    - (a) Range: -160" to +60"
    - (b) Provides ECCS and Reactor Protection System actuation and/or trip signals.
    - (c) Calibrated at normal operating temperature and pressure.
3. Shutdown Range
    - (a) Range: 0" to +400"
    - (b) Used for following level during flood up.
    - (c) Calibrated to read correctly when cold (120F) and 0 psig.
4. Upset Range
    - (a) Range: 0" to +180"
    - (b) Used following abnormal level increases during transient conditions.
    - (c) Calibrated to read correctly at normal operating temperature and pressure.
5. Fuel Zone
    - (a) Range: Information Scale, -150" to +50" (Referenced to the top of active fuel); Second Scale Common Referenced to Wide Range, -310” to –110”.
    - (b) Used by operators during accident/transients to take emergency operating procedure actions.

d) Flow Rates - Flow rates are monitored at many points in the reactor. The following is a list of critical flow rates available to the operator:

Recirculation Loop Flow
Total Core Flow
Steam Line Flow
Total Steam Line Flow
ECCS Injection Flows
Feedwater Flow

e) Containment/Drywell Temperature and Pressure - Temperature and pressure data for the containment and drywell are available to the Main Control Room operators.

5.2.4 Onsite Fire Detection Instrumentation

The fire protection system is designed to provide an adequate supply of water or other chemicals to points throughout the plant where fire protection may be required. Diversified fire-alarm and fire-suppression type systems are selected to suit the particular areas being protected or the hazards which could be encountered. The fire protection water is drawn from the ultimate heat sink that is sized to include 900,000 gallons of water for fire protection. The fire protection system consists of two 100% capacity diesel-driven fire pumps (primary fire protection system water supply), one connection to the plant service water, a dedicated pressure maintenance jockey pump, and the associated piping, valves, and hydrants.

Chemical fire-fighting systems, such as CO2 and Halon 1301, are also provided in areas, where water systems are not practical to suppress fires. Appropriate instrumentation and controls are provided for the proper operation of the fire detection, annunciation, and fire-fighting systems.

The fire-protection system is discussed in detail in USAR Subsection 9.5.1 and in the Clinton Station Fire Protection Evaluation Report, located in the USAR, Appendix E.

5.2.5 Facilities and Equipment for Offsite Monitoring

Consult the station specific Offsite Dose Calculation Manual (ODCM) for the most current location for fixed continuous air samplers and TLD locations. These fixed air samplers and TLD locations are maintained by Clinton Power Station personnel.

5.2.6 Site Hydrological Characteristics

The hydrological characteristics of the Clinton Station vicinity are described in Section 2.4 of the USAR. The site is located 6 miles east of the city of Clinton, DeWitt County in central Illinois. The site and its environs consist primarily of the generating station, Clinton Lake, woodlands, pasture land, cultivated farmland, and the recreational areas. The condenser cooling water is provided from the U-shaped cooling lake (Clinton Lake) that has been formed by construction of a dam just downstream from the confluence of North Fork of Salt Creek with Salt Creek. Clinton Lake has a surface area at normal lake level (690 feet mean sea level) of approximately 4895 acres with an average depth of about 15.6 feet. Clinton Lake is totally within the site property boundary. The station facilities and the 3.4-mile discharge flume occupy about 150 acres and 130 acres, respectively. The station is located between the two fingers of the lake with a station grade elevation of 736 feet and plant floor elevation of 737 feet. The station circulating water screen house is located on the North Fork finger of the lake with the circulating water discharging back into the Salt Creek finger through a discharge flume.

5.2.6.1 Flood Design Considerations

The cooling lake is designed to withstand the effects of a probable maximum storm occurring over the entire drainage basin above the dam site.

Results of the hydrologic analyses discussed in USAR Subsections 2.4.3 and 2.4.8 show that a probable maximum flood runoff into the lake routed through the spillways will raise the lake water level to elevation 708.8 feet at the dam site. The backwater effect along the North Fork finger will raise the probable maximum flood water level at the station site to elevation 708.9 feet. Superimposing the wind wave effect due to a sustained 40 mph wind acting on the probable maximum water level will result in wave run-up elevations of 711.9 feet and 713.8 feet for significant waves and maximum (1%) waves, respectively, at the station site. The station’s Seismic Category I structures with grade elevation of 736 feet will not be affected by the probable maximum flood design conditions. The circulating water screen house is designed to withstand the effects of probable maximum flood.

The maximum run-up elevation at the dam for significant waves due to a sustained 40 mph wind acting on the probable maximum water level is elevation 711.0 feet. The top of the dam is at elevation 711.8 feet. In the Salt Creek basin, there are no existing or proposed dams upstream from the Clinton Station; therefore flood waves induced from dam failures that affect the safety-related structures are considered impossible.

Massive landslide from the valley walls into the cooling lake caused by a seismic disturbance is not possible because of lack of susceptible topographic and geological features. Thick glacial till available in the site precludes the possibility of massive landslides that can produce flood waves greater in magnitude than the probable maximum flood conditions and coincident wind wave effects.

Flooding due to tsunami is not possible at the site.

Based on considerations and studies made, the probable maximum flood condition in the lake is considered the controlling event. All the safety-related structures are protected against this event.

5.3 Protective Facilities and Equipment

The on-site assembly area is the Outage Control Center and the 762' elevation of the Radwaste Building as described in Section 4 of this Annex.

5.4 First Aid and Medical Facilities

Clinton Power Station has an inplant decontamination room located on the 737' of the Radwaste building. This room is provided with a sink and shower for decontamination purposes.

First aid kits, stretchers, sinks, eyewashes, and emergency showers have been placed in strategic locations throughout the station.

Medical treatment given to injured persons at the station is of a "first aid" nature. When more professional care is needed, injured persons are transported to a local hospital or clinic. John Warner Hospital in Clinton, Illinois is the primary supporting medical facility for injured persons who are contaminated with radioactivity. Decatur Memorial Hospital in Decatur, Illinois is the supporting Trauma Center for injured persons who are contaminated with radioactivity.

Decatur Memorial Hospital in Decatur (a Level 1 trauma center) is the backup medical facility for ill/injured persons suffering from severe spinal injuries or who have received an acute overexposure, where blood chemistry must be monitored and/or the possibilities of radiation sickness exist.

Appendix 1: NUREG-0654 Cross-Reference

Annex Section	**NUREG-0654**	**Annex Section**	**NUREG-0654**
1.0	Part I, Section A	Figure 4-1	Part II, Section J.10.m
1.1	Part I, Section C	Figure 4-2	Part II, Section J.8
1.2	Part I, Section D	Figure 4-3	Part II, Section J.5
1.3	Part II, Section A.1	4.4	Part II, Section J.2 & 3
Figure 1-1	Part I, Section D	Table 4-1	Part II, Section J.10.b
2.0	Part II, Section A.4	5.1	Part II, Section H.1 & G.3
2.1	Part II, Section A.3	5.2.1	Part II, Section H.5.a & 8
		5.2.2	Part II, Section H.5.b & I.2
3.0	Part II, Section D	5.2.3	Part II, Section H.5.c
		5.2.4	Part II, Section H.5.d
4.1	Part II, Section E.1 & J.7	5.2.5	Part II, Section H.6.b & 7
4.2	Part II, Section I.2 & 3	5.2.6	Part II, Section H.5.a & 6.a
4.3	Part II, Section J.10.m	5.3	Part II, Section J.1-5
4.3.1	Part II, Section E.6	5.4	Part II, Section L.1 & 2
4.3.2	Part II, Section J.8		
4.4	Part II, Section J.1-5		

Appendix 2: Station Letters of Agreement

1. DeWitt County Sheriff's Office - law enforcement
2. John Warner Hospital of Clinton - medical services
3. Decatur Memorial Hospital - medical services
4. Clinton Fire Department - fire protection
5. Clinton Ambulance - ambulance services
6. Sargent & Lundy – technical services
7. Horton Field House – relocation center
8. Monticello High School – relocation center
9. Richland Community College – relocation center
10. National Weather Service – weather forecasts

Attachment 4

EP-AA-1004

"Exelon Nuclear Standardized Radiological Emergency Plan Annex for Dresden Station"

Revision 23

EP-AA-1004
Revision 23

EXELON NUCLEAR

RADIOLOGICAL EMERGENCY PLAN ANNEX FOR DRESDEN STATION

Submitted: Kevin Appel Date: 10/10/07
Midwest Region Emergency Preparedness Manager

Authorized: Jim Meister Date: 10/12/07
Vice President – Operations Support

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APPENDIXES

REVISION HISTORY

Revision 0; July 1980	Revision 6i; February 1996	Revision 15; September 12, 2002
Revision 1; April 1981	Revision 6j; June 1996	Revision 16; July 31, 2003
Revision 2; June 1982	Revision 6k; January 1997	Revision 17; August 27, 2003
Revision 3; September 1984	Revision 6l; February 1997	Revision 18, December 2004
Revision 4; March 1986	Revision 6m; May 1997	Revision 19, May 2005
Revision 5; February 1987	Revision 6n; January 5, 1998	Revision 20, December 2005
Revision 6; January 1991	Revision 6p; August 14, 1998	Revision 21, November 2006
Revision 6a; July 1992	Revision 7; May 13, 1999	Revision 22, February, 2007
Revision 6b; August 1992	Revision 8; February 11, 2000	
Revision 6c; November 1992	Revision 9; May 22, 2000	
Revision 6d; April 1993	Revision 10; January 8, 2001	
Revision 6e; December 1993	Revision 11; May 3, 2001	
Revision 6f; January 1994	Revision 12; October 8, 2001	
Revision 6g; January 1995	Revision 13; October 31, 2001	
Revision 6h; December 1995	Revision 14; January 3, 2002	

Section 1: Introduction

As required in the conditions set forth by the Nuclear Regulatory Commission (NRC) for the operating licenses for the Exelon Nuclear Stations, the management of Exelon recognizes its responsibility and authority to operate and maintain the nuclear power stations in such a manner as to provide for the safety of the general public.

The Exelon Emergency Preparedness Program consists of the Exelon Nuclear Standardized Emergency Plan (Emergency Plan), Station Annexes, Emergency Plan Implementing Procedures, and associated program administrative documents. The Emergency Plan outlines the basis for response actions that would be implemented in an emergency. Planning efforts common to all Exelon Nuclear stations are encompassed within the Emergency Plan.

This document serves as the Dresden Station Annex and contains information and guidance that is unique to the station. This includes Emergency Action Levels (EALs), and facility geography and location for a full understanding and representation of the station’s emergency response capabilities. The Station Annex is subject to the same review and audit requirements as the Emergency Plan.

1.1 Facility Description

Dresden Station, Units 1, 2 and 3, is located in the Goose Lake Township of Grundy County in northeastern Illinois. Unit 1 is in permanent shutdown (see Figure 1-1).

The plant consists of three Boiling Water Reactor (BWR) Nuclear Steam Supply Systems (NSSS) and turbine generators provided by General Electric Company. Unit 1 is a dual cycle boiling water reactor designed for a power output of 700 MWt and has officially been retired as of August 31, 1984. Units 2 and 3 are equipped with nuclear steam supply systems (NSSS) designed for a power output of 2957 MWt.

The station property consists of a 953 acre tract of land with boundaries generally following the Illinois River to the north, the Kankakee River on the south and east and the Elgin, Joliet and Eastern Railway right-of-way on the west. Exelon is the sole owner of the 953 acre tract subject only to an easement of the U.S. Government for an access road to Dresden Island Lock and Dam maintained and operated by the U.S. Corps. of Engineers. This road traverses the site from north to south ~ 0.8 mile west of the plant.

In addition to ownership of the 953 acre tract, Exelon Nuclear also leases approximately 17 acres in two narrow strips of river frontage located near the northeast corner of the site from the State of Illinois. The terms of the lease provide that these "buffer" strips shall remain idle.

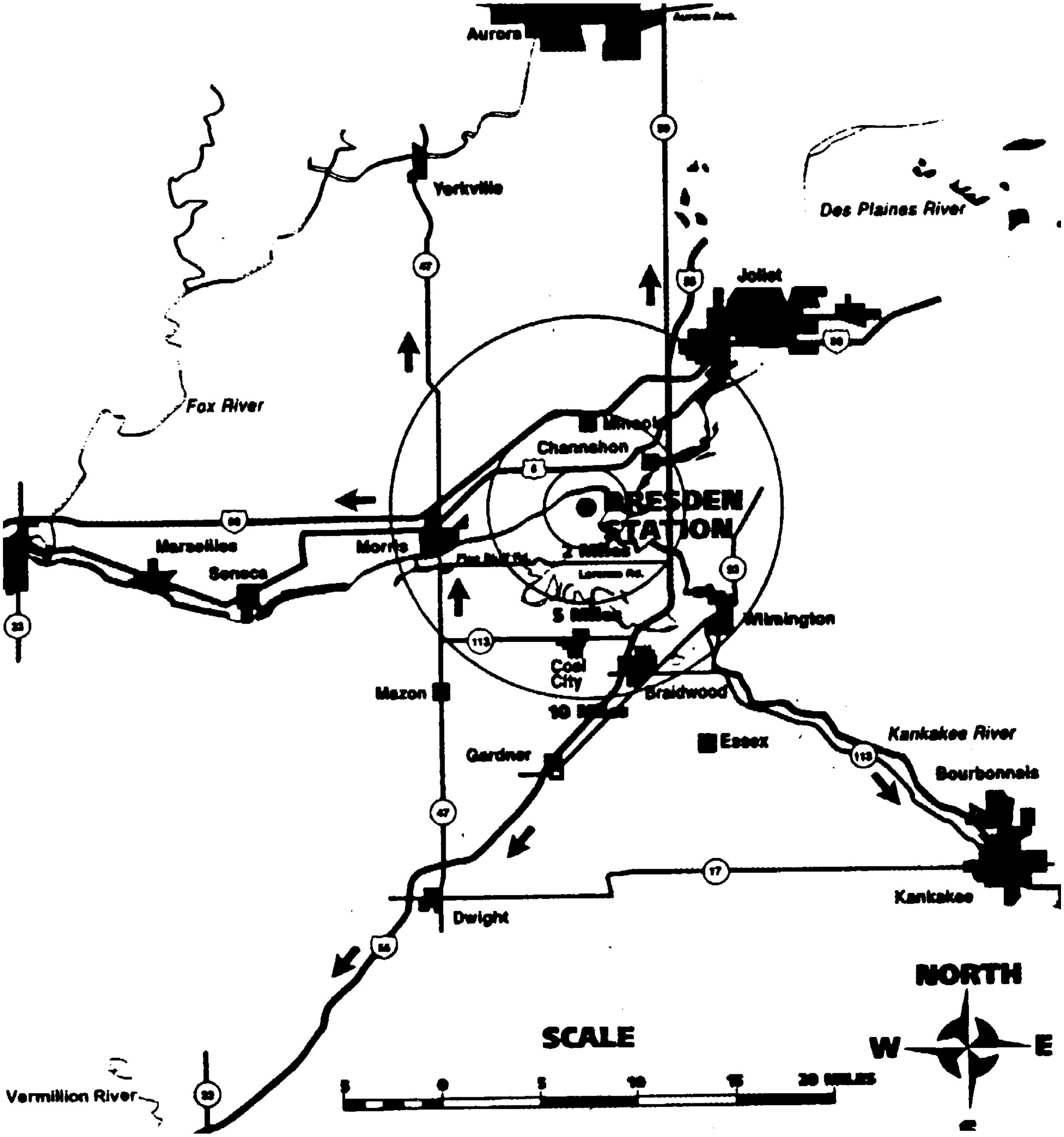
For more specific site location information, refer to the Station UFSAR.

1.2 Emergency Planning Zone

The plume exposure Emergency Planning Zone (EPZ) for Dresden Station is an area surrounding the Station with a radius of about ten miles. (Exact boundaries are determined by the State of Illinois). Refer to Figure 1-1.

The ingestion pathway EPZ for Dresden Station is an area surrounding the station with a radius of about 50 miles.

Figure 1-1: Dresden Station Location and 10 Mile EPZ



Section 2: Organizational Control of Emergencies

Initial response to any emergency is by the normal plant organization present at the site. This organization includes positions that are onsite 24 hours per day and is described in Section B.1 of the Emergency Plan.

Once an emergency is declared, the Emergency Response Organization is activated according to Section B.4 of the Emergency Plan and Implementing Procedures.

2.1 Non-Exelon Nuclear Support Groups

Exelon Nuclear has contractual agreements with several companies whose services would be available in the event of a radiological emergency. These agencies and their available services are listed in Appendix 3 of the Emergency Plan.

Emergency response coordination with governmental agencies and other support organizations is discussed in Section A of the Emergency Plan.

Agreements exist on file at Dresden Station with several support agencies. These agencies and their support roles are listed in Appendix 2, Station Letters of Agreement.

Section 3: Classification of Emergencies

3.1 General

Section D of the Exelon Nuclear Standardized Emergency Plan divides the types of emergencies into four Emergency Classification Levels (ECLs). The first four are the UNUSUAL EVENT, ALERT, SITE AREA EMERGENCY, and GENERAL EMERGENCY. These ECLs are entered by meeting the Emergency Action Level (EAL) Threshold Values provided in this section of the Annex. 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. Depending on the severity of an event, prior to returning to a standard day-to-day organization, a state or phase called RECOVERY may be entered to provide dedicated resources and organization in support of restoration and communication activities following the termination of the emergency.

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 Guideline 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 acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

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

RECOVERY: Recovery can be considered as a phase of the emergency and is entered by meeting emergency termination criteria provided in EP-AA-111 Emergency Classification and Protective Action Recommendations.

An emergency is classified by assessing plant conditions and comparing abnormal conditions to Initiating Conditions and Threshold Values for each Emergency Action Level.

Individuals responsible for the classification of events will refer to the Initiating Condition and Threshold Values on the matrix of the appropriate station Standardized Emergency Plan Annex (this document). This matrix will contain Initiating Conditions, EAL Threshold Values, Mode Applicability Designators, appropriate EAL numbering system, and additional guidance necessary to classify events. It may be provided as a user aid.

The matrix is set up in four Recognition Categories. The first is designated as "R" and relates to Abnormal Radiological Conditions / Abnormal Radiological Effluent Releases. The second is designated as "F" and relates to Fission Product Barrier Degradation. The third is designated as "M" and relates to System Malfunctions. The fourth is designated as "H" and relates to Hazards and Other Conditions.

The matrix is designed to provide an evaluation of the Initiating Conditions from the worst conditions (General Emergencies) on the left to the relatively less severe conditions on the right (Unusual Events). Evaluating conditions from left to right will reduce the possibility that an event will be under classified. All Recognition Categories should be reviewed for applicability prior to classification.

The Initiating Conditions are coded with a two letter and one number code. The first letter is the Recognition Category designator, the second letter is the Classification Level, “U” for (NOTIFICATION OF) UNUSUAL EVENT, “A” for ALERT, “S” for SITE AREA EMERGENCY and “G” for GENERAL EMERGENCY. The EAL number is a sequential number for that Recognition Category series. All Initiating Conditions that are describing the severity of a common condition (series) will have the same number.

The EAL number may then be used to reference a corresponding page(s), which provides the basis information pertaining to the Initiating Condition:

- Threshold Value
- Mode Applicability
- Basis

Emergency Action Levels are the measurable, observable detailed conditions that must be met in order to classify the event. Classification is not to be made without referencing, comparing and satisfying the Threshold Values specified in the Emergency Action Levels.

A list of definitions is provided as part of this document for terms having specific meaning to the Emergency Action Levels. Site specific definitions are provided for terms with the intent to be used for a particular Initiating Condition/Threshold Value and may not be applicable to other uses of that term at other sites, the Emergency Plan or procedures.

References are also included to documents that were used to develop the EAL Threshold Values.

References to the Emergency Director means the person in Command and Control as defined in the Emergency Plan. Classification of emergencies is a non-delegable responsibility of Command and Control for the onsite facilities with responsibility assigned to the Shift Emergency Director (Control Room Shift Manager) or the Station Emergency Director (TSC). Classification of emergencies remains the responsibility of the applicable onsite facility even after Command and Control is transferred to the Corporate Emergency Director (EOF).

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

Indications used for monitoring and evaluation of plant conditions include the normally used instrumentation, backup or redundant instrumentation, and the use of other parameters that provide information that supports determination if an EAL threshold value has been reached. When an EAL refers to a specific instrument or indication that is determined to be inaccurate or unavailable, then alternate indications shall be used to monitor the specified condition.

During an event that results in changing parameters trending towards an EAL classification, and instrumentation that was available to monitor this parameter becomes unavailable or the parameter goes off scale, the parameter should be assumed to have been exceeded consistent with the trend and the classification made if there are no other direct or indirect means available to determine if the threshold has not been exceeded.

EALs are for unplanned events. A planned evolution involves 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. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL Threshold Value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

When two or more Emergency Action Levels are determined, declaration will be made on the highest classification level for the Unit. When both units are affected, the highest classification for the Station will be used for notification purposes and both units' classification levels will be noted.

3.2 Mode Applicability

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

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 Fission Product Barrier Matrix EALs are applicable only to events that initiate in Hot Shutdown or higher.

If there is a change in 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.

3.3 Emergency Director Judgment

Emergency Director Judgment EALs are provided in the Hazards and Other Condition Affecting Plant Safety section and on the Fission Product Barrier Matrix. Both of the Emergency Director Judgment EALs have specific criteria for when they should be applied.

The Hazards Section Emergency Director Judgment EALs are intended to address unanticipated conditions which are not addressed explicitly by other EALs but warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under specific emergency classifications (UE, Alert, SAE or GE).

The FPB Matrix ED Judgment EALs are intended to include unanticipated conditions, which are not addressed explicitly by any of the other FPB threshold values, but warrant determination because conditions exist that fall under the broader definition for a significant Loss or Potential Loss of the barrier (equal to or greater than the defined FPB threshold values).

3.4 Fission Product Barrier Restoration

Fission Product Barriers (FPBs) are not treated the same as EAL threshold values. Conditions warranting declaration of the loss or potential loss of a Fission Product Barrier may occur resulting in a specific classification. The condition that caused the loss or potential loss declaration could be rectified as the result of Operator action, automatic actions, or designed plant response. Barriers will be considered re-established when there are direct verifiable indications (containment penetration or open valve has been isolated, coolant sample results, etc) that the barrier has been restored and is capable of mitigating future events.

The reestablishment of a fission product barrier does not alter or lower the existing classification. Entry into Termination/Recovery phase is still required for exiting the present classification. However the reestablishment of the barrier should be considered in determining future classifications should plant conditions or events change.

3.5 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 five or more persons violently protesting station operations or activities at the site.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

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

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

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

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

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute 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., 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.

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

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.

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

NORMAL LEVELS: Normal levels can be considered as 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.

OPERATING MODES	REACTOR MODE SWITCH POSITION	TEMP
(1) Power Operation:	Run	N/A
(2) Startup:	Refuel [(a)] or Startup/Hot Standby	N/A
(3) Hot Shutdown [(a)]:	Shutdown	> 212° F
(4) Cold Shutdown [(a)]:	Shutdown	≤ 212° F
(5) Refueling [(b)]:	Shutdown or Refuel	N/A
(D) Defueled:	All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).	

[(a)] All reactor vessel head closure bolts fully tensioned.

[(b)] One or more reactor vessel head closure bolts less than fully tensioned.

Hot Matrix – applies in modes (1), (2), and (3)

Cold Matrix – applies in modes (4), (5), and (D)

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

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

SABOTAGE: A 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.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations >10% Reactor Power change.

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 isolated from the Control Room.

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

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

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

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

Emergency Action Level Technical Basis Page Index

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	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs that exceeds or is expected to exceed **1.50E+07 uCi/sec** for **≥ 15 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs that exceeds or is expected to exceed **1.50E+06 uCi/sec** for **≥ 15 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on any effluent monitor **> 200 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**. **OR** 2. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs **> 1.05E+06 uCi/sec** for **≥ 15 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on any effluent monitor **> 2 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**. **OR** 2. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs **> 6.03E+05 uCi/sec** for **≥ 60 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration **≥ 60 minutes**.

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

<table>
<tr><th></th><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="5">**Abnormal Rad Levels / Radiological Effluent**</td></tr>
<tr>
<td rowspan="2">**Abnormal Rad Levels**</td>
<td></td>
<td><table><tr><th>Table R1
Fuel Handling Incident Radiation Monitors</th></tr><tr><td>• Refuel Floor High Range ARM Station #2(4)
• Fuel Pool Radiation Monitor</td></tr></table></td>
<td>**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 D
EAL Threshold Values:
1. VALID isolation of Reactor Building Vent due to damage to irradiated fuel.
OR
2. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1.
OR
3. Water level drop in the Reactor Refueling Cavity, the Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.</td>
<td>**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 D
EAL Threshold Values:
1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:
• Refueling Cavity water level **< 466 in**. (Refuel Outage Reactor Vessel and Cavity Level Instrument LI 2(3)-263-114).
OR
• Spent Fuel Pool water level **< 19 ft.** above the fuel (**33 ft. 9 in.** indicated level).
OR
• Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal
AND
b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1.
OR
2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.</td>
</tr>
<tr>
<td><table><tr><th>Table R2
Areas Requiring Continuous Occupancy</th></tr><tr><td>• Main Control Room (Unit 2 ARM Station #22)
• Central Alarm Station (by survey)
• Secondary Alarm Station (by survey)
• Radwaste Control Room (Unit 2 ARM Station #31)
• Gatehouse (by survey)</td></tr></table></td>
<td><table><tr><th>Table R3
Areas Requiring Infrequent Access</th></tr><tr><td>• HPCI Cubicle
• East and West LPCI Pump Areas
• East and West CRD Module Areas
• Vessel Instrument Rack Area
• RWCU Area
• Isolation Condenser Area</td></tr></table></td>
<td>**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown 1 2 3 4 5 D
EAL Threshold Values:
1. VALID radiation monitor or survey readings **>15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions.
OR
2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.</td>
<td>**RU3** Fuel clad degradation. 1 2 3
EAL Threshold Values:
1. Offgas system isolation due to VALID Offgas Radiation Monitor high Trip.
OR
2. Specific coolant activity **> 4.0 uCi/gm** Dose Equivalent I-131.</td>
</tr>
</table>

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

Fission Product Barrier Matrix **Hot Matrix**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
FG1 Loss of ANY two barriers AND Loss or Potential Loss of third barrier. 1 2 3	**FS1** Loss or Potential Loss of ANY two barriers. 1 2 3	**FA1** ANY Loss or ANY Potential Loss of either Fuel Clad or RCS. 1 2 3	**FU1** ANY Loss or ANY Potential Loss of Containment. 1 2 3

Sub-Category	FC - Fuel Clad		RC – Reactor Coolant System		CT - Containment	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
1. RCS Activity →	Coolant activity **> 300 uCi/gm** Dose Equivalent I-131	None	None	None	None	None
2. RPV Water Level →	1. RPV level **< –164 in**. without at least one core spray pump **> 4750 gpm** **OR** 2. RPV level **< –191 in.**	RPV level **< - 143 in.** (TAF).	RPV level **< -143 in.** (TAF).	None	None	Plant conditions indicate that Primary Containment flooding is required.
3. Drywell Pressure →	None	None	1. Drywell Pressure **> 2.0 psig.** **AND** 2. Drywell Pressure rise due to RCS leakage.	None	1. Rapid unexplained drop in Drywell Pressure following initial pressure rise. **OR** 2. Drywell pressure response not consistent with LOCA conditions.	1. Drywell pressure **≥ 62 psig** and rising. **OR** 2. a. Drywell or torus hydrogen concentration **≥ 6%.** **AND** b. Drywell or torus oxygen concentration **≥ 5%.**
4. RCS Leakrate →	None	None	1. UNISOLABLE Main Steam Line (MSL) break as indicated by the failure of both MSIVs in ANY one line to close. **AND** 2. a. High MSL Flow AND High Steam Tunnel Temperature. **OR** b. Direct report of steam release.	1. RCS leakage **> 50 gpm** inside the drywell. **OR** 2. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> DEOP 300-1, Maximum Normal** operating levels.		
5. Hi Cont/Drywell Rad →	Drywell radiation monitor reading **> Fuel Cladding Loss Threshold**, Table F1.	None	1. Drywell Radiation monitor reading **> 100 R/hr.** **AND** 2. Indications of RCS leakage into the Drywell.	None	None	Drywell radiation monitor reading **> Containment Potential Loss Threshold**, Table F2.
6. Breach/Bypass →			None	None	1. a. Failure of all isolation valves in any one line to close. **AND** b. Downstream pathway to the environment exists. **OR** 2. Intentional venting/purging of Primary Containment per EOPs or SAMGs due to accident conditions. **OR** 3. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> DEOP 300-1, Maximum Safe** operating levels.	
7. ED Judgment. →	Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Table F2 Drywell Radiation Thresholds

Time After Shutdown (hrs)	Containment Potential Loss (R/hr)
≤ 2	1.60 E+03
> 2 to 4	1.35 E+03
> 4 to 8	1.20 E+03
> 8 to 16	1.00 E+03
> 16 to 23	8.75 E+02
> 23	8.60 E+02

Table F1 Drywell Radiation Thresholds

Time After Shutdown (hrs)	Fuel Cladding Loss (R/hr)
≤ 2	6.70 E+02
> 2 to 4	5.90 E+02
> 4 to 8	5.10 E+02
> 8 to 16	4.30 E+02
> 16 to 23	3.90 E+02
> 23	3.80 E+02

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
AC Electrical Distribution	**MG1** Prolonged loss of all offsite power and prolonged loss of all onsite AC power to essential busses. 1 2 3 **EAL Threshold Values:** 1. Loss of power to Reserve Auxiliary Transformer TR-22(32) and Unit Auxiliary Transformer TR-21(31). **AND** 2. Failure of DG 2(3), shared DG 2/3 and SBO DG 2(3) emergency diesel generators to supply power to unit ECCS busses. **AND** 3. a. Restoration of at least one unit ECCS bus within 4 hours is **not** likely. **OR** b. RPV level **cannot** be determined to be **> -143 in.** (TAF).	**MS1** Loss of all offsite power and loss of all onsite AC power to essential busses. 1 2 3 **EAL Threshold Values:** 1. Loss of power to Reserve Auxiliary Transformer TR-22(32) and Unit Auxiliary Transformer TR-21(31). **AND** 2. Failure of DG 2(3), shared DG 2/3 and SBO DG 2(3) emergency diesel generators to supply power to unit ECCS busses. **AND** 3. Failure to restore power to at least one Unit ECCS bus within **15 minutes** from the time of loss of both offsite and onsite AC power.	**MA1** AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout. 1 2 3 **EAL Threshold Values:** 1. AC power capability to unit ECCS busses reduced to only one of the following power sources for **> 15 minutes**: • Reserve Auxiliary Transformer TR-22 (TR-32) • Unit Auxiliary Transformer TR-21 (TR-31) • Unit Emergency Diesel Generator DG 2 (3) • Shared Emergency Diesel Generator DG 2/3 • Station Blackout Diesel Generator SBO DG 2 (3) • Unit Crosstie Breakers **AND** 2. Any additional single power source failure will result in unit blackout.	**MU1** Loss of all offsite power to essential busses for greater than 15 minutes. 1 2 3 4 5 **EAL Threshold Values:** Loss of power to Reserve Auxiliary Transformer TR-22(32) AND Unit Auxiliary Transformer TR-21(31) for **> 15 minutes**.
RPS / Inadvertent Criticality	**MG3** Failure of the Reactor Protection System to complete an automatic scram and manual scram was NOT successful and there is indication of an extreme challenge to the ability to cool the core. 1 2 **EAL Threshold Values:** 1. Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**: a. Reactor power remains **> 6% APRM.** **OR** b. Torus temperature **> 110°F** and boron injection required for reactivity control. **AND** 2. a. RPV level cannot be restored and maintained **> -164 in**. **OR** b. Heat Capacity Limit (DEOP 200-1 Fig. M) exceeded.	**MS3** Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded and manual scram was not successful. 1 2 **EAL Threshold Values:** Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**: 1. Reactor power remains **> 6% APRM.** **OR** 2. Torus Bulk Temperature **> 110°F** and boron injection required for reactivity control.	**MA3** Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a reactor protection system setpoint has been exceeded. 1 2 **EAL Threshold Values:** 1. A Reactor Protection System setpoint was exceeded. **AND** 2. Automatic scram did not reduce reactor power **< IRM Range 7 and lowering**.	**MU3** Inadvertent criticality. 3 4 5 **EAL Threshold Values:** An UNPLANNED extended positive period observed on nuclear instrumentation.
DC Power		**MS4** Loss of all vital DC power. 1 2 3 **EAL Threshold Values:** Loss of all Vital DC Power based on **< 105 VDC** on 125 VDC battery busses #2 and #3 for **> 15 minutes**.		

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
Heat Sink		**MS5** Complete loss of heat removal capability. 1 2 3 **EAL Threshold Values:** Heat Capacity Limit (DEOP 200-1 Fig. M) exceeded.		
Annunciators		**MS6** Inability to monitor a SIGNIFICANT TRANSIENT in progress. 1 2 3 **EAL Threshold Values:** 1. Loss of most **(approximately 75%)** safety system annunciators (Table M2). **AND** 2. Indications needed to monitor safety functions (Table M3) are unavailable. **AND** 3. SIGNIFICANT TRANSIENT in progress (Table M4). **AND** 4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MA6** UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable. 1 2 3 **EAL Threshold Values:** 1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes.** **AND** 2. a. SIGNIFICANT TRANSIENT in progress (Table M4). **OR** b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MU6** UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes. 1 2 3 **EAL Threshold Values:** 1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** 2. UNPLANNED loss of most **(approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.
RCS Leakage				**MU7** RCS leakage. 1 2 3 **EAL Threshold Values:** 1. Unidentified or pressure boundary leakage **> 10 gpm**. **OR** 2. Identified leakage **> 25 gpm**.

Table M2 - Control Room Panels	Table M3 - Safety Functions and Related Systems	Table M4 - Significant Transients
• 902(3)-3 • 902(3)-5 • 902(3)-8	• 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) • Fission Product Barriers	• Turbine trip • Reactor scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10%** Reactor Power change

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
Communications				**MU10** UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5 **EAL Threshold Values:** 1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations. **OR** 2. Loss of all Table M6 **Offsite** communications capability.
T. S. Time				**MU11** Inability to reach required shutdown within Technical Specification limits. 1 2 3 **EAL Threshold Values:** Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.

Table M6 - Communications Capability

System	Onsite	Offsite
Plant Radio System	X	
Plant Paging System	X	
Sound Power Phones	X	
In-Plant Telephones	X	
All telephone lines (commercial and microwave)		X
NARS		X
ENS		X
Satellite Phones		X
HPN		X
Cellular Phones		X

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

Hazards and Other Conditions Affecting Plant Safety

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, "Threat Assessment". **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 D **EAL Threshold Values:** 1. Control Room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per DSSP 0100-CR in **< 30 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 D **EAL Threshold Values:** Entry into DSSP 0100-CR for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Reactor Building • Aux Electric Room • Control Room • Diesel Generator Rooms • 4-KV ECCS Switchgear Area • Battery Rooms • CRD & CCSW Pump Rooms • Turbine Building Cable Tunnel • Turbine Building Safe Shutdown Areas • Crib House	**Table H3** **Internal Flooding Areas** • Condenser Pits • Condensate Pump Rooms • Containment Cooling Service Water Vaults • Crib House • East Corner Room • West Corner Room	**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by seismic instrumentation > **0.10g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or sustained high winds **> 100 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in any Table H3 area as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment. **OR** 6. a. High river water level **> 513 ft.** **OR** b. Low river water level **< 501 ft. 6 in.**	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. Seismic event identified by any **TWO** of the following: • Earthquake felt in plant. • Seismic event confirmed by station seismic monitor procedure. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained **(>15 minutes)** high winds **> 100 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode. **OR** 6. River level transients potentially affecting safe operation of the plant: a. High river level **> 509 ft**. **OR** b. Report of substantial reduction in river level by site personnel and confirmation by the Corps of Engineers that Dresden Lock and Dam has failed.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas		**Table H2** **Vital Areas** • Reactor Building • Aux Electric Room • Control Room • Diesel Generator Rooms • 4-KV ECCS Switchgear Area • Battery Rooms • CRD & CCSW Pump Rooms • Turbine Building Cable Tunnel • Turbine Building Safe Shutdown Areas • Crib House	**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
E. D. Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 D **EAL Threshold Values:** 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.
ISFSI Events				**HU9** Damage to a loaded cask CONFINEMENT BOUNDARY. 1 2 3 4 5 D **EAL Threshold Values:** 1. Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY. **OR** 2. Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY. **OR** 3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask MPC CONFINEMENT BOUNDARY.

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

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	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs that exceeds or is expected to exceed **1.50E+07 uCi/sec** for **≥ 15 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs that exceeds or is expected to exceed **1.50E+06 uCi/sec** for **≥ 15 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1 VALID reading on any effluent monitor **> 200 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**. **OR** 2. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs **> 1.05E+06 uCi/sec** for **≥ 15 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on any effluent monitor **> 2 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**. **OR** 2. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs **> 6.03E+05 uCi/sec** for **≥ 60 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration **≥ 60 minutes**.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Abnormal Rad Levels		**Table R1** **Fuel Handling Incident Radiation Monitors** • Refuel Floor High Range ARM Station #2(4) • Fuel Pool Radiation Monitor	**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID isolation of Reactor Building Vent due to damage to irradiated fuel. **OR** 2. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1. **OR** 3. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.	**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by: • Refueling Cavity water level **< 466 in**. (Refuel Outage Reactor Vessel and Cavity Level Instrument LI 2(3)-263-114). **OR** • Spent Fuel Pool water level **< 19 ft.** above the fuel (**33 ft. 9 in.** indicated level). **OR** • Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal. **AND** b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1. **OR** 2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.
	Table R2 **Areas Requiring Continuous Occupancy** • Main Control Room (Unit 2 ARM Station #22) • Central Alarm Station (by survey) • Secondary Alarm Station (by survey) • Radwaste Control Room (Unit 2 ARM Station #31) • Gatehouse (by survey)	**Table R3** **Areas Requiring Infrequent Access** • HPCI Cubicle • East and West LPCI Pump Areas • East and West CRD Module Areas • Vessel Instrument Rack Area • RWCU Area • Isolation Condenser Area	**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID radiation monitor or survey readings **>15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions. **OR** 2. VALID radiation monitor or survey readings **>2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
AC Electrical Distribution			**MA2** Loss of all offsite power and loss of all onsite AC power to essential busses. 4 5 D **EAL Threshold Values:** 1. Loss of power to Reserve Auxiliary Transformer TR-22(32) and Unit Auxiliary Transformer TR-21(31). **AND** 2. Failure of DG 2(3), shared DG 2/3 and SBO DG 2(3) emergency diesel generators to supply power to unit ECCS busses. **AND** 3. Failure to restore power to at least one unit ECCS bus within **15 minutes** from the time of loss of both offsite and onsite AC power.	**MU1** Loss of all offsite power to essential busses for greater than 15 minutes. 1 2 3 4 5 **EAL Threshold Values:** Loss of power to Reserve Auxiliary Transformer TR-22(32) AND Unit Auxiliary Transformer TR-21(31) for **> 15 minutes**.
RPS				**MU3** Inadvertent criticality. 3 4 5 **EAL Threshold Values:** An UNPLANNED extended positive period observed on nuclear instrumentation.
DC Power				**MU4** UNPLANNED loss of required DC power for greater than 15 minutes. 4 5 **EAL Threshold Values:** 1. UNPLANNED Loss of all required Vital DC Power based on **< 105 VDC** indications on 125 VDC battery busses #2 and #3. **AND** 2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.
Heat Sink			**MA5** Inability to maintain plant in Cold Shutdown with irradiated fuel in the RPV. 4 5 **EAL Threshold Values:** 1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 212°F** for > Table M1 duration. **OR** 2. UNPLANNED RPV pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.	**MU5** UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV. 4 5 **EAL Threshold Values:** 1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 212° F.** **OR** 2. Loss of all RCS temperature AND RPV level indication for **> 15 minutes**.

Table M1 – RCS Reheat Duration Thresholds

RCS	Secondary Containment Closure	Duration
Intact	N/A	60 minutes*
Not Intact	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.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
RCS Leakage / Inventory	**MG8** Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged with irradiated fuel in the RPV. 4 5 **EAL Threshold Values:** 1. Loss of RPV inventory per Table M5 indications. **AND** 2. a. RPV level **< -143 in**. (TAF) for **> 30 minutes**. **OR** b. RPV level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following: • Refuel Floor Hi Range ARM > **3000 mR/hr** or off-scale high. • Erratic Source Range Monitor indication. **AND** 3. Containment is challenged as indicated by one or more of the following: • Primary containment Hydrogen concentration **≥ 6%** and Oxygen concentration **≥ 5%.** • Drywell pressure **≥ 62 psig.** • Primary and Secondary CONTAINMENT CLOSURE not established. • Any Secondary Containment radiation monitors **> DEOP 300-1 Maximum Safe** operating level.	**MS8** Loss of RCS/RPV inventory affecting core decay heat removal capability. 4 **EAL Threshold Values:** 1. **Without** Primary or Secondary CONTAINMENT CLOSURE established: a. RPV level **< - 60 in.** **OR** b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory per Table M5 indications. **OR** 2. **With** Primary or Secondary CONTAINMENT CLOSURE established: a. RPV level **< -143 in**. (TAF). **OR** b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory as evidenced by either of the following: • Per Table M5 indications. • Erratic Source Range Monitor indication.	**MA8** Loss of RCS/RPV inventory with irradiated fuel in the RPV. 4 5 **EAL Threshold Values:** 1. Loss of RCS/RPV inventory as indicated by RPV level **< - 54 in.** **OR** 2. a. Loss of RPV inventory per Table M5 indications. **AND** b. RCS/RPV level unknown for **> 15 minutes.** **Table M5 – Indications of RCS Leakage** • Unexplained floor or equipment sump level rise • Unexplained Torus level rise • Unexplained vessel make up rate rise • Observation of leakage	**MU8** RCS leakage. 4 **EAL Threshold Values:** RPV level **cannot** be restored and maintained **> 0 in.**
RCS Leakage / Inventory		**MS9** Loss of RPV Inventory affecting core decay heat removal capability with irradiated fuel in the RPV. 5 **EAL Threshold Values:** 1. **Without** Secondary CONTAINMENT CLOSURE established: a. RPV inventory as indicated by RPV level **< - 60 in**. **OR** b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following: • Refuel Floor Hi Range ARM > **3000 mR/hr** or off-scale high. • Erratic Source Range Monitor indication. **OR** 2. **With** Secondary CONTAINMENT CLOSURE established: a. RPV inventory as indicated by RPV level **< - 143 in.** (TAF). **OR** b. RPV level unknown with Indication of core uncovery as evidenced by one or more of the following: • Refuel Floor Hi Range ARM > **3000 mR/hr** or off-scale high. • Erratic Source Range Monitor indication.		**MU9** UNPLANNED loss of RCS inventory with irradiated fuel in the RPV. 5 **EAL Threshold Values:** 1. UNPLANNED RPV level drop below the RPV flange for **≥ 15 minutes.** **OR** 2. a. Loss of RPV inventory per Table M5 indications. **AND** b. RPV level unknown.

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

<table>
<tr><th></th><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="5">System Malfunction</td></tr>
<tr><td>Communications</td><td></td><td></td><td>
<table>
<tr><th colspan="3">Table M6 - Communications Capability</th></tr>
<tr><th>System</th><th>Onsite</th><th>Offsite</th></tr>
<tr><td>Plant Radio System</td><td>X</td><td></td></tr>
<tr><td>Plant Paging System</td><td>X</td><td></td></tr>
<tr><td>Sound Power Phones</td><td>X</td><td></td></tr>
<tr><td>In-Plant Telephones</td><td>X</td><td></td></tr>
<tr><td>All telephone lines (commercial and microwave)</td><td></td><td>X</td></tr>
<tr><td>NARS</td><td></td><td>X</td></tr>
<tr><td>ENS</td><td></td><td>X</td></tr>
<tr><td>Satellite Phones</td><td></td><td>X</td></tr>
<tr><td>HPN</td><td></td><td>X</td></tr>
<tr><td>Cellular Phones</td><td></td><td>X</td></tr>
</table>
</td><td>MU10 UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5
<u>EAL Threshold Values:</u>
1. Loss of all Table M6 Onsite communications capability affecting the ability to perform routine operations.
OR
2. Loss of all Table M6 Offsite communications capability.</td></tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, "Threat Assessment". **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** 1. Control Room evacuation has been initiated. **AND** 2. Control of the plant **<u>cannot</u>** be established per DSSP 0100-CR in **< 30 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 D **<u>EAL Threshold Values:</u>** Entry into DSSP 0100-CR for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Reactor Building • Aux Electric Room • Control Room • Diesel Generator Rooms • 4-KV ECCS Switchgear Area • Battery Rooms • CRD & CCSW Pump Rooms • Turbine Building Cable Tunnel • Turbine Building Safe Shutdown Areas • Crib House	**Table H3** **Internal Flooding Areas** • Condenser Pits • Condensate Pump Rooms • Containment Cooling Service Water Vaults • Crib House • East Corner Room • West Corner Room	**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by seismic instrumentation **> 0.10g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or sustained high winds **> 100 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding in that results in **EITHER**: a. Degraded safety system performance in any Table H3 area as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment. **OR** 6. a. High river water level **> 513 ft**. **OR** b. Low river water level **< 501 ft. 6 in.**	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. Seismic event identified by any **TWO** of the following: • Earthquake felt in plant. • Seismic event confirmed by station seismic monitor procedure. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained **(>15 minutes)** high winds **> 100 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode. **OR** 6. River level transients potentially affecting safe operation of the plant: a. High river level **> 509 ft**. **OR** b. Report of substantial reduction in river level by site personnel and confirmation by the Corps of Engineers that Dresden Lock and Dam has failed.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas		**Table H2** **Vital Areas** - Reactor Building - Aux Electric Room - Control Room - Diesel Generator Rooms - 4-KV ECCS Switchgear Area - Battery Rooms - CRD & CCSW Pump Rooms - Turbine Building Cable Tunnel - Turbine Building Safe Shutdown Areas - Crib House	**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
E. D. Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 D **EAL Threshold Values:** 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.
ISFSI Events				**HU9** Damage to a loaded cask CONFINEMENT BOUNDARY. 1 2 3 4 5 D **EAL Threshold Values:** 1. Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY. **OR** 2. Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY. **OR** 3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask MPC CONFINEMENT BOUNDARY.

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

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RG1

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs that exceeds or is expected to exceed **1.50 E+07 uCi/sec** for **≥ 15 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate).

   OR

2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**:
   a. **> 1000 mRem TEDE**

      OR

   b. **> 5000 mRem CDE Thyroid**

   OR

3. Field survey results at or beyond site boundary indicate **EITHER**:
   a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RG1 (cont)

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in a PAG dose being reached. Assumptions and DAPAR inputs are provided in calculation EP-EAL-0604.

The same value is used for ground level (Rx Bldg Vent) and elevated (Chimney) release points. An elevated release may not affect offsite areas as close to plant as ground level release; however, use of ground level values provides conservative estimates for exposure (cloud shine) to an overhead plume (EPA-400, section 5.6.1).

The sum of both units' monitors provides the total station release rate.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of dose assessments are available when the classification is made (i.e., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RG1 (cont)

Basis (cont):

Threshold #2 Basis:

The TEDE (1000 mRem) and the CDE Thyroid (5000 mRem) doses are set at the EPA PAG Limits.

The “site boundary” is defined by an approximately 800-meter (1/2 mile) radius around the plant. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release has been stopped and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AG1
2. EP-AA-112-500 Emergency Environmental Monitoring
3. Exelon DAPAR version 3.1
4. EP-MW-110-200 Dose Assessment
5. ODCM Section 12.4 Gaseous Effluents and Total Dose
6. DOP 1700-10, Obtaining And Calculating A Gaseous Release Rate From the Unit 2/3 Chimney, Unit 1 Chimney and Unit 2/3 combined Reactor Vent Using the Eberline Control Terminal
7. EP-EAL-0604, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Dresden Station
8. DEOP 300-2, Radioactivity Release Control

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RS1

Initiating Condition:

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs that exceeds or is expected to exceed **1.50 E+06 uCi/sec** for **≥ 15 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate.)

   OR

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

   a. **> 100 mRem TEDE**

      OR

   b. **> 500 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RS1 (cont)

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in 10% of a PAG dose being reached. Assumptions and DAPAR inputs are provided in calculation EP-EAL-0604.

The sum of both units' monitors provides the total station release rate.

The same value is used for ground level (Rx Bldg Vent) and elevated (Chimney) release points. An elevated release may not affect offsite areas as close to plant as ground level release; however, use of ground level values provides conservative estimates for exposure (cloud shine) to an overhead plume (EPA-400, section 5.6.1).

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of dose assessments are available when the classification is made (i.e., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RS1 (cont)

Basis (cont):

Threshold #2 Basis:

The TEDE (100 mRem) and the CDE Thyroid (500 mRem) doses are set at 10% of the EPA PAG Limits.

The “site boundary” is defined by an approximately 800-meter (1/2 mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release has been stopped and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AS1
2. EP-AA-112-500 Emergency Environmental Monitoring
3. Exelon DAPAR version 3.1
4. EP-MW-110-200 Dose Assessment
5. ODCM Section 12.4 Gaseous Effluents
6. DOP 1700-10, Obtaining And Calculating A Gaseous Release Rate From the Unit 2/3 Chimney, Unit 1 Chimney and Unit 2/3 combined Reactor Vent Using the Eberline Control Terminal
7. EP-EAL-0604, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Dresden Station
8. DEOP 300-2, Radioactivity Release Control

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading on any effluent monitor **> 200 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**.

   OR

2. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs **> 1.05E+06 uCi/sec** for **≥ 15 minutes** (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM** Limit with a release duration of **≥ 15 minutes**.

Basis:

UNPLANNED: As used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (i.e., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

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

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA1 (cont)

Basis (cont):

Threshold #1 Basis:

The threshold addresses radioactivity releases (liquid or gaseous) that for whatever reason cause effluent radiation monitor readings to exceed two hundred times the alarm setpoint established by the radioactive discharge permit. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the Offsite Dose Calculation Manual (ODCM) to warn of a release that is not in compliance with the Radiological Effluent Technical Specifications (RETS). Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

An elevated monitor reading while the effluent flow path is isolated is NOT considered to be a VALID reading.

The Liquid Radwaste Discharge Monitor (Rad Monitor 2/3-2001-948) high alarm setpoint is typically calculated based on historical activity and dilution flow data. Typically, the alarm setpoint is based on a value fifty times the highest expected monitor response during discharge, which is still well below the ODCM concentration limit.

Threshold #2 Basis:

Dresden incorporates 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 ODCM. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

This EAL addresses a potential or actual drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. However, the effluent monitor Alert value for gaseous effluents was reduced to a value one half way between the Unusual Event value and the Site Area Emergency value to ensure sequential classifications. Detailed calculations provided in EP-EAL-0604. The sum of the readings from the Unit 2/3 Rx Building SPING (2/3-1740-203) and the Unit 2/3 Chimney SPING (2/3-1740-202) provides a total station gaseous effluent release rate. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM. The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA1 (cont)

Basis (cont):

Threshold #3 Basis:

Confirmed sample analyses in excess of two hundred times the site ODCM limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by increasing the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mRem/yr basis of the 10 CFR 20 limits for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be approximately 10 mR/hr. The required release duration was reduced to 15 minutes in recognition of the increased severity.

Grab samples are used to determine release concentrations or rates to confirm meter readings or when the effluent monitors are not in service or other alarms indicate the need for sampling. Maximum instantaneous relesae rate limits are calculated in accordance with the ODCM. These are indicated on approved discharge permits.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AA1
2. Sargent & Lundy calculation ATD-0223, Rev. 0, 1/12/93
3. ODCM Section 12.3 Liquid Effluents
4. ODCM Section 12.4 Gaseous Effluents
5. DOP 1700-10, Obtaining And Calculating A Gaseous Release Rate From the Unit 2/3 Chimney, Unit 1 Chimney and Unit 2/3 combined Reactor Vent Using the Eberline Control Terminal
6. UNIT 2/3 DAN 2223-6 A-12 "2/3 RADWASTE DISCHARGE HIGH RADIATION"
7. UNIT 2/3 DOP 2000-110, Radioactive Waste Discharge to River With the Off-Stream Liquid Effluent Monitor Operable
8. UNIT 2/3 DOP 2000-109, Waste Surge Tank Batching for a Radwaste River Discharge
9. Structural Drawing B-01A Composite Site Plan Dresden Station Units 1, 2 & 3
10. EP-EAL-0604, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Dresden Station
11. DEOP 300-2, Radioactivity Release Control

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading on any effluent monitor **> 2 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**.

   OR

2. The sum of VALID readings on the Unit 2/3 Rx Bldg and Unit 2/3 Chimney SPINGs **> 6.03E+05 uCi/sec** for **≥ 60 minutes**. (as determined by DOP 1700-10 or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

Basis:

<u>UNPLANNED</u>: As used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (i.e., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

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

The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU1 (cont)

Basis (cont):

Threshold #1 Basis:

The effluent release paths are monitored for radioactivity prior to the flow reaching the point where it would mix with the process flow to the environment. Prior to initiating batch releases, the discharge volume is sampled and analyzed for radioactivity. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate. Radiation monitor alarm setpoints are established to isolate the process flow at the point determined by the discharge permit. These limits are based on the Offsite Dose Calculation Manual ODCM.

An elevated monitor reading while the effluent flow path is isolated is NOT considered to be a VALID reading.

The Liquid Radwaste Discharge Monitor (Rad Monitor 2/3-2001-948) high alarm setpoint is typically calculated based on historical activity and dilution flow data. Typically, the alarm setpoint is based on a value fifty times the highest expected monitor response during discharge, which is still well below the ODCM concentration limit.

Threshold #2 Basis:

This EAL addresses a potential drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 2, regulatory commitments for an extended period of time. The sum of both gaseous effluent monitor readings provides a total station release rate. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM, per assumptions/inputs provided in EP-EAL-0604.

The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

Threshold #3 Basis:

Confirmed sample analyses in excess of two times the site ODCM limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times ODCM for 30 minutes does not exceed this EAL. Grab samples are used to determine release concentrations or rates to confirm meter readings or when the effluent monitors are not in service. Maximum instantaneous release rate limits are calculated in accordance with the ODCM. These are indicated on approved discharge permits.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU1 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 AU1
2. Sargent & Lundy calculation ATD-0223, Rev. 0, 1/12/93
3. ODCM Section 12.3 Liquid Effluents
4. ODCM Section 12.4 Gaseous Effluents
5. DOP 1700-10, Obtaining And Calculating A Gaseous Release Rate From the Unit 2/3 Chimney, Unit 1 Chimney and Unit 2/3 combined Reactor Vent Using the Eberline Control Terminal
6. UNIT 2/3 DAN 2223-6 A-12 "2/3 RADWASTE DISCHARGE HIGH RADIATION"
7. UNIT 2/3 DOP 2000-110, Radioactive Waste Discharge to River With the Off-Stream Liquid Effluent Monitor Operable
8. UNIT 2/3 DOP 2000-109, Waste Surge Tank Batching for a Radwaste River Discharge
9. EP-EAL-0604, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Dresden Station
10. DEOP 300-2, Radioactivity Release Control

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA2

Initiating Condition:

Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID isolation of Reactor Building Vent due to damage to irradiated fuel.

   OR

2. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1.

Table R1 **Fuel Handling Incident Radiation Monitors**
• Refuel Floor High Range ARM Station #2(4) • Fuel Pool Radiation Monitor

   OR

3. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.

Basis:

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

Threshold #1 and #2 Basis:

Uncovering spent fuel represents a substantial degradation of the level of safety of the plant and warrants an Alert classification. Time is available to take corrective actions. 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, no prompt fatalities are predicted and the risk of injury is low. 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 fuel uncovery begins. The value of 1000 mR/hr was conservatively chosen for classification purposes.

Radiation monitor readings are used to provide indication of fuel uncovery and/or fuel damage. High monitor readings associated with the transfer or relocation of a source, stored in or near the pool or readings responding to a planned evolution such as removal of the reactor head or equipment relocation are not classified under this threshold since the reading would not be indicative of fuel uncovery and/or fuel damage.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA2 (cont)

Basis (cont):

Dropping heavy loads onto the spent fuel can cause significant damage to the spent fuel and an Alert is also warranted under these conditions provided that the above radiation monitor threshold readings are reached.

Threshold #3 Basis:

When the RPV head is removed and the Refuel Outage Reactor Vessel and Cavity Level instrumentation, LI 2(3)-263-114, is calibrated to indicate levels as high as the refuel floor elevation, remote indication of Refueling Cavity water level is available in the Control Room. Spent Fuel Pool water level drops can be directly monitored only by visual observation. If the Spent Fuel Pool is in communication with the Refueling Cavity, however, remote indication of Spent Fuel Pool water level from the bottom of the fuel transfer canal to the refueling floor is provided by the Refuel Outage Reactor Vessel and Cavity Level instrumentation, LI 2(3)-263-114. Even so, uncovery of spent fuel seated in the Spent Fuel Pool storage racks unknown remotely because the bottom of the fuel transfer canal is above the elevation of the top of the storage racks. Any fuel that becomes uncovered while suspended from the refuel grapple may be indicated on LI 2(3)-263-114 but, without report of the vertical position of the grapple, fuel uncovery cannot be determined. Thus, any fuel that becomes uncovered while suspended from the refuel grapple may not be indicated. Without report of the vertical position of the grapple, fuel uncovery cannot be determined. Visual observation, therefore, provides the only viable mechanism of determining if spent fuel in the Refueling Cavity or Spent Fuel Pool will be uncovered.

This EAL applies to irradiated fuel requiring water coverage and is not intended to address spent fuel, which is licensed for dry storage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AA2
2. DAN 902(3)-3 C-16(E-16) Reactor Building Fuel Pool Hi Radiation
3. DAN 902(3)-3 B-1 Refuel Floor Hi Radiation
4. DAN 902(3)-3 A-3(F-14) Reactor building Vent Hi-Hi Radiation
5. UFSAR 9.1
6. DAN 902(3)-4 D-24 Fuel Pool Skimmer Tank Level Lo
7. DIP 0260-01 Refuel Outage Reactor Vessel and Cavity Level Instrumentation
8. DFP 0850-01 Water Level Loss in SFP or Cavity
9. DOP 1900-03 Reactor Cavity, Dryer/Separator Storage Pit and Fuel Pool Level Control

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU2

Initiating Condition:

Unexpected rise in plant radiation.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:

- Refueling Cavity water level **< 466 in.** (Refuel Outage Reactor Vessel and Cavity Level Instrument LI 2(3)-263-114).

  OR

- Spent Fuel Pool water level **< 19 ft.** above the fuel (**33 ft. 9 in.** indicated level).

  OR

- Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity,Spent Fuel Pool or Fuel Transfer Canal.

AND

b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1.

Table R1 **Fuel Handling Incident Radiation Monitors**
• Refuel Floor High Range ARM Station #2(4) • Fuel Pool Radiation Monitor

OR

2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU2 (cont)

Basis:

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

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

NORMAL LEVELS: Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Threshold #1 Basis:

During refueling when the RPV head is removed, the Refuel Outage Reactor Vessel and Cavity Level instrumentation, LI 2(3)-263-114, is installed in place of the normal Wide Range instrumentation to indicate water level to the elevation of the refuel floor. With the refueling cavity in connected with the Spent Fuel Pool through the fuel transfer canal, uncontrolled inventory loss can be remotely monitored.

The Refueling Cavity includes the fuel transfer canal. When the Refueling Cavity is flooded to normal level, water level is approximately one foot below the refuel floor. Technical Specifications require Reactor Cavity water level be maintained at least 23 ft. above the top of the RPV flange (190 in. + 276 in.) or 466 in. when irradiated fuel or control rods are being handled within the RPV. During refueling when the RPV head is removed, plant procedures (i.e., DGP 02-02, Reactor Vessel Slow Fill, etc.) provide alternate level monitoring capabilities when the normal level instrumentation is unavailable for the desired level range or the head vent piping is removed. LI 2(3)-263-114 is located on Panel 902(3)-4 and indicates from instrument zero in the RPV to the maximum refuel floor water level (469 in. above instrument zero). In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed. DIP 0260-01, Refuel Outage Reactor Vessel and Cavity Level Instrumentation, illustrates various RPV and Refueling Cavity elevations referenced to instrument zero.

Technical Specifications require the Spent Fuel Pool water level be maintained at least 19 ft. over the top of the irradiated fuel assemblies seated in the pool racks.

Since no remote indication of Spent Fuel Pool water level exists, drops in Spent Fuel Pool water level can normally be detected only through visual observation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU2 (cont)

Basis (cont):

Threshold #2 Basis:

Valid elevated area radiation levels usually have long lead times relative to the potential for radiological release beyond the site boundary, thus impact to public health and safety is very low.

This EAL addresses unplanned increases in radiation levels inside the plant. These radiation levels represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AU2
2. RP-AA-203 Exposure Control and Authorization
3. Technical Specifications 3.7.8
4. Technical Specifications 3.9.6
5. UFSAR 9.1
6. DAN 902(3)-4 D-24 Fuel Pool Skimmer Tank Level Lo
7. DIP 0260-01 Refuel Outage Reactor Vessel and Cavity Level Instrumentation
8. DFP 0850-01 Water Level Loss in SFP or Cavity
9. DOP 1900-03 Reactor Cavity, Dryer/Separator Storage Pit and Fuel Pool Level Control
10. DGP 02-02, Reactor Vessel Slow Fill
11. DAN 902(3)-3 C-16(E-16) Reactor Building Fuel Pool Hi Radiation
12. DAN 902(3)-3 B-1 Refuel Floor Hi Radiation
13. DAN 902(3)-3 A-3(F-14) Reactor building Vent Hi-Hi Radiation

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA3

Initiating Condition:

Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions:

Table R2 – Areas Requiring Continuous Occupancy
• Main Control Room (Unit 2 ARM Station #22) • Central Alarm Station (by survey) • Secondary Alarm Station (by survey) • Radwaste Control Room (Unit 2 ARM Station #31) • Gatehouse (by survey)

OR

2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.

Table R3 – Areas Requiring Infrequent Access
• HPCI Cubicle • East and West LPCI Pump Areas • East and West CRD Module Areas • Vessel Instrument Rack Area • RWCU Area • Isolation Condenser Area

Basis:

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

RECOGNITION CATEGORY ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA3 (cont)

Basis (cont):

Threshold #1 Basis:

This EAL addresses increased radiation levels that impede necessary access to operating stations requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. Areas requiring continuous occupancy include the Main Control Room, the central alarm station (CAS) and the secondary security alarm station (SAS). The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations.

The value of 15 mR/hr is derived from the General Design Criteria (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. A 30 day duration implies an event potentially more significant than an Alert.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EALs may be involved. For example, a dose rate of 15 mR/hr in the Main Control Room may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This EAL is not intended to apply to anticipated temporary radiation increases due to planned events (i.e., radwaste container movement, depleted resin transfers, etc.).

Threshold #2 Basis:

This EAL addresses increased radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Typically areas requiring infrequent access to maintain plant safety functions include plant VITAL AREAS. Area radiation levels above 2000 mR/hr are indicative of radiation fields that may limit personnel access to equipment, the operation of which may be needed to assure adequate core cooling or shutdown the reactor.

The dose rate threshold selected is based on site administrative limits.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA3 (cont)

Basis (cont):

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved. For example, a dose rate of 2000 mR/hr may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation increases due to planned events (i.e., radwaste container movement, depleted resin transfers, etc.) or pre-existing radiation areas for which radiological controls already exist. The concern of this threshold is the unanticipated rise in radiation levels that results in unplanned restrictions to areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AA3
2. DOP 1800-01 Area Radiation Monitors
3. FSAR Section 3.2 Classification of Structures, Components and Systems
4. General Arrangement Drawings M-3, M-4, M-4A, M-5 and M-10
5. DEOP 300-2, Radioactivity Release Control

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU3

Initiating Condition:

Fuel clad degradation.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Offgas system isolation due to VALID Offgas radiation monitor high trip.

   OR

2. Specific coolant activity **> 4.0 uCi/gm** Dose Equivalent I-131.

Basis:

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

Threshold #1 Basis:

During unit operation, the steam jet air ejectors (SJAEs) remove all non-condensable gases from the main condenser including air in-leakage and disassociated products originating in the reactor and exhausts them to the offgas holdup volume. A rise in offgas activity could therefore indicate damage to the fuel cladding, a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

The gas from the main condenser normally includes relatively low levels of radioactivity. If radioactivity of the gas reaches the Off Gas Rad Monitor Hi-Hi annunciator setpoint and the Offgas isolation timer is not reset, the Offgas system isolates (i.e. chimney isolation valve auto closes) after a fifteen-minute time delay. The fifteen-minute time delay is allotted for operator action to reduce the offgas radiation levels and exclude transient conditions.

The modifier “VALID” is appropriate because there are several conditions that may cause the monitor to alarm that are not related to fuel clad degradation and therefore should not result in the declaration of an Unusual Event.

RU3 (cont)

Basis Reference(s):

Threshold #2 Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding coolant Technical Specifications for iodine spiking. The specific iodine activity ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) accident is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits.

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 Reactor Coolant activity unless another cause is known, i.e., Reactor Coolant System chemical decontamination evolution (during shutdown) is ongoing with resulting high activity levels.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU4
2. Technical Specifications 3.4.6
3. DAN 902(3)-3 C-2(D-2) Off Gas Rad Monitor Hi-Hi
4. Technical Specifications 3.7.6, Main Condenser Offgas
5. DGA 16 Coolant High Activity/Fuel Element Failure

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FG1

Initiating Condition:

Loss of ANY two barriers AND Loss or Potential Loss of the third barrier.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the General Emergency classification level each barrier is weighted equally.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FS1

Initiating Condition:

Loss or Potential Loss of ANY two barriers.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Site Area Emergency classification level, each barrier is weighted equally.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FA1

Initiating Condition:

ANY Loss or ANY Potential Loss of either Fuel Clad or RCS.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Alert classification level, 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FU1

Initiating Condition:

ANY Loss or ANY Potential Loss of Containment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

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 EAL 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC1 – Loss

Initiating Condition:

Primary coolant activity level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.

Basis:

Loss Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

300 uCi/gm Dose Equivalent I-131 is well above that expected for iodine spikes and corresponds, generically, to about 2% to 5% fuel cladding damage. When reactor coolant activity reaches this level, significant clad damage has occurred and thus the Fuel Cladding barrier is considered lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. UFSAR Section 9.3.2.1
3. DGA-16, Coolant High Activity / Fuel Element Failure

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC2 – Loss or Potential Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. RPV level **< – 164 in.** without at least one core spray pump **> 4750 gpm.**

   OR

2. RPV level **< – 191 in**.

POTENTIAL LOSS

RPV level **< – 143 in.** (TAF).

Basis:

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the Technical Support Guidelines (TSG).

Loss Basis:

When primary containment flooding is required, all symptom-based EOPs (DEOPs) are exited and the SAMGs are entered in order to restore and maintain cooling to the core and any core debris. Since it may not be possible to recover the core inside the RPV, flooding the primary containment to the elevation of the top of active fuel in the drywell may be required. EOPs require primary containment flooding when:

- DEOP 100, RPV Control:

  Cannot restore level above -164 in. and hold it there and neither Core Spray loop flow is at or above 4,750 gpm.

  -164 in. is the Minimum Steam Cooling RPV Water Level (MSCRWL). At or above the MSCRWL, the covered portion of the core generates sufficient steam to prevent any cladding temperature in the uncovered part of the core from exceeding 1500° F (threshold temperature for fuel rod perforation).

  OR

  Cannot restore level at or above -191 in. and hold it there.

  -191 in. is the top of the jet pump risers or ~2/3 core height. Water level held at or above this elevation with the specified design Core Spray loop flow satisfies the core cooling requirements of the Design Basis Accident LOCA event.

- DEOP 400-5, Failure to Scram: Cannot restore level above -164 in.
- DEOP 400-1, RPV Flooding: Core damage is occurring.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC2 – Loss or Potential Loss (cont)

Basis (cont):

Potential Loss Basis:

Core submergence is the preferred method of core cooling and as such, the failure to re-establish RPV level above the top of active fuel for an extended period of time could lead to significant fuel damage.

An RPV level reading of -143 in. indicates RPV level is at the top of active fuel (TAF). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling).

If core uncovery is threatened, the DEOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops below TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. DEOP 100 RPV Control
3. DEOP 400-5 Failure to Scram
4. DEOP 400-1 RPV Flooding
5. DEOP 0010-00 Guidelines for Use of Dresden Emergency Operating Procedures and Severe Accident Management Guidelines

FC5 – Loss

Initiating Condition:

Drywell radiation monitoring.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Drywell radiation monitor reading **> Fuel Cladding Loss Threshold**, Table F1.

Table F1 – Drywell Radiation Thresholds

Time After Shutdown (hours)	Fuel Cladding Loss (R/hr)
≤ 2	6.70 E+02
> 2 to 4	5.90 E+02
> 4 to 8	5.10 E+02
> 8 to 16	4.30 E+02
> 16 to 23	3.90 E+02
> 23	3.80 E+02

Basis:

The drywell radiation monitor readings specified in Table F1 provide values that indicate the release of reactor coolant into the drywell with elevated activity indicative of fuel damage (~2%). The values are a function of time after shutdown and were derived using Core Damage Assessment Methodology (CDAM) with 2% clad damage, no drywell sprays in operation and a LOCA depressurized system. The reading is calculated assuming the instantaneous release and dispersal of the above reactor coolant noble gas and iodine inventory into the drywell atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations allowed within Technical Specifications (including iodine spiking) and are therefore indicative of fuel damage (approximately 2% - 5% cladding failure).

During at power (including ATWS) conditions the value listed for the "< 2 hours after shutdown" row is used as an indication of fuel damage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.

Basis:

The Emergency Director judgment fuel cladding loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Fuel Cladding barrier. The inability to monitor fuel cladding integrity should also be considered as a factor in judging that the Fuel Cladding barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC2 – Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

RPV level **< - 143 in.** (TAF).

Basis:

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the Technical Support Guidelines (TSG).

Loss Basis:

RPV level reading of -143 in. indicates RPV level is at the top of active fuel (TAF). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment barriers, and initiation of all ECCS. If RPV level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of any unplanned loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a loss of the RCS barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. DEOP100 RPV Control
3. DEOP 0010-00 Guidelines for Use of Dresden Emergency Operating Procedures and Severe Accident Management Guidelines

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC3 – Loss

Initiating Condition:

Drywell pressure.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. Drywell pressure **> 2.0 psig**.

   AND

2. Drywell pressure rise due to RCS leakage.

Basis:

The drywell pressure value is the drywell high pressure ECCS initiation setpoint (2.0 psig analytical limit) and is therefore indicative of a Loss of Coolant Accident (LOCA) event that requires ECCS response. Elevated drywell pressure also causes a reactor scram (2.0 psig) and is an entry condition to DEOP 100, RPV Control, and DEOP 200-1, Primary Containment Control. Normal primary containment pressure control functions (i.e., operation of drywell cooling, Standby Gas Treatment system, etc.) are specified in DEOP 200-1 in advance of less desirable but more effective functions (i.e., operation of drywell or torus sprays, etc.).

In the Dresden design basis, primary containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the increasing pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control primary containment vent/purge.

The second threshold focuses the fission product barrier loss threshold on a failure of the RCS instead of the non-LOCA malfunctions that may adversely affect primary containment pressure.

Therefore:

- Drywell pressure greater than 2.0 psig with corollary indications (increase drywell temperature with drywell cooling system normal) should therefore be considered a loss of RCS.
- Loss of drywell cooling that results in greater than 2.0 psig should not be considered a loss of RCS.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC3 – Loss (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Technical Specifications Table 3.3.5.1-1
3. DAN 902(3)-5 D-11
4. DEOP 100 RPV Control
5. DEOP 200-1 Primary Containment Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC4 – Loss or Potential Loss

Initiating Condition:

RCS leak rate.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. UNISOLABLE Main Steam Line (MSL) break as indicated by the failure of both MSIVs in ANY one line to close.

   AND

2 a. High MSL Flow AND High Steam Tunnel Temperature.

   OR

   b. Direct report of steam release.

POTENTIAL LOSS

1. RCS leakage **> 50 gpm** inside the drywell.

   OR

2. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> DEOP 300-1 Maximum Normal** operating levels.

Basis:

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Loss Basis:

High Steam Flow and High Steam Tunnel Temperature Annunciators are both indications of a Main Steam Line Break. Both of these parameters will cause a signal for closure of the MSIVs. Should both valves in any one line fail to isolate, this event would be considered a Loss of the RCS.

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside primary containment exists when flow is not prevented by downstream isolations. In the case of a failure of both Main Steam Isolation Valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required.

Direct report of steam release is meant to provide an alternate means of classification if the Hi Steam Flow Annunciator or the Hi Steam Tunnel Temperature Annunciator fails to operate and the observation of conditions indicates a Main Steam Line Break in the judgment of the Emergency Director. This is not meant to cause a declaration based on leaks such as valve packing leaks where the consequences offsite would be negligible.

Basis (cont):

Potential Loss Threshold #1 Basis:

The potential loss of RCS based on leakage is set at a level indicative of a small breach of the RCS but which is well within the makeup capability of normal and emergency high-pressure systems. Core uncovery is not a significant concern for a 50 gpm leak; however, break propagation leading to significantly larger loss of inventory is possible. RCS leakage inside the drywell is normally determined by monitoring drywell equipment and floor drain sump pumpout rates. This method of monitoring leakage may be isolated as part of the drywell isolation, and thus may be unavailable. If primary system leak rate information is unavailable other indicators of drywell leakage should be used. When the reactor is at power, it is expected that an RCS leak of this magnitude will increase drywell pressure above the isolation setpoint. Inventory loss events, such as a stuck open SRV, should not be considered when referring to “RCS leakage” because they are not indications of a break, which could propagate.

Potential Loss Threshold #2 Basis:

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of an unisolable primary system leakage outside the primary containment. The maximum normal values define this RCS threshold because it is the maximum normal operating value and signifies the onset of abnormal system operation. When parameters reach this level, equipment failure or mis-operation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in DEOP-300-1, Secondary Containment Control.

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the reactor building since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (i.e. room flooding, high area temperatures, reports of steam in the Reactor Building, an unexpected rise in Feedwater flowrate, or unexpected Main Turbine Control Valve closure) may indicate that a primary system is discharging into the Reactor Building.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC4 – Loss or Potential Loss (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. M-12, M-345, Main steam piping
3. Technical Specifications 3.4.4 RCS Operational LEAKAGE
4. Technical Specifications Section 3.4.5, RCS Leakage Detection Instrumentation
5. DAN 902(3)-4 A-17 DRYWELL EQUIP SUMP LVL HI-HI
6. DAN 902(3)-4 H-18 DRYWELL FLOOR DRN SUMP LVL HI-HI
7. DOA 0040-01 SLOW LEAK
8. DOP 2000-24 DRYWELL SUMP OPERATION
9. DEOP 300, Secondary Containment Control
10. UFSAR Section 5.2.5

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC5 – Loss

Initiating Condition:

Drywell Radiation Monitoring

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Drywell Radiation monitor reading **> 100 R/hr.**

   AND

2. Indications of RCS leakage into the Drywell.

Basis:

The drywell radiation monitor reading is a value that indicates a significant release of reactor coolant to the Drywell. A reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the Drywell atmosphere. Conservative estimates (high RCS uCi/cc) indicated that the readings from release of the normal RCS inventory would be ~ 100 R/hr. The reading is less than that specified for Fuel Cladding barrier Loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations.

Indication of a RCS leak into the drywell is added to qualify the radiation monitor indication to avoid declaring the loss of RCS barrier for situations where the radiation rise is not due to primary a system leak. For situations that involve failure of the Fuel Clad barrier alone, radiation monitor readings would rise due to shine and potentially giving a false indication of a loss of the RCS barrier. Therefore this EAL contains a qualifier to preclude over classification of the event if only fuel clad barrier failed.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Calc. EP-EAL-0611

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.

Basis:

The Emergency Director judgment RCS loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the RCS barrier. The inability to monitor RCS integrity should also be considered as a factor in judging that the RCS barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT2 – Potential Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

POTENTIAL LOSS

Plant conditions indicate that Primary Containment Flooding is required.

Basis:

Potential Loss Basis:
When primary containment flooding is required, all EOPs (DEOPs) are exited and the SAMGs are entered in order to restore and maintain cooling to the core and any core debris. Since it may not be possible to recover the core inside the RPV, flooding the primary containment to the elevation of the top of active fuel in the drywell may be required.

The EOP conditions requiring primary containment flooding represent imminent core melt sequences that, if not corrected, could lead to RPV failure and increased potential for containment failure.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. DEOP 100 RPV Control
3. DEOP 400-5 Failure to Scram
4. DEOP 400-1 RPV Flooding

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT3 – Loss or Potential Loss

Initiating Condition:

Drywell pressure.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. Rapid unexplained drop in Drywell Pressure following initial pressure rise.

   OR

2. Drywell pressure response not consistent with LOCA conditions.

POTENTIAL LOSS

1. Drywell pressure **≥ 62 psig** and rising.

   OR

2. a. Drywell or torus hydrogen concentration **≥ 6%.**

      AND

   b. Drywell or torus oxygen concentration **≥ 5%.**

Basis:

Loss Threshold #1 Basis:

Rapid unexplained loss of pressure (i.e., not attributable to drywell sprays, torus sprays or condensation effects) following an initial pressure rise indicates a loss of containment integrity.

Loss Threshold #2 Basis:

Drywell pressure should rise as a result of mass and energy release into the containment from a LOCA. Thus, drywell pressure response not consistent with LOCA conditions indicates a loss of containment integrity. This indicator relies on operator recognition of an unexpected response for the condition and therefore does not include a specific pressure value or trend. Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. For example, blowdown mass flowrate may be only 60-80% of the analyzed rate. The unexpected response is important because it is the indicator for a containment bypass condition.

Potential Loss Threshold #1 Basis:

When the Primary Containment design pressure is challenged, primary containment venting is required even if offsite radioactivity release rate limits will be exceeded. This condition, if compounded by further plant degradation may challenge primary containment integrity and is, therefore, an appropriate threshold for potential loss of the Primary Containment barrier.

CT3 – Loss or Potential Loss (cont)

Basis (cont):

A Drywell pressure of 62 psig is based on the containment/drywell design pressure. If the containment design pressure is exceeded this represents a challenge to the containment structure because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a potential loss of the containment barrier even if a breach has NOT occurred.

Potential Loss Threshold #2 Basis:

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

Except for brief periods during plant startup and shutdown, oxygen concentration in the primary containment is maintained at insignificant levels by nitrogen inertion. The specified values for this potential loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen concentration) and readily recognizable because 6% hydrogen is above the hydrogen monitor alarm setpoint (2%) and the Primary Containment Control EOP entry condition. The minimum global deflagration hydrogen/oxygen concentrations (6% and 5%, respectively) require intentional primary containment venting, which is defined to be a barrier loss under Primary Containment barrier Loss CT6.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. UFSAR 6.2.1.3.2.1
3. UFSAR Table 6.2-3
4. UFSAR 15.6.5
5. UFSAR 6.2.1.1
6. DEOP 200-1 Primary Containment Control
7. DEOP 200-2 Hydrogen Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT5 – Potential Loss

Initiating Condition:

Significant radioactive inventory in Containment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

POTENTIAL LOSS

Drywell radiation monitor reading **> Containment Potential Loss Threshold**, Table F2.

Table F2 - Drywell Radiation Thresholds

Time After Shutdown (hours)	Containment Potential Loss (R/hr)
≤ 2	1.60 E+03
> 2 to 4	1.35 E+03
> 4 to 8	1.20 E+03
> 8 to 16	1.00 E+03
> 16 to 23	8.75 E+02
> 23	8.60 E+02

Basis:

The drywell radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the Fuel Cladding barrier. NUREG-1228 “Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents” states that such readings do not exist when the amount of cladding damage is less than 20%. The values are a function of time after shutdown and were derived using Core Damage Assessment Methodology (CDAM) assuming 20% clad damage, no drywell sprays in operation and a LOCA depressurized system. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a significant failure into the reactor coolant has occurred.

During at power (including ATWS) conditions the value listed for the “< 2 hours after shutdown” row is used as an indication of fuel damage.

Regardless of whether the Primary Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Primary Containment barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT6 – Loss

Initiating Condition:

Containment isolation failure or bypass.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

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

   AND

   b. A downstream pathway to the environment exists.

   OR

2. Intentional venting/purging of Primary Containment per EOPs or SAMGs due to accident conditions.

   OR

3. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> DEOP 300-1, Maximum Safe** operating levels.

Basis:

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Threshold #1 Basis:

This threshold addresses failure of open isolation devices that should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway to the environment. The concern is the unisolable open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of primary containment integrity.

Failure of containment isolation valves to isolate with a downstream pathway to the environment is only a concern during an accident. If this condition exists during normal Power Operation, a Technical Specification Action Statement will address it. However, during accident conditions, this will represent a breach of Primary Containment.

Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include unisolable Main steamline breaks, HPCI steamline breaks, unisolable RWCU system breaks, and unisloable containment atmosphere vent paths. Minor release paths such as instrument and sample lines are not considered under this threshold.

Examples of “downstream pathway to the Environment” could be through Turbine/Condenser, or direct release to the Turbine Building or Reactor Building.

CT6 – Loss (cont)

Basis (cont):

The breach is NOT isolable from the Control Room if an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation from the Control Room should be made prior to the accident classification. If Operator actions from the Control Room are successful, then this IC is not applicable. Credit is NOT given for Operator actions taken in-plant (outside the Control Room) to isolate the leak.

This EAL is intended to cover containment isolation failures allowing a direct flow path to the environment such as failure of both MSIVs to close with open valves downstream to the turbine or to the condenser, even if these systems are not breached.

Threshold #2 Basis:

Intentional venting of the primary containment to the secondary containment and/or the environment per the DEOPs/SAMGs due to accident conditions is considered a loss of the Primary Containment barrier.

Threshold #3 Basis:

The presence of elevated general area temperatures and/or area radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. Temperatures and radiation levels beyond their maximum safe operating temperatures are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This EAL threshold addresses problematic discharges outside primary containment that may not originate from a high-energy line break.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. DEOP 200-1 Primary Containment Control
3. DEOP 200-2 Hydrogen Control
4. DEOP 500-4 Containment Venting
5. DEOP 300-1 Secondary Containment Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Containment barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment barrier.

Basis:

The Emergency Director judgment Containment loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Containment barrier. The inability to monitor Containment parameters should also be considered as a factor in judging that the Containment barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG1

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of power to Reserve Auxiliary Transformer TR-22(32) and Unit Auxiliary Transformer TR-21(31).

   AND

2. Failure of DG 2(3), shared DG 2/3 and SBO DG 2(3) emergency diesel generators to supply power to unit ECCS busses.

   AND

3. a. Restoration of at least one unit ECCS bus within 4 hours is **not** likely.

      OR

   b. RPV level **cannot** be determined to be **> -143 in.** (TAF).

Basis:

Loss of all AC power to ECCS busses compromises the availability of all plant safety systems. Prolonged loss of all AC power may lead to loss of Fuel Cladding, RCS and Primary Containment barriers. The one-hour interval to restore AC power to either unit ECCS bus is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout."

The likelihood of restoring at least one ECCS bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

Units 2 and 3 each have two independent sources of offsite power available:

- Unit 2 – The normal source of offsite power is supplied by the 345-kV switchyard through TR86 and Reserve Auxiliary Transformer (RAT) TR-22. The alternate source of offsite power is supplied by the 345-kV switchyard through the 4160-V crosstie between busses 24-1 and 34-1 or busses 23-1 and 33-1.
- Unit 3 – The normal source of offsite power is supplied by the 345-kV switchyard through RAT TR-32. The alternate source of offsite power is supplied by the 345-kV switchyard through the 4160-V crosstie between busses 34-1 and 24-1 or safety busses 23-1 and 33-1.

Basis (cont):

- Onsite AC power sources to each unit include the UAT powered from the unit main generator and emergency diesel generators: DG 2(3), shared DG 2/3 and the station blackout diesel generators SBO DG 2(3). The station blackout system is a non-safety-related, independent source of additional onsite emergency AC power. Each SBO DG is connectable, but not normally connected, to the safe shutdown equipment on one unit, but can also be connected to the opposite unit via the 4160-V crossties.

In addition to the above independent sources, another source from offsite is available by virtue of removable links in the main generator isolated phase bus. When removed, a unit auxiliary transformer 21(31) can be backfed from the 345-kV system through the main power transformer. This source is similar to items 2 and 3 discussed above; however, it is available only when the unit is shut down and the generator disconnected (due to the time required to effect the backfeed, this source is likely only to be available when previously configured).

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the Technical Support Guidelines (TSG).

A reading of 143 in. below instrument zero indicates RPV level is at the top of active fuel (TAF). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops below TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

Basis (cont):

Consideration should be given to operable loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not operable on the energized bus, the bus should not be considered operable. The Emergency Director should address how the loss of power to the affected unit may affect the operation of the other unit due to the possible loss of power for common or shared systems.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SG1
2. UFSAR 8.3
3. 12E-2302A, Station Key Diagram 4160V and 480V Switchgears Part 1
4. DOA-6400-01, 138-kV System and 345-kV Alternate Supply Failure
5. DOA 6500-01 4-KV Bus Failure
6. UFSAR Fig. 9.5-14 Single-Line Electrical Diagram of Station Blackout Generator Ties to Plant Auxiliary Electric System
7. UFSAR 9.5.9
8. DOP 6620-05, Powering Unit 2(3) 4-KV Busses via the SBO D/G 2(3)
9. DGA-12 Partial or Complete Loss of AC Power
10. DEOP100 RPV Control
11. DEOP 0010-00 Guidelines for Use of Dresden Emergency Operating Procedures and Severe Accident Management Guidelines

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS1

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of power to Reserve Auxiliary Transformer TR-22(32) and Unit Auxiliary Transformer TR-21(31).

   AND

2. Failure of DG 2(3), shared DG 2/3 and SBO DG 2(3) emergency diesel generators to supply power to unit ECCS busses.

   AND

3. Failure to restore power to at least one ECCS bus within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power compromises all plant safety systems and represents failures of plant functions required for the protection of the public. Units 2 and 3 each have two independent sources of offsite power available:

- Unit 2 – The normal source of offsite power is supplied by the 345-kV switchyard through TR86 and Reserve Auxiliary Transformer (RAT) TR-22. The alternate source of offsite power is supplied by the 345-kV switchyard through the 4160-V crosstie between busses 24-1 and 34-1 or busses 23-1 and 33-1.
- Unit 3 – The normal source of offsite power is supplied by the 345-kV switchyard through RAT TR-32. The alternate source of offsite power is supplied by the 345-kV switchyard through the 4160-V crosstie between busses 34-1 and 24-1 or safety busses 23-1 and 33-1.

An additional source of offsite power is available when the main generator is off-line by backfeeding through the main power transformers and UATs. The backfeed operation must be manually performed and involves removal of flexible link connections between the main generator and the main power transformers and UATs (due to the time required to effect the backfeed, this source is likely only to be available when previously configured).

Onsite AC power sources to each unit include the UAT powered from the unit main generator and emergency diesel generators: DG 2(3), shared DG 2/3 and the station blackout diesel generators SBO DG 2(3). The station blackout system is a non-safety-related, independent source of additional onsite emergency AC power. Each SBO DG is connectable, but not normally connected, to the safe shutdown equipment on one unit, but can also be connected to the opposite unit via the 4160-V crossties.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont):

Consideration should be given to available loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not available on the energized bus, the bus should not be considered available.

The fifteen-minute interval begins from the time of loss of both onsite and offsite AC power and was selected as a threshold to exclude transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS1
2. UFSAR 8.3
3. 12E-2302A, Station Key Diagram 4160V and 480V Switchgears Part 1
4. DOA-6400-01, 138-kV System and 345-kV Alternate Supply Failure
5. DOA 6500-01 4KV Bus Failure
6. UFSAR Fig. 9.5-14 Single-Line Electrical Diagram of Station Blackout Generator Ties to Plant Auxiliary Electric System
7. UFSAR 9.5.9
8. DOP 6620-05, Powering Unit 2(3) 4KV Busses via the SBO D/G 2(3)
9. DGA-12 Partial or Complete Loss of AC Power

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA1

Initiating Condition:

AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. AC power capability to unit ECCS busses reduced to only one of the following power sources for **> 15 minutes**:
   - Reserve auxiliary transformer TR-22(TR-32)
   - Unit auxiliary transformer TR-21(TR-31)
   - Unit Emergency Diesel Generator DG 2(3)
   - Shared Emergency Diesel Generator DG 2/3
   - Station Blackout Diesel Generator SBO DG 2(3)
   - Unit crosstie breakers

   AND

2. Any additional single power source failure will result in unit blackout.

Basis:

Capability: (pertaining to electrical power supplies) is equipment that is available to provide and maintain AC power at the required voltage and frequency for the required load.

The reduction of available reliable power sources to a condition in which any additional single failure will result in a Unit Blackout is a substantial degradation in the level of safety of the plant. A Unit Blackout is a loss of AC power to all unit ECCS busses. Dresden blackout coping duration is four hours.

The listed power supplies take into account sources that, if unavailable, establish single-failure vulnerability. This EAL allows for the use of the unit crosstie breakers if they are the only source of power to the affected unit. The Emergency Director must consider the use of the crosstie breakers and the consequent demand on the unaffected unit.

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA5
2. UFSAR 8.3
3. 12E-2302A, Station Key Diagram 4160V and 480V Switchgears Part 1
4. DOA-6400-01, 138 KV System and 345 KV Alternate Supply Failure
5. DOA 6500-01 4KV Bus Failure
6. UFSAR Fig. 9.5-14 Single-Line Electrical Diagram of Station Blackout Generator Ties to Plant Auxiliary Electric System
7. UFSAR 9.5.9 Station Blackout System
8. DOP 6620-05, Powering Unit 2(3) 4KV Busses via the SBO D/G 2(3)
9. DGA-12 Partial or Complete Loss of AC Power

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU1

Initiating Condition:

Loss of all offsite power to essential busses for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3, 4, 5

EAL Threshold Values:

Loss of power to Reserve Auxiliary Transformer TR-22(32) AND Unit Auxiliary Transformer TR-21(31) for **> 15 minutes**.

Basis:

The Essential busses are the safety-related, ECCS busses 23-1(33-1) and 24-1(34-1). Units 2 and 3 are each provided offsite power as described below:

- Unit 2 – The normal source of offsite power is supplied by the 345-kV switchyard through TR86 and Reserve Auxiliary Transformer (RAT) TR-22.
- Unit 3 – The normal source of offsite power is supplied by the 345-kV switchyard through RAT TR-32.

Each ECCS bus can be powered by emergency diesel generators: DG 2(3), shared DG 2/3 and the station blackout diesel generators SBO DG 2(3). The station blackout system is a non-safety-related, independent source of additional onsite emergency AC power. Each SBO DG is connectable, but not normally connected, to the safe shutdown equipment on one unit, but can also be connected to the opposite unit via the 4160-V crossties.

Loss of offsite power causes a reactor scram and primary containment isolation. Emergency diesel generators DG2(3) and DG2/3 should automatically start and be available to carry the essential loads for each affected unit. Balance of plant systems that would assist in plant operations (i.e., condensate pumps, etc.) may be unavailable due to the loss of power. The station blackout diesel generators must be manually started and connected to the appropriate safe shutdown loads.

A loss of offsite AC power reduces the 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.

The intent of this EAL is to declare an Unusual Event when offsite power has been lost and the emergency diesel generators have successfully started and energized the ECCS busses. The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU1 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU1 & CU3
2. UFSAR 8.3
3. 12E-2302A, Station Key Diagram 4160V and 480V Switchgears Part 1
4. DOA-6400-01, 138 kV System and 345 kV Alternate Supply Failure
5. DOA 6500-01 4kV Bus Failure
6. UFSAR Fig. 9.5-14 Single-Line Electrical Diagram of Station Blackout Generator Ties to Plant Auxiliary Electric System
7. UFSAR 9.5.9
8. DOP 6620-05, Powering Unit 2(3) 4kV Busses via the SBO D/G 2(3)
9. DGA-12 Partial or Complete Loss of AC Power

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA2

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability:

4, 5, D

EAL Threshold Values:

1. Loss of power to Reserve Auxiliary Transformer TR-22(32) and Unit Auxiliary Transformer TR-21(31).

   AND

2. Failure of DG 2(3), shared DG 2/3 and SBO DG 2(3) emergency diesel generators to supply power to unit ECCS busses.

   AND

3. Failure to restore power to at least one unit ECCS bus within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power when in Cold Shutdown, Refueling or Defueled modes compromises safety systems required for decay heat removal and represents a substantial degradation of the level of safety of the plant. An Alert declaration (instead of a Site Area Emergency under EAL MS1) is appropriate in these modes because post-shutdown, decay heat energy levels offer more time to restore AC power to heat removal systems than the levels present when the reactor is in Power Operation, Startup or Hot Shutdown mode. Thus, the threat to the protection of the health and safety of the public is less severe.

Units 2 and 3 each have two independent sources of offsite power available:

- Unit 2 – The normal source of offsite power is supplied by the 345-kV switchyard through TR86 and Reserve Auxiliary Transformer (RAT) TR-22. The alternate source of offsite power is supplied by the 345-kV switchyard through the 4160-V crosstie between busses 24-1 and 34-1 or busses 23-1 and 33-1.
- Unit 3 – The normal source of offsite power is supplied by the 345-kV switchyard through RAT TR-32. The alternate source of offsite power is supplied by the 345-kV switchyard through the 4160-V crosstie between busses 34-1 and 24-1 or safety busses 23-1 and 33-1.

An additional source of offsite power is available when the main generator is off-line by backfeeding through the main power transformers and UATs. The backfeed operation must be manually performed and involves removal of flexible link connections between the main generator and the main power transformers and UATs (due to the time required to effect the backfeed, this source is likely only to be available when previously configured).

MA2 (cont)

Basis (cont):

Onsite AC power sources to each unit include the UAT powered from the unit main generator and emergency diesel generators: DG 2(3), shared DG 2/3 and the station blackout diesel generators SBO DG 2(3). The station blackout system is a non-safety-related, independent source of additional onsite emergency AC power. Each SBO DG is connectable, but not normally connected, to the safe shutdown equipment on one unit, but can also be connected to the opposite unit via the 4160-V crossties.

Consideration should be given to available loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not available on the energized bus, the bus should not be considered available.

The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA3
2. UFSAR 8.3
3. 12E-2302A, Station Key Diagram 4160V and 480V Switchgears Part 1
4. DOA-6400-01, 138 KV System and 345 KV Alternate Supply Failure
5. DOA 6500-01 4KV Bus Failure
6. UFSAR Fig. 9.5-14
7. UFSAR 9.5.9
8. DOP 6620-05, Powering Unit 2(3) 4KV Busses via the SBO D/G 2(3)
9. DGA-12 Partial or Complete Loss of AC Power

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG3

Initiating Condition:

Failure of the Reactor Protection System to complete an automatic scram and manual scram was NOT successful and there is indication of an extreme challenge to the ability to cool the core.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**:
   a. Reactor power remains **> 6% APRM.**

      OR

   b. Torus temperature **> 110° F AND** boron injection required for reactivity control.

   AND

2. a. RPV level cannot be restored and maintained **> - 164 in**.

      OR

   b. Heat Capacity Limit (DEOP 200-1 Fig. M) exceeded.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed The reactor power threshold (6%) is approximately equal to the APRM downscale trip setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 6% power. Classification at the General Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG3 (cont)

Basis (cont):

The torus water temperature criterion (110°F) is the Boron Injection Initiation Temperature (BIIT). The BIIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the torus heats the suppression pool to the Heat Capacity Limit (HCL). If torus temperature exceeds the BIIT, reactor power is heating the torus and the suppression pool cooling may be inadequate or incapable of performing its design function.

The second condition of this EAL indicates either:

- An extreme challenge to the ability to cool the core as indicated when RPV level cannot be maintained above -164 in. The specified water level is the Minimum Steam Cooling RPV Water Level (MSCRWL). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500° F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core cooling could be a precursor of a core melt sequence.

  RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the Technical Support Guidelines (TSG).
- An extreme challenge to the primary containment as indicated when heat cannot be removed from the primary containment resulting in elevated torus bulk temperature. The Heat Capacity Limit is the highest torus bulk temperature from which a blowdown will not raise torus bottom pressure above the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer from the RPV to the primary containment is within the capacity of the primary containment vent. (When the PCPL is challenged, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded.) The Heat Capacity Limit is a function of RPV pressure and torus bulk temperature and is a measure of the maximum heat load that the primary containment can withstand. Plant parameters in excess of the Heat Capacity Limit could be a precursor of primary containment failure. The Heat Capacity Limit is given in Detail M of DEOP 200-1, Primary Containment Control.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SG2
2. DEOP 100 RPV Control
3. DEOP 400-5 Failure to Scram
4. DEOP 200-1 Primary Containment Control

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS3

Initiating Condition:

Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded and manual scram was NOT successful.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**:

1. Reactor power remains **> 6% APRM.**

   OR

2. Torus bulk temperature **> 110° F AND** boron injection required for reactivity control.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed The reactor power threshold (6%) is approximately equal the APRM downscale trip setpoint and to the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 6% power.

The torus water temperature criterion is the Boron Injection Initiation Temperature (BIIT). The BIIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the torus heats the suppression pool to the Heat Capacity Limit (HCL). If torus temperature exceeds the BIIT, reactor power is heating the torus and the suppression pool cooling may be inadequate or incapable of performing its design function.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS3 (cont.)

Basis (cont.):

Classification at the Site Area Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS2
2. DEOP 100 RPV Control
3. DEOP 400-5 Failure to Scram
4. DEOP 200-1 Primary Containment Control

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA3

Initiating Condition:

Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a reactor protection system setpoint has been exceeded.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. A Reactor Protection System setpoint was exceeded.

   AND

2. Automatic scram did not reduce reactor power to **< IRM Range 7 and lowering**.

Basis:

This condition indicates a failure of the automatic reactor protection system to successfully scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. Site-specific indication of reactor shutdown is included as the criteria of whether the scram was successful when required. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue.

A successful scram has occurred when there is sufficient rod insertion to bring the reactor subcritical (< IRM Range 7 and lowering).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA3 (cont)

Basis (cont):

The second condition of this EAL indicates a failure of the automatic RPS scram function to rapidly insert a sufficient number of control rods to achieve reactor shutdown. The CRD system backup scram valves and the Alternate Rod Insertion (ARI) system provide automatic, alternate methods of completing the scram function. These backups, however, insert control rods at a much slower rate than the automatic RPS scram function. For the purpose of emergency classification at the Alert level, reactor shutdown achieved by automatic backup scram valve operation and ARI initiation does not constitute a successful RPS automatic scram.

Following any automatic RPS scram signal DEOP 400-5, Failure to Scram, prescribes insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Alert.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded, or first-out annunciators), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA2
2. Technical Specifications Table 3.3.1.1-1
3. DEOP 100 RPV Control
4. DEOP 400-5 Failure to Scram

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU3

Initiating Condition:

Inadvertent criticality.

Operating Mode Applicability:

3, 4, 5

EAL Threshold Values:

An UNPLANNED extended positive period observed on nuclear instrumentation.

Basis:

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

The term “extended” is used in order to allow exclusion of expected short-term positive periods from planned fuel bundle or control rod movements during core alteration. These short-term positive periods are the result of the rise in neutron population due to subcritical multiplication.

This EAL includes criticality events that occur in Cold Shutdown or Refueling modes (NUREG1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events as well as inadvertent criticalities occurring in Hot Shutdown mode. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification.

This condition can be identified using:

- SRM Channel 21, 22, 23 and 24 period meters on Panel 902(3)-5
- Amber short period lights on Panel 902(3)-5
- Annunciator 902(3)-5 E-4, SRM SHORT PERIOD
- Computer Point 975, "SRM SHORT PERIOD".

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU8 & CU8
2. Technical Specifications 3.3.1.2
3. DOS 0700-01 SRM Functional Test
4. DAN 902(3)-5 E-4 SRM Short Period

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS4

Initiating Condition:

Loss of all vital DC power.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Loss of all vital DC power based on **< 105 VDC** on 125 VDC battery busses #2 and #3 for **> 15 minutes**.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of Containment integrity when there is significant decay heat and sensible heat in the reactor system.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. A total of four (250 VDC safety related, 250 VDC non safety related, 125 VDC and 24/48 VDC) station battery systems are provided for each unit. One safety related 250-V power battery is provided to serve the larger loads, such as DC motor-driven pumps and valves. Two shared 125 VDC batteries are provided to supply the power required for exit lighting and all DC control functions such as that required for control of the 4160 VAC breakers, 480 VAC breakers, various control relays, and annunciators. The 125 VDC battery is, therefore, the battery of interest for this EAL. An alternate 125 VDC battery is provided to allow rated discharge testing of the unit 125 VDC battery while both units remain at power. The alternate 125 VDC battery is also available, in accordance with Technical Specifications, if the unit 125 VDC battery is inoperable.

The ampere-hour capacity of each battery is adequate to supply expected essential loads following station trip and loss of all AC power without battery terminal voltage falling below 105 VDC terminal voltage, the minimum discharge level.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS3
2. UFSAR 8.3.2
3. DOA 6900-02(3) Failure of Unit 2(3) 125 VDC Power Supply
4. Technical Specification B.3.8.4, DC Power Sources - Operating

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU4

Initiating Condition:

UNPLANNED loss of required DC power for greater than 15 minutes.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. UNPLANNED Loss of all required Vital DC power based on **< 105 VDC** indication on 125 VDC battery busses #2 and #3.

   AND

2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.

Basis:

<u>UNPLANNED:</u> A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

“Unplanned activities” is included in this EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, plants perform maintenance on a bus-related basis during shutdown periods.

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown, refueling or defueled operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. A total of four (250 VDC safety related, 250 VDC non safety related, 125 VDC and 24/48 VDC) station battery systems are provided for each unit. One safety related 250-V power battery is provided to serve the larger loads, such as DC motor-driven pumps and valves. Two shared 125 VDC batteries are provided to supply the power required for exit lighting and all DC control functions such as that required for control of the 4160 VAC breakers, 480 VAC breakers, various control relays, and annunciators. The 125 VDC battery is, therefore, the battery of interest for this EAL. An alternate 125 VDC battery is provided to allow rated discharge testing of the unit 125 VDC battery while both units remain at power. The alternate 125 VDC battery is also available, in accordance with Technical Specifications, if the unit 125 VDC battery is inoperable.

The ampere-hour capacity of each unit 125 VDC battery is adequate to supply expected essential loads following station trip and loss of all AC power, without the battery terminal voltage falling below the minimal discharge level (105 VDC).

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU7
2. UFSAR 8.3.2
3. DOA 6900-02(3) Failure of Unit 2(3) 125 VDC Power Supply
4. Technical Specification B.3.8.4, DC Power Sources - Operating

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS5

Initiating Condition:

Complete loss of heat removal capability.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Heat Capacity Limit (DEOP 200-1 Fig. M) exceeded.

Basis:

Plant parameters associated with the Heat Capacity Limit (DEOP 200-1 Fig. M) are RPV pressure, torus water level and torus bulk temperature. The Heat Capacity Limit is the highest torus bulk temperature from which a blowdown will not raise torus bottom pressure above the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer from the RPV to the primary containment is within the capacity of the primary containment vent. When the PCPL is challenged, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded.

The Heat Capacity Limit is a function of RPV pressure and torus bulk temperature and is a measure of the maximum heat load that the primary containment can withstand. If DEOP actions to control torus bulk temperature and RPV pressure below the Heat Capacity Limit are unsuccessful, RPV blowdown is required. The Heat Capacity Limit has been implemented as a single, bounding curve in the DEOPs, valid for all torus water levels at or below 17 ft. Alternate curves for other torus water levels are included in the Technical Support Guidelines.

Heat up of the torus to the Heat Capacity Limit signals the loss of functions required to maintain hot shutdown, including the ultimate heat sink. It also infers an RPV blowdown that could be caused by a loss of the RCS barrier. If compounded by further plant degradation, the event may challenge primary containment integrity.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS4
2. DEOP 200-1 Primary Containment Control

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5

Initiating Condition:

Inability to maintain plant in cold shutdown with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 212° F** for > Table M1 duration.

Table M1 – RCS Reheat Duration Thresholds		
RCS	**Secondary Containment Closure**	**Duration**
Intact	N/A	60 minutes*
Not Intact	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. UNPLANNED RPV pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.

Basis:

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

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

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 main steam line nozzle plugs, etc.).

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as decay heat removal system design and RPV level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5 (cont)

Basis (cont):

Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212° F), such as:

- Panel 902(3)-21, TR 2(3)-263-104, Rx Vessel Metal Temp
- Panel 902(3)-4, TR 2(3)-260-11, Recirc Loop Temps (representative of the most restrictive beltline region metal temperatures.)
- 902(3)-4, TR 2(3)-263-105, Vessel Flange & Shell Temps

Threshold #1 Basis:

The first condition in Table M1 addresses complete loss of functions required for core cooling for greater than sixty minutes during Refueling and Cold Shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (i.e., no freeze seals or main steam line nozzle plugs, etc.). With secondary containment closure established, a low-pressure barrier to fission product release exists. In this condition, containment status is of less importance than the status of RCS integrity because the RCS is intact and providing a high-pressure barrier to fission product release. The sixty-minute interval should allow sufficient time to restore cooling without a substantial degradation in plant safety. The asterisk highlights the note at the bottom of the table. The note indicates that the first condition is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the sixty-minute interval.

The second condition in Table M1 addresses the complete loss of functions required for core cooling for greater than twenty minutes during Refueling and Cold Shutdown modes when secondary containment closure is established but RCS integrity is not established or RPV inventory is reduced. RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (i.e., no freeze seals or main steam line nozzle plugs, etc.).

The allowed twenty-minute interval is 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 barrier to fission product release is established (i.e., secondary containment closure). The asterisk highlights the note at the bottom of the table. The note indicates that the second threshold is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the twenty-minute interval.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5 (cont)

Basis (cont):

The third condition in Table M1 addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when containment closure, and RCS integrity are not established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (i.e., no freeze seals or main steam line nozzle plugs, etc.). No delay time is allowed for this condition because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

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

Threshold #2 Basis:

The 10 psig pressure rise due to loss of decay heat removal infers an intact RCS with uncontrolled RPV temperature rise in excess of the Technical Specification cold shutdown limit (212° F) for which MA5 Threshold #1 would permit up to sixty minutes to restore RCS cooling before declaration of an Alert.

This EAL therefore covers situations in which it is determined that, due to high decay heat loads, the time provided to reestablish temperature control should be less than sixty minutes.

NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

RPV pressure is indicated by PI 2(3)-0263-156 (Panel 902(3)-3) or computer points C200(C300), C265(C365), C274(C374) and C267(C367).

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA4
2. Technical Specifications 3.6.1.1
3. Technical Specifications 3.6.4.1
4. OU-AA-103 Shutdown Safety Management Program
5. DGP 02-01 Unit Shutdown
6. DOA 0201-04 Loss of Vessel Flange, Shell, or Recirculation (Recirc) Loop Temperature Recorders During Heatup or Cooldown
7. DGP 02-02 Reactor Vessel Slow Fill
8. DIS 0263-19 Reactor Wide Range Pressure Transmitter Calibration Eq. Maintenance Inspection

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU5

Initiating Condition:

UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 212° F**.

   OR

2. Loss of all RCS temperature AND RPV level indication for **> 15 minutes**.

Basis:

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

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, 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. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the fuel cladding may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. 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 Reactor Vessel level so that escalation to the Alert under EAL MA5 will occur if required.

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in RCS/Reactor Vessel temperatures depending on the time since shutdown.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont):

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 Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling modes, the second condition of this EAL would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication.

Reactor Vessel water level is normally monitored using the following instruments:

- Wide Range (+330 to -70 in.)
- Medium Range (+60 to -60 in.)
- Narrow Range (+60 to -60 in.)
- Fuel Zone (+60 to -340 in.)

Detail A of DEOP 100, RPV Control, indicates when an instrument may be used for RPV level indication when EOPs are entered. DEOP 0010-00 provides secondary methods of determining RPV level.

During shutdown conditions, the Wide Range is the primary instrument for monitoring RPV level as the RPV is flooded in preparation for vessel head removal and refueling operations. Plant procedures (i.e., DGP 02-02, Reactor Vessel Slow Fill, etc.) provide alternate level monitoring capabilities when the normal level instrumentation is unavailable for the desired level range or the head vent piping is removed. The Refuel Outage Reactor Vessel and Cavity Level instrumentation, LI 2(3)-263-114, is installed in place of the normal Wide Range instrumentation. This instrumentation is located on Panel 902(3)-4 and indicates from instrument zero in the RPV to the maximum refuel floor water level (469 in. above instrument zero).

In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed. Figure 2 of DIP 0260-01, Refuel Outage Reactor Vessel and Cavity Level Instrumentation, illustrates various RPV and Refueling Cavity elevations referenced to instrument zero.

Several instruments and computer points are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212° F), such as:

- Panel 902(3)-21, TR 2(3)-263-104, Rx Vessel Metal Temp
- Panel 902(3)-4, TR 2(3)-260-11, Recirc Loop Temps (representative of the most restrictive beltline region metal temperatures.)
- 902(3)-4, TR 2(3)-263-105, Vessel Flange & Shell Temps

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU5 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU4
2. Technical Specifications Table 1.1-1
3. DGP 02-01 UNIT SHUTDOWN
4. DOA 0201-04, Loss of Vessel Flange, Shell, or Recirculation (Recirc) Loop Temperature Recorders During Heatup or Cooldown
5. DGP 02-02 Reactor Vessel Slow Fill
6. DEOP 0010-00 Guidelines for Use or Dresden Emergency Operating Procedures and Severe Accident Management Guidelines
7. DEOP 100 RPV Control, Table A
8. DIP 0260-01 Refuel Outage Reactor Vessel and Cavity Level Instrumentation

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6

Initiating Condition:

Inability to monitor a SIGNIFICANT TRANSIENT in progress.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of most **(approximately 75%)** safety system annunciators (Table M2).

Table M2 – Control Room Panels
• 902(3)-3 • 902(3)-5 • 902(3)-8

AND

2. Indications needed to monitor safety functions (Table M3) are unavailable.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

AND

3. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Turbine trip • Reactor scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10 %** Reactor Power change

AND

4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6 (cont)

Basis:

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change , or (5) thermal power oscillations >10% Reactor Power change.

Planned and unplanned actions are not differentiated since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not a factor.

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in a 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.

A Site Area Emergency exists if the Control Room staff cannot monitor safety functions needed for protection of the public. Indications needed to monitor safety functions necessary for protection of the public must include Control Room indications; computer generated indications i.e. and dedicated annunciation capability. The specific parameters should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled and in a coolable geometry, remove heat from the core, and maintain the reactor coolant system and containment intact. These parameters are monitored and controlled in the symptom-based emergency operating procedures (DEOPs).

Symptoms of a loss of annunciators can be:

- ALARM POTENTIAL FAILURE or ANNUNCIATOR DC POWER FAILURE alarms on one or more panels
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

Station procedures provide instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring frequency as determined by the Unit Supervisor.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS6
2. DEOP 100 RPV Control
3. DEOP 200-1 Primary Containment Control
4. UFSAR 7.5.2
5. UFSAR 7.5.3
6. DOP 9900-201 Restarting the Process Computer
7. DOP 9900-205 Safety Parameter Display System (SPDS)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 902(3)-3 • 902(3)-5 • 902(3)-8

OR

b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

AND

2. a. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Turbine trip • Reactor scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10 %** Reactor Power change

OR

b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis:

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

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations > 10% Reactor Power change.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

For this EAL "most" is approximately 75% of the safety system annunciators or indicators being lost, resulting in 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the Reactor Control, ECCS, and Electrical panel annunciation or indication equipment. The availability of computer based indication equipment is considered.

Symptoms of a loss of annunciators can be:

- ALARM POTENTIAL FAILURE or ANNUNCIATOR DC POWER FAILURE alarms on one or more panels
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

Station procedures provide instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The fifteen-minute interval offers time to recover from transient or momentary power losses.

MA6 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA4
2. UFSAR 7.5.2
3. UFSAR 7.5.3
4. DOP 9900-201 Restarting the Process Computer
5. DOP 9900-205 Safety Parameter Display System (SPDS)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 902(3)-3 • 902(3)-5 • 902(3)-8

OR

2. UNPLANNED loss of most **(approximately 75%)** of indicators associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

Basis:

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

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in a 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the Reactor Control, ECCS, and Electrical panel annunciation or indication equipment. The availability of computer based indication equipment is considered.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU6 (cont)

Basis (cont):

Symptoms of a loss of annunciators can be:

- ALARM POTENTIAL FAILURE or ANNUNCIATOR DC POWER FAILURE alarms on one or more panels
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

Station procedures provide instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The fifteen-minute interval offers time to recover from transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU3
2. UFSAR 7.5.2
3. UFSAR 7.5.3
4. DOP 9900-201 Restarting the Process Computer
5. DOP 9900-205 Safety Parameter Display System (SPDS)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU7

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

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

   OR

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

Basis:

The conditions of this EAL threshold may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. Only leakage inside the drywell qualifies toward exceeding Threshold #1. Various indications may be used to identify or verify potential leakage from the RCS into the drywell. They include drywell sump flow indications, drywell temperature and pressure changes, drywell air cooler cooling water differential temperature changes, and drywell atmosphere activity level changes. Unit 2(3) Appendix A Unit NSO Daily Surveillance Log provides direction for determining RCS leakage.

The 10 gpm value for unidentified leakage was selected because it is observable with normal Control Room measurement of sump pumpout rates. It is consistent with the Technical Specification threshold for leaks beyond which increased risk of crack propagation exists.

The 25 gpm value for identified leakage is set at a higher value because of the significance of identified leakage in comparison to unidentified or pressure boundary leakage.

No classification under this threshold is made for relief valve operation or leakage.

Both threshold values are observable on Control Room instrumentation and do not require a mass balance calculation.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU7 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU5
2. UFSAR 5.2.5
3. Technical Specifications 3.4.4
4. Technical Specifications 3.4.5
5. Unit 2(3) Appendix A Unit NSO Daily Surveillance Log
6. DAN 902(3)-4 A-17 Drywell Equip Sump Lvl HI-HI
7. DAN 902(3)-4 H-18 Drywell Floor Drn Sump Lvl HI-HI
8. DOA 0040-01 Slow Leak
9. DOP 2000-24 Drywell Sump Operation
10. DGP 02-02 Reactor Vessel Slow Fill

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8

Initiating Condition:

Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Torus level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

AND

2. a. RPV level **< - 143 in**. (TAF) for **> 30 minutes**.

   OR

   b. RPV level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following:

   - Refuel Floor Hi Range ARM **> 3000 mR/hr** or off-scale high.
   - Erratic Source Range Monitor indication.

AND

3. Containment is challenged as indicated by one or more of the following:

- Primary containment Hydrogen concentration **≥ 6%** and Oxygen concentration **≥ 5%**.
- Drywell pressure **≥ 62 psig**.
- Primary and Secondary CONTAINMENT CLOSURE not established.
- Any Secondary Containment radiation monitor **> DEOP 300-1 Maximum Safe** operating level.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8 (cont)

Basis (cont):

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

Threshold #1 and #2 Basis:

This EAL represents the inability to restore and maintain RPV level to above the top of active fuel, -143 in. (TAF). Fuel damage is probable if core uncovery is prolonged and submergence cannot be restored and maintained. Available decay heat will cause boiling and further drop RPV level.

This EAL is 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, (i.e., decay heat removal system design, etc.) can have a significant affect on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

When RPV level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated drywell radiation or unexplained rise in drywell floor or equipment drain sump pumpout rate. Detail A of DEOP 100, RPV Control, provides guidance on determining if RPV level can be monitored. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors (SRM CH 21, CH 22, CH 23, or CH 24) can be used as a tool for making such determinations.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The Fuel Handling Area Radiation Monitor indication of > 3000 mR/hr is based on calculation EP-EAL-0501.

Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8 (cont)

Basis (cont):

Threshold #3 Basis:

Four conditions are associated with the challenge to containment integrity:

- When hydrogen and oxygen concentrations in primary containment reach or exceed the deflagration limits, imminent loss of the Primary Containment barrier exists. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred.
- The primary containment design pressure (62 psig) is well in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation.
- Containment Closure provides a barrier to the release of radioactivity to the environment. When this barrier is not established with prolonged core uncovery, the health and safety of the public may be threatened.
- The secondary containment area radiation level is the DEOP Maximum Safe Operating level. The Maximum Safe Operating radiation level is based on the highest radiation level at which neither equipment necessary for the safe shutdown of the plant will fail nor personnel access necessary for the safe shutdown of the plant will be precluded. The maximum safe operating radiation level is based on personnel exposure rather than equipment operability concerns.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CG1
2. DEOP 100 RPV Control
3. Technical Specifications 3.3.1.2
4. DOS 0700-01 SRM Functional Test
5. DAN 902(3)-5 E-4 SRM Short Period
6. DEOP 200-1 Primary Containment Control
7. DEOP 200-2 Hydrogen Control
8. DEOP 300-1 Secondary Containment Control
9. USAR Table 6.2-1
10. EP-EAL-0501

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS8

Initiating Condition:

Loss of RCS/RPV inventory affecting core decay heat removal capability.

Operating Mode Applicability:

4

EAL Threshold Values:

1. **Without** Primary or Secondary CONTAINMENT CLOSURE established:

   a. RPV level **< - 60 in.**

   OR

   b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Torus level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

OR

2. **With** Primary or Secondary CONTAINMENT CLOSURE established:

   a. RPV level **< -143 in**. (TAF).

   OR

   b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory as evidenced by either of the following:

   - Per Table M5 indications.
   - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

Basis (cont):

Threshold #1 Basis:

Under the conditions specified by this threshold, continued drop in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV. If a low-pressure boundary to fission product release does not exist (i.e., neither primary nor secondary containment closure is not established), the RPV level associated with this threshold is six inches below the low-pressure ECCS actuation setpoint (i.e., - 54 in. - 6 in. = - 60 in.). If primary or secondary containment closure is established, a low-pressure boundary to fission product release exists and RPV level can drop to the top of active fuel, -143 in. (TAF), before a Site Area Emergency declaration is required.

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 drop and potential core uncovery.

In Cold Shutdown, the decay heat available to raise RPV temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding 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.

Threshold #2 Basis:

This threshold is 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, (i.e., decay heat removal system design, etc.) can have a significant impact on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

The thirty-minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis Reference(s):

1. NEI 99-01, Rev. 4 CS1
2. Technical Specifications Table 3.3.5.1 1
3. UFSAR 5.2.5
4. DEOP 0010-00 Guidelines for Use or Dresden Emergency Operating Procedures and Severe Accident Management Guidelines
5. DEOP 100 RPV Control
6. Unit 2(3) Appendix A Unit NSO Daily Surveillance Log
7. DAN 902(3)-4 A-17 Drywell Equip Sump Lvl HI-HI
8. DAN 902(3)-4 H-18 Drywell Floor Drn Sump Lvl HI-HI
9. DOA 0040-01 Slow Leak
10. DOP 2000-24 Drywell Sump Operation
11. Technical Specifications 3.3.1.2

12 DOS 0700-01 SRM Functional Test

13. DAN 902(3)-5 E-4 SRM Short Period

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8

Initiating Condition:

Loss of RCS/RPV inventory with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. Loss of RCS/RPV inventory as indicated by RPV level **< - 54 in.**

   OR

2. a. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise
• Unexplained Torus level rise
• Unexplained vessel make up rate rise
• Observation of leakage or inventory loss

   AND

   b. RCS/RPV level unknown for **> 15 minutes.**

Basis:

This EAL serves as a precursor 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 drop and potential core uncovery. The low-pressure ECCS actuation setpoint is 54 in. below RPV instrument zero. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

In Cold Shutdown mode, the decay heat available to raise RPV temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding 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.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8 (cont)

Basis (cont):

In Cold Shutdown mode, the RCS will normally be intact and standard RPV inventory and RPV level monitoring means are available. In the Refueling mode, the RCS is not intact and RPV level and inventory are monitored by different means. In the Refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

RPV level is normally monitored using the following instruments:

- Wide Range (+330 to -70 in.)
- Medium Range (+60 to -60 in.)
- Narrow Range (+60 to -60 in.)
- Fuel Zone (+60 to -340 in.)

In the second condition of this EAL, all RPV level indication would be unavailable. Detail A of DEOP 100, RPV Control, provides guidance on determining if RPV level can be monitored. RPV inventory loss, therefore, must be detected by alternate means (i.e., drywell floor and equipment drain sump pumpout rates). Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency duration.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA1 & CA2
2. Technical Specifications Table 3.3.5.1 1
3. UFSAR 5.2.5
4. DEOP 0010-00 Guidelines for Use or Dresden Emergency Operating Procedures and Severe Accident Management Guidelines
5. DEOP 100 RPV Control
6. Unit 2(3) Appendix A Unit NSO Daily Surveillance Log
7. DAN 902(3)-4 A-17 Drywell Equip Sump Lvl HI-HI
8. DAN 902(3)-4 H-18 Drywell Floor Drn Sump Lvl HI-HI
9. DOA 0040-01 Slow Leak
10. DOP 2000-24 Drywell Sump Operation

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU8

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

4

EAL Threshold Values:

RPV level **cannot** be restored and maintained **> 0 in.**

Basis:

The inability to restore and maintain level after reaching the RPS low level scram setpoint infers a degradation of the level of safety at the plant.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU1
2. Technical Specifications Table 3.3.5.1-1
3. UFSAR 5.2.5
4. DEOP 0010-00 Guidelines for Use or Dresden Emergency Operating Procedures and Severe Accident Management Guidelines
5. DEOP 100 RPV Control
6. Unit 2(3) Appendix A Unit NSO Daily Surveillance Log
7. DAN 902(3)-4 A-17 Drywell Equip Sump Lvl HI-HI
8. DAN 902(3)-4 H-18 Drywell Floor Drn Sump Lvl HI-HI
9. DOA 0040-01 Slow Leak
10. DOP 2000-24 Drywell Sump Operation

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS9

Initiating Condition:

Loss of RPV inventory affecting core decay heat removal capability with irradiated fuel in the RPV.

Operating Mode Applicability:

5

EAL Threshold Values:

1. **Without** Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< - 60 in**.

      OR

   b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
      - Refuel Floor Hi Range ARM **> 3000 mR/hr** or off-scale high.
      - Erratic Source Range Monitor indication.

   OR

2. **With** Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< - 143 in.** (TAF).

      OR

   b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
      - Refuel Floor Hi Range ARM **> 3000 mR/hr** or off-scale high.
      - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

Threshold #1 and #2 Basis:

Under the refueling conditions specified in this threshold, prolonged loss of the ability to monitor RPV level in conjunction with indirect indications of inventory loss infer a continued drop in RPV level and loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV.

Basis (cont):

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

In the refueling mode, when RPV level indication is unavailable, the inventory loss must be detected by drywell floor and equipment drain sump pumpout rates or erratic Source Range Monitor indication. DEOP 100, RPV Control, provides guidance on determining if RPV level can be monitored. Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to core shine, scattering and radiation bounce off of the solid surfaces in the area will result in readings on the Refuel floor Rad monitor > 3000 mR/hr. This threshold radiation value is based on calculations documented in EP-EAL-0501.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CS2
2. Technical Specifications Table 3.3.5.1-1
3. UFSAR 5.2.5
4. DEOP 0010-00 Guidelines for Use or Dresden Emergency Operating Procedures and Severe Accident Management Guidelines
5. DEOP 100 RPV Control
6. Unit 2(3) Appendix A Unit NSO Daily Surveillance Log
7. DAN 902(3)-4 A-17 Drywell Equip Sump Lvl HI-HI
8. DAN 902(3)-4 H-18 Drywell Floor Drn Sump Lvl HI-HI
9. DOA 0040-01 Slow Leak
10. DOP 2000-24 Drywell Sump Operation
11. Technical Specifications 3.3.1.2

12 DOS 0700-01 SRM Functional Test

13. DAN 902(3)-5 E-4 SRM Short Period

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU9

Initiating Condition:

UNPLANNED loss of RCS inventory with irradiated fuel in the RPV.

Operating Mode Applicability:

5

EAL Threshold Values:

1. UNPLANNED RPV level drop below the RPV flange for **≥ 15 minutes**.

   OR

2. a. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Torus level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

AND

b. RPV level unknown.

Basis:

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

Thresholds #1 Basis:

The RPV flange is 190 in. above instrument zero and normally indicated by the Wide Range instrument (+330 to -70 in.). Plant procedures (i.e., DGP 02-02, Reactor Vessel Slow Fill, etc.) provide alternate level monitoring capabilities when the normal level instrumentation is unavailable for the desired level range or the head vent piping is removed. The Refuel Outage Reactor Vessel and Cavity Level instrumentation, LI 2(3)-263-114, is installed in place of the normal Wide Range instrumentation. This instrumentation is located on Panel 902(3)-4 and indicates from instrument zero in the RPV to the maximum refuel floor water level (469 in. above instrument zero). In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed. DIP 0260-01, Refuel Outage Reactor Vessel and Cavity Level Instrumentation, illustrates various RPV and Refueling Cavity elevations referenced to instrument zero.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont):

This threshold is applicable only in the Refueling mode and addresses loss of inventory to below the RPV flange during refueling operations. Refueling operations that drop RPV level below the RPV flange are carefully planned and procedurally controlled. An Unusual Event is appropriate because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant.

The fifteen-minute interval provides a reasonable time frame to restore level using one or more of the redundant means of refill that should be available. If RPV level cannot be restored in this interval, a more serious condition may exist.

Thresholds #2 Basis:

In the second condition of this threshold, all RPV level indication would be unavailable. Detail A of DEOP 100, RPV Control, indicates when an instrument may be used for RPV level indication when EOPs are entered. RPV inventory loss, therefore, must be detected by alternate means (i.e., drywell floor and equipment drain sump pumpout rates). Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU2
2. UFSAR 5.2.5
3. Technical Specifications 3.4.4
4. Technical Specifications 3.4.5
5. Unit 2(3) Appendix A Unit NSO Daily Surveillance Log
6. DAN 902(3)-4 A-17 Drywell Equip Sump Lvl HI-HI
7. DAN 902(3)-4 H-18 Drywell Floor Drn Sump Lvl HI-HI
8. DOA 0040-01 Slow Leak
9. DOP 2000-24 Drywell Sump Operation
10. DGP 02-02 Reactor Vessel Slow Fill
11. DEOP 0010-00 Guidelines for Use or Dresden Emergency Operating Procedures and Severe Accident Management Guidelines
12. DEOP 100 RPV Control, Table A
13. DIP 0260-01 Refuel Outage Reactor Vessel and Cavity Level Instrumentation
14. Technical Specifications Table 3.3.5.1 1

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU10

Initiating Condition:

UNPLANNED loss of all onsite or offsite communications capabilities.

Operating Mode Applicability:

1, 2, 3, 4, 5

EAL Threshold Values:

1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations.

   OR

2. Loss of all Table M6 **Offsite** communications capability.

Table M6 - Communications Capability

System	Onsite	Offsite
Plant Radio System	X	
Plant Paging System	X	
Sound Power Phones	X	
In-Plant Telephones	X	
All telephone lines (commercial and microwave)		X
NARS		X
ENS		X
Satellite Phones		X
HPN		X
Cellular Phones		X

Basis:

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

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems with offsite authorities or personnel. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This should include ENS, FAX transmissions and dedicated phone systems. This EAL is applicable only when extraordinary means are being utilized to make communications possible (i.e., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.).

MU10 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU6 & CU6
2. EP-MW-124-1001 Facilities Inventories and Equipment Tests
3. UFSAR 9.5.2
4. DOA 0010-14 Loss of Off-Site Telephone Communication Systems

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU11

Initiating Condition:

Inability to reach required shutdown within Technical Specification limits.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a prescribed shutdown mode when the Technical Specification configuration cannot be restored. The plant is within its safety envelope when being shut down 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 completion period elapses under Technical Specifications and is not related to how long a condition may have existed.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU2
2. Dresden Technical Specifications

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HG1

Initiating Condition:

Security event resulting in loss of physical control of the facility.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

A HOSTILE FORCE has taken control of:

1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1).

Table H1 - Safety Functions and Related Systems
• 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) • Fission Product Barriers

OR

2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).

Basis:

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.

Threshold #1 Basis

This threshold encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions. As a result, equipment control cannot be transferred to and operated from another location.

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

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.

Threshold #2 Basis

This threshold addresses loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely because there is freshly off-loaded fuel in the pool. The condition “freshly off-loaded reactor fuel in pool” equates to fuel off-loaded within the last 120 days in NF-AA-310 Special Nuclear Material And Core Component Movement.

HG1 (cont)

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HG1
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. DOA 0010-18, Escalated Security Event / Hostile Force Intrusion
5. NF-AA-310, Special Nuclear Material And Core Component Movement
6. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS1

Initiating Condition:

Site attack.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (i.e., violent acts between individuals in the OWNER CONTROLLED AREA).

PROTECTED AREA: Area which normally encompasses all controlled areas within the security protected area fence.

This class of security events represents an escalated threat to plant safety above that contained in the Alert ICs (HA1 and HA2) in that a hostile force has progressed from the OWNER CONTROLLED AREA to the Protected Area.

Although Nuclear Power Plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for Offsite Response Organizations (ORO) 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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS1 (cont)

Basis (cont):

This EAL is intended to address the potential for a very rapid progression of events due to a dedicated attack. It is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs. HOSTILE ACTION identified above encompasses various acts including:

- Air attack (LARGE AIRCRAFT impacting the protected area)
- Land-based attack (HOSTILE FORCE penetrating protected area)
- Waterborne attack (HOSTILE FORCE on water penetrating protected area)
- BOMBs breeching the protected area

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs.

This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for ORO to be notified and to activate in order to be better prepared to respond should protective actions become necessary. If not previously notified by NRC that the LARGE AIRCRAFT impact 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.

LARGE AIRCRAFT is meant to be an 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.

This EAL addresses the immediacy of a threat to impact site vital areas within a relatively short time. The fact that the site is under serious attack with minimal time available for additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HS4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. DOA 0010-18, Escalated Security Event / Hostile Force Intrusion

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA1

Initiating Condition:

Notification of an airborne attack threat.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LARGE AIRCRAFT is meant to be an 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.

The intent of this EAL is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations (ORO) and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Alert. This EAL is met when a plant receives information regarding a LARGE AIRCRAFT attack threat from NRC and the LARGE AIRCRAFT is less than 30 minutes away from the plant.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from such an attack. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for ORO to be notified and encouraged to activate (if they do not normally) to be better prepared should it be necessary to consider further actions.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA7
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. DOA 0010-18, Escalated Security Event / Hostile Force Intrusion

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU1

Initiating Condition:

Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. A credible site-specific security threat notification as determined per SY-AA-101-132, “Threat Assessment.”

   OR

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

Basis:

Threshold #1 Basis

The intent of this threshold is to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat.

The determination of “credible” is made through use of information found in the Station Security Plan or SY-AA-101-132, “Threat Assessment” procedure.

Threshold #2 Basis

The intent of this threshold is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Unusual Event. This threshold is met when a plant receives information regarding an aircraft threat from NRC. Should the threat involve a LARGE AIRCRAFT (LARGE AIRCRAFT is meant to be an aircraft with the potential for causing significant damage to the plant), then escalation to Alert via HA1 would be appropriate if the LARGE AIRCRAFT is less than 30 minutes away from the plant. The status and size of the plane may be provided by NORAD through the NRC. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
6. DOA 0010-18, Escalated Security Event / Hostile Force Intrusion

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA2

Initiating Condition:

Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (i.e., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This EAL is intended to address the potential for a very rapid progression of events due to an attack including:

- Air attack (LARGE AIRCRAFT impacting the OCA)
- Land-based attack (HOSTILE FORCE progressing across licensee property or directing projectiles at the site)
- Waterborne attack (HOSTILE FORCE on water attempting forced entry or directing projectiles at the site)
- BOMBs

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA2 (cont)

Basis (cont):

This EAL is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne terrorist attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for Offsite Response Organizations to be notified and to activate in order to be better prepared to respond should protective actions become necessary.

If not previously notified by NRC that the LARGE AIRCRAFT impact 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. LARGE AIRCRAFT is meant to be an 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.

This IC/EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time. The fact that the site is an identified attack candidate with minimal time available for further preparation requires a heightened state of readiness and implementation of protective measures that can be effective (onsite evacuation, dispersal or sheltering) before arrival or impact.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA8
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
6. DOA 0010-18, Escalated Security Event / Hostile Force Intrusion

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS3

Initiating Condition:

Confirmed security event in a plant VITAL AREA

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.

Basis:

VITAL AREA: is any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC (HA3).

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a Station’s security. Only those events that involve Actual or Likely Major failures of plant functions needed for protection of the public need to be considered. The following events would not normally meet this requirement; (i.e., Failure by a Member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Station Security Plan.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HS1
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. DOA 0010-18, Escalated Security Event / Hostile Force Intrusion

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA3

Initiating Condition:

Confirmed security event in a plant PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.

Basis:

PROTECTED AREA: is an area which normally encompasses all controlled areas within the security protected area fence.

This class of security events represents an escalated threat to plant safety above that contained in the Unusual Event.

Multi-unit stations with shared safety functions should further consider how this IC may affect more than one unit and how this may be a factor in escalating the emergency class.

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a station's security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered. The following events would not normally meet this requirement; (i.e., failure by a member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HA4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. DOA 0010-18, Escalated Security Event / Hostile Force Intrusion

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU3

Initiating Condition:

Confirmed security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.

Basis:

Reference is made to Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

This threshold is based on Station Security Plan – Appendix C. 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.

Consideration should be given to the following types of events when evaluating an event against the criteria of the Station Security Plan: CIVIL DISTURBANCE, and STRIKE ACTION.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. DOA 0010-18, Escalated Security Event / Hostile Force Intrusion
6. DOA 0010-13, Security Threat

HS4

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Control room evacuation has been initiated.

   AND

2. Control of the plant **cannot** be established per DSSP 0100-CR in **< 30 minutes**.

Basis:

The 30 minute time period starts when either:

a. Control of the plant is no longer maintained in the Main Control Room

   OR

b. The last Operator has left the Main Control Room.

The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. The 30 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 determination of whether or not control is established outside of the Main Control Room is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the site-specific time for transfer that the licensee has control of the plant. Transfer of control to locations outside the Control Room is considered established when the Shift Manager has determined that the operators are capable of controlling reactivity, core cooling and heat sink functions.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HS2
2. DSSP 0100-CR Hot Shutdown Procedure - Control Room Evacuation

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA4

Initiating Condition:

Control Room evacuation has been initiated.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Entry into DSSP 0100-CR for Control Room evacuation.

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency operations centers are necessary. Procedure DSSP 0100-CR specifies conditions under which Control Room evacuation may be necessary.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA5
2. DSSP 0100-CR Hot Shutdown Procedure - Control Room Evacuation

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA5

Initiating Condition:

Natural and destructive phenomena affecting the plant VITAL AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by seismic instrumentation **> 0.10 g**.

   AND

   b. Confirmed by **EITHER**:

   - Earthquake felt in plant.
   - National Earthquake Center.

   OR

2. Tornado or sustained high winds **> 100 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

   OR

3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems:

   OR

4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area.

Table H2 – Vital Areas
• Reactor Building • Aux Electric Room • Control Room • Diesel Generator Rooms • 4-KV ECCS Switchgear Area • Battery Rooms • CRD & CCSW Pump Rooms • Turbine Building Cable Tunnel • Turbine Building Safe Shutdown Areas • Crib House

OR

HA5 (cont)

EAL Threshold Value(s) (cont):

5. Uncontrolled flooding that results in **EITHER**:

   a. Degraded safety system performance in any Table H3 area as indicated in the Control Room.

Table H3 – Internal Flooding Areas
• Condenser Pits • Condensate pump Rooms • Containment Cooling Service Water Vaults • Crib House • East Corner Room • West Corner Room

   OR

   b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.

OR

6. a. High river water level **> 513 ft**.

   OR

   b. Low river water level **< 501 ft. 6 in.**

Basis:

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

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

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

Threshold #1 Basis:

This threshold addresses events that may have resulted in a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this threshold to assess the actual magnitude of the damage.

HA5 (cont)

Basis (cont):

This threshold is based on seismic ground acceleration in excess of 0.1 g for the UFSAR Operating Basis Earthquake (OBE). Seismic events of this magnitude are a factor of ten greater that the Unusual Event threshold of EAL HU5 and can cause damage to plant safety functions.

The seismic monitor is located in the Auxiliary Electric Equipment Room.

Confirmation from the National Earthquake center shall not delay declaration in the presence of valid confirming indications.

Threshold #2 Basis:

This threshold addresses events that may have resulted a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The Alert classification is appropriate if visible damage is observed or relevant plant parameters indicate that the performance of safety systems in these areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

110 mph is the UFSAR design basis wind speed. Station structures are designed to withstand wind loads which may exist if sustained wind speeds reach or exceed 110 mph. Wind loads in excess of this magnitude can cause damage to safety functions. Since the wind speed instrument can only measure up to 100 mph, the threshold has been reduced to this value.

Threshold #3 Basis:

This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes, or impact of projectiles into a Table H2 area. This threshold addresses vehicle crashes that challenge the operability of systems necessary for safe shutdown of the plant. Table H2 areas include Category 1 structures and those Category 2 structures that contain Category 1 Systems and components.

The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Table H2 areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

Threshold #4 Basis:

This threshold addresses the threat to safety-related equipment imposed by missiles generated by main turbine rotating component failures.

Basis (cont):

It is consistent with the definition of an ALERT in that, if missiles have damaged or penetrated areas containing safety-related equipment, the potential exists for substantial degradation of the level of safety of the plant.

Threshold #5 Basis:

This threshold addresses the effect of internal flooding that has resulted in degraded performance of safety systems or has created industrial safety hazards (i.e., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant.

“Uncontrolled” as used in this threshold describes a condition where water is entering an area from an unplanned evolution. This flooding may have been caused by internal events such as component failures, equipment misalignment, and fire suppression system actuation or outage activity mishaps. Water entering an area, which resulted in degraded performance of safety systems within the area due to wetting or submergence, would meet the intent of this threshold. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding.” Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if indications of degraded system performance is available or a shock hazard is known to exist.

The Internal Flooding Areas listed in Table H3 include areas containing systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Threshold #6 Basis:

The probable maximum flood (PMF) produces a peak flood to 528 ft el. This is significantly above the grade elevation (517 ft) and the lowest opening leading to safety-related equipment (509 ft el.). When the water level reaches 509 ft el. the reactors are shutdown, the drywells are deinerted, and the vessels are flooded and cooled to cold shutdown conditions as quickly as possible. If the water level reaches the EAL threshold value (513 ft), reactor cooldown is transferred to the Isolation Condensers, which thereafter maintain a safe shutdown condition until the flood waters recede and plant startup can be initiated.

Minimum river water levels to assure pump suction are:

- Circulating Water Pumps: 490 ft el.
- Service Water Pumps: 494 ft el.
- CCSW Pumps: 501 ft 6 in.

Basis (cont):

- Unit 2/3 Fire Pump: 498 ft 6 in.

The low river water level threshold (501 ft. 6 in.) is based on the most limiting pump suction requirement (CCSW).

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA1
2. DOA 0010-03 Earthquakes
3. UFSAR 3.7.4
4. DIS 0020-01 Seismic Recorder Functional Testing, Data Retrieval, and Initialization
5. UFSAR 3.3.1.1
6. DOA 0010-02 Tornado Warning/Severe Winds
7. UFSAR 3.2
8. General Arrangement Drawings 3, 4, 4A, 5 and 10
9. Drawing M-1D Plant Development Plan
10. UFSAR 4.4.1.1
11. DOA 0010-01 Dresden Lock and Dam Failure
12. UFSAR 9.2.5.3.1
13. DOA 0010-04 Floods
14. DOA 0040-02 Localized Flooding In Plant
15. UFSAR 3.5.3

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU5

Initiating Condition:

Natural and destructive phenomena affecting the PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Seismic event identified by any **TWO** of the following:
   - Earthquake felt in plant.
   - Seismic event confirmed by station seismic monitor procedure.
   - National Earthquake Center.

   OR

2. Report by plant personnel of tornado striking or sustained (>15 minutes) high winds **> 100 mph**, within PROTECTED AREA boundary.

   OR

3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area.

Table H2 – Vital Areas
• Reactor Building • Aux Electric Room • Control Room • Diesel Generator Rooms • 4-KV ECCS Switchgear Area • Battery Rooms • CRD & CCSW Pump Rooms • Turbine Building Cable Tunnel • Turbine Building Safe Shutdown Areas • Crib House

   OR

4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

   OR

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU5 (cont)

EAL Threshold Values (cont):

5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode.

Table H3 – Internal Flooding Areas
• Condenser Pits
• Condensate pump Rooms
• Containment Cooling Service Water Vaults
• Crib House
• East Corner Room
• West Corner Room

OR

6. River level transients potentially affecting safe operation of the plant:
   a. High river level **> 509 ft.**

      OR

   b. Report of substantial reduction in river level by site personnel and confirmation by the Corp. of Engineers that Dresden Lock and Dam has failed.

Basis:

<u>PROTECTED AREA:</u> An area which normally encompasses all controlled areas within the security protected area fence.

Threshold #1 Basis:

A felt earthquake may be the first indication to Control Room personnel that a seismic event is occurring and warrants the emergency declaration because the seismograph does not provide a readout or alarm to the Control Room when actuated. The seismic monitor is located in the Auxiliary Electric Equipment Room at Dresden Station, which is a strong-motion seismograph that actuates at a sensed earthquake threshold of 0.01g. Seismic events of this magnitude are ~1/10 of the Alert event threshold (OBE) of EAL HA5 in which it is assumed the earthquake can cause damage to plant safety functions.

Basis (cont):

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. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01 g.”

Confirmation from the National Earthquake center shall not delay declaration in the presence of valid confirming indications.

Threshold #2 Basis:

This threshold is based on the assumption that a tornado striking (touching down) or design force winds (> 110 mph) within the Protected Area boundary may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. Since the wind speed instrument can only measure up to 100 mph, the threshold has been reduced to this value.

The Protected Area boundary is within the security isolation zone and is defined in the Station Security Plan – Appendix C. Verification of a tornado is obtained by direct observation and reporting by station personnel. “Sustained” wind speeds exist for 15 minutes or longer. Wind speed is obtained from meteorological data in the Control Room.

Threshold #3 Basis:

In this context, a "vehicle crash" is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

Threshold #4 Basis:

This threshold is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this threshold to classify minor operational leakage.

HU5 (cont)

Basis (cont):

Threshold #5 Basis:

“Uncontrolled” as used in this threshold describes a condition where water is entering the area from an unplanned evolution. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if a potential to affect safety related equipment needed for the current operating mode exists.

This threshold addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, fire suppression system actuation or outage activity mishaps. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding.” The Internal Flooding Areas of concern for the Unusual Event declaration are those areas in the plant that have the potential to affect safety related equipment needed for the current operating mode including:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Threshold #6 Basis:

The possible maximum flood (PMF) produces a peak flood to 528 ft el. This is significantly above the grade elevation (517 ft) and the high river water level threshold, which is the lowest opening leading to safety-related equipment (509 ft el.). When this level is reached, the reactors are shutdown, the drywells are deinerted, and the vessels are flooded and cooled to cold shutdown conditions as quickly as possible. If the water level reaches 513 ft el., reactor cooldown is transferred to the Isolation Condensers, which thereafter maintain a safe shutdown condition until the flood waters recede and plant startup can be initiated.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU1
2. DOA 0010-03 Earthquakes
3. UFSAR 3.7.4
4. DIS 0020-01 Seismic Recorder Functional Testing, Data Retrieval, and Initialization
5. UFSAR 3.3.1.1
6. DOA 0010-02 Tornado Warning/Severe Winds
7. UFSAR 3.2
8. General Arrangement Drawings 3, 4, 4A, 5 and 10
9. Drawing M-1D Plant Development Plan
10. UFSAR 4.4.1.1

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU5 (cont)

Basis Reference(s) (cont):

11. DOA 0010-01 Dresden Lock and Dam Failure
12. UFSAR 9.2.5.3.1
13. DOA 0010-04 Floods
14. DOA 0040-02 Localized Flooding In Plant
15. UFSAR 3.5.3
16. Drawing B-01A Composite Site Plan
17. DOP 5320-05 Hydrogen System Leak Location
18. DAN 902(3)-7 B-4 Turbine Trip Vacuum Lo
19. DAN 902(3)-7 E-11 H2 Seal Oil Trouble

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA6

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, D

EAL Threshold Values:

1. FIRE or EXPLOSION in any Table H2 area.

Table H2 – Vital Areas
• Reactor Building • Aux Electric Room • Control Room • Diesel Generator Rooms • 4-KV ECCS Switchgear Area • Battery Rooms • CRD & CCSW Pump Rooms • Turbine Building Cable Tunnel • Turbine Building Safe Shutdown Areas • Crib House

AND

2. a. Affected safety system parameter indications show degraded performance.

   OR

   b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.

Basis:

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

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

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

HA6 (cont)

Basis (cont):

The areas listed in Table H2 house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Personnel access to these areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses FIRES and EXPLOSIONS that challenge the operability of systems necessary for safe shutdown of the plant.

The only FIRES and EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment required for safe shutdown. Visual observation of damage infers the ability to approach or enter the affected areas. Lacking the ability to adequately inspect the area for damage, the Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this EAL. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

A steam line break or steam EXPLOSION that damages permanent structures or equipment in one of these areas would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA2
2. UFSAR 3.2
3. UFSAR 3.5.3
4. General Arrangement Drawings M-3, M-4, M-4A, M-5 and M-10

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU6

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

OR

2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

Table H2 – Vital Areas
• Reactor Building • Aux Electric Room • Control Room • Diesel Generator Rooms • 4-KV ECCS Switchgear Area • Battery Rooms • CRD & CCSW Pump Rooms • Turbine Building Cable Tunnel • Turbine Building Safe Shutdown Areas • Crib House

OR

3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

Basis:

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

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU6 (cont)

Basis (cont):

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

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

Thresholds #1 and #2 Basis:

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins with a credible notification that a fire is occurring or indication of a valid fire detection system alarm. A verified alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm.

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 EAL HA7, Release of Toxic or Flammable Gases. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under EAL HA7.

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

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished (i.e., smoldering waste paper basket, etc.). Such fires are excluded from consideration in this threshold since they have no safety consequence.

Threshold #3 Basis:

The only EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment in the PROTECTED AREA.

A steam line break or steam EXPLOSION that damages permanent structures or equipment in a PROTECTED AREA would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU2
2. UFSAR 3.2
3. Drawing M-1D Plant Development Plan

HA7

Initiating Condition:

Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).

Table H2 – Vital Areas
• Reactor Building • Aux Electric Room • Control Room • Diesel Generator Rooms • 4-KV ECCS Switchgear Area • Battery Rooms • CRD & CCSW Pump Rooms • Turbine Building Cable Tunnel • Turbine Building Safe Shutdown Areas • Crib House

Basis:

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

Basis (cont)

Values for LFL for common gases at Dresden Station are:

- Propane – 2.2% (BOC Gasses MSDS)
- Hydrogen – 4% (Air Liquide Safety Data Sheet)
- Acetylene – 2.2% (BOC Gasses MSDS)

This EAL is based on toxic, asphyxiant, or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Toxic or flammable gases detected outside of these areas need not be considered for this EAL unless there is a spread of the gasses into one of these areas.

Threshold #1:

Declaration should not be delayed for confirmation from atmospheric testing if it is reasonable to conclude that the IDLH concentrations have been met (e.g., documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards).

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 this EAL. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under this EAL.

The first condition is met if measurement of toxic gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a Table H2 area. Non-Toxic Gases which displace oxygen (site examples; Halon or Nitrogen) to a life threatening level due to asphyxiation (oxygen deprivation) should also be considered for this EAL.

An Asphyxiant is a material 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.

Threshold #2:

The second condition is met when the flammable gas concentration in a Table H2 area exceeds the lower flammability limit. Flammable gases such as hydrogen and acetylene are routinely used to maintain plant systems (hydrogen – main generator cooling, reactor coolant chemistry control) or repair equipment/components (acetylene - welding). This condition addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a Table H2 area has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage or personnel injury. Once it has been determined that an uncontrolled release of flammable gas is occurring, sampling must be done to determine if the gas concentration exceeds the lower flammability limit.

HA7 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA3
2. UFSAR 3.5.3
3. UFSAR 3.2
4. General Arrangement Drawings M-3, M-4, M-4A, M-5 and M-10

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU7

Initiating Condition:

Release of toxic or flammable gases deemed detrimental to normal operation of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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:

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.

This EAL is based on the existence of uncontrolled releases of toxic, asphyxiant, or flammable gas affecting plant operations or the health of plant personnel. The release may have originated within the Protected Area boundary, or it may have originated offsite and subsequently drifted inside the Protected Area boundary. Offsite events (i.e., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on normal plant operations.

It is intended that releases of toxic, asphyxiant, or flammable gases are of sufficient quantity and the release point of such gases is such that safe plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for safe plant operation. The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect safe plant operations or plant personnel safety.

An Asphyxiant is a material 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.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU3
2. UFSAR 3.5.3
3. UFSAR 3.2
4. General Arrangement Drawings 3, 4, 4A, 5 and 10
5. DOA-0010-12, Toxic Gas/Chemical Release from Nearby Chemical Facilities

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HG8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (i.e., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels (≥ 1 Rem TEDE or ≥ 5 Rem CDE Thyroid) outside the site boundary.

Basis Reference(s):

1. NEI 99-01, Rev 4 HG2
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS8

Initiating Condition:

Other Conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (i.e., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels ($>$ 1 Rem TEDE or $>$ 5 Rem CDE Thyroid) outside the site boundary.

Basis Reference(s):

1. NEI 99-01, Rev 4 HS3
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

<u>HOSTILE ACTION:</u> An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (i.e., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA6
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

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:

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency operating procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU5
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU9

Initiating Condition:

Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY.

   OR

2. Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY.

   OR

3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask MPC CONFINEMENT BOUNDARY.

Basis:

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

The CONFINEMENT BOUNDARY consists of the Multi-Purpose Canister (MPC) shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds. A Unusual Event in this EAL is based on a loaded fuel storage cask CONFINEMENT BOUNDARY being violated leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

Threshold #1 and #2

The results of the HI-STORM 100 and HI-STAR 100 Final Safety Analysis Reports (FSARs) were used to develop the list of natural phenomena events and accident conditions. These EALs address responses to a dropped cask, a tipped over cask, EXPLOSION, missile damage, fire damage or natural phenomena affecting a cask (i.e., seismic event, tornado, flood, etc.). The cask FSARs require that the cask and in some cases the MPC be inspected to determine if the cask or MPC may have been affected as a result of a natural phenomena event or accident condition. The inspections are performed to the extent practical to assess the potential damage to the cask and/or the confinement boundary. If it were determined during the assessment that the MPC CONFINEMENT BOUNDARY was damaged such that boundary integrity is in question, declaration under this EAL would be necessitated.

HU9 (cont)

Basis (cont):

Threshold #3

Any condition not detailed as an EAL threshold value, which, in the judgment of the Emergency Director, is a potential degradation in the level of safety of the ISFSI. Emergency Director judgment is to be based on known conditions and the expected response to mitigating activities within a short time period.

Basis Reference(s):

1. NEI 99-01, Rev. 4 E-HU1
2. HI-STORM 100 FSAR Rev 3
3. HI-STAR 100 FSAR Rev. 1

Section 4: Emergency Measures

Exelon Nuclear emergency response actions are the same for all nuclear stations and are thus covered by Section E of the Emergency Plan.

4.1 Notification of the Emergency Organization

Standard NARS notifications for the Dresden Station are made to the State of Illinois Emergency Management Agency (IEMA). At the Dresden Station, if a General Emergency is the initiating event, the Emergency Director is responsible for notifying the following additional Illinois and local agencies:

- Grundy County EOC
- Kendall County EOC
- Will County EOC
- Grundy County Sheriff's Office
- Kendall County Sheriff's Office
- Will County Sheriff's Office

4.2 Assessment Actions

Throughout each emergency situation, continuing assessment will occur. Assessment actions at Dresden Station may include an evaluation of plant conditions; in-plant, onsite, and initial offsite radiological measurements; and initial estimates of offsite doses. Core damage information is used to refine dose assessments and confirm or extend initial protective action recommendations. Dresden Station utilizes NEDC-33045P-A, Revision 0, (2001) as the basis for the methodology for post-accident core damage assessment. This methodology utilizes real-time plant indications. In addition, Dresden Station may use samples of plant fluids and atmospheres as inputs to the CDAM (Core Damage Assessment Methodology) program for core damage estimation.

4.3 Protective Actions for the Offsite Public

To aid Control Room personnel during a rapidly developing emergency situation, Figure 4-1, "Protective Action Recommendation (PAR) Determination Flowchart for Dresden Station" has been developed based on Section J.10.m of the Emergency Plan.

4.3.1 Alert and Notification System (ANS) Sirens

This ANS consists of a permanently installed outdoor notification system within a ten mile radius around the station. The ten-mile radius around the station is a mix of agriculture and industry with a relatively low population distribution. The ANS, as installed, consists of mechanical and electronic sirens that will cover this entire area with a minimum sound level of 60 db. Additionally, the ANS will cover the heavily populated areas within the ten-mile radius around the station with a minimum sound level of 70 db to ensure complete coverage.

The ANS sirens are controlled and monitored on a daily basis, by a computerized telemetry system. The daily monitoring assures early failure detection and therefore maximizes system operability and reliability.

4.3.2 Evacuation Time Estimates

The evacuation time estimates were developed per the requirements of NUREG-0654, and to support the Illinois Plan For Radiological Accidents (IPRA) - Dresden Volume II. The purpose of the evacuation time estimates is to assess the postulated evacuation times for the Dresden Station Emergency Planning Zone (EPZ).

The evacuation time estimate data was updated per a study performed by Earth Tech. Inc. documented in their report dated December, 2003 entitled “Evacuation Time Estimates for the Dresden Station Plume Exposure Pathway Emergency Planning Zone.”

The evacuation times are based on a detailed consideration of the EPZ roadway network and population distribution. The information in Table 4-1 presents representative evacuation times for daytime and nighttime scenarios, for summer and winter seasons, and under various weather conditions for the evacuation of various areas around the Dresden Station, once a decision has been made to evacuate. The evacuation times noted include notification, mobilization, and travel time. These times are for the general population which include permanent population and special facilities (schools, nursing homes, hospitals, and recreational areas). Table 4-2 provides information on the scenario population distribution (by Subarea) that was used for this study. Table 4-3 provides a representation of the Subarea Locations in relation to the EPZ.

4.4 Protective Actions for Onsite Personnel

Dresden Station has a siren system to assemble personnel during emergency conditions. Upon hearing a continuous two (2) minute siren, all personnel not having emergency assignments have been instructed to assemble in predesignated assembly areas. Refer to Figure 4-2.

Assembly of site personnel, for purposes of accountability and possible evacuation, is initiated per the requirements of Section J of the Emergency Plan. Accountability of site personnel is handled by the Dresden Station security force.

If a site evacuation of non-essential personnel is required by Section J of the Emergency Plan, personnel will be either relocated and monitored at the Relocation Centers or sent home if there is no release or radiological or safety concerns. The designated relocation centers for Dresden Station are:

- Mazon Relocation Center, Mazon, Illinois
- LaSalle County Nuclear Power Station, LaSalle Co. Illinois
- Braidwood Station, Braceville, Illinois

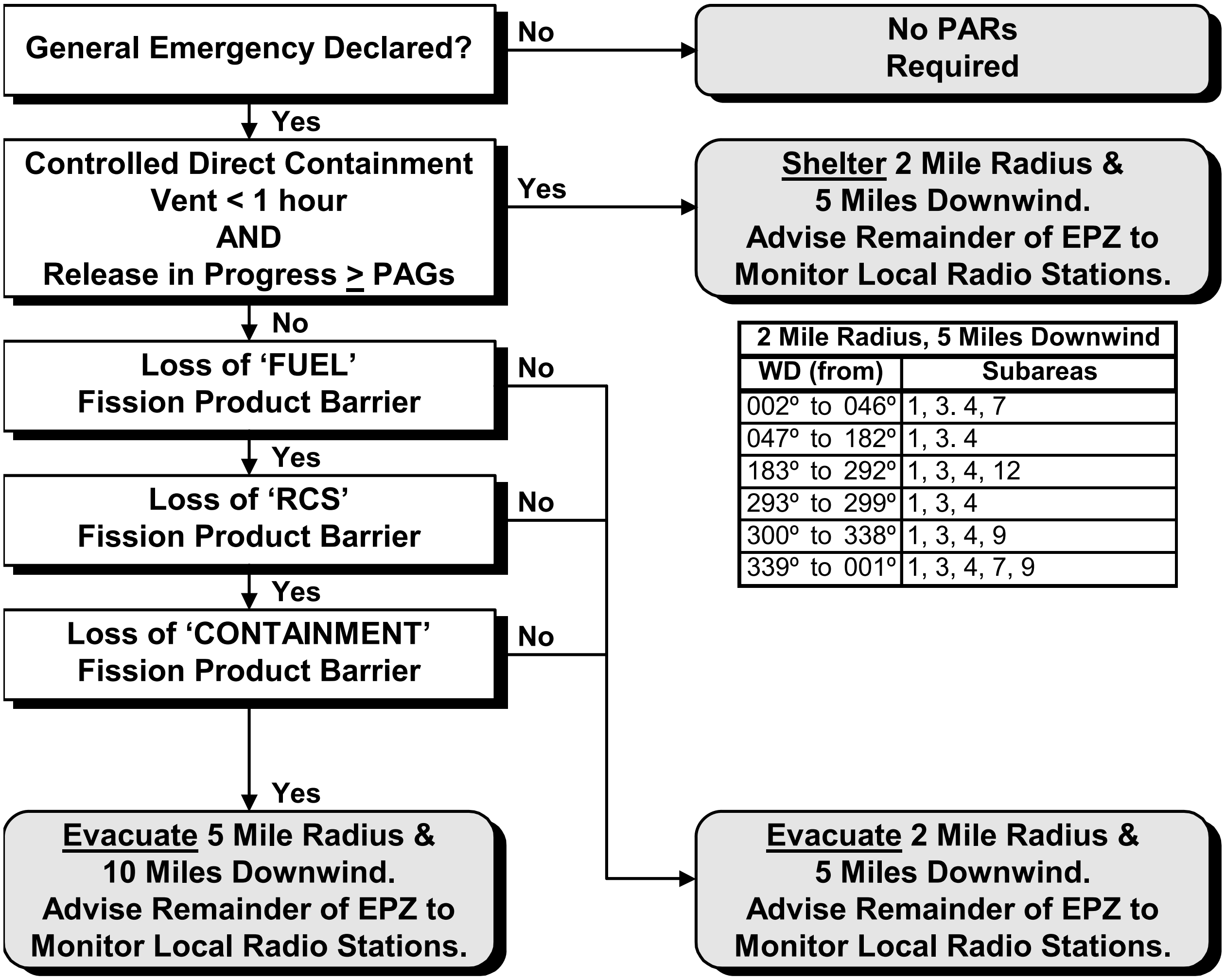
For evacuation routes, refer to EP-AA-113-F-19.

Traffic control for onsite areas will be handled by the Dresden Nuclear Power Station security force, if necessary.

Equipment and personnel would be available at LaSalle County Station, Braidwood Station, and the Mazon Relocation Center for monitoring and decontamination of evacuated personnel. If major decontamination, follow-up, or bioassay samples are necessary, those persons would be sent to LaSalle County Station or Braidwood Station.

Other emergency measures common to all nuclear stations are discussed in the Emergency Plan.

Figure 4-1: Dresden Station PAR Determination Flowchart



2 Mile Radius, 5 Miles Downwind	
WD (from)	**Subareas**
002° to 046°	1, 3. 4, 7
047° to 182°	1, 3. 4
183° to 292°	1, 3, 4, 12
293° to 299°	1, 3, 4
300° to 338°	1, 3, 4, 9
339° to 001°	1, 3, 4, 7, 9

5 Mile Radius, 10 Miles Downwind	
WD (from)	**Subareas**
002° to 026°	1, 3, 4, 7, 9, 10, 11, 12
027° to 044°	1, 3, 4, 7, 8, 9, 11, 12
045° to 068°	1, 3, 4, 7, 8, 9, 12
069° to 083°	1, 2, 3, 4, 5, 7, 8, 9, 12
084° to 092°	1, 2, 3, 4, 5, 7, 9, 12
093° to 112°	1, 2, 3, 4, 7, 9, 12
113° to 143°	1, 2, 3, 4, 6, 7, 9, 12
144° to 177°	1, 3, 4, 6, 7, 9, 12
178° to 199°	1, 3, 4, 6, 7, 9, 12, 13
200° to 225°	1, 3, 4, 7, 9, 12, 13
226° to 249°	1, 3, 4, 7, 9, 12, 13, 14
250° to 266°	1, 3, 4, 7, 9, 12, 14
267° to 286°	1, 3, 4, 7, 9, 12, 14, 15
287° to 321°	1, 3, 4, 7, 9, 12, 15
322° to 344°	1, 3, 4, 7, 9, 12, 15, 16
345° to 353°	1, 3, 4, 7, 9, 10, 12, 15, 16
354° to 001°	1, 3, 4, 7, 9, 10, 12, 16

2 Mile Radius, 5 Miles Downwind	
WD (from)	**Subareas**
002° to 046°	1, 3. 4, 7
047° to 182°	1, 3. 4
183° to 292°	1, 3, 4, 12
293° to 299°	1, 3, 4
300° to 338°	1, 3, 4, 9
339° to 001°	1, 3, 4, 7, 9

Figure 4-2: Dresden Onsite Assembly Areas and Emergency Response Facilities

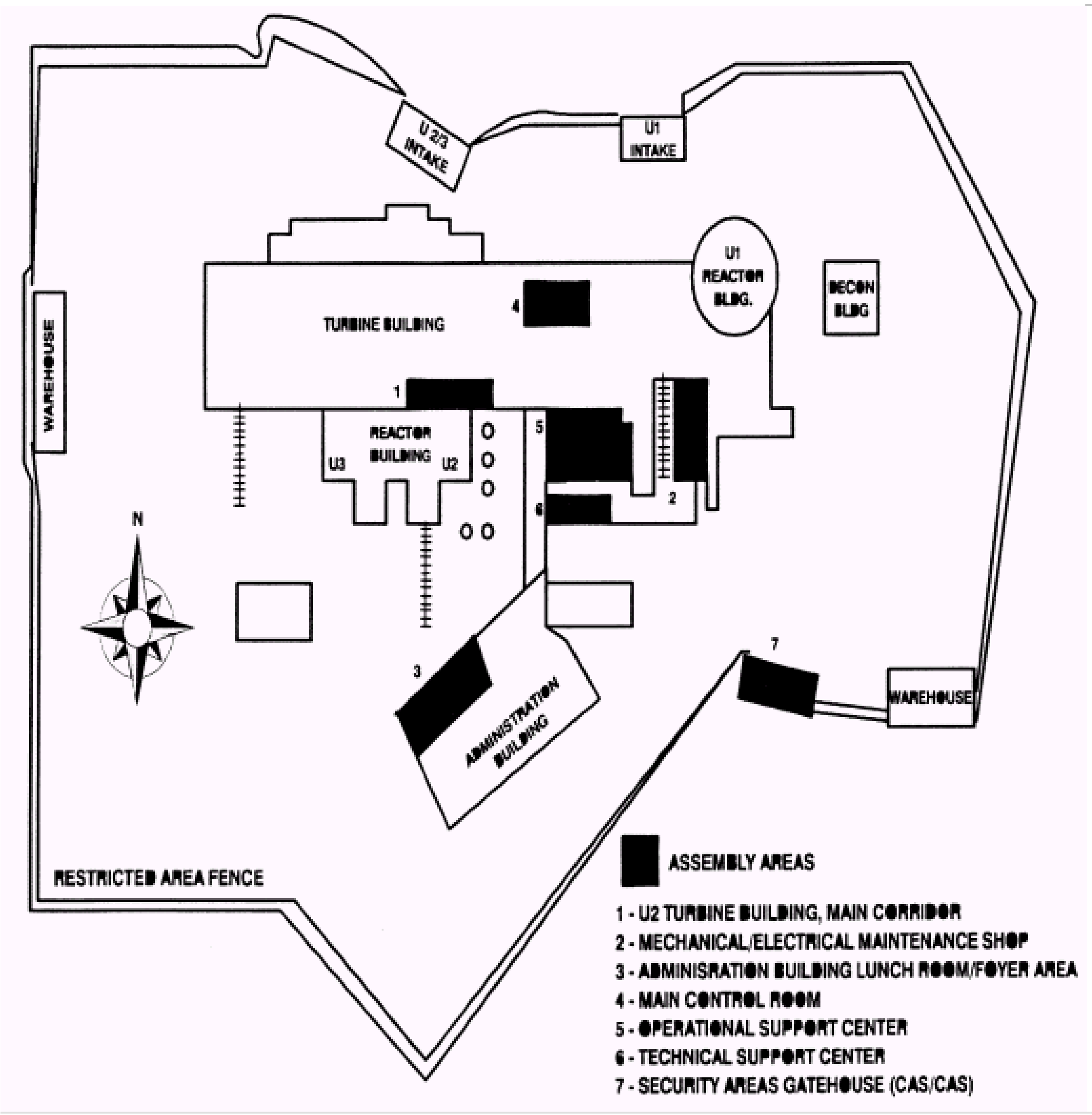


Table 4-1: Dresden Evacuation Time Estimates (in minutes) (1)

	Summer Daytime		Summer Nighttime		Winter Daytime		Winter Nighttime	
PAR Evacuation Zone	**fair**	**adverse**	**fair**	**adverse**	**fair**	**adverse**	**fair**	**adverse**
2 Mile Radius & 5 Miles Downwind								
WD 002 to 046 [1, 3, 4, 7]	190	215	125	130	180	220	120	130
WD 047 to 182 [1, 3, 4]	185	210	120	125	175	215	115	125
WD 183 to 292 [1, 3, 4, 12]	190	215	125	130	180	220	120	130
WD 293 to 299 [1, 3, 4]	185	210	120	125	175	215	115	125
WD 300 to 338 [1, 3, 4, 9]	185	210	120	125	175	215	115	125
WD 339 to 001 [1, 3, 4, 7, 9]	190	215	125	130	180	220	120	130
5 Mile Radius & 10 Miles Downwind								
WD 002 to 026 [5R, 10, 11]	190	220	130	135	180	225	125	135
WD 027 to 044 [5R, 8, 11]	190	220	130	135	180	225	125	135
WD 045 to 068 [5R, 8]	190	220	130	135	180	225	125	135
WD 069 to 083 [5R, 2, 5, 8]	190	220	130	135	180	230	125	135
WD 084 to 092 [5R, 2, 5]	190	220	130	135	180	230	125	135
WD 093 to 112 [5R, 2]	190	220	130	135	180	225	125	135
WD 113 to 143 [5R, 2, 6]	190	220	130	135	180	225	125	135
WD 144 to 177 [5R, 6]	190	220	130	135	180	225	125	135
WD 178 to 199 [5R, 6, 13]	215	265	140	170	200	280	125	175
WD 200 to 225 [5R, 13]	215	265	140	170	200	280	125	175
WD 226 to 249 [5R, 13, 14]	215	265	140	170	200	280	125	175
WD 250 to 266 [5R, 14]	190	220	130	140	180	225	125	135
WD 267 to 286 [5R, 14, 15]	190	220	130	140	185	230	130	140
WD 287 to 321 [5R, 15]	190	220	130	135	180	225	125	135
WD 322 to 344 [5R, 15, 16]	195	220	130	135	180	225	125	135
WD 345 to 353 [5R, 10, 15, 16]	195	220	130	135	190	235	125	140
WD 354 to 001 [5R, 10, 16]	195	220	130	135	180	225	125	135
Full EPZ	215	265	140	170	200	280	130	175

(1) Times are rounded to the nearest 5 minutes

(2) Subareas in brackets. See Table 4-3 for Subarea locations. PAR evacuation zones per EP-AA-111

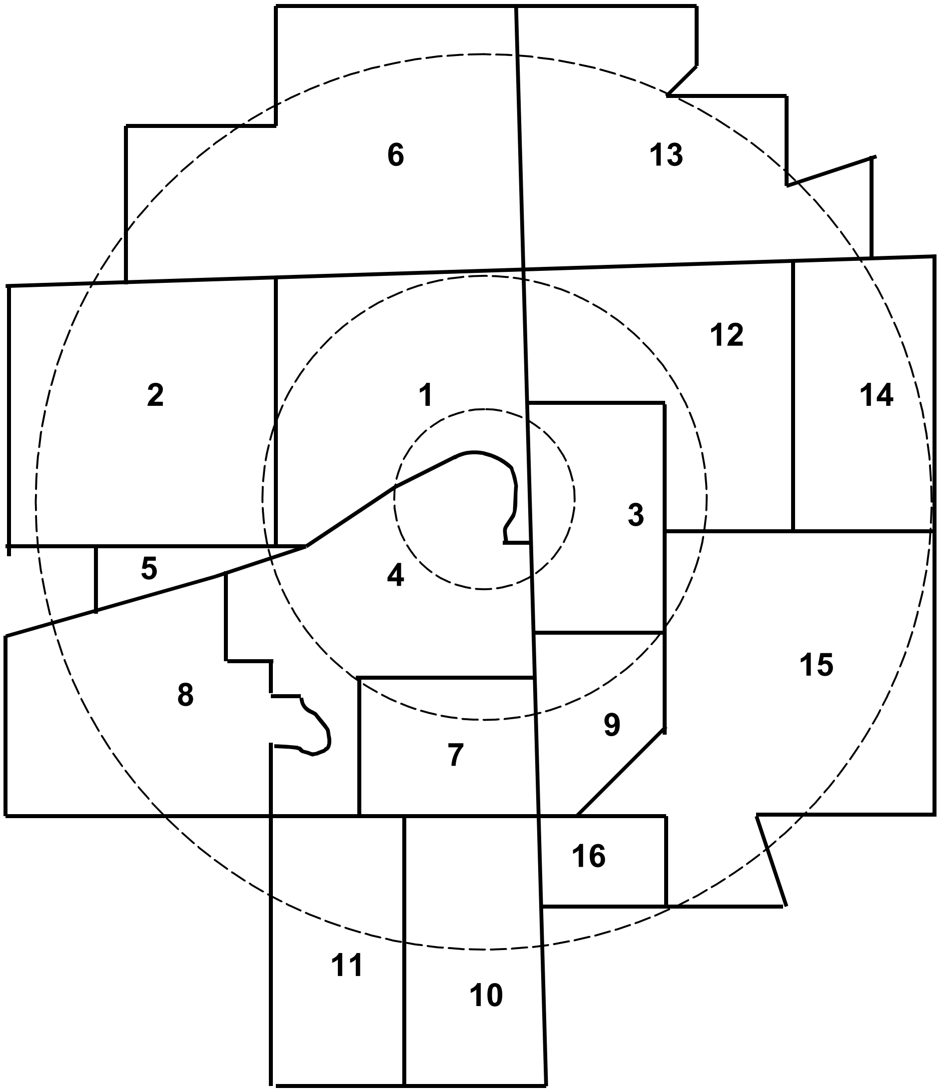
(3) "5R" designates all Subareas within 5-mile radius (Subareas 1, 3, 4, 7, 9, 12)

(4) WD is the direction (in degrees) from which the wind is blowing (00 or 360 represents a wind from north to south)

Table 4-2: Dresden Scenario Population Distribution By Subarea

	Summer				Winter			
	Daytime		Nighttime		Daytime		Nighttime	
Subarea	Population	Vehicles	Population	Vehicles	Population	Vehicles	Population	Vehicles
1	5,541	2,561	4,708	1,835	8,530	2,826	4,672	1,823
2	6,012	2,458	5,584	2,198	6,020	2,231	5,152	1,990
3	2,706	1,222	1,988	766	2,386	1,095	1,838	707
4	4,770	2,190	3,562	1,302	2,850	1,715	2,026	903
5	10,681	4,465	8,358	3,131	12,068	4,142	8,219	3,083
6	1,059	412	1,013	375	1,033	402	1,003	373
7	5,105	1,913	4,791	1,749	4,751	1,636	4,019	1,489
8	2,539	939	2,539	939	2,523	927	2,491	923
9	1,659	691	1,588	650	429	179	376	152
10	5,174	1,857	4,925	1,823	6,827	2,053	4,895	1,813
11	242	90	242	90	242	90	242	90
12	12,028	5,584	8,933	3,689	11,526	4,868	8,582	3,525
13	25,200	11,957	20,309	8,210	28,272	12,410	18,632	7,403
14	1,901	720	1,885	704	2,266	756	1,862	696
15	10,012	3,828	8,542	3,031	9,048	2,952	6,509	2,365
16	8,832	3,337	7,322	2,764	6,647	2,036	4,839	1,804
EPZ total	103,462	44,223	86,290	33,254	105,418	40,315	75,357	29,137

Table 4-3: Dresden Subarea Locations



Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities

Refer to Figure 4-2 for the location of the Dresden Station Control Room, Technical Support Center (TSC), and Operations Support Center (OSC) within the Station's Protected Area boundary.

5.1.1 Station Control Room

The Dresden Station Control Room is the initial onsite center of emergency control. The Dresden Station Units 2 and 3 Control Rooms are located at the 534' level at the east end of the Unit 2/3 Turbine Building.

5.1.2 Technical Support Center (TSC)

A Technical Support Center is located on the southwest corner of the Service Building at Elevation 518'. The TSC fully meets the requirements of Section H.1.b of the E-Plan.

5.1.3 Operational Support Center (OSC)

The Operational Support Center is located in the Radiation Protection Area and the Work Control Area. The OSC conforms to the requirements of Section H.1.c of the Emergency Plan and is the location to which operations support personnel will report during an emergency and from which they will be dispatched for assignments in support of emergency operations.

5.2 Assessment Resources

5.2.1 Onsite Meteorological Monitoring Instrumentation

The meteorological tower, located approximately 3000 ft. west of the reactor building, is 400 ft. high and is instrumented at three levels. The 35 ft., 150 ft. and 300 ft. levels correspond to the elevations of the possible points of airborne effluent release. Wind speed and wind direction are measured at all three elevations. Ambient temperature is measured at the 35 ft. level and differential temperatures referenced to 35 ft. are measured at 150 ft. and 300 ft. Precipitation is also measured at the site.

The onsite meteorological monitoring program is covered in the contract specification and vendor procedures of the meteorological monitoring contractor. These data are used to generate wind roses and to provide estimates of airborne concentrations of gaseous effluents.

5.2.1.1 Instrumentation

The meteorological tower is instrumented with equipment that conforms to the recommendations of Regulatory Guide 1.23 and ANSI/ANS 2.5 (1984). The equipment is placed on booms oriented into the generally prevailing wind at the site. Equipment signals are brought to an instrument building with controlled environmental conditions. The building at the base of the tower houses the analog and digital recording equipment, signal conditioners, and other equipment used to process and re-transmit the data to the end point users.

5.2.1.2 Meteorological Measurement Program During a Disaster

Cooperation between the corporate office and the meteorological contractor assures that a timely restoration of any outage can be made. Emergency field visits to the site are made as quickly as possible after detection of a failure.

Should a disaster of sufficient magnitude occur to destroy the tower structure, a contract is maintained to have a temporary tower erected within 72 hours, weather conditions permitting. Further, the meteorological contractor maintains two levels of sensors (wind speed, wind direction and temperature) in a state of readiness for use on the temporary tower.

Additionally, Exelon Nuclear's existing instrumentated towers at other nuclear sites provide a measurement network with multiple backup opportunities.

Meteorological data is available to the station Control Room, Technical Support Center, and Emergency Operations Facility for use in the Dose Assessment Computer Model for estimating the environmental impact of unplanned releases of radioactivity from the station.

5.2.2 Onsite Radiation Monitoring Equipment

5.2.2.1 Radiation Monitoring System

Onsite radiation monitoring systems at Dresden can be categorized into four systems:

- A process radiological monitoring and sample system;
- An effluent radiological monitoring and sampling system;
- An airborne radioactive monitoring system;
- An area radiation monitoring system; and
- A supply of portable survey and counting equipment.

5.2.2.2 Radiological Noble Gas Effluent Monitoring

A wide range monitor is installed in the effluent stream that enters the main chimneys and the reactor building vents. These wide range monitors have a range of $1x10^{-7}$ uCi/cc to $1x10^{5}$ uCi/cc.

The method of converting instrument readings to release rates will be determined after the energy responses of the detector are obtained. Due to system design, the monitors give an estimate of a release. Actual releases will be determined by periodically collecting grab samples, counting the samples collected and calculating the releases.

5.2.2.3 Radioiodine and Particulate Effluent Monitoring

Effluent sampling media are analyzed in the Station counting room. Silver based cartridges are available to reduce the interference of noble gases.

5.2.2.4 High-Range Containment Radiation Monitors

Two high range containment radiation monitors are installed on each of Dresden's units. The range of these monitors is from 1 R/hr to 10^8 R/hr.

5.2.2.5 In-plant Iodine Instrumentation

Dresden Station has the capability to sample and determine iodine concentrations in the plant using charcoal cartridges and gamma ray spectroscopy. Portable monitors may be used to measure increasing levels of iodine during emergency conditions.

5.2.3 Onsite Process Monitors

Adequate monitoring capability exists to properly assess the plant status for the modes of operation. The operability of the post-accident instrumentation ensures information is available on selected plant parameters to monitor and assess important variables following an accident. Instrumentation is available to monitor the parameters and ranges given in the Dresden Station Technical Specifications.

Station procedures have been developed which would aid personnel in recognizing inadequate core cooling using applicable instrumentation.

5.2.4 Onsite Fire Detection Instrumentation

Dresden Station has a fire protection system that is designed to quickly detect any fires; annunciating locally and in the Control Room. The fire detection system is designed to applicable National Fire Protection Association (NFPA) standards. The majority of the detectors consist of electrically supervised ionization smoke detectors.

5.2.5 Facilities and Equipment for Offsite Monitoring

Consult Chapter 11 of the station specific Offsite Dose Calculation Manual (ODCM) for the most current location for fixed continuous air samplers and TLD.

5.2.6 Site Hydrological Characteristics

The hydrological characteristics of Dresden Station are described in Section 2.4 of the Dresden UFSAR. The Dresden site at the confluence of the DesPlaines and Kankakee rivers is at the location considered to divide the upper and lower parts of the Illinois River system. The normal river pool elevation controlled at the adjacent Dresden Island Lock and Dam is nominally 505 feet. In December 1982, the Dresden site was subjected to flood waters that exceeded 509 feet establishing a maximum historical flood elevation. Nominal ground elevation is about 516 feet at the location of the principal structures of Units 2 and 3, and design plant grade is 517 feet. Consequently the probability of flooding critical areas of the site is remote.

Spillway capacity at the Dresden Island Lock and Dam is well in excess of the estimated maximum instantaneous flow of the Illinois River. The site elevation is well above the vast valley storage area upstream from the dam.

River system flow data applicable to the site are more than adequate to meet the cooling water requirements of the two operating units, to assure the availability of sufficient quantities of water for dilution of all radioactive liquid wastes discharged into the Illinois River within the limits in 10 CFR 20, and to reduce concentrations to approximately one one-thousandth of the maximum permissible concentration in the river below the point of discharge from the station.

The closest point downstream of the station where the Illinois River is used as a source of domestic water is Peoria which is 100 miles downstream. At this point the combined effects of dilution, mixing, radioactive decay, and deposition of radioactivity on the river bottom will have rendered the contribution of radioactivity by the station negligible in relation to that present in the Illinois River from other sources.

5.2.6.1 Probable Maximum Flood on Streams and Rivers

Since the site probable maximum flood (PMF) elevation of 528'-0" is above the plant grade (elevation 517'0") and above the lowest opening leading to safety-related equipment (elevation 509'-0"), the safe operation of the plant during the PMF is accomplished via implementation of the flood emergency procedures.

5.2.6.2 Potential Dike and Dam Failures, Seismically Induced

An earth dam of the type specified usually does not collapse in its entirety. A break occurs and widens as the water washes through the break. This tends to prolong the time it would take to empty the lake; nevertheless, instantaneous dike losses have been considered since the dikes are not constructed to Class I criteria.

The Dresden lock and dam are concrete structures that are operated and maintained by the U.S. Army Corps of Engineers. Operations response procedures are in place to deal with loss of the cooling lake and/or the lock and dam.

5.2.6.3 Ice Effects

An 8-foot diameter deicing line connects the discharge canal headworks and the crib house forebay. Its high point is in the headworks at elevation 495'-0" and its low point is in the forebay at elevation 489'-0". A slide gate valve is used to isolate the deicing line when not in use.

5.2.6.4 Cooling Lake

The purpose of the cooling lake is to provide adequate cooling of the circulating and service water before discharge to the Illinois River. The water discharged to the river must meet state requirements. The lake is connected to the intake and discharge canals for Units 2 and 3 by two canals (the "hot" and "cold" canals). The level of the lake is maintained by a concrete spillway located adjacent to the lift station and between the cold canal and the north end of the lake. The spillway is equipped with weir gates that can be lowered to block some of the spillover to maintain the level of the lake. The weir gates can be manually operated if necessary. Operations response procedures are in place regarding loss of lake level control and/or loss of the lift station.

A discussion of the groundwater resources and aquifers in the vicinity of Dresden Station is discussed in the Final Environmental Statement.

5.3 Protective Facilities and Equipment

The onsite assembly areas for Dresden Station are described in Section 4 of this annex. The offsite evacuation assembly areas for Dresden Station are discussed in Section 4 of this annex. These areas are outside the plume exposure Emergency Planning Zone and equipped with monitoring, decontamination and bioassay capabilities.

5.4 First Aid and Medical Facilities

Dresden Station has an inplant decon room located in a room adjacent to the Radiation Protection Office. This room is provided with a sink, decontamination shower, and a supply cabinet.

First aid kits, stretchers, sinks, eyewashes, and emergency showers have been placed in strategic locations throughout the station.

Medical treatment given to injured persons at the station is of a "first aid" nature. When more professional care is needed, injured persons are transported to a local hospital or clinic. Morris Hospital in Morris, Illinois is the Dresden Station primary supporting medical facility for radioactively contaminated injured persons. Provena St. Joseph Medical Center in Joliet, Illinois is the backup medical facility for evaluation and treatment of persons suffering from traumatic injury, medical illness, or radiation exposure and uptake.

Appendix 1: NUREG-0654 Cross-Reference

Annex Section	NUREG-0654
1.0	Part I, Section A
1.1	Part I, Section C
1.2	Part I, Section D
Figure 1-1	Part I, Section D
2.0	Part II, Section A.4
2.1	Part II, Section A.3
3.0	Part II, Section D
4.1	Part II, Section E.1 & J.7
4.2	Part II, Section I.2 & 3
4.3	Part II, Section J.10.m
4.3.1	Part II, Section E.6
4.3.2	Part II, Section J.8
4.4	Part II, Section J.1-5
Figure 4-1	Part II, Section J.10.m
Figure 4-2	Part II, Section J.5
4.4	Part II, Section J.2 & 3
Table 4-1	Part II, Section J.8
Table 4-2	Part II, Section J.10.b
5.1	Part II, Section H.1 & G.3
5.2.1	Part II, Section H.5.a & 8
5.2.2	Part II, Section H.5.b & I.2
5.2.3	Part II, Section H.5.c
5.2.4	Part II, Section H.5.d
5.2.5	Part II, Section H.6.b & 7
5.2.6	Part II, Section H.5.a & 6.a
5.3	Part II, Section J.1-5
5.4	Part II, Section L.1 & 2

Appendix 2: Station Letters of Agreement

1. US Army Corps of Engineers - provide information regarding failure of or problems with the Dresden Lock and Dam.
2. Will County Sheriff's Office - provides services of law enforcement.
3. Grundy County Sheriff - services of law enforcement.
4. Morris Hospital of Morris, Illinois, acts as the primary supporting medical facility for Dresden Station.
5. General Electric Midwest Fuel Reprocessing Plant - Health Physics support - instrumentation and limited technical assistance.
6. Coal City Fire Protection and Ambulance District - Fire protection and advanced life support for transportation of accident victims.

Attachment 5

EP-AA-1005

"Exelon Nuclear Standardized Radiological Emergency Plan Annex for LaSalle Station"

Revision 25

Exelon
Nuclear

EP-AA-1005
Revision 25

EXELON NUCLEAR

RADIOLOGICAL EMERGENCY PLAN ANNEX FOR LASALLE STATION

Submitted: Kevin Appel Date: 10/10/07
Midwest Region Emergency Preparedness Manager

Authorized: Jim Meister Date: 10/12/07
Vice President – Operations Support

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APPENDIXES

REVISION HISTORY

Revision 0: July 1980	Revision 6k: January 5, 1998	Revision 22, September, 2006
Revision 1: April 1981	Revision 6l: August 14, 1998	Revision 23, not implemented
Revision 2: December 1981	Revision 7; May 13, 1999	Revision 24, May 2007
Revision 3: October 1984	Revision 8; January 8, 2001	
Revision 4: March 1986	Revision 9; May 7, 2001	
Revision 5: February 1987	Revision 10; October 8, 2001	
Revision 6: March 1991	Revision 11; October 31, 2001	
Revision 6a: November 1992	Revision 12; January 3, 2002	
Revision 6b: March 1993	Revision 13; July 22, 2002	
Revision 6c: December 1993	Revision 14; September 09, 2002	
Revision 6d: January 1994	Revision 15; June 30, 2003	
Revision 6e: October 1994	Revision 16; August 27, 2003	
Revision 6f: December 1995	Revision 17, December 2004	
Revision 6g: January 1996	Revision 18, May 2005	
Revision 6h: February 1996	Revision 19, September 2005	
Revision 6i: June 1996	Revision 20, January, 2006	
Revision 6j: February 1997	Revision 21, March, 2006	

Section 1: Introduction

As required in the conditions set forth by the Nuclear Regulatory Commission (NRC) for the operating licenses for the Exelon Nuclear Stations, the management of Exelon recognizes its responsibility and authority to operate and maintain the nuclear power stations in such a manner as to provide for the safety of the general public.

The Exelon Emergency Preparedness Program consists of the Exelon Nuclear Standardized Emergency Plan (Emergency Plan), Station Annexes, emergency plan implementing procedures, and associated program administrative documents. The Emergency Plan outlines the basis for response actions that would be implemented in an emergency. Planning efforts common to all Exelon Nuclear stations are encompassed within the Emergency Plan.

This document serves as the LaSalle Station Annex and contains information and guidance that is unique to the station. This includes Emergency Action Levels (EALs), and facility geography and location for a full understanding and representation of the station’s emergency response capabilities. The Station Annex is subject to the same review and audit requirements as the Emergency Plan.

1.1 Facility Description

The LaSalle Station, Units 1 and 2, is located in Brookfield Township of LaSalle County in northeastern Illinois.

It is approximately 55 direct-line miles southwest of Chicago and 20 miles west of Dresden Station. The plant is on flat terrain about 220 feet above the Illinois River channel which traverses north central Illinois some 3 1/2 miles to the north of the site. Figure LA 1-1 shows the site location.

The LaSalle Station utilizes two single cycle forced circulation Boiling Water Reactors (BWR), each rated at3489 MWt. . The design electrical rating of each unit is 1154 MWe; the net output is 1114 MWe from each General Electric (GE) turbine generator. The Nuclear Steam System Supplier (NSSS) was GE (Nuclear Energy Division). The entire plant, except for the NSSS, was designed by Sargent & Lundy (S & L) Engineers.

The containment design employs the BWR Mark II concept of over/under pressure suppression with multiple downcomers connecting the reactor drywell to the water filled pressure suppression chamber. The primary containment is a steel lined, post-tensioned, concrete enclosure, housing the reactor and the suppression pool. This primary containment is entirely enclosed in the reinforced concrete reactor building that is the secondary containment structure.

The power generation complex includes several contiguous buildings, including two Reactor Buildings, an Auxiliary Building (housing the Control Room), the Turbine Building, Diesel Generator Buildings, the Radwaste Building, the Service Building, and the Offgas Building. Other buildings such as the gatehouse, warehouses, etc., are also located in the general plant area.

A Lake Screen House on the intake flume is located about 800 feet east of the main building complex. A small river screen house, located on the Illinois River, provides makeup water to the cooling lake for the LaSalle Station.

Condenser cooling for the station is provided from a perched cooling lake of 2058 acres. The ultimate heat sink for emergency core cooling is a submerged pond and intake flume of 458 acre-feet capacity that underlies the cooling lake and the natural grade of the site.

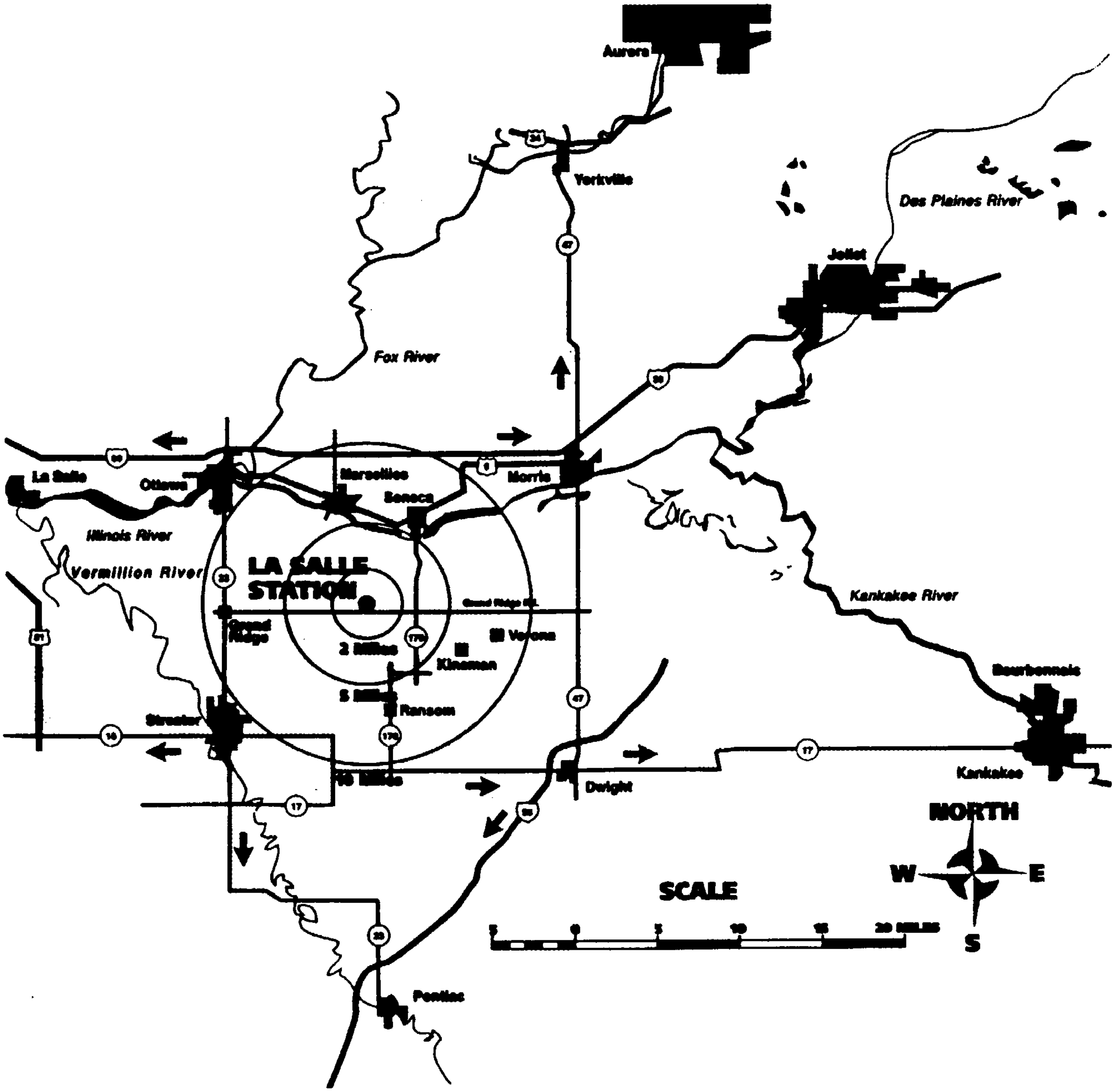
The LaSalle Station utilizes a single vent stack for elevated release of all gaseous waste. Liquid radwaste is stored for decay prior to release to the Illinois River or concentrated to solid waste for controlled disposal at regulated storage sites. For more specific site location information, refer to the Station UFSAR.

1.2 Emergency Planning Zone

The plume exposure Emergency Planning Zone (EPZ) for LaSalle Station is an area surrounding the Station with a radius of about ten miles. (Exact boundaries are determined by the State of Illinois). Refer to Figure 1-1.

The ingestion pathway EPZ for LaSalle Station is an area surrounding the Station with a radius of about 50 miles.

Figure 1-1: LaSalle Station Location and 10 Mile EPZ



Section 2: Organizational Control of Emergencies

Initial response to any emergency is by the normal plant organization present at the site. This organization includes positions that are onsite 24 hours per day and is described in Section B.1 of the Emergency Plan.

Once an emergency is declared, the Emergency Response Organization is activated according to Section B.4 of the Emergency Plan and Implementing Procedures.

2.1 Non-Exelon Nuclear Support Groups

Exelon Nuclear has contractual agreements with several companies whose services would be available in the event of a radiological emergency. These agencies and their available services are listed in Appendix 3 of the Emergency Plan.

Emergency response coordination with governmental agencies and other support organizations is discussed in Section A of the Emergency Plan.

Agreements exist on file at LaSalle Station with several support agencies. These agencies and their support roles are listed in Appendix 2, Station Letters of Agreement.

Section 3: Classification of Emergencies

3.1 General

Section D of the Exelon Nuclear Standardized Emergency Plan divides the types of emergencies into four Emergency Classification Levels (ECLs). The first four are the UNUSUAL EVENT, ALERT, SITE AREA EMERGENCY, and GENERAL EMERGENCY. These ECLs are entered by meeting the Emergency Action Level (EAL) Threshold Values provided in this section of the Annex. 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. Depending on the severity of an event, prior to returning to a standard day-to-day organization, a state or phase called RECOVERY may be entered to provide dedicated resources and organization in support of restoration and communication activities following the termination of the emergency.

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 Guideline 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 acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

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

RECOVERY: Recovery can be considered as a phase of the emergency and is entered by meeting emergency termination criteria provided in EP-AA-111 Emergency Classification and Protective Action Recommendations.

An emergency is classified by assessing plant conditions and comparing abnormal conditions to Initiating Conditions and Threshold Values for each Emergency Action Level.

Individuals responsible for the classification of events will refer to the Initiating Condition and Threshold Values on the matrix of the appropriate station Standardized Emergency Plan Annex (this document). This matrix will contain Initiating Conditions, EAL Threshold Values, Mode Applicability Designators, appropriate EAL numbering system, and additional guidance necessary to classify events. It may be provided as a user aid.

The matrix is set up in four Recognition Categories. The first is designated as "R" and relates to Abnormal Radiological Conditions / Abnormal Radiological Effluent Releases. The second is designated as "F" and relates to Fission Product Barrier Degradation. The third is designated as "M" and relates to System Malfunctions. The fourth is designated as "H" and relates to Hazards and Other Conditions.

The matrix is designed to provide an evaluation of the Initiating Conditions from the worst conditions (General Emergencies) on the left to the relatively less severe conditions on the right (Unusual Events). Evaluating conditions from left to right will reduce the possibility that an event will be under classified. All Recognition Categories should be reviewed for applicability prior to classification.

The Initiating Conditions are coded with a two letter and one number code. The first letter is the Recognition Category designator, the second letter is the Classification Level, “U” for (NOTIFICATION OF) UNUSUAL EVENT, “A” for ALERT, “S” for SITE AREA EMERGENCY and “G” for GENERAL EMERGENCY. The EAL number is a sequential number for that Recognition Category series. All Initiating Conditions that are describing the severity of a common condition (series) will have the same number.

The EAL number may then be used to reference a corresponding page(s), which provides the basis information pertaining to the Initiating Condition:

- Threshold Value
- Mode Applicability
- Basis

Emergency Action Levels are the measurable, observable detailed conditions that must be met in order to classify the event. Classification is not to be made without referencing, comparing and satisfying the Threshold Values specified in the Emergency Action Levels.

A list of definitions is provided as part of this document for terms having specific meaning to the Emergency Action Levels. Site specific definitions are provided for terms with the intent to be used for a particular Initiating Condition/Threshold Value and may not be applicable to other uses of that term at other sites, the Emergency Plan or procedures.

References are also included to documents that were used to develop the EAL Threshold Values.

References to the Emergency Director means the person in Command and Control as defined in the Emergency Plan. Classification of emergencies is a non-delegable responsibility of Command and Control for the onsite facilities with responsibility assigned to the Shift Emergency Director (Control Room Shift Manager) or the Station Emergency Director (TSC). Classification of emergencies remains the responsibility of the applicable onsite facility even after Command and Control is transferred to the Corporate Emergency Director (EOF).

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

Indications used for monitoring and evaluation of plant conditions include the normally used instrumentation, backup or redundant instrumentation, and the use of other parameters that provide information that supports determination if an EAL threshold value has been reached. When an EAL refers to a specific instrument or indication that is determined to be inaccurate or unavailable, then alternate indications shall be used to monitor the specified condition.

During an event that results in changing parameters trending towards an EAL classification, and instrumentation that was available to monitor this parameter becomes unavailable or the parameter goes off scale, the parameter should be assumed to have been exceeded consistent with the trend and the classification made if there are no other direct or indirect means available to determine if the threshold has not been exceeded.

EALs are for unplanned events. A planned evolution involves 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. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL Threshold Value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

When two or more Emergency Action Levels are determined, declaration will be made on the highest classification level for the Unit. When both units are affected, the highest classification for the Station will be used for notification purposes and both units' classification levels will be noted.

3.2 Mode Applicability

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

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 Fission Product Barrier Matrix EALs are applicable only to events that initiate in Hot Shutdown or higher.

If there is a change in 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.

3.3 Emergency Director Judgment

Emergency Director Judgment EALs are provided in the Hazards and Other Condition Affecting Plant Safety section and on the Fission Product Barrier Matrix. Both of the Emergency Director Judgment EALs have specific criteria for when they should be applied.

The Hazards Section Emergency Director Judgment EALs are intended to address unanticipated conditions which are not addressed explicitly by other EALs but warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under specific emergency classifications (UE, Alert, SAE or GE).

The FPB Matrix ED Judgment EALs are intended to include unanticipated conditions, which are not addressed explicitly by any of the other FPB threshold values, but warrant determination because conditions exist that fall under the broader definition for a significant Loss or Potential Loss of the barrier (equal to or greater than the defined FPB threshold values).

3.4 Fission Product Barrier Restoration

Fission Product Barriers (FPBs) are not treated the same as EAL threshold values. Conditions warranting declaration of the loss or potential loss of a Fission Product Barrier may occur resulting in a specific classification. The condition that caused the loss or potential loss declaration could be rectified as the result of Operator action, automatic actions, or designed plant response. Barriers will be considered re-established when there are direct verifiable indications (containment penetration or open valve has been isolated, coolant sample results, etc) that the barrier has been restored and is capable of mitigating future events.

The reestablishment of a fission product barrier does not alter or lower the existing classification. Entry into Termination/Recovery phase is still required for exiting the present classification. However the reestablishment of the barrier should be considered in determining future classifications should plant conditions or events change.

3.5 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 five or more persons violently protesting station operations or activities at the site.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

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

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

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

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

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute 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, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

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.

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

NORMAL LEVELS: Normal levels can be considered as 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.

OPERATING MODES	REACTOR MODE SWITCH POSITION	TEMP
(1) Power Operation:	Run	N/A
(2) Startup:	Refuel [(a)] or Startup/Hot Standby	N/A
(3) Hot Shutdown [(a)]:	Shutdown	> 200° F
(4) Cold Shutdown [(a)]:	Shutdown	≤ 200° F
(5) Refueling [(b)]:	Shutdown or Refuel	N/A
(D) Defueled:	All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).	

[(a)] All reactor vessel head closure bolts fully tensioned.

[(b)] One or more reactor vessel head closure bolts less than fully tensioned.

Hot Matrix – applies in modes (1), (2), and (3)

Cold Matrix – applies in modes (4), (5), and (D)

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

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

SABOTAGE: A 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.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations >10% Reactor Power change.

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 isolated from the Control Room.

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

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

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

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

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General		Site Area		Alert		Unusual Event	
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Fuel Clad		RCS		Containment	
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MG1	3-74	MS1	3-77	MA1	3-79	MU1	3-81
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	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Vent Stack and SBGT WRGMs that exceeds or is expected to exceed **3.70E+08 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** **NOTE:** If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Vent Stack and SBGT WRGMs that exceeds or is expected to exceed **3.70E+07 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on any effluent monitor **> 200 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**. **OR** 2. The sum of VALID readings on the Vent Stack and SBGT WRGMs is **> 1.90E+07 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on any effluent monitor **> 2 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**. **OR** 2. The sum of VALID readings on the Vent Stack and SBGT WRGMs is > **9.66E+05 uCi/sec** for **≥ 60 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Abnormal Rad Levels			**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading **> 1000 mR/hr** on radiation monitor ARM 0D21-K604A. **OR** 2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in Irradiated Fuel becoming uncovered.	**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by: • Refueling Cavity water level **< 340 in.** on shutdown range. **OR** • Spent Fuel Pool water level **< 21ft. 4 in**. **OR** • Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal. **AND** b. UNPLANNED VALID Area Radiation Monitor reading rise on refuel radiation monitor ARM 0D21-K604A. **OR** 2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.
	Table R1 **Areas Requiring Continuous Occupancy** • Main Control Room (1(2)D18-K751A-D) • Central Alarm Station (by survey) • Secondary Alarm Station (by survey) • TSC (if staffed) (Panel 0PLC1J ARM Channel 4-10) • Radwaste Control Room (Panel 0PLC1J ARM Channel 4-5) • Remote Shutdown Panels (1(2)D21-K601F)	**Table R2** **Areas Requiring Infrequent Access** • RB Sample (K601G) • Aux Building Containment Purge (K602I) • Reactor Building HCU Modules (K601C,D) • RHR Heat Exchanger Rooms (K602E,F) • RCIC Room (K602G) • HPCS Room (K601H) • SBGT (K602A)	**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID radiation monitor or survey readings **>15 mR/hr** in areas requiring continuous occupancy (Table R1) to maintain plant safety functions. **OR** 2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R2) which will impede necessary access and threaten safe operation of the plant.	**RU3** Fuel clad degradation. 1 2 3 **EAL Threshold Values:** 1. Offgas system isolation due to VALID offgas post-treatment radiation monitor signal. **OR** 2. Specific coolant activity **> 4.0 uCi/gm** Dose Equivalent I-131.

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

Fission Product Barrier Matrix

Hot Matrix

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
FG1 Loss of ANY two barriers AND Loss or Potential Loss of third barrier. 1 2 3	**FS1** Loss or Potential Loss of ANY two barriers. 1 2 3	**FA1** ANY Loss or ANY Potential Loss of either Fuel Clad or RCS. 1 2 3	**FU1** ANY Loss or ANY Potential Loss of Containment. 1 2 3

<table>
<tr><th rowspan="2">Sub-Category</th><th colspan="2">FC - Fuel Clad</th><th colspan="2">RC – Reactor Coolant System</th><th colspan="2">CT - Containment</th></tr>
<tr><th>Loss</th><th>Potential Loss</th><th>Loss</th><th>Potential Loss</th><th>Loss</th><th>Potential Loss</th></tr>
<tr><td>1. RCS Activity →</td><td>Coolant activity > 300 uCi/gm Dose Equivalent I-131</td><td>None</td><td>None</td><td>None</td><td>None</td><td>None</td></tr>
<tr><td>2. RPV Water Level →</td><td>1. RPV level < –185 in. without adequate core spray.
OR
2. RPV level < –210 in.</td><td>RPV level < -161 in. (TAF).</td><td>RPV level < -161 in. (TAF).</td><td>None</td><td>None</td><td>Plant conditions indicate that Primary Containment Flooding is required.</td></tr>
<tr><td>3. Drywell Pressure →</td><td>None</td><td>None</td><td>1. Drywell pressure > 1.77 psig.
AND
2. Drywell pressure rise due to RCS leakage.</td><td>None</td><td>1. Rapid unexplained drop in Drywell pressure following initial pressure rise.
OR
2. Drywell pressure response not consistent with LOCA conditions.</td><td>1. Drywell pressure ≥ 45 psig and rising.
OR
2. a. Drywell or suppression chamber hydrogen concentration ≥ 6%.
AND
b. Drywell or suppression chamber oxygen concentration ≥ 5%.</td></tr>
<tr><td>4. RCS Leakrate →</td><td>None</td><td>None</td><td>1. UNISOLABLE Main Steam Line (MSL) break as indicated by the failure of both MSIVs in ANY one line to close.
AND
2. a. High MSL Flow AND High Steam Tunnel Temperature.
OR
b. Direct report of steam release.</td><td>1. RCS leakage > 50 gpm inside the drywell.
OR
2. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels > LGA-002 Maximum Normal operating levels.</td><td colspan="2">
<table>
<tr><th colspan="2">Table F2 Drywell Radiation Thresholds</th></tr>
<tr><th>Time After Shutdown (hrs)</th><th>Containment Potential Loss (R/hr)</th></tr>
<tr><td>≤ 2</td><td>4.35 E+02</td></tr>
<tr><td>> 2 to 4</td><td>3.75 E+02</td></tr>
<tr><td>> 4 to 8</td><td>3.15 E+02</td></tr>
<tr><td>> 8 to 16</td><td>2.60 E+02</td></tr>
<tr><td>> 16 to 23</td><td>2.30 E+02</td></tr>
<tr><td>> 23</td><td>2.25 E+02</td></tr>
</table>
</td></tr>
<tr><td>5. Hi Cont/Drywell Rad →</td><td>Drywell radiation monitor reading > Fuel Cladding Loss Threshold, Table F1.</td><td>None</td><td>1. Drywell Radiation monitor reading > 100 R/hr
AND
2. Indications of RCS leakage into the Drywell.</td><td>None</td><td>None</td><td>Drywell radiation monitor reading > Containment Potential Loss Threshold, Table F2.</td></tr>
<tr><td>6. Breach/Bypass →</td><td colspan="2">
<table>
<tr><th colspan="2">Table F1 Drywell Radiation Thresholds</th></tr>
<tr><th>Time After Shutdown (hrs)</th><th>Fuel Cladding Loss (R/hr)</th></tr>
<tr><td>≤ 2</td><td>1.90 E+02</td></tr>
<tr><td>> 2 to 4</td><td>1.65 E+02</td></tr>
<tr><td>> 4 to 8</td><td>1.40 E+02</td></tr>
<tr><td>> 8 to 16</td><td>1.12 E+02</td></tr>
<tr><td>> 16 to 23</td><td>9.90 E+01</td></tr>
<tr><td>> 23</td><td>9.65 E+01</td></tr>
</table>
</td><td>None</td><td>None</td><td>1. a. Failure of all isolation valves in any one line to close.
AND
b. A downstream pathway to the environment exists.
OR
2. Intentional venting/purging of Primary Containment per EOPs or SAMGs due to accident conditions.
OR
3. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels > LGA-002, Maximum Safe operating levels.</td><td>None</td></tr>
<tr><td>7. ED Judgment→</td><td>Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.</td><td>Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.</td><td>Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.</td><td>Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.</td><td>Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.</td><td>Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.</td></tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
AC Electrical Distribution	**MG1** Prolonged loss of all offsite power and prolonged loss of all onsite AC power to Division 1 and Division 2 essential busses. 1 2 3 **EAL Threshold Values:** 1. Loss of power to System Auxiliary Transformer 142(242) and Unit Auxiliary Transformer 141(241). **AND** 2. Failure of DG 0 and DG 1A(2A) emergency diesel generators to supply power to unit ECCS busses. **AND** 3. a. Restoration of either Unit ECCS bus (excluding Division 3) within 4 hours is **not** likely. **OR** b. RPV level **cannot** be determined to be **> -150 in.** on WR at RSP.	**MS1** Loss of all offsite power and loss of all onsite AC power to Division 1 and Division 2 essential busses. 1 2 3 **EAL Threshold Values:** 1. Loss of power to System Auxiliary Transformer 142(242) and Unit Auxiliary Transformer 141(241). **AND** 2. Failure of DG 0 and DG 1A(2A) emergency diesel generators to supply power to unit ECCS busses. **AND** 3. Failure to restore power to at least one Unit ECCS bus (excluding Division 3) within **15 minutes** from the time of loss of both offsite and onsite AC power.	**MA1** AC power capability to Division 1 and Division 2 essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout. 1 2 3 **EAL Threshold Values:** 1. AC power capability to unit ECCS busses (excluding Division 3) reduced to only one of the following power sources for **> 15 minutes**: • System Auxiliary Transformer 142(242) • Unit Auxiliary Transformer 141(241) • Unit Emergency Diesel Generator 1A(2A) • Shared Emergency Diesel Generator DG 0 • Other unit SAT via crosstie breakers **AND** 2. Any additional single power source failure will result in unit blackout.	**MU1** Loss of all offsite power to Division 1 and Division 2 essential busses for greater than 15 minutes. 1 2 3 4 5 **EAL Threshold Values:** Loss of power to System Auxiliary Transformer 142(242) **AND** Unit Auxiliary Transformer 141(241) for **> 15 minutes**.
RPS / Inadvertent Criticality	**MG3** Failure of the Reactor Protection System to complete an automatic scram and manual scram was **NOT** successful and there is indication of an extreme challenge to the ability to cool the core. 1 2 **EAL Threshold Values:** 1. Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by reactor power **> 3% APRM.** **AND** 2. a. RPV level cannot be restored and maintained **> - 150 in**. on WR (– 185 in. on FZ if WR not available). **OR** b. Heat Capacity Limit (LGA-003 Fig. H) exceeded.	**MS3** Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded and manual scram was **NOT** successful. 1 2 **EAL Threshold Values:** Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by reactor power **> 3% APRM.**	**MA3** Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded. 1 2 **EAL Threshold Values:** 1. A Reactor Protection System setpoint was exceeded. **AND** 2 Automatic scram did not reduce reactor power < **40 on IRM Range 7**.	**MU3** Inadvertent criticality. 3 4 5 **EAL Threshold Values:** An UNPLANNED extended positive period observed on nuclear instrumentation.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
DC Power		**MS4** Loss of all vital DC power. 1 2 3 **EAL Threshold Values:** Loss of all vital DC power based on **< 108 VDC** on 125 VDC battery busses 111Y(211Y) and 112Y(212Y) for **> 15 minutes**.		
Heat Sink		**MS5** Complete loss of heat removal capability. 1 2 3 **EAL Threshold Values:** Heat Capacity Limit Curve (LGA-003) exceeded.		
Annunciators		**MS6** Inability to monitor a SIGNIFICANT TRANSIENT in progress. 1 2 3 **EAL Threshold Values:** 1. Loss of most **(approximately 75%)** safety system annunciators (Table M2). **AND** 2. Indications needed to monitor safety functions (Table M3) are unavailable. **AND** 3. SIGNIFICANT TRANSIENT in progress (Table M4). **AND** 4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MA6** UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable. 1 2 3 **EAL Threshold Values:** 1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes**. **AND** 2. a. SIGNIFICANT TRANSIENT in progress (Table M4). **OR** b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MU6** UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes. 1 2 3 **EAL Threshold Values:** 1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** 2. UNPLANNED loss of most **(approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.
RCS Leakage				**MU7** RCS leakage. 1 2 3 **EAL Threshold Values:** 1. Unidentified or pressure boundary leakages **> 10 gpm**. **OR** 2. Identified leakage **> 25 gpm**.
		Table M2 - Control Room Panels	**Table M3 - Safety Functions and Related Systems**	**Table M4 - Significant Transients**
		• 1(2)H13-P601 • 1(2)H13-P603 • 1(2)PM01J	• 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) • Fission Product Barriers	• Turbine trip • Reactor scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10%** Reactor Power change

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

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<tr><td>Satellite Phones</td><td></td><td>X</td></tr>
<tr><td>Cellular Phones</td><td></td><td>X</td></tr>
</table>
</td><td>MU10 UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5

<u>EAL Threshold Values:</u>
1. Loss of all Table M6 Onsite communications capability affecting the ability to perform routine operations.
OR
2. Loss of all Table M6 Offsite communications capability.</td></tr>
<tr><td>T. S. Time</td><td></td><td></td><td></td><td>MU11 Inability to reach required shutdown within Technical Specification limits. 1 2 3
<u>EAL Threshold Values:</u>
Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.</td></tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, "Threat Assessment." **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evac		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 D **EAL Threshold Values:** 1. Control Room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per LOA-RX 101 (201) in **< 15 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 D **EAL Threshold Values:** Entry into LOA-RX-101(201) for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Reactor Building • Control Room • Auxiliary Building • Diesel Generator Rooms • Switchgear and Battery Rooms • Remote Shutdown Rooms • CSCS Pump Rooms • LSH (for 0E12-F300 access only)	**Table H3** **Internal Flooding Areas** • RCIC Room • B/C RHR Room • HPCS Room • A RHR Room • RB Raceway	**HA5** Natural and destructive phenomena affecting a VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by seismic instrumentation **> 0.10 g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 90 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in any Table H3 area as indicated in the Control Room. b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event as indicated by Station seismic monitoring procedures **> 0.01g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained **(> 15 minutes)** high winds **> 90 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm: **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

Hazards and Other Conditions Affecting Plant Safety

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Toxic / Flammable Gas	**Table H2** **Vital Areas** - Reactor Building - Control Room - Auxiliary Building - Diesel Generator Rooms - Switchgear and Battery Rooms - Remote Shutdown Rooms - CSCS Pump Rooms - LSH (for 0E12-F300 access only)		**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 D **EAL Threshold Values:** 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.

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

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	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Vent Stack and SBGT WRGMs that exceeds or is expected to exceed **3.70E+08 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Vent Stack and SBGT WRGMs that exceeds or is expected to exceed **3.70E+07 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on any effluent monitor **> 200 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**. **OR** 2. The sum of VALID readings on the Vent Stack and SBGT WRGMs is **> 1.90E+07 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on any effluent monitor **> 2 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**. **OR** 2. The sum of VALID readings on the Vent Stack and SBGT WRGMs is **> 9.66E+05 uCi/sec** for **≥ 60 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Abnormal Rad Levels			**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading **> 1000 mR/hr** on radiation monitor ARM 0D21-K604A. **OR** 2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in Irradiated Fuel becoming uncovered.	**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by: • Refueling Cavity water level **< 340 in.** on shutdown range. **OR** • Spent Fuel Pool water level **< 21 ft. 4 in**. **OR** • Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal. **AND** b. UNPLANNED VALID Area Radiation Monitor reading rise on refuel radiation monitor ARM 0D21-K604A. **OR** 2. UNPLANNED VALID Area Radiation Monitor readings rise by a factor of **1000** over NORMAL LEVELS.
	Table R1 **Areas Requiring Continuous Occupancy** • Main Control Room (1(2)D18-K751A-D) • Central Alarm Station (by survey) • Secondary Alarm Station (by survey) • TSC (if staffed) (Panel 0PLC1J ARM Channel 4-10) • Radwaste Control Room (Panel 0PLC1J ARM Channel 4-5) • Remote Shutdown Panels (1(2)D21-K601F)	**Table R2** **Areas Requiring Infrequent Access** • RB Sample (K601G) • Aux Building Containment Purge (K602I) • Reactor Building HCU Modules (K601C,D) • RHR Heat Exchanger Rooms (K602E,F) • RCIC Room (K602G) • HPCS Room (K601H) • SBGT (K602A)	**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID radiation monitor or survey readings **>15 mR/hr** in areas requiring continuous occupancy (Table R1) to maintain plant safety functions. **OR** 2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R2) which will impede necessary access and threaten safe operation of the plant.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunction				
AC Electrical Distribution				**MU1** Loss of all offsite power to Division 1 and Division 2 essential busses for greater than 15 minutes. 1 2 3 4 5 **EAL Threshold Values:** Loss of power to System Auxiliary Transformer 142(242) **AND** Unit Auxiliary Transformer 141(241) for **> 15 minutes**.
			MA2 Loss of all offsite power and loss of all onsite AC power to Division 1 and Division 2 essential busses. 4 5 D **EAL Threshold Values:** 1. Loss of AC power to System Auxiliary Transformer 142(242) and Unit Auxiliary Transformer 141(241). **AND** 2. Failure of DG 0 and DG 1A (2A) emergency diesel generators to supply power to unit ECCS busses. **AND** 3. Failure to restore power to at least one unit ECCS bus (excluding Division 3**) within 15 minutes** from the time of loss of both offsite and onsite AC power.	
RPS				**MU3** Inadvertent criticality. 3 4 5 **EAL Threshold Values:** An UNPLANNED extended positive period observed on nuclear instrumentation.
DC Power				**MU4** UNPLANNED loss of required DC power for greater than 15 minutes. 4 5 **EAL Threshold Values:** 1. UNPLANNED loss of all required vital DC Power based on **< 108 VDC** indication on 125 VDC battery busses 111Y(211Y) and 112Y(212Y). **AND** 2. Failure to restore power to at least one required DC bus **within 15 minutes** from the time of loss.

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

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

System Malfunction

Heat Sink

GENERAL EMERGENCY

SITE AREA EMERGENCY

Table M1 – RCS Reheat Duration Thresholds

RCS	Secondary Containment Closure	Duration
Intact	N/A	60 minutes*
Not Intact	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.

ALERT

MA5 Inability to maintain plant in Cold Shutdown with irradiated fuel in the RPV. 4 5

EAL Threshold Values:

1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200°F** for > Table M1 duration.

   OR

2. UNPLANNED RPV pressure rise > **10 psig** as a result of temperature rise due to loss of decay heat removal.

UNUSUAL EVENT

MU5 UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV. 4 5

EAL Threshold Values:

1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F.**

   OR

2. Loss of all RCS temperature AND RPV level indication for **> 15 minutes**.

RCS Leakage / Inventory

GENERAL EMERGENCY

MG8 Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged with irradiated fuel in the RPV. 4 5

EAL Threshold Values:

1. Loss of RPV inventory per Table M5 indications.

   AND

2. a. RPV level **< -161 in.** (TAF) on FZ for **> 30 minutes.**

   OR

   b. RPV level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following:
   - Refuel floor Rad monitor 0D21-K604A indicates > **3000 mR/hr** or off-scale high.
   - Erratic Source Range Monitor indication.

   AND

3. Containment is challenged as indicated by one or more of the following:
   - Primary containment Hydrogen concentration **≥ 6%** and Oxygen concentration **≥ 5%.**
   - Drywell pressure **≥ 45 psig.**
   - Primary and Secondary CONTAINMENT CLOSURE not established.
   - Any Secondary Containment radiation monitor **> LGA-002 Maximum Safe** operating levels.

SITE AREA EMERGENCY

MS8 Loss of RCS/RPV inventory affecting core decay heat removal capability. 4

EAL Threshold Values:

1. **WITHOUT** Primary or Secondary CONTAINMENT CLOSURE established:

   a. RPV level **< - 150 in**.

   OR

   b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory per Table M5 indications.

   OR

2. **WITH** Primary or Secondary CONTAINMENT CLOSURE established:

   a RPV level **< -161 in**. (TAF).

   OR

   b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory as evidenced by either of the following:
   - Per Table M5 indications.
   - Erratic Source Range Monitor indication.

ALERT

MA8 Loss of RCS/RPV inventory with irradiated fuel in the RPV. 4 5

EAL Threshold Values:

1. Loss of RCS/RPV inventory as indicated by RPV level **< -147 in**. WR.

   OR

2. a. Loss of RPV inventory per Table M5 indications.

   AND

   b. RCS/RPV level unknown for **> 15 minutes.**

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise
• Unexplained Suppression Pool level rise
• Unexplained vessel make up rate rise
• Observation of leakage or inventory loss

UNUSUAL EVENT

MU8 RCS leakage. 4

EAL Threshold Values:

RPV level **cannot** be restored and maintained **> 11.0 in.**

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

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<td>**RCS Leakage / Inventory**</td>
<td></td>
<td>**MS9** Loss of RPV inventory affecting core decay heat removal capability with irradiated fuel in the RPV. 5
EAL Threshold Values:
1. **WITHOUT** Secondary CONTAINMENT CLOSURE established:
a. RPV level **< - 150**.
OR
b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
• Refuel Floor Rad Monitor 0D21-K604A indicates > **3000 mR/hr** or off-scale high.
• Erratic Source Range Monitor indication.
OR
2. **WITH** Secondary CONTAINMENT CLOSURE established:
a. RPV level **< -161 in**. (TAF).
OR
b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
• Refuel Floor Rad Monitor 0D21-K604A indicates > **3000 mR/hr** or off-scale high.
• Erratic Source Range Monitor indication.</td>
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• Unexplained Suppression Pool level rise
• Unexplained vessel make up rate rise
• Observation of leakage or inventory loss</td></tr>
</table></td>
<td>**MU9** UNPLANNED loss of RCS inventory with irradiated fuel in the RPV. 5
EAL Threshold Values:
1. UNPLANNED RPV level drop below the RPV flange for **> 15 minutes.**
OR
2. a. Loss of RPV inventory per Table M5 indications.
AND
b. RPV level unknown.</td>
</tr>
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<tr><td>HPN</td><td></td><td>X</td></tr>
<tr><td>Satellite Phones</td><td></td><td>X</td></tr>
<tr><td>Cellular Phones</td><td></td><td>X</td></tr>
</table></td>
<td>**MU10** UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5
EAL Threshold Values:
1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations.
OR
2. Loss of all Table M6 **Offsite** communications capability.</td>
</tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, “Threat Assessment.” **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a Security Event as determined from Station Security Plan – Appendix C.
C. R. Evac		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 D **EAL Threshold Values:** 1. Control room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per LOA-RX-101(201) in **< 15 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 D **EAL Threshold Values:** Entry into LOA-RX-101(201) for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Reactor Building • Control Room • Auxiliary Building • Diesel Generator Rooms • Switchgear and Battery Rooms • Remote Shutdown Rooms • CSCS Pump Rooms • LSH (for 0E12-F300 access only)	**Table H3** **Internal Flooding Areas** • RCIC Room • B/C RHR Room • HPCS Room • A RHR Room • RB Raceway	**HA5** Natural and destructive phenomena affecting a VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by seismic instrumentation **> 0.10 g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 90 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in any Table H3 area as indicated in the Control Room. b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event as indicated by station seismic monitoring procedures **> 0.01g**. **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained **(> 15 minutes)** high winds **> 90 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas	**Table H2** **Vital Areas** • Reactor Building • Control Room • Auxiliary Building • Diesel Generator Rooms • Switchgear and Battery Rooms • Remote Shutdown Rooms • CSCS Pump Rooms • LSH (for 0E12-F300 access only)		**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 D **EAL Threshold Values:** 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.

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

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RG1

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Vent Stack and SBGT WRGMs that exceeds or is expected to exceed **3.70 E+08 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 1000 mRem TEDE**

      OR

   b. **> 5000 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RG1 (cont)

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events that may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in a PAG dose being reached. Assumptions and DAPAR inputs are provided in Calc. EP-EAL-0605 (reference 9).

The sum of both units’ monitors provides the total station release rate.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

Threshold #2 Basis:

The TEDE (1000 mRem) and the CDE Thyroid (5000 mRem) doses are set at the EPA PAG Limits.

The “site boundary” is defined by an approximately 800-meter (1/2-mile) radius around the plant. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RG1 (cont)

Basis (cont):

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release has been stopped and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev 4 AG1
2. EP-AA-112-500, Emergency Environmental Monitoring
3. Exelon DAPAR version 3.1
4. EP-MW-110-200 Dose Assessment
5. ODCM Section 12.4 Gaseous Effluents and Total Dose
6. UFSAR Section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems
7. EP-EAL-0605, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values LaSalle Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RS1

Initiating Condition:

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Vent Stack and SBGT WRGMs that exceeds or is expected to exceed **3.70 E+07 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 100 mRem TEDE**

      OR

   b. **> 500 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RS1 (cont)

Basis (cont):

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events that may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in 10% of a PAG dose being reached. Assumptions and DAPAR inputs are provided in Calc. EP-EAL-0605 (reference 9).

The sum of both units' monitors provides the total station release rate.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

Threshold #2 Basis:

The TEDE (100 mRem) and the CDE Thyroid (500 mRem) doses are set at 10% of the EPA PAG Limits.

The "site boundary" is defined by an approximately 800-meter (1/2-mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

RS1 (cont)

Basis (cont):

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release has been stopped and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev 4 AS1
2. EP-AA-112-500, Emergency Environmental Monitoring
3. Exelon DAPAR version 3.1
4. EP-MW-110-200 Dose Assessment
5. ODCM Section 12.4, Gaseous Effluents and Total Dose
6. UFSAR Section 11.5,
7. EP-EAL-0605, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values LaSalle Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading on any effluent monitor **> 200 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**.

   OR

2. The sum of VALID readings on the Vent Stack and SBGT WRGMs is **> 1.90 E+07 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.

Basis:

UNPLANNED: As used in this context, 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.

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

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

Basis (cont):

Threshold #1 Basis:

The threshold addresses radioactivity releases (liquid or gaseous) that for whatever reason cause effluent radiation monitor readings to exceed two hundred times the alarm setpoint established by the radioactive discharge permit. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the Offsite Dose Calculation Manual (ODCM) to warn of a release that is not in compliance with the Radiological Effluent Technical Specifications (RETS). Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

An elevated monitoring reading while the effluent flow path is isolated is NOT considered to be a VALID reading.

The Liquid Radwaste Discharge Monitor (LCRM 0D18-K606) measures the radioactivity in the radwaste effluent discharge line before mixing with the lake blowdown flow prior to entering the river. Prior to discharge to the lake blowdown from the discharge tanks, the liquid is sampled and analyzed for radioactivity. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate. A HI-HI radiation setpoint or INOP closes the lake blowdown radwaste discharge flow control valve 0WL067. A setpoint is established so that automatic valve closure on the radwaste discharge line occurs before 10 CFR 20 limits are reached for drinking water in the river downstream of the plant.

Threshold #2 Basis:

LaSalle incorporates 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 ODCM. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

This EAL addresses a potential or actual drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. However, the effluent monitor Alert value for gaseous effluents was reduced to 10 X ODCM limits to ensure sequential classifications. The sum of both gaseous effluent monitor readings provides a total station release rate because Unit 1 and Unit 2 discharge through the same monitors. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM. Assumptions and DAPAR inputs are provided in Calc. EP-EAL-0605 (reference 13). The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA1 (cont)

Basis (cont):

Threshold #3 Basis:

Confirmed sample analyses in excess of two hundred times the site ODCM limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by increasing the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mRem/yr basis of the 10 CFR 20 limits for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be approximately 10 mRem/hr. The required release duration was reduced to 15 minutes in recognition of the increased severity.

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

Basis Reference(s):

1. NEI 99-01, Rev 4 AA1
2. Sargent & Lundy calculation ATD-0227, Rev. 0, 1/12/93
3. S&L Calculation L-002356 1999
4. Calculation 3-PR-10, Rev. 1 2/9/93
5. ODCM Section 12.3, Liquid Effluents
6. ODCM Section 12.4, Gaseous Effluents and Total Dose
7. UFSAR Section 11.5
8. LCP-140-7, Analysis Of Radwaste Discharge Tanks 1(2)WF05T and Determination Of Discharge Flowrate And Liquid Radwaste Effluent Monitor Response
9. Structural Drawing S-01A Composite Site Plan LaSalle Station Units 1 & 2
10. LAP-1800-4, Chemistry Department Improved Technical Specifications, Technical Requirements Manual, TRM Appendixes, Offsite Dose Calculation Manual (ODCM) LaSalle Annex Check Lists
11. LYP-1200-2, Instantaneous Airborne Releases – 10 CFR 20 Design Objectives
12. EP-EAL-0605, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values LaSalle Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading on any effluent monitor **> 2 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**.

   OR

2. The sum of VALID readings on the Vent Stack and SBGT WRGMs is **> 9.66 E+05 uCi/sec** for **≥ 60 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

Basis:

<u>UNPLANNED:</u> As used in this context, 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.

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

The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

Basis (cont):

Threshold #1 Basis:

The effluent release paths are monitored for radioactivity prior to the flow reaching the point where it would mix with the process flow to the environment. Prior to initiating batch releases, the discharge volume is sampled and analyzed for radioactivity. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate. Radiation monitor alarm setpoints are established to automatically isolate the process flow at the point determined by the discharge permit. These limits are based on the Offsite Dose Calculation Manual ODCM.

An elevated monitoring reading while the effluent flow path is isolated is NOT considered to be a VALID reading.

The Liquid Radwaste Discharge Monitor (LCRM 0D18-K606) measures the radioactivity in the radwaste effluent discharge line before mixing with the lake blowdown flow prior to entering the river. Prior to discharge to the lake blowdown from the discharge tanks, the liquid is sampled and analyzed for radioactivity. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate. A HI-HI radiation setpoint or INOP closes the lake blowdown radwaste discharge flow control valve 0WL067. A setpoint is established so that automatic valve closure on the radwaste discharge line occurs before 10 CFR 20 limits are reached for drinking water in the river downstream of the plant.

Threshold #2 Basis:

This EAL addresses a potential drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 2, regulatory commitments for an extended period of time. The sum of both gaseous effluent monitor readings provides a total station release rate because Unit 1 and Unit 2 discharge through the same monitors. The gaseous effluent value was determined on a per-station basis using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM. Assumptions and DAPAR inputs are provided in Calc. EP-EAL-0605 (reference 10).

The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU1 (cont)

Basis (cont):

Threshold #3 Basis:

Confirmed sample analyses in excess of two times the site ODCM limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times ODCM for 30 minutes does not exceed this EAL. Grab samples are used to determine release concentrations or rates to confirm meter readings or when the effluent monitors are not in service. Maximum instantaneous release rate limits are calculated in accordance with the ODCM. These are indicated on approved discharge permits.

Basis Reference(s):

1. NEI 99-01, Rev 4 AU1
2. Sargent & Lundy calculation ATD-0227, Rev. 0, 1/12/93
3. S&L Calculation (L-002356) 1999
4. Calculation 3-PR-10, Rev. 1 2/9/93
5. ODCM Section 12.3, Liquid Effluents
6. ODCM Section 12.4, Gaseous Effluents and Total Dose

7. UFSAR Section 11.5
8. LCP-140-7, Analysis of Radwaste Discharge Tanks 1(2)WF05T and Determination of Discharge Flowrate and Liquid Radwaste Effluent Monitor Response
9. EP-EAL-0605, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values LaSalle Station
10. LAP-1800-4, Chemistry Department Improved Technical Specifications, Technical Requirements Manual, TRM Appendixes, Offsite Dose Calculation Manual (ODCM) LaSalle Annex Check Lists
11. LYP-1200-2, Instantaneous Airborne Releases – 10 CFR 20 Design Objectives

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA2

Initiating Condition:

Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading **> 1000 mR/hr** on the radiation monitor ARM 0D21-K604A.

   OR

2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in Irradiated Fuel becoming uncovered.

Basis:

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

Threshold #1 Basis:

Uncovering spent fuel represents a substantial degradation of the level of safety of the plant and warrants an Alert classification. Time is available to take corrective actions. 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, no prompt fatalities are predicted and the risk of injury is low. 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 fuel uncovery begins. The value of 1000 mR/hr was conservatively chosen for classification purposes.

Radiation monitor readings are used to provide indication of fuel uncovery and/or fuel damage. High monitor readings associated with the transfer or relocation of a source, stored in or near the pool or readings responding to a planned evolution such as removal of the reactor head or equipment relocation are not classified under this threshold since the reading would not be indicative of fuel uncovery and/or fuel damage.

Dropping heavy loads onto the spent fuel can cause significant damage to the spent fuel and an Alert is also warranted under these conditions provided that the above radiation monitor threshold readings are reached.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA2 (cont)

Basis (cont):

Threshold #2 Basis:

When the RPV head is removed and the Shutdown Range RPV level instrument range is expanded to indicate levels as high as the refuel floor elevation, remote indication of Refueling Cavity water level is available in the Control Room. Once Spent Fuel Pool water level drops below the low level alarm setpoint (842 ft. 1 in. el.), further drops can be monitored only by visual observation unless the Spent Fuel Pool is in communication with the Refueling Cavity. Even so, uncovery of spent fuel seated in the Spent Fuel Pool storage racks cannot be monitored remotely because the bottom of the fuel transfer canal is above the elevation of the top of the storage racks. Any fuel that becomes uncovered while suspended from the refuel grapple may be indicated on the Shutdown Range instrument but, without report of the vertical position of the grapple, fuel uncovery cannot be determined. Visual observation, therefore, provides the only viable mechanism of determining if spent fuel in the Fuel Pool or Refueling Cavity will be uncovered.

This EAL applies to irradiated fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

Basis Reference(s):

1. NEI 99-01, Rev 4 AA2
2. LOA-FH-001 Irradiated Fuel Assembly Damage
3. LOR-1(2)H13-P601-B108, Refuel Floor Area High Range/Low Range Radiation - High
4. LOR-1(2)H13-P601-E205/F205, Fuel Pool Radiation High High
5. Technical Specification 3.9.6
6. LOP-SF-06 Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup
7. LOA-FC-101(201), Unit 1(2) Fuel Pool Cooling System Abnormal
9. LOA-AR-101(201), Area Radiation Monitoring System Abnormal
10. Technical Specification 3.7.8 Spent Fuel Storage Pool Water Level

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU2

Initiating Condition:

Unexpected rise in plant radiation.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:

   - Refueling Cavity water level **< 340 in.** on shutdown range.

     OR

   - Spent Fuel Pool water level **< 21 ft. 4 in**.

     OR

   - Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal.

   AND

   b. UNPLANNED VALID Area Radiation Monitor reading rise on refuel radiation monitor ARM 0D21-K604A.

   OR

2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU2 (cont)

Basis:

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

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

NORMAL LEVELS: Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Threshold #1 Basis:

During refueling when the RPV head is removed, the Shutdown Range RPV level instrument is recalibrated to indicate water level to the elevation of the refuel floor. With the refueling cavity in communication with the Spent Fuel Pool through the fuel transfer canal, uncontrolled inventory loss can be remotely monitored with this instrument.

The Refueling Cavity includes the fuel transfer canal. Technical Specifications require Reactor Cavity water level be maintained at least 22 ft. above the top of the RPV flange (22 ft. + 819 ft. 8-3/4 in. el.) or 841 ft. 8-3/4 in. el. when irradiated fuel or control rods are being handled within the RPV. This elevation corresponds to 340 in. on the Shutdown Range instrument when calibrated for floodup conditions. During refueling when the RPV head is removed, the Shutdown Range RPV level instrument range is expanded to indicate water level to the elevation of the refuel floor. In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed. Attachment C of LOP-SF-06, Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup, provides a cross-reference of indicated level to plant elevation.

Indication of Spent Fuel Pool water level is provided at Panel 1(2)PM10J on 1(2)LI-FC043. Drops in Spent Fuel Pool water level can normally be detected through visual observation or the existence of the Spent Fuel Pool low level alarm (Fuel Pool Level Low/High).

Spent Fuel Pool water level should be maintained at least 21 ft. 4 in. over the top of the irradiated fuel assemblies seated in the pool racks. This corresponds to 841 ft. 7-7/8 in. el.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU2 (cont)

Basis (cont):

Threshold #2 Basis

Valid elevated area radiation levels usually have long lead times relative to the potential for radiological release beyond the site boundary, thus impact to public health and safety is very low.

This EAL addresses unplanned increases in radiation levels inside the plant. These radiation levels represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant.

Basis Reference(s):

1. NEI 99-01, Rev 4 AU2
2. LRP-5800-3 Radiation Monitoring Alarm/Trip Setpoint Determination
3. RP-AA-203, Exposure Control and Authorization
4. Technical Specification 3.7.8, Spent Fuel Storage Pool Water Level
5. Technical Specification 3.9.6, Reactor Pressure Vessel (RPV) Water Level — Irradiated Fuel
6. Technical Specification 3.9.7, Reactor Pressure Vessel (RPV) Water Level — New Fuel or Control Rods
7. LOP-SF-06 Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup
8. LOA-FC-101(201), Unit 1(2) Fuel Pool Cooling System Abnormal

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA3

Initiating Condition:

Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R1) to maintain plant safety functions:

Table R1 – Areas Requiring Continuous Occupancy
• Main Control Room (1(2)D18-K751A-D) • Central Alarm Station (by survey) • Secondary Alarm Station (by survey) • TSC (if staffed) (Panel 0PLC1J ARM Channel 4-10) • Radwaste Control Room (Panel 0PLC1J ARM Channel 4-5) • Remote Shutdown Panels (1(2)D21-K601F)

OR

2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R2) which will impede necessary access and threaten safe operation of the plant.

Table R2 – Areas Requiring Infrequent Access
• RB Sample (K601G) • Aux Building Containment Purge (K602I) • Reactor Building HCU Modules (K601C, D) • RHR Heat Exchanger Rooms (K602E, F) • RCIC Room (K602G) • HPCS Room (K601H) • SBGT (K602A)

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA3 (cont)

Basis:

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

Threshold #1 Basis:

This EAL addresses increased radiation levels that impede necessary access to operating stations requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. Areas requiring continuous occupancy include the Main Control Room, the central alarm station (CAS), the secondary security alarm station (SAS), the Radwaste Control Room, and the TSC (if staffed). The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations. The Remote Shutdown Panels are also listed in the UFSAR as an area that may require continuous occupancy.

The value of 15 mR/hr is derived from the General Design Criteria (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. A 30 day duration implies an event potentially more significant than an Alert.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EALs may be involved. For example, a dose rate of 15 mR/hr in the Main Control Room may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This EAL is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.).

Threshold #2 Basis:

This EAL addresses increased radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Typically areas requiring infrequent access to maintain plant safety functions include plant VITAL AREAS. Area radiation levels above 2000 mR/hr are indicative of radiation fields that may limit personnel access to equipment, the operation of which may be needed to assure adequate core cooling or shutdown the reactor.

RA3 (cont)

Basis (cont):

The dose rate threshold selected is based on site administrative limits.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved. For example, a dose rate of 2000 mR/hr may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.) or pre-existing radiation areas for which radiological controls already exist. The concern of this threshold is the unanticipated rise in radiation levels that results in unplanned restrictions to areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown.

Basis Reference(s):

1. NEI 99-01, Rev 4 AA3
2. LRP-5800-3, Radiation Monitoring Alarm/Trip Setpoint Determination
3. LIS-AR-105 (205)A-D, Main Control Room Radiation Monitor Channel A Calibration
4. UFSAR Section 3.8
4. UFSAR Section 12.3.2.5

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU3

Initiating Condition:

Fuel clad degradation.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Offgas system isolation due to VALID Offgas post-treatment radiation monitor signal.

   OR

2. Specific coolant activity **> 4.0 uCi/gm** Dose Equivalent I-131.

Basis:

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

Threshold #1 Basis:

During unit operation, the steam jet air ejectors (SJAEs) remove all non-condensable gases from the main condenser including air in-leakage and disassociated products originating in the reactor and exhausts them to the offgas holdup volume. A rise in offgas activity could therefore indicate damage to the fuel cladding, a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

The modifier "VALID" is appropriate because there are several conditions that may cause the monitor to alarm that are not related to fuel clad degradation and therefore should not result in the declaration of an Unusual Event.

Threshold #2 Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding coolant Technical Specifications for iodine spiking. The specific iodine activity ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) accident is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits.

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 Reactor Coolant activity unless another cause is known, e.g., Reactor Coolant System chemical decontamination evolution (during shutdown) is ongoing with resulting high activity levels.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU3 (cont)

Basis Reference(s):

1. NEI 99-01, Rev 4 SU4
2. Technical Specifications 3.4.8
3. LOR 1(2)N62-P600-B501, Off Gas Pre-Treatment Radiation Monitor High-High Radiation
4. LRP-5820-34, Off-Gas Pre and Post Treatment Monitor Alarm and Trip Setpoints
5. LIS-OG-102(202), Steam Jet Air Ejector Off Gas (Pretreatment) Radiation Monitor Calibration
6. LOA-AR-101(201), Area Radiation Monitoring System Abnormal
7. LAP-1800-4, Chemistry Department Improved Technical Specifications, Technical Requirements Manual, TRM Appendixes, Offsite Dose Calculation Manual (ODCM) LaSalle Annex Check Lists

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FG1

Initiating Condition:

Loss of ANY two barriers AND Loss or Potential Loss of the third barrier.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the General Emergency classification level each barrier is weighted equally.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FS1

Initiating Condition:

Loss or Potential Loss of ANY two barriers.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Site Area Emergency classification level, each barrier is weighted equally.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FA1

Initiating Condition:

ANY Loss or ANY Potential Loss of either Fuel Clad or RCS.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Alert classification level, 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FU1

Initiating Condition:

ANY Loss or ANY Potential Loss of Containment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

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 EAL 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC1 – Loss

Initiating Condition:

Primary coolant activity level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.

Basis:

Loss Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

300 uCi/gm Dose Equivalent I-131 is well above that expected for iodine spikes and corresponds, generically, to about 2% to 5% fuel cladding damage. When reactor coolant activity reaches this level, significant clad damage has occurred and thus the Fuel Cladding barrier is considered lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC2 – Loss or Potential Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. RPV level **< – 185 in.** without adequate core spray.

   OR

2. RPV level **< – 210 in**.

POTENTIAL LOSS

RPV level **< – 161 in.** (TAF).

Basis:

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

Loss Basis:

When primary containment flooding is required, all symptom-based EOPs (LGAs) are exited and the LSAMGs are entered in order to restore and maintain cooling to the core and any core debris. Since it may not be possible to recover the core inside the RPV, flooding the primary containment to the elevation of the top of active fuel in the drywell may be required.

When water level is below TAF, adequate core cooling of the uncovered fuel can be provided by one of two methods - by steam cooling from steam created lower in the core or by core spray cooling from the ECCS core spray systems in Design Basis Accident LOCA conditions. Steam cooling from steam created lower in the core is established by maintaining RPV water level above the Minimum Steam Cooling RPV Water level (-185). In LOCA conditions, when design spray flow requirements are satisfied (see LGA-001) and RPV water level is at or above the elevation of the jet pump suctions (-210), the covered portion of the core is cooled by submergence while the uncovered portion is cooled by the spray flow.

Potential Loss Basis:

Core submergence is the preferred method of core cooling and as such, the failure to re-establish RPV level above the top of active fuel for an extended period of time could lead to significant fuel damage.

FC2 – Loss or Potential Loss (cont)

Basis (cont):

An RPV level reading of -161 in. indicates RPV level is at the top of active fuel (TAF). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling).

If core uncovery is threatened, the EOPs (LGAs) specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops below TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. LGA-001, RPV Control
3. LGA-010, Failure to Scram
4. LGA-005, RPV Flooding
5. LPGP-CALC-02, EOP & SAMG Calculation Control -- Setpoints and Calculation Results

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC5 – Loss

Initiating Condition:

Drywell radiation monitoring.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Drywell radiation monitor reading **> Fuel Cladding Loss Threshold**, Table F1.

Table F1 – Drywell Radiation Thresholds

Time After Shutdown (hours)	Fuel Cladding Loss (R/hr)
≤ 2	1.90 E+02
> 2 to 4	1.65 E+02
> 4 to 8	1.40 E+02
> 8 to 16	1.12 E+02
> 16 to 23	9.90 E+01
> 23	9.65 E+01

Basis:

The drywell radiation monitor readings specified in Table F1 provide values that indicate the release of reactor coolant into the drywell with elevated activity indicative of fuel damage (~2%). The values are a function of time after shutdown and were derived using Core Damage Assessment Methodology (CDAM) with 2% clad damage, no drywell sprays in operation and a LOCA depressurized system. The reading is calculated assuming the instantaneous release and dispersal of the above reactor coolant noble gas and iodine inventory into the drywell atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations allowed within Technical Specifications (including iodine spiking) and are therefore indicative of fuel damage (approximately 2% - 5% cladding failure).

During at power (including ATWS) conditions the value listed for the “< 2 hours after shutdown” row is used as an indication of fuel damage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.

Basis:

The Emergency Director judgment fuel cladding loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Fuel Cladding barrier. The inability to monitor fuel cladding integrity should also be considered as a factor in judging that the Fuel Cladding barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC2 – Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

RPV level **< - 161 in.** (TAF).

Basis:

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

Loss Basis:

RPV level reading of -161 in. on FZ indicates RPV level is at the top of active fuel (TAF). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment barriers, and initiation of all ECCS. If RPV level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of any unplanned loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a loss of the RCS barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. LPGP-CALC-02, EOP & SAMG Calculation Control -- Setpoints and Calculation Results
3. LGA-001, RPV Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC3 – Loss

Initiating Condition:

Drywell pressure.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. Drywell pressure **> 1.77 psig**.

   AND

2. Drywell pressure rise due to RCS leakage.

Basis:

The drywell pressure value is the drywell high pressure ECCS initiation setpoint (1.77 psig) and is therefore indicative of a Loss of Coolant Accident (LOCA) event that requires ECCS response. Elevated drywell pressure also causes a reactor scram and is an entry condition to LGA-001, RPV Control, and LGA-003, Primary Containment Control. Normal primary containment pressure control functions (e.g., operation of drywell cooling, Primary Containment Vent and Purge system, etc.) are specified in LGA-003 in advance of less desirable but more effective functions (e.g., operation of drywell or suppression chamber sprays, etc.).

In the LaSalle design basis, primary containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the increasing pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control primary containment vent/purge.

The second threshold focuses the fission product barrier loss threshold on a failure of the RCS instead of the non-LOCA malfunctions that may adversely affect primary containment pressure.

Therefore:

- Drywell pressure greater than 1.77 psig with corollary indications (drywell temperature, humidity) should therefore be considered a loss of RCS.
- Loss of drywell cooling that results in greater than 1.77 psig should not be considered a loss of RCS.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. UFSAR Table 3.3.5.1-1
3. Technical Specifications Table 3.3.5.1-1
4. LGA-001, RPV Control
5. LGA-003, Primary Containment Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC4 – Loss or Potential Loss

Initiating Condition:

RCS leak rate.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. UNISOLABLE Main Steam Line (MSL) break as indicated by the failure of both MSIVs in ANY one line to close.

   AND

2 a. High MSL Flow **AND** High Steam Tunnel Temperature.

   OR

   b. Direct report of steam release.

POTENTIAL LOSS

1. RCS leakage **> 50 gpm** inside the drywell.

   OR

2. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> LGA-002 Maximum Normal** operating levels.

Basis:

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Loss Basis:

High Steam Flow and High Steam Tunnel Temperature Annunciators are both indications of a Main Steam Line Break. Both of these parameters will cause a signal for closure of the MSIVs. Should both valves in any one line fail to isolate, this event would be considered a Loss of the RCS.

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside primary containment exists when flow is not prevented by downstream isolations. In the case of a failure of both Main Steam Isolation Valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required.

Direct report of steam release is meant to provide an alternate means of classification if the Hi Steam Flow Annunciator or the Hi Steam Tunnel Temperature Annunciator fails to operate and the observation of conditions indicates a Main Steam Line Break in the judgment of the Emergency Director. This is not meant to cause a declaration based on leaks such as valve packing leaks where the consequences offsite would be negligible.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC4 – Loss or Potential Loss (cont)

Basis (cont):

Potential Loss Threshold #1 Basis:

The potential loss of RCS based on leakage is set at a level indicative of a small breach of the RCS but which is well within the makeup capability of normal and emergency high-pressure systems. Core uncovery is not a significant concern for a 50 gpm leak; however, break propagation leading to significantly larger loss of inventory is possible. RCS leakage inside the drywell is normally determined by monitoring drywell equipment and floor drain sump pumpout rates. LOP-NB-03, Troubleshooting Drywell Leakage, provides detailed instructions for determining the type and location of leakage in the drywell and is implemented, as needed, to monitor trends whenever drywell leakage is elevated or increasing. The normal method of monitoring leakage may be isolated as part of the drywell isolation, and thus may be unavailable. If primary system leak rate information is unavailable, LOP-NB-03 provides alternate indicators of RCS leakage. Inventory loss events, such as a stuck open SRV, should not be considered when referring to “RCS leakage” because they are not indications of a break, which could propagate.

Potential Loss Threshold #2 Basis:

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. The maximum normal values define this RCS threshold because it is the maximum normal operating value and signifies the onset of abnormal system operation. When parameters reach this level, equipment failure or misoperation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in LGA-002, Secondary Containment Control.

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the Reactor Enclosure since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the Reactor Enclosure, an unexpected rise in Feedwater flowrate, or unexpected Main Turbine Control Valve closure) may indicate that a primary system is discharging into the Reactor Building.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. UFSAR Section 5.4.9
3. Technical Specifications 3.4.5 RCS Operational LEAKAGE
4. UFSAR Section 5.2.5
5. LOP-NB-03, Troubleshooting Drywell Leakage
6. LGA-002, Secondary Containment Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC5 – Loss

Initiating Condition:

Drywell Radiation Monitoring

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Drywell Radiation monitor reading **> 100 R/hr**.

   AND

2. Indications of RCS leakage into the Drywell.

Basis:

The drywell radiation monitor reading is a value that indicates a significant release of reactor coolant to the drywell. A reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the drywell atmosphere. Conservative estimates (high RCS uCi/cc) indicated that the readings from release of the normal RCS inventory would be ~ 100 R/hr. The reading is less than that specified for Fuel Cladding barrier Loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations.

Indication of a RCS leak into the drywell is added to qualify the radiation monitor indication to avoid declaring the loss of RCS barrier for situations where the radiation rise is not due to primary a system leak. For situations that involve failure of the Fuel Clad barrier alone, radiation monitor readings would rise due to shine and potentially giving a false indication of a loss of the RCS barrier. Therefore this EAL contains a qualifier to preclude over classification of the event if only fuel clad barrier failed.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Calc. EP-EAL-0611

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.

Basis:

The Emergency Director judgment RCS loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the RCS barrier. The inability to monitor RCS integrity should also be considered as a factor in judging that the RCS barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT2 – Potential Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

POTENTIAL LOSS

Plant conditions indicate that Primary Containment Flooding is required.

Basis:

Potential Loss Basis:

When primary containment flooding is required, all EOPs (LGAs) are exited and the SAMGs are entered in order to restore and maintain cooling to the core and any core debris. Since it may not be possible to recover the core inside the RPV, flooding the primary containment to the elevation of the top of active fuel in the drywell may be required.

The EOP conditions requiring primary containment flooding represent imminent core melt sequences that, if not corrected, could lead to RPV failure and increased potential for containment failure.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. LGA-001, RPV Control
3. LGA-010, Failure to Scram
4. LGA-005, RPV Flooding

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT3 – Loss or Potential Loss

Initiating Condition:

Drywell pressure.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. Rapid unexplained drop in Drywell pressure following initial pressure rise.

   OR

2. Drywell pressure response not consistent with LOCA conditions.

POTENTIAL LOSS

1. Drywell pressure **≥ 45 psig** and rising.

   OR

2. a. Drywell or suppression chamber hydrogen concentration **≥ 6%.**

      AND

   b. Drywell or suppression chamber oxygen concentration **≥ 5%.**

Basis:

Loss Threshold #1Basis:

Rapid unexplained loss of pressure (i.e., not attributable to drywell sprays, suppression chamber sprays or condensation effects) following an initial pressure rise indicates a loss of containment integrity.

Loss Threshold #2 Basis:

Drywell pressure should rise as a result of mass and energy release into the containment from a LOCA. Thus, drywell pressure response not consistent with LOCA conditions indicates a loss of containment integrity. This indicator relies on operator recognition of an unexpected response for the condition and therefore does not include a specific pressure value or trend. Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. The unexpected response is important because it is the indicator for a containment bypass condition.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT3 – Loss or Potential Loss (cont)

Basis (cont):

Potential Loss Threshold #1 Basis:

When the Primary Containment design pressure is challenged, primary containment venting is required even if offsite radioactivity release rate limits will be exceeded. This condition, if compounded by further plant degradation may challenge primary containment integrity and is, therefore, an appropriate threshold for potential loss of the Primary Containment barrier.

A Drywell pressure of 45 psig is based on the containment/drywell design pressure. If the containment design pressure is exceeded this represents a challenge to the containment structure because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a potential loss of the containment barrier even if a breach has NOT occurred.

Potential Loss Threshold #2 Basis:

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

Except for brief periods during plant startup and shutdown, oxygen concentration in the primary containment is maintained at insignificant levels by nitrogen inertion. The specified values for this potential loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen concentration) and readily recognizable because 6% hydrogen is above the hydrogen monitor alarm setpoint (2%) and the Primary Containment Control EOP entry condition. The minimum global deflagration hydrogen/oxygen concentrations (6% and 5%, respectively) require intentional primary containment venting, which is defined to be a barrier loss under Primary Containment barrier CT6.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. UFSAR 15.6.5
3. UFSAR Section 6.2
4. LGA-003, Primary Containment Control
5. LGA-011, Hydrogen Control
6. LaSalle PSTG Section 5B, Hydrogen Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT5 – Potential Loss

Initiating Condition:

Significant radioactive inventory in Containment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

POTENTIAL LOSS

Drywell radiation monitor reading **> Containment Potential Loss Threshold**, Table F2

Table F2 – Drywell Radiation Thresholds	
Time After Shutdown (hours)	**Containment Potential Loss (R/hr)**
≤ 2	4.35 E+02
> 2 to 4	3.75 E+02
> 4 to 8	3.15 E+02
> 8 to 16	2.60 E+02
> 16 to 23	2.30 E+02
> 23	2.25 E+02

Basis:

The drywell radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the Fuel Cladding barrier. NUREG-1228 “Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents” states that such readings do not exist when the amount of cladding damage is less than 20%. The values are a function of time after shutdown and were derived using Core Damage Assessment Methodology (CDAM) assuming 20% clad damage, no drywell sprays in operation and a LOCA depressurized system. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a significant failure into the reactor coolant has occurred.

During at power (including ATWS) conditions the value listed for the “< 2 hours after shutdown” row is used as an indication of fuel damage.

Regardless of whether the Primary Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Primary Containment barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT6 – Loss

Initiating Condition:

Containment isolation failure or bypass.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

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

   AND

   b. A downstream pathway to the environment exists.

   OR

2. Intentional venting/purging of Primary Containment per EOPs or SAMGs due to accident conditions.

   OR

3. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> LGA-002, Maximum Safe** operating levels.

Basis:

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Threshold #1 Basis:

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway to the environment. The concern is the unisolable open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of primary containment integrity.

Failure of containment isolation valves to isolate with a downstream pathway to the environment is only a concern during an accident. If this condition exists during normal Power Operation, a Technical Specification Action Statement will address it. However, during accident conditions, this will represent a breach of Primary Containment.

Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include unisolable Main, RCIC steamline breaks, unisolable RWCU system breaks, and unisolable containment atmosphere vent paths. Minor release paths such as instrument and sample lines are not considered under this threshold.

Examples of “downstream pathway to the Environment” could be through Turbine/Condenser, or direct release to the Turbine Enclosure or Reactor Enclosure.

CT6 – Loss (cont)

Basis (cont):

The breach is NOT isolable from the Control Room if an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation from the Control Room should be made prior to the accident classification. If Operator actions from the Control Room are successful, then this IC is not applicable. Credit is NOT given for Operator actions taken in-plant (outside the Control Room) to isolate the leak.

This EAL is intended to cover containment isolation failures allowing a direct flow path to the environment such as failure of both MSIVs to close with open valves downstream to the turbine or to the condenser, even if these systems are not breached.

Threshold #2 Basis:

Intentional venting of the primary containment to the secondary containment and/or the environment per the EOPs/SAMG due to accident conditions is considered a loss of the Primary Containment barrier.

Threshold #3 Basis:

The presence of elevated general area temperatures and/or area radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. Temperatures and radiation levels beyond their maximum safe operating temperatures are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This EAL threshold addresses problematic discharges outside primary containment that may not originate from a high-energy line break.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. LGA-002, Secondary Containment Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis:

The Emergency Director judgment Containment loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Containment barrier. The inability to monitor Containment parameters should also be considered as a factor in judging that the Containment barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG1

Initiating Condition:

Prolonged loss of all offsite power and prolonged loss of all onsite AC power to Division 1 and Division 2 essential busses.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of power to System Auxiliary Transformer 142(242) and Unit Auxiliary Transformer 141(241).

   AND

2. Failure of DG 0 and DG 1A(2A) emergency diesel generators to supply power to unit ECCS busses.

   AND

3. a. Restoration of either unit ECCS bus (excluding Division 3) within 4 hours is **not** likely.

      OR

   b. RPV level **cannot** be determined to be **> - 150 in.** on WR at RSP.

Basis:

Loss of all AC power to ECCS busses compromises the availability of all plant safety systems. Prolonged loss of all AC power may lead to loss of Fuel Cladding, RCS and Primary Containment barriers. The four-hour interval to restore AC power to either unit ECCS bus is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout.”

The likelihood of restoring at least one ECCS bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

MG1 (cont)

Basis (cont):

Division 1 and Division 2 ECCS busses are provided with four independent sources of AC power:

1. A normal offsite source from the 345-kV system through the system auxiliary transformer 142(242) directly to busses 141Y(241Y) and 142Y(242Y).
2. An emergency offsite source from the 345-kV system through the system auxiliary transformer of the opposite unit to busses 141Y(241Y) and 142Y(242Y) via unit tie breakers ACB 1414 and ACB 2414 with bus 241Y(141Y) and ACB 1424 and ACB 2424 with bus 242Y(142Y).
3. A reserve onsite source, available during unit operation, from the unit through the unit auxiliary transformer 141(241) to busses 141Y(241Y) and 142Y(242Y) via bus tie breakers ACB 1415(ACB 2415) and ACB 1425(ACB 2425) with busses 141X(241X) and 142X(242X) respectively.
4. A standby onsite source which is provided from the onsite diesel generators: DG 0 to bus 141Y or 241Y and DG 1A(2A) to bus 142Y(242Y) or a power source lineup per LOA-AP-101(201) which DG1A to power bus 242Y and DG2A to power bus 142Y.

In addition to the above four independent sources, a fifth source from offsite is available by virtue of removable links in the main generator isolated phase bus. When removed, a unit auxiliary transformer 141(241) can be backfed from the 345-kV system through the main power transformer. This source is similar to items 2 and 3 discussed above; however, it is available only when the unit is shut down and the generator disconnected (due to the time required to effect the backfeed, this source is likely only to be available when previously configured).

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG1 (cont)

Basis (cont):

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

A reading of –150 in. below instrument zero indicates RPV level is near the top of active fuel (TAF). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops below TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

In blackout conditions, the RPV level can only be read at the RSP.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SG1
2. UFSAR 8.1
3. LOA-AP-101(201), Unit 1(2) AC Power System Abnormal
4. UFSAR 15.9
5. LPGP-CALC-02, EOP & SAMG Calculation Control -- Setpoints and Calculation Results
6. LGA-001, RPV Control
7. LGA-010, Failure to Scram

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS1

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to Division 1 and Division 2 essential busses.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of power to System Auxiliary Transformer 142(242) and Unit Auxiliary Transformer 141(241).

   AND

2. Failure of DG 0 and DG 1A(2A) emergency diesel generators to supply power to unit ECCS busses.

   AND

3. Failure to restore power to at least one Unit ECCS bus (excluding Division 3) within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power compromises all plant safety systems and represents failures of plant functions required for the protection of the public. Division 1 and Division 2 ECCS busses are provided with four independent sources of AC power:

1. A normal offsite source from the 345-kV system through the system auxiliary transformer 142(242) directly to busses 141Y(241Y) and 142Y(242Y).
2. An emergency offsite source from the 345-kV system through the system auxiliary transformer of the opposite unit to busses 141Y(241Y) and 142Y(242Y) via unit tie breakers ACB 1414 and ACB 2414 with bus 241Y(141Y) and ACB 1424 and ACB 2424 with bus 242Y(142Y).
3. A reserve onsite source, available during unit operation, from the unit through the unit auxiliary transformer 141(241) to busses 141Y(241Y) and 142Y(242Y) via bus tie breakers ACB 1415(ACB 2415) and ACB 1425(ACB 2425) with busses 141X(241X) and 142X(242X) respectively.
4. A standby onsite source which is provided from the onsite diesel generators: DG 0 to bus 141Y or 241Y and DG 1A(2A) to bus 142Y(242Y) or a power source lineup per LOA-AP-101(201) which DG1A to power bus 242Y and DG2A to power bus 142Y.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS1 (cont)

Basis (cont):

In addition to the above four independent sources, a fifth source from offsite is available by virtue of removable links in the main generator isolated phase bus. When removed, a unit auxiliary transformer 141(241) can be backfed from the 345-kV system through the main power transformer. This source is similar to items 2 and 3 discussed above; however, it is available only when the unit is shut down and the generator disconnected. (Due to the time required to effect the backfeed, this source is likely only to be available when previously configured.)

Consideration should be given to available loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not available on the energized bus, the bus should not be considered available.

The fifteen-minute interval begins from the time of loss of both onsite and offsite AC power and was selected as a threshold to exclude transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS1
2. UFSAR 8.1.
3. LOA-AP-101(201), Unit 1(2) AC Power System Abnormal

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA1

Initiating Condition:

AC power capability to Division 1 and Division 2 essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. AC power capability to unit ECCS busses (excluding Division 3) reduced to only one of the following power sources for **> 15 minutes**:
   - System Auxiliary Transformer 142(242)
   - Unit Auxiliary Transformer 141(241)
   - Unit Emergency Diesel Generator 1A(2A)
   - Shared Emergency Diesel Generator DG 0
   - Other unit SAT via crosstie breakers

   AND
2. Any additional single power source failure will result in unit blackout.

Basis:

Capability: (pertaining to electrical power supplies) is equipment that is available to provide and maintain AC power at the required voltage and frequency for the required load.

The reduction of available reliable power sources to a condition in which any additional single failure will result in a Unit Blackout is a substantial degradation in the level of safety of the plant. A Unit Blackout is a loss of AC power to 141Y(241Y) and 142Y(242Y) busses. LaSalle blackout coping duration is four hours.

The listed power supplies take into account sources that, if unavailable, establish single-failure vulnerability. This EAL allows for the use of the unit crosstie breakers if they are the only source of power to the affected unit. The Emergency Director must consider the use of the crosstie breakers and the consequent demand on the unaffected unit.

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA1 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA5
2. UFSAR 8.1
3. LOA-AP-101(201) Unit 1(2) AC Power System Abnormal
4. UFSAR 15.9

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU1

Initiating Condition:

Loss of all offsite power to Division 1 and Division 2 essential busses for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3, 4, 5

EAL Threshold Values:

Loss of power to System Auxiliary Transformer 142(242) **AND** Unit Auxiliary Transformer 141(241) for **> 15 minutes**.

Basis:

1. The Essential busses are the safety-related, 4160-VAC Division 1 ECCS bus 141Y(241Y), Division 2 ECCS bus 142Y(242Y). Division 1 and Division 2 ECCS busses are provided with offsite power from the 345-kV system through the system auxiliary transformer 142(242) directly to busses 141Y(241Y) and 142Y(242Y).

There is also a standby onsite source which is provided from the onsite diesel generators: DG 0 to bus 141Y or 241Y and DG 1A(2A) to bus 142Y(242Y).

Loss of all offsite power causes a reactor scram and primary containment isolation. Emergency Diesel Generators DG 0 and DG 1A(2A) should automatically start and be available to carry the essential loads for each affected unit. Balance of plant systems that would assist in plant operations (e.g., condensate pumps, etc.) may be unavailable due to the loss of power.

A loss of offsite AC power reduces the 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.

The intent of this EAL is to declare an Unusual Event when offsite power has been lost and the emergency diesel generators have successfully started and energized their respective ESF busses. The fifteen-minute interval was selected as a threshold to exclude transient power losses

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU1 & CU3
2. UFSAR 8.1
3. LOA-AP-101(201), Unit 1(2) AC Power System Abnormal

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RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA2

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to Division 1 and Division 2 essential busses.

Operating Mode Applicability:

4, 5, D

EAL Threshold Values:

1. Loss of power to System Auxiliary Transformer 142(242) and Unit Auxiliary Transformer 141(241).

   AND

2. Failure of DG 0 and DG 1A(2A) emergency diesel generators to supply power to unit ECCS busses.

   AND

3. Failure to restore power to at least one unit ECCS bus (excluding Division 3) **within 15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power when in Cold Shutdown, Refueling or Defueled modes compromises safety systems required for decay heat removal and represents a substantial degradation of the level of safety of the plant. An Alert declaration (instead of a Site Area Emergency under EAL MS1) is appropriate in these modes because post-shutdown, decay heat energy levels offer more time to restore AC power to heat removal systems than the levels present when the reactor is in Power Operation, Startup or Hot Shutdown mode. Thus, the threat to the protection of the health and safety of the public is less severe.

Division 1 and Division 2 ECCS busses are provided with three independent sources of AC power while the unit is shut down:

1. A normal offsite source from the 345-kV system through the system auxiliary transformer 142(242) directly to busses 141Y(241Y) and 142Y(242Y).
2. An emergency offsite source from the 345-kV system through the system auxiliary transformer of the opposite unit to busses 141Y(241Y) and 142Y(242Y) via unit tie breakers ACB 1414 and ACB 2414 with bus 241Y(141Y) and ACB 1424 and ACB 2424 with bus 242Y(142Y).
3. A standby onsite source which is provided from the onsite diesel generators: DG 0 to bus 141Y or 241Y and DG 1A(2A) to bus 142Y(242Y) or Power source lineup to allow DG1A to power bus 242Y and DG2A to power bus 142Y per LOA-AP-101(201).

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA2 (cont)

Basis (cont):

In addition to the above three independent sources, a fourth source from offsite is available by virtue of removable links in the main generator isolated phase bus. When removed, a unit auxiliary transformer 141(241) can be backfed from the 345-kV system through the main power transformer. This source is similar to items 2 and 3 discussed above (due to the time required to effect the backfeed, this source is likely only to be available when previously configured).

Consideration should be given to available loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not available on the energized bus, the bus should not be considered available.

The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA3
2. UFSAR 8.3
3. LOA-AP-101(201), Unit 1(2) AC Power System Abnormal

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG3

Initiating Condition:

Failure of the Reactor Protection System to complete an automatic scram and manual scram was **NOT** successful and there is indication of an extreme challenge to the ability to cool the core.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by reactor power **> 3% APRM.**

   AND

2. a. RPV level cannot be restored and maintained **> -150 in.** on WR (-185 in. on FZ if WR not available).

      OR

   b. Heat Capacity Limit (LGA-003 Fig. H) exceeded.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed The reactor power threshold (3%) is approximately equal to the APRM downscale setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, Suppression Pool temperature trend) can be used to determine if reactor power is greater than 3% power. Classification at the General Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG3 (cont)

Basis (cont):

The second condition of this EAL indicates either:

- An extreme challenge to the ability to cool the core as indicated when RPV level cannot be held above -150 in. on WR (-185in. on FZ if WR instrument is not available). The specified water levels are the Minimum Steam Cooling RPV Water Level (MSCRWL). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core cooling could be a precursor of a core melt sequence.

  RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

- An extreme challenge to the primary containment as indicated when heat cannot be removed from the primary containment resulting in elevated suppression pool temperature. The Heat Capacity Limit is the highest suppression pool temperature from which a blowdown will not raise drywell pressure above 60 psig before the rate of energy transfer from the RPV to the primary containment is within the capacity of the primary containment vent. (Before drywell pressure reaches 60 psig, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded.) The Heat Capacity Limit is a function of RPV pressure and suppression pool temperature and level and is a measure of the maximum heat load that the primary containment can withstand. Plant parameters in excess of the Heat Capacity Limit could be a precursor of primary containment failure. The Heat Capacity Limit is given in Fig. H of LGA-003, Primary Containment Control.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SG2
2. LGA-001, RPV Control
3. LGA-010, Failure to Scram
4. LGA-003, Primary Containment Control

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS3

Initiating Condition:

Failure of Reactor Protection System Instrumentation to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded and manual scram was **NOT** successful.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by reactor power **> 3% APRM**.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed The reactor power threshold (3%) is the APRM downscale trip setpoint and is approximately equal to the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, Suppression Pool temperature trend) can be used to determine if reactor power is greater than 3% power.

Classification at the Site Area Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS2
2. LGA-001, RPV Control
3. LGA-010, Failure to Scram
4. LGA-003, Primary Containment Control

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA3

Initiating Condition:

Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. A Reactor Protection System setpoint was exceeded.

   AND

2. Automatic scram did not reduce reactor power to **< 40 on IRM Range 7**.

Basis:

This condition indicates a failure of the automatic reactor protection system to successfully scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. Site-specific indication of reactor shutdown is included as the criteria of whether the scram was successful when required. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue.

A successful scram has occurred when there is sufficient rod insertion to bring the reactor subcritical (< on 40 IRM Range 7).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function.

The second condition of this EAL indicates a failure of the automatic RPS scram function to rapidly insert a sufficient number of control rods to achieve reactor shutdown. The CRD system backup scram valves and the Alternate Rod Insertion (ARI) system provide automatic, alternate methods of completing the scram function. These backups, however, insert control rods at a much slower rate than the automatic RPS scram function. For the purpose of emergency classification at the Alert level, reactor shutdown achieved by automatic backup scram valve operation and ARI initiation does not constitute a successful RPS automatic scram.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA3 (cont)

Basis (cont):

Following any automatic RPS scram signal LGA-001, RPV Control and/or LGA-010, Failure to Scram, prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Alert.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded, or first-out annunciators), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA2
2. LGA-001, RPV Control
3. LGA-010, Failure to Scram
4. Technical Specifications Table 3.3.1.1-1

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU3

Initiating Condition:

Inadvertent criticality.

Operating Mode Applicability:

3, 4, 5

EAL Threshold Values:

An UNPLANNED extended positive period observed on nuclear instrumentation.

Basis:

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

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

This EAL includes criticality events that occur in Cold Shutdown or Refueling modes (NUREG1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events as well as inadvertent criticalities occurring in Hot Shutdown mode. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification.

This condition can be identified using:

- SRM Channel A-D period meters 1(2)C51-R601A-D and 1(2)C51-K600A-D
- Amber short period lights
- Annunciator (A306) "SRM SHORT PERIOD" on panel 1(2)H13-P603.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU8 & CU8
2. LIS-NR-301(401), Unit 1(2) Source Range Monitor Rod Block Functional Test
3. LGP-1(2)-1, Normal Unit Startup
4. LGP-1(2)-S1, Master Startup Checklist

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS4

Initiating Condition:

Loss of all vital DC power.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Loss of all vital DC power based on **< 108 VDC** on 125 VDC battery busses 111Y(211Y) and 112Y(212Y) for **> 15 minutes**.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of Containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. Each unit has two vital 125 VDC batteries. Each 125 VDC battery is sized to supply control power requirements of the switchgear and logic circuitry of one of the three safety related divisions and are, therefore, the batteries of interest for this EAL.

125 VDC busses 111Y and 112Y (for Unit 1) are mutually redundant for Unit 1. Similarly, busses 211Y and 212Y are mutually redundant for Unit 2. This design allows for the single failure or loss of one redundant DC bus on each unit during simultaneous accident and loss-of-offsite-power conditions without adversely affecting the safe shutdown capability of the plant. Bus ties are provided so that the nonredundant DC busses of Unit 1 and Unit 2 can be interconnected during maintenance and testing operations for the battery and/or battery charger associated with either bus 111Y or 211Y and bus 112Y or 212Y.

The ampere-hour capacity of each battery is adequate to supply expected essential loads following station trip and loss of all AC power without battery terminal voltage falling below 105 VDC terminal voltage, the minimum discharge level.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS3
2. UFSAR 8.3.2
3. LOA-DC-101(201) Unit 1(2) DC Power System Failure

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU4

Initiating Condition:

UNPLANNED loss of required DC power for greater than 15 minutes.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. UNPLANNED loss of all required vital DC power based on **< 108 VDC** indication on 125 VDC battery busses 111Y(211Y) and 112Y(212Y).

   AND

2. Failure to restore power to at least one required DC bus **within 15 minutes** from the time of loss.

Basis:

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

“Unplanned activities” is included in this EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, plants perform maintenance on a bus-related basis during shutdown periods.

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown, refueling or defueled operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. Each unit has two vital 125 VDC batteries. Each 125 VDC battery is sized to supply control power requirements of the switchgear and logic circuitry of one of the two safety related divisions and are, therefore, the batteries of interest for this EAL.

125 VDC busses 111Y and 112Y (for Unit 1) are mutually redundant for Unit 1. Similarly, busses 211Y and 212Y are mutually redundant for Unit 2. This design allows for the single failure or loss of one redundant DC bus on each unit during simultaneous accident and loss-of-offsite-power conditions without adversely affecting the safe shutdown capability of the plant. Bus ties are provided so that the nonredundant DC busses of Unit 1 and Unit 2 can be interconnected during maintenance and testing operations for the battery and/or battery charger associated with either bus 111Y or 211Y and bus 112Y or 212Y.

The ampere-hour capacity of each battery is adequate to supply expected essential loads following station trip and loss of all AC power without battery terminal voltage falling below 105 VDC terminal voltage, the minimum discharge level.

MU4 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU7
2. UFSAR 8.3.2
3. LOA-DC-101(201), Unit 1(2) DC Power System Failure

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS5

Initiating Condition:

Complete loss of heat removal capability.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Heat Capacity Limit Curve (LGA-003) exceeded.

Basis:

Plant parameters associated with the Heat Capacity Temperature Limit (HCTL Curve - LGA-003) are RPV pressure, suppression pool water level and suppression pool temperature. The HCTL is the highest suppression pool temperature from which a blowdown will not raise drywell pressure above 60 psig before the rate of energy transfer from the RPV to the primary containment is within the capacity of the primary containment vent. Before drywell pressure reaches 60 psig, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded. The HCTL is a function of RPV pressure and suppression pool temperature and is a measure of the maximum heat load which the primary containment can withstand. If LGA actions to control suppression chamber bulk suppression pool temperature and RPV pressure below the HCTL are unsuccessful, RPV blowdown is required.

Heat up of the suppression pool to the HCTL signals the loss of functions required to maintain hot shutdown, including the ultimate heat sink. It also infers an RPV blowdown that could be caused by a loss of the RCS barrier. If compounded by further plant degradation, the event may challenge primary containment integrity.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS4
2. LGA-003, Primary Containment Control

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA5

Initiating Condition:

Inability to maintain plant in cold shutdown with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F** for > Table M1 duration.

Table M1 – RCS Reheat Duration Thresholds

RCS	Secondary Containment Closure	Duration
Intact	N/A	60 minutes*
Not Intact	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. UNPLANNED RPV pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.

Basis:

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

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

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 main steam line nozzle plugs, etc.).

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as decay heat removal system design and RPV level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Basis (cont):

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F), such as:

- Reactor Pressure Vessel Metal Temperature Recorder on panel 1(2)H22-P007
- Reactor vessel bottom head outside surface metal temperature as recorded on Channel 007 (Tag "RVBH") of 1(2)C11-R018
- Bottom Head (RVBH) Temp 1(2)H22-P007 Ch 7
- Recirculation Pump Suction Temperature Recorder 1(2)B33-R650 on Control Room Panel 1(2)H13-P602
- Reactor Coolant Temperature Computer Point B741 (Cleanup System Inlet Temp)
- Process Computer Video Display Window 47

Threshold #1 Basis:

The first condition in Table M1 addresses complete loss of functions required for core cooling for greater than sixty minutes during Refueling and Cold Shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.). With secondary containment established, a low-pressure barrier to fission product release exists. In this condition, containment status is of less importance than the status of RCS integrity because the RCS is intact and providing a high-pressure barrier to fission product release. The sixty-minute interval should allow sufficient time to restore cooling without a substantial degradation in plant safety. The asterisk highlights the note at the bottom of the table. The note indicates that the first condition is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the sixty-minute interval.

The second condition in Table M1 addresses the complete loss of functions required for core cooling for greater than twenty minutes during Refueling and Cold Shutdown modes when secondary containment is established but RCS integrity is not established or RPV inventory is reduced. RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.).

The allowed twenty-minute interval is 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 barrier to fission product release is established (i.e., secondary containment closure). The asterisk highlights the note at the bottom of the table. The note indicates that the second condition is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the twenty-minute interval.

MA5 (cont)

Basis (cont):

The third condition in Table M1 addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when secondary containment, and RCS integrity are not established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.). No delay time is allowed for this condition because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary unplanned excursion above 200° F when the heat removal function is available.

Threshold #2 Basis:

The 10 psig pressure rise due to loss of decay heat removal infers an intact RCS with uncontrolled RPV temperature rise in excess of the Technical Specification cold shutdown limit (200°F) for which MA5 Threshold #1 would permit up to sixty minutes to restore RCS cooling before declaration of an Alert. This EAL therefore covers situations in which it is determined that, due to high decay heat loads, the time provided to reestablish temperature control should be less than sixty minutes.

NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

RPV pressure is indicated by 1(2)C34-R605, 1(2)B21-R664 or 1(2)B21-R884A or B.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA4
2. Technical Specifications 3.6.1.1
3. Technical Specifications 3.6.4.1
4. OU-AA-103, Shutdown Safety
5. OU-LA-104, Shutdown Safety Management Program
6. LGP-1-S1, Master Startup Checklist
7. LGP-1-1, Normal Unit Startup
8. LOR 1(2)H13-P601-C204, RHR Shutdown Cooling Line High Temperature

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU5

Initiating Condition:

UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 200° F**.

   OR

2. Loss of all RCS temperature AND RPV level indication for **> 15 minutes**.

Basis:

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

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, 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. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the fuel cladding may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. 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 Reactor Vessel level so that escalation to the Alert under EAL MA5 will occur if required.

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in RCS/Reactor Vessel temperatures depending on the time since shutdown.

Basis (cont):

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 Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling modes, the second condition of this EAL would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication.

Reactor Vessel water level is normally monitored using the following instruments:

- Shutdown Range (0 to +400 in.)
- Upset Range (0 to +180 in.)
- Narrow Range (0 to +60 in.)
- Wide Range (-150 to +60 in.)
- Fuel Zone (-311 to -111 in.)

Detail I of LGA-001, RPV Control, indicates when an instrument may be used for RPV level indication when EOPs are entered.

During shutdown conditions, the Shutdown Range and Upset Range are the primary instruments for monitoring RPV level as the RPV is flooded in preparation for vessel head removal and refueling operations. Plant procedures (e.g., LOP-SF-06, Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup, etc.) provide alternate level monitoring capabilities when the normal level instrumentation is unavailable for the desired level range or the head vent piping is removed. In order to expand the indicating range of the Shutdown and Upset Range, temporary reference leg signals (normal or floodup) are applied. The signal simulates one of two predetermined levels, based on the desired water level in the RPV or Refueling Cavity. These two elevations are:

- Normal – Reference leg at 831 ft. 6-5/8 in. el.– This is the elevation of the normal fill for the reference legs and provides a direct readout of water level from instrument zero in the RPV to above the vessel flange.
- Floodup – Reference leg at 845 ft. 7 in. el. – This provides level monitoring in the range from instrument zero in the RPV to the level of the refuel floor.

In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed. Attachments to LOP-SF-06 provide a cross-reference of indicated level to plant elevation.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU5 (cont)

Basis (cont):

Several instruments and computer points are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200° F), such as:

- Reactor Pressure Vessel Metal Temperature Recorder on panel 1(2)H22-P007
- Reactor vessel bottom head outside surface metal temperature as recorded on Channel 007 (Tag "RVBH") of 1(2)C11-R018
- Bottom Head (RVBH) Temp 1(2)H22-P007 Ch 7
- Recirculation Pump Suction Temperature Recorder 1(2)B33-R650 on Control Room Panel 1(2)H13-P602
- Reactor Coolant Temperature Computer Point B741 (Cleanup System Inlet Temp)
- Process Computer Video Display Window 47

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU4
2. Technical Specifications Table 1.1-1
3. LGP-1-S1, Master Startup Checklist
4. LGP-1-1, Normal Unit Startup
5. LGA-001, RPV Control
6. LPGP-PSTG-01S03 Plant Specific Technical Guidelines Section 3 – Cautions
7. LOP-SF-06, Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup
8. LOR 1(2)H13-P601-C204, RHR Shutdown Cooling Line High Temperature

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS6

Initiating Condition:

Inability to monitor a SIGNIFICANT TRANSIENT in progress.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of most **(approximately 75%)** safety system annunciators (Table M2).

Table M2 – Control Room Panels
• 1(2)H13-P601 • 1(2)H13-P603 • 1(2)PM01J

AND

2. Indications needed to monitor safety functions (Table M3) are unavailable.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

AND

3. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Turbine trip • Reactor scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10 %** Reactor Power change

AND

4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS6 (cont)

Basis:

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations > 10% Reactor Power change.

Planned and unplanned actions are not differentiated since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not a factor.

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in a 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.

A Site Area Emergency exists if the Control Room staff cannot monitor safety functions needed for protection of the public. 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 parameters should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled and in a coolable geometry, remove heat from the core, and maintain the reactor coolant system and containment intact. These parameters are monitored and controlled in the symptom-based emergency operating procedures (LGAs).

Symptoms of a loss of annunciators can be:

- Loss of Division 1 or 2 annunciator power
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

LOA-AN-101(201), Loss of Annunciators, provides instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS6 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS6
2. LGA-001, RPV Control
2. LGA-003, Primary Containment Control
4. LOA-AN-101(201), Loss of Annunciators
5. LEP-AN-01, Annunciator Troubleshooting and Testing
6. LOP-CX-01, On Demand Functions Of The Plant Process Computer
7. LOP-CX-02, Safety Parameter Display System (SPDS)

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 1(2)H13-P601 • 1(2)H13-P603 • 1(2)PM01J

OR

b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

AND

2. a. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Turbine trip • Reactor scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10 %** Reactor Power change

OR

b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

Basis:

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

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations > 10% Reactor Power change.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

For this EAL "most" is approximately 75% of the safety system annunciators or indicators being lost, resulting in a 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the Reactor Control, ECCS, and Electrical panel annunciation or indication equipment. The availability of computer based indication equipment is considered.

Symptoms of a loss of annunciators can be:

- Loss of Division 1 or 2 annunciator power
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

LOA-AN-101(201), Loss of Annunciators, provides instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The fifteen-minute interval offers time to recover from transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA4
2. LOA-AN-101(201), Loss of Annunciators
3. LEP-AN-01, Annunciator Troubleshooting and Testing
4. LOP-CX-01, On Demand Functions Of The Plant Process Computer
5. LOP-CX-02, Safety Parameter Display System (SPDS)

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 1(2)H13-P601 • 1(2)H13-P603 • 1(2)PM01J

OR

2. UNPLANNED loss of most **(approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

Basis:

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

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in a 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the Reactor Control, ECCS, and Electrical panel annunciation or indication equipment. The availability of computer based indication equipment is considered.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU6 (cont)

Basis (cont):

Symptoms of a loss of annunciators can be:

- Loss of Division 1 or 2 annunciator power
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

LOA-AN-101(201), Loss of Annunciators, provides instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The fifteen-minute interval offers time to recover from transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU3
2. LOA-AN-101(201), Loss of Annunciators
3. LEP-AN-01, Annunciator Troubleshooting and Testing
4. LOP-CX-01, On Demand Functions Of The Plant Process Computer
5. LOP-CX-02, Safety Parameter Display System (SPDS)

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU7

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

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

   OR

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

Basis:

The conditions of this EAL threshold may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. Only leakage inside the drywell qualifies toward exceeding Threshold #1. Various indications may be used to identify or verify potential leakage from the RCS into the drywell. They include drywell sump flow indications, drywell temperature and pressure changes, drywell air cooler cooling water differential temperature changes, and drywell atmosphere activity level changes. LOP-NB-03, Troubleshooting Drywell Leakage, provides direction for determining RCS leakage.

The 10 gpm value for unidentified leakage was selected because it is observable with normal Control Room measurement of sump pumpout rates. It is consistent with the Technical Specification threshold for leaks beyond which increased risk of crack propagation exists.

The 25 gpm value for identified leakage is set at a higher value because of the significance of identified leakage in comparison to unidentified or pressure boundary leakage.

No classification under this threshold is made for relief valve operation or leakage.

Both threshold values are observable on Control Room instrumentation and do not require a mass balance calculation.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU5
2. Technical Specifications 3.4. 5
3. UFSAR 5.2.5
4. LOP-NB-03, Troubleshooting Drywell Leakage
6. LGA-001, RPV Control

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG8

Initiating Condition:

Loss of RCS/RPV inventory affecting fuel clad Integrity with containment challenged with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Suppression Pool level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

AND

2. a. RPV level **< -161 in**. (TAF) on FZ for **> 30 minutes**.

   OR

   b. RPV level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following:

   - Refuel floor Rad monitor 0D21-K604A indicates **> 3000 mR/hr** or off-scale high.
   - Erratic Source Range Monitor Indication.

AND

3. Containment is challenged as indicated by one or more of the following:
   - Primary containment Hydrogen concentration **≥ 6%** and Oxygen concentration **≥ 5%.**
   - Drywell pressure **≥ 45 psig**.
   - Primary and Secondary CONTAINMENT CLOSURE not established.
   - Any Secondary Containment radiation monitor **> LGA-002 Maximum Safe** operating level.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

Basis (cont):

Threshold #1 and #2 Basis:

This EAL represents the inability to restore and maintain RPV level to above the top of active fuel, -161 in. on FZ. Fuel damage is probable if core uncovery is prolonged and submergence cannot be restored and maintained. Available decay heat will cause boiling and further drop RPV level.

This EAL is 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, (e.g., decay heat removal system design, etc.) can have a significant affect on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

When RPV level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated drywell radiation or unexplained rise in drywell floor or equipment drain sump pumpout rate. LGA-001, RPV Control, provides guidance on determining if RPV level can be monitored. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors (SRM Channels A-D) can be used as a tool for making such determinations.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The Fuel Handling Area Radiation Monitors 1(2) 1743 - A or B indication of > 3000 mR/hr. is based on calculation EP-EAL-0501.

Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

Threshold #3 Basis:

Four conditions are associated with the challenge to containment integrity:

- When hydrogen and oxygen concentrations in primary containment reach or exceed the deflagration limits, imminent loss of the Primary Containment barrier exists. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred.
- The primary containment design pressure (45 psig) is well in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MG8 (cont)

Basis (cont):

- Primary Containment Closure and Secondary Containment Closure provides a barrier to the release of radioactivity to the environment. When this barrier is not established with prolonged core uncovery, the health and safety of the public may be threatened.
- The secondary containment area radiation level is the LGA Maximum Safe Operating level. The Maximum Safe Operating radiation level is based on the highest radiation level at which neither equipment necessary for the safe shutdown of the plant will fail nor personnel access necessary for the safe shutdown of the plant will be precluded.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CG1
2. LGA-001, RPV Control
3. Technical Specifications 3.6.1.1
4. Technical Specifications 3.6.4.1
5. LGA-003, Primary Containment Control
6. LGA-011, Hydrogen Control
7. LaSalle PSTG Section 5B, Hydrogen Control
8. LGA-002, Secondary Containment Control
9. UFSAR 3.6.2
10. LIS-NR-301(401), Unit 1(2) Source Range Monitor Rod Block Functional Test

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS8

Initiating Condition:

Loss of RCS/RPV inventory affecting core decay heat removal capability

Operating Mode Applicability:

4

EAL Threshold Values:

1. **WITHOUT** Primary or Secondary CONTAINMENT CLOSURE established:
    a. RPV level **< - 150 in.**

    OR

    b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Suppression Pool level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

OR

2. **WITH** Primary or Secondary CONTAINMENT CLOSURE established:
    a. RPV level **< -161 in**. (TAF).

    OR

    b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory as evidenced by either of the following:
    - Per Table M5 indications.
    - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

Basis (cont):

Threshold #1 Basis:

Under the conditions specified by this threshold, continued drop in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV. If a low-pressure boundary to fission product release does not exist (i.e., neither primary nor secondary containment closure is not established), the RPV level associated with this threshold is three inches below the low-pressure ECCS actuation setpoint (i.e., - 147 in. - 3 in. = - 150 in. WR that is the bottom range of the indication). If primary or secondary containment closure is established, a low-pressure boundary to fission product release exists and RPV level can drop to the top of active fuel, -161 in. on FZ, before a Site Area Emergency declaration is required. 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 drop and potential core uncovery.

In Cold Shutdown, the decay heat available to raise RPV temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding 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.

Threshold #2 Basis:

This threshold is 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, (e.g., decay heat removal system design, etc.) can have a significant impact on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

When RPV level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated drywell radiation or unexplained rise in drywell floor or equipment drain sump pumpout rate. LGA-001, RPV Control, provides guidance on determining if RPV level can be monitored. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors (SRM Channels A-D) can be used as a tool for making such determinations.

The thirty-minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS8 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 CS1
2. LGA-001, RPV Control
3. Technical Specifications Table 3.3.5.1-1
4. LOP-SF-06, Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup
5. LGA-001, RPV Control
6. UFSAR 5.2.5
7. LOP-NB-03, Troubleshooting Drywell Leakage
8. LIS-NR-301(401), Unit 1(2) Source Range Monitor Rod Block Functional Test
9. LGP-1-1, Normal Unit Startup
10. LGP-1-S1, Master Startup Checklist

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA8

Initiating Condition:

Loss of RCS/RPV inventory with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. Loss of RCS/RPV inventory as indicated by RPV level **< -147 in**. WR.

   OR

2. a. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Suppression Pool level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

   AND

   b. RCS/RPV level unknown for **> 15 minutes.**

Basis:

This EAL serves as a precursor 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 drop and potential core uncovery.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

In Cold Shutdown mode, the decay heat available to raise RPV temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding 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.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MA8 (cont)

Basis (cont):

In Cold Shutdown mode, the RCS will normally be intact and standard RPV inventory and RPV level monitoring means are available. In the Refueling mode, the RCS is not intact and RPV level and inventory are monitored by different means. In the Refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

RPV level is normally monitored using the following instruments:

- Shutdown Range (0 to +400 in.)
- Upset Range (0 to +180 in.)
- Narrow Range (0 to +60 in.)
- Wide Range (-150 to +60 in.)
- Fuel Zone (-311 to -111 in.)

In the second condition of this EAL, all RPV level indication would be unavailable. LGA-001 RPV Control provides guidance on determining if RPV level can be monitored. RPV inventory loss, therefore, must be detected by alternate means (i.e., drywell floor and equipment drain sump pumpout rates). Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency duration.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA1 & CA2
2. LOP-SF-06, Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup
3. LGA-001, RPV Control
4. Technical Specifications Table 3.3.5.1-1
5. LGA-001, RPV Control
6. UFSAR 5.2.5
7. LOP-NB-03, Troubleshooting Drywell Leakage

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU8

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

4

EAL Threshold Values:

RPV level **cannot** be restored and maintained **> 11.0 in.**

Basis:

The inability to restore and maintain level after reaching this setpoint (Level 3 – Low Level Scram Setpoint = 11.0 in.) infers a degradation of the level of safety at the plant.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU1
2. LOP-SF-06, Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup
3. LGA-001, RPV Control
4. Technical Specifications Table 3.3.1.1-1.
5. LPGP-CALC-2
6. UFSAR 5.2.5

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MS9

Initiating Condition:

Loss of RPV inventory affecting core decay heat removal capability with irradiated fuel in the RPV.

Operating Mode Applicability:

5

EAL Threshold Values:

1. **WITHOUT** Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< - 150 in**.

      OR

   b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
      - Refuel Floor Rad Monitor 0D21-K604A indicates **> 3000 mR/hr** or off-scale high.
      - Erratic Source Range Monitor indication.

   OR

2. **WITH** Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< - 161 in.** (TAF).

      OR

   b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
      - Refuel Floor Rad Monitor 0D21-K604A indicates **> 3000 mR/hr** or off-scale high.
      - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

Threshold #1 and #2 Basis:

Under the refueling conditions specified in this threshold, prolonged loss of the ability to monitor RPV level in conjunction with indirect indications of inventory loss infer a continued drop in RPV level and loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV.

Basis (cont):

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

In the refueling mode, when RPV level indication is unavailable, the inventory loss must be detected by drywell floor and equipment drain sump pumpout rates or erratic Source Range Monitor indication. Detail I of LGA-001, RPV Control, provides guidance on determining if RPV level can be monitored. Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to core shine, scattering and radiation bounce off of the solid surfaces in the area will result in readings on the Refuel floor Rad monitor 0D21-K604A > 3000 mR/hr. This threshold radiation value is based on calculations documented in EP-EAL-0501.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CS2
2. LOP-SF-06, Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup
3. LGA-001, RPV Control
4. Technical Specifications Table 3.3.5.1-1
5. UFSAR 5.2.5
6. LOP-NB-03, Troubleshooting Drywell Leakage
7. LIS-NR-301(401), Unit 1(2) Source Range Monitor Rod Block Functional Test
8. LGP-1-1, Normal Unit Startup
9. LGP-1-S1, Master Startup Checklist

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU9

Initiating Condition:

UNPLANNED loss of RCS inventory with irradiated fuel in the RPV.

Operating Mode Applicability:

5

EAL Threshold Values:

1. UNPLANNED RPV level drop below the RPV flange for **> 15 minutes**.

   OR

2. a. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Suppression Pool level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

   AND

   b. RPV level unknown.

Basis:

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

Threshold #1 Basis:

The RPV flange is at 819 ft. 8-3/4 in. el. or 217.5 in. above instrument zero. RPV level at this plant elevation is normally indicated by the Shutdown Range instrument (0 in. to +400 in.). With the RPV head removed, the Shutdown Range and Upset Range instruments are calibrated for floodup conditions and indicate reactor cavity water levels as high as the refuel floor. When calibrated for floodup conditions, the Shutdown Range instrument reads ~55 in. at the RPV flange and the Upset Range instrument reads ~45in. Visual observation of water level in the refueling cavity and RPV is also used during refuel operations.

This threshold is applicable only in the Refueling mode and addresses loss of inventory to below the RPV flange during refueling operations. Refueling operations that drop RPV level below the RPV flange are carefully planned and procedurally controlled. An Unusual Event is appropriate because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant.

The fifteen-minute interval provides a reasonable time frame to restore level using one or more of the redundant means of refill that should be available. If RPV level cannot be restored in this interval, a more serious condition may exist.

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU9 (cont)

Basis (cont):

Threshold #2 Basis:

In the second condition of this threshold, all RPV level indication would be unavailable. LGA Detail I provides guidance on determining if RPV level can be monitored. RPV inventory loss, therefore, must be detected by alternate means (i.e., drywell floor and equipment drain sump pumpout rates). Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU2
2. Technical Specifications 3.4.5
3. UFSAR 5.2.5
4. LOP-NB-03, Troubleshooting Drywell Leakage
5. LOP-SF-06, Filling the Reactor, Reactor Well and Dryer/Separator Pit Through Feedwater with Suppression Pool Cleanup
6. LGA-001, RPV Control
7. Technical Specifications Table 3.3.5.1-1

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU10

Initiating Condition:

UNPLANNED loss of all onsite or offsite communications capabilities.

Operating Mode Applicability:

1, 2, 3, 4, 5

EAL Threshold Values:

1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations.

   OR

2. Loss of all Table M6 **Offsite** communications capability.

Table M6 - Communications Capability		
System	**Onsite**	**Offsite**
Plant radio system	X	
Plant paging system	X	
Sound power phones	X	
In-plant telephones	X	
All telephone lines (commercial and microwave)		X
NARS		X
ENS		X
HPN		X
Satellite Phones		X
Cellular Phones		X

Basis:

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

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems with offsite authorities or personnel. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

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

MU10 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU6 & CU6
2. EP-MW-124-1001 Facilities Inventories and Equipment Tests
3. LAP-100-37, Station Communications
4. LOP-CQ-02, Intercom/Loud Speaker System
5. LOP-CQ-03, Sound Powered Telephone System
6. LOP-CQ-04, Intra-Plant Radio System
7. OP-AA-104-101, Communications

RECOGNITION CATEGORY
SYSTEM MALFUNTIONS

MU11

Initiating Condition:

Inability to reach required shutdown within Technical Specification limits.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a prescribed shutdown mode when the Technical Specification configuration cannot be restored. The plant is within its safety envelope when being shut down 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 completion period elapses under Technical Specifications and is not related to how long a condition may have existed.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU2
2. LaSalle Technical Specifications

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1

Initiating Condition:

Security event resulting in loss of physical control of the facility.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

A HOSTILE FORCE has taken control of:

1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1).

Table H1 - Safety Functions and Related Systems
• 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) • Fission Product Barriers

OR

2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).

Basis:

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.

Threshold #1 Basis

This threshold encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions. As a result, equipment control cannot be transferred to and operated from another location.

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

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.

Threshold #2 Basis

This threshold addresses loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely because there is freshly off-loaded fuel in the pool. The condition “freshly off-loaded reactor fuel in pool” equates to fuel off-loaded within the last 120 days in NF-AA-310 Special Nuclear Material And Core Component Movement.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HG1
2. LOA-RX-101(201) Unit 1(2) Control Room Evacuation Abnormal
3. SY-AA-101-132, Threat Assessment
4. Station Security Plan – Appendix C
5. LOA-SY-001, Security Abnormal Procedure
6. NF-AA-310, Special Nuclear Material And Core Component Movement

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1

Initiating Condition:

Site attack.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert ICs (HA1 and HA2) in that a hostile force has progressed from the OWNER CONTROLLED AREA to the Protected Area.

Although Nuclear Power Plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for Offsite Response Organizations (ORO) 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.

HS1 (cont)

Basis (cont):

This EAL is intended to address the potential for a very rapid progression of events due to a dedicated attack. It is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs. HOSTILE ACTION identified above encompasses various acts including:

- Air attack (LARGE AIRCRAFT impacting the protected area)
- Land-based attack (HOSTILE FORCE penetrating protected area)
- Waterborne attack (HOSTILE FORCE on water penetrating protected area)
- BOMBs breeching the protected area

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs.

This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for ORO to be notified and to activate in order to be better prepared to respond should protective actions become necessary. If not previously notified by NRC that the LARGE AIRCRAFT impact 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.

LARGE AIRCRAFT is meant to be an 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.

This EAL addresses the immediacy of a threat to impact site vital areas within a relatively short time. The fact that the site is under serious attack with minimal time available for additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HS4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. LOA-SY-001, Security Abnormal Procedure

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition:

Notification of an airborne attack threat.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LARGE AIRCRAFT is meant to be an 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.

The intent of this EAL is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations (ORO) and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Alert. This EAL is met when a plant receives information regarding a LARGE AIRCRAFT attack threat from NRC and the LARGE AIRCRAFT is less than 30 minutes away from the plant.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from such an attack. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for OROs to be notified and encouraged to activate (if they do not normally) to be better prepared should it be necessary to consider further actions.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA7
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. LOA-SY-001, Security Abnormal Procedure

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

Initiating Condition:

Confirmed terrorism security event which indicates a potential degradation in the Level of safety of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. A credible site-specific security threat notification as determined per SY-AA-101-132, "Threat Assessment."

   OR

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

Basis:

Threshold #1 Basis

The intent of this threshold is to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat.

The determination of "credible" is made through use of information found in the Station Security Plan or SY-AA-101-132, "Threat Assessment" procedure.

Threshold #2 Basis

The intent of this threshold is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Unusual Event. This threshold is met when a plant receives information regarding an aircraft threat from NRC. Should the threat involve a LARGE AIRCRAFT (LARGE AIRCRAFT is meant to be an aircraft with the potential for causing significant damage to the plant), then escalation to Alert via HA1 would be appropriate if the LARGE AIRCRAFT is less than 30 minutes away from the plant. The status and size of the plane may be provided by NORAD through the NRC. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
6. LOA-SY-001, Security Abnormal Procedure

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

Initiating Condition:

Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This EAL is intended to address the potential for a very rapid progression of events due to an attack including:

- Air attack (LARGE AIRCRAFT impacting the OCA)
- Land-based attack (HOSTILE FORCE progressing across licensee property or directing projectiles at the site)
- Waterborne attack (HOSTILE FORCE on water attempting forced entry or directing projectiles at the site)
- BOMBs

This EAL is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2 (cont)

Basis (cont):

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne terrorist attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for Offsite Response Organizations to be notified and to activate in order to be better prepared to respond should protective actions become necessary.

If not previously notified by NRC that the LARGE AIRCRAFT impact 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. LARGE AIRCRAFT is meant to be an 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.

This IC/EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time. The fact that the site is an identified attack candidate with minimal time available for further preparation requires a heightened state of readiness and implementation of protective measures that can be effective (onsite evacuation, dispersal or sheltering) before arrival or impact.

Basis Reference(s):

1. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security Based Events, HA8
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
6. LOA-SY-001, Security Abnormal Procedure

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS3

Initiating Condition:

Confirmed security event in a plant VITAL AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC (HA3).

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a Station's security. Only those events that involve Actual or Likely Major failures of plant functions needed for protection of the public need to be considered. The following events would not normally meet this requirement; (e.g., Failure by a Member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Station Security Plan.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HS1
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. LOA-SY-001, Security Abnormal Procedure

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

Initiating Condition:

Confirmed security event in a plant PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Unusual Event.

Multi-unit stations with shared safety functions should further consider how this IC may affect more than one unit and how this may be a factor in escalating the emergency class.

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a station’s security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered. The following events would not normally meet this requirement; (e.g., failure by a member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HA4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. LOA-SY-001, Security Abnormal Procedure

RECOGNITION CATEGORY HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU3

Initiating Condition:

Confirmed security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.

Basis:

Reference is made to Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

This threshold is based on Station Security Plan – Appendix C. 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.

Consideration should be given to the following types of events when evaluating an event against the criteria of the Station Security Plan: CIVIL DISTURBANCE, and STRIKE ACTION.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. LOA-SY-001, Security Abnormal Procedure

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS4

Initiating Condition:

Control Room Evacuation has been initiated and Plant Control cannot be established.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Control Room evacuation has been initiated.

   AND

2. Control of the plant **<u>cannot</u>** be established per LOA-RX-101(201) in **< 15 minutes**.

Basis:

The 15 minute time period starts when either:

a. Control of the plant is no longer maintained in the Main Control Room.

OR

b. The last Operator has left the Main Control Room.

The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. 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 determination of whether or not control is established outside of the Main Control Room is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the site-specific time for transfer that the licensee has control of the plant. Transfer of control to locations outside the Control Room is considered established when the Shift Manager has determined that the operators are capable of controlling reactivity, core cooling and heat sink functions.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HS2
2. LOA-RX-101(201), Unit 1(2) Control Room Evacuation Abnormal

RECOGNITION CATEGORY HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

Initiating Condition:

Control Room evacuation has been initiated.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Entry into LOA-RX-101(201) for Control Room evacuation.

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency operations centers are necessary. Procedure LOA-RX-101(201) Control Room Inaccessibility specifies conditions under which Control Room evacuation may be necessary.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA5
2. LOA-RX-101(201) Unit 1(2) Control Room Evacuation Abnormal

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5

Initiating Condition:

Natural and destructive phenomena affecting a VITAL AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by seismic instrumentation **> 0.10 g**.

   AND

   b. Confirmed by **EITHER**:

   - Earthquake felt in plant.
   - National Earthquake Center.

OR

2. Tornado or high winds **> 90 mph** within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

OR

3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

OR

4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area.

Table H2 – Vital Areas
• Reactor Building • Control Room • Auxiliary Building • Diesel Generator Rooms • Switchgear and Battery Rooms • Remote Shutdown Rooms • CSCS Pump Rooms • LSH (for 0E12-F300 access only)

OR

HA5 (cont)

EAL Threshold Values (cont):

5. Uncontrolled flooding that results in **EITHER**:

   a. Degraded safety system performance in any Table H3 area as indicated in the Control Room.

Table H3 – Internal Flooding Areas
• RCIC Room • B/C RHR Room • HPCS Room • A RHR Room • RB Raceway

   OR

   b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.

Basis:

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

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

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

Threshold #1 Basis:

This threshold addresses events that may have resulted in a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this threshold to assess the actual magnitude of the damage.

This threshold is based on seismic ground acceleration in excess of 0.1 g for the UFSAR Operating Basis Earthquake (OBE). Seismic events of this magnitude are greater than the Unusual Event threshold of EAL HU5 and can cause damage to plant safety functions.

Basis (cont):

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

Threshold #2 Basis:

This threshold addresses events that may have resulted in a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The Alert classification is appropriate if visible damage is observed and relevant plant parameters indicate that the performance of safety systems in these areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

The 90 mph threshold is the UFSAR design basis wind speed. Sustained winds present a more severe loading on the buildings than a gust.

Threshold #3 Basis:

This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes, or impact of projectiles into a Table H2 area. This threshold addresses vehicle crashes that challenge the operability of systems necessary for safe shutdown of the plant. Table H2 areas include Category 1 structures and those Category 2 structures that contain Category 1 Systems and components.

The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Table H2 areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

Threshold #4 Basis:

This threshold covers threats to safety related equipment imposed by missiles generated by failure of the main turbine. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Basis (cont):

Threshold #5 Basis:

This threshold addresses the effect of internal flooding that has resulted in degraded performance of safety systems or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant.

“Uncontrolled” as used in this threshold describes a condition where water is entering an area from an unplanned evolution. This flooding may have been caused by internal events such as component failures, equipment misalignment, and fire suppression system actuation or outage activity mishaps. Water entering an area, which resulted in degraded performance of safety systems within the area due to wetting or submergence, would meet the intent of this threshold. Minor leaks, such as valve packing or instrument line breaks would not constitute “'Uncontrolled Flooding.” Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if indications of degraded system performance is available or a shock hazard is known to exist.

The Internal Flooding Areas listed in Table H3 include areas listed in LGA-002 (Table W) for Max. Safe Water Level containing systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Potential internal flooding sources are:

- Circulating Water System
- CSCS Equipment Cooling Water System (RH WS, DG Cooling Water)
- Service Water System
- Failure of the Suppression Pool
- Fire protection
- Potable Water

The Illinois River is more than 180 ft. below the plant grade and does not affect safety-related systems and structures.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA1
2. UFSAR 3.3
3. UFSAR 3.4
4. UFSAR 3.7

Basis Reference(s) (cont):

5. UFSAR 3.8
6. UFSAR 3.11
7. LOR-1PM10J-B503 Seismic Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE) Level Exceeded
8. LOA-TORN-001, High Winds/Tornado
9. Drawing S-01A, Composite Site Plan
10. LOA-FLD-001, Flooding
11. Drawing M-24, Flood Plan
12. LGA 002, Secondary Containment Control

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition:

Natural and destructive phenomena affecting the PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. a. Seismic event as indicated by station seismic monitoring procedures **> 0.01g**.

   AND

   b. Confirmed by **EITHER**:

   - Earthquake felt in plant.
   - National Earthquake Center.

   OR

2. Report by plant personnel of tornado striking or sustained **(> 15 minutes)** high winds **> 90 mph**, within PROTECTED AREA boundary.

   OR

3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area.

Table H2 – Vital Areas
• Reactor Building • Control Room • Auxiliary Building • Diesel Generator Rooms • Switchgear and Battery Rooms • Remote Shutdown Rooms • CSCS Pump Rooms • LSH (for 0E12-F300 access only)

   OR

4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

   OR

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont)

EAL Threshold Values (cont):

5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode.

Table H3 – Internal Flooding Areas
• RCIC Room • B/C RHR Room • HPCS Room • A RHR Room • RB Raceway

Basis:

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

Threshold #1 Basis:

This threshold is based on the strong-motion seismograph actuation level which is the sensed earthquake threshold of 0.01 g. Seismic events of this magnitude are ~1/10 of the Alert event threshold (OBE) of EAL HA5 in which it is assumed the earthquake can cause damage to plant safety functions.

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

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. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01 g.”

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis (cont):

Threshold #2 Basis:

90 mph is the UFSAR design basis wind speed. Station structures are designed to withstand wind loads which may exist if sustained wind speeds reach or exceed 90 mph. Wind loads in excess of this magnitude can cause damage to safety functions. Verification of a tornado is obtained by direct observation and reporting by station personnel. “Sustained” wind speeds exist for 15 minutes or longer. Wind speed is obtained from meteorological data in the Control Room.

Threshold #3 Basis:

In this context, a "vehicle crash" is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

Threshold #4 Basis:

This threshold is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this threshold to classify minor operational leakage.

Threshold #5 Basis:

“Uncontrolled” as used in this threshold describes a condition where water is entering the area from an unplanned evolution. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if a potential to affect safety related equipment needed for the current operating mode exists.

This threshold addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, fire suppression system actuation or outage activity mishaps. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding.” The Internal Flooding Areas of concern for the Unusual Event declaration are those areas listed in Table H3 – Internal Flooding Areas that have the potential to affect safety related equipment needed for the current operating mode.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU1
2. UFSAR 3.3
3. UFSAR 3.4
4. UFSAR 3.7
5. UFSAR 3.8
6. UFSAR 3.11

Basis Reference(s) (cont):

7. LOR-1PM10J-B504, Seismic Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE) Level Exceeded
8. LOA-TORN-001, High Winds/Tornado
9. Drawing S-01A, Composite Site Plan
10. LOA-FLD-001, Flooding
11. Drawing M-24, Flood Plan
12. LOA-HY-01(02), Unit 1(2) Generator Hydrogen System Abnormal
13. LOP-HY-06, Hydrogen System Leak Detection
14. LOR-1PM03J-B511, Condenser Vacuum Low
15. LOR-1H13-P603-B201, Division 1 Main Condenser Vacuum Low
16. LOR-1H13-P603-B212, Division II Main Condenser Vacuum Low
17. LOR-2PM03J-B511, Condenser Vacuum Low
19. LOR-2H13-P603-B201, Division 1 Main Condenser Vacuum Low
10. LOR-2H13-P603-B212, Division II Main Condenser Vacuum Low
20. LGA 002, Secondary Containment Control

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6

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, D

EAL Threshold Values:

1. FIRE or EXPLOSION in any Table H2 area.

Table H2 – Vital Areas
• Reactor Building • Control Room • Auxiliary Building • Diesel Generator Rooms • Switchgear and Battery Rooms • Remote Shutdown Rooms • CSCS Pump Rooms • LSH (for 0E12-F300 access only)

AND

2. a. Affected safety system parameter indications show degraded performance.

   OR

   b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.

Basis:

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.

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

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

Basis (cont):

The areas listed in Table H2 house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Personnel access to these areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses FIRES and EXPLOSIONS that challenge the operability of systems necessary for safe shutdown of the plant.

The only FIRES and EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment required for safe shutdown. Visual observation of damage infers the ability to approach or enter the affected areas. Lacking the ability to adequately inspect the area for damage, the Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this EAL. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

A steam line break or steam EXPLOSION that damages permanent structures or equipment in one of these areas would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA2
2. UFSAR 3.8
3. LOA-FP-101(201), Fire Protection System Abnormal

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

   OR

2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

Table H2 – Vital Areas
• Reactor Building • Control Room • Auxiliary Building • Diesel Generator Rooms • Switchgear and Battery Rooms • Remote Shutdown Rooms • CSCS Pump Rooms • LSH (for 0E12-F300 access only)

   OR

3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

Basis:

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

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.

HU6 (cont)

Basis (cont):

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

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

Thresholds #1 and #2 Basis:

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins with a credible notification that a fire is occurring or indication of a valid fire detection system alarm. A verified alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm.

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 EAL HA7, Release of Toxic or Flammable Gases. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under EAL HA7.

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

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket, etc.). Such fires are excluded from consideration in this threshold since they have no safety consequence.

Threshold #3 Basis:

The only EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment in the PROTECTED AREA.

A steam line break or steam EXPLOSION that damages permanent structures or equipment in a PROTECTED AREA would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU2
2. UFSAR 3.8
3. LOA-FP-101(201), Fire Protection System Abnormal

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA7

Initiating Condition:

Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

   OR

2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).

Table H2 – Vital Areas
• Reactor Building • Control Room • Auxiliary Building • Diesel Generator Rooms • Switchgear and Battery Rooms • Remote Shutdown Rooms • CSCS Pump Rooms • LSH (for 0E12-F300 access only)

Basis:

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

Basis (cont):

Values for LFL for common gases at LaSalle Station are:

- Propane – 2.2% (BOC Gasses MSDS)
- Hydrogen – 4% (Air Liquide Safety Data Sheet)
- Acetylene – 2.2% (BOC Gasses MSDS)

This EAL is based on toxic, asphyxiant, or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Toxic or flammable gases detected outside of these areas need not be considered for this EAL unless there is a spread of the gasses into one of these areas.

Threshold #1:

Declaration should not be delayed for confirmation from atmospheric testing if it is reasonable to conclude that the IDLH concentrations have been met (e.g., documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards).

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 this EAL. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under this EAL.

The first condition is met if measurement of toxic gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a Table H2 area. Non-Toxic Gases which displace oxygen (site examples; Halon or Nitrogen) to a life threatening level due to asphyxiation (oxygen deprivation) should also be considered for this EAL.

An Asphyxiant is a material 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.

Threshold #2:

The second condition is met when the flammable gas concentration in a Table H2 area exceeds the lower flammability limit. Flammable gases such as hydrogen and acetylene are routinely used to maintain plant systems (hydrogen – main generator cooling, reactor coolant chemistry control) or repair equipment/components (acetylene - welding). This condition addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a Table H2 area has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage or personnel injury.

Basis (cont):

Once it has been determined that an uncontrolled release of flammable gas is occurring, sampling must be done to determine if the gas concentration exceeds the lower flammability limit.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA3
2. UFSAR 9.5
3. LAP-911-2, Hazardous Materials (Hazmat) Release Field Actions
4. LAP-911-3, Hazardous Materials (Hazmat) Release Control Room Actions

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU7

Initiating Condition:

Release of toxic or flammable gases deemed detrimental to normal operation of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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:

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.

This EAL is based on the existence of uncontrolled releases of toxic, asphyxiant, or flammable gas affecting plant operations or the health of plant personnel. The release may have originated within the Protected Area boundary, or it may have originated offsite and subsequently drifted inside the Protected Area boundary. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on normal plant operations.

It is intended that releases of toxic, asphyxiant, or flammable gases are of sufficient quantity and the release point of such gases is such that safe plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for safe plant operation. The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect safe plant operations or plant personnel safety.

An Asphyxiant is a material 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.

Basis Reference(s) (cont):

1. NEI 99-01, Rev. 4 HU3
2. UFSAR 9.53. UFSAR 2.2
4. LAP-911-2, Hazardous Materials (Hazmat) Release Field Actions
5. LAP-911-3, Hazardous Materials (Hazmat) Release Control Room Actions

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels ($>$ 1 Rem TEDE or $>$ 5 Rem CDE Thyroid) outside the site boundary.

Basis Reference(s):

1. NEI 99-01, Rev 4 HG2
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS8

Initiating Condition:

Other Conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

Basis Reference(s):

1. NEI 99-01, Rev 4 HS3
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA6
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

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:

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency operating procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU5
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)

Section 4: Emergency Measures

Exelon Nuclear emergency response actions are the same for all nuclear stations and are thus covered by Section E of the Emergency Plan.

4.1 Activation and Notification of the Emergency Organization

Standard NARS notifications for the LaSalle Station are made to the State of Illinois Emergency Management Agency (IEMA). At the LaSalle Station, if a General Emergency is the initiating event, the Emergency Director is responsible for notifying the following offsite agencies:

- LaSalle County
- Grundy County

4.2 Assessment Actions

Throughout each emergency situation, continuing assessment will occur. Assessment actions at LaSalle Station may include an evaluation of plant conditions; in-plant, onsite, and initial offsite radiological measurements; and initial estimates of offsite doses. Core damage information is used to refine dose assessments and confirm or extend initial protective action recommendations. LaSalle Station utilizes NEDC-33045P-A, Revision 0, (2001) as the basis for the methodology for post-accident core damage assessment. This methodology utilizes real-time plant indications. In addition, LaSalle Station may use samples of plant fluids and atmospheres as inputs to the CDAM (Core Damage Assessment Methodology) program for core damage estimation.

4.3 Protective Actions for the Offsite Public

To aid Control Room personnel during a rapidly developing emergency situation, Figure 4-1, "Protective Action Recommendation (PAR) Determination Flowchart for LaSalle Station" has been developed based on Section J.10.m of the Emergency Plan.

4.3.1 Alert and Notification System (ANS) Sirens

The ANS consists of a permanently installed outdoor notification system within the ten mile radius around the station. The ten mile radius around the station is primarily an agricultural area with a population density below 2000 persons per square mile. The ANS as installed consists of mechanical and electronic sirens which will cover this entire area with a minimum sound level of 60 db. Additionally, the ANS will cover the heavily populated areas within the ten mile radius around the station with a minimum sound level of 70 db to ensure complete coverage.

Once the public has tuned to designated radio stations in an emergency, detailed instructional messages will be given to the public. State and local procedures provide for these messages.

4.3.2 Evacuation Time Estimates

The evacuation time estimates were developed per the requirements of NUREG-0654, and to support the Illinois Plan For Radiological Accidents (IPRA) - LaSalle Station Volume III. The purpose of the evacuation time estimates is to assess postulated evacuation times for the LaSalle Station Emergency Planning Zone (EPZ).

The evacuation time estimate data was updated per a study performed by Earth Tech. Inc. documented in their report dated June, 2005 entitled "Update of Evacuation Time Estimates for the Plume Exposure Pathway Emergency Planning Zone for LaSalle County Nuclear Generating Station."

An updated set of evacuation time estimates (ETEs) for LaSalle County Nuclear Generating Station has been developed, using population data from the 2000 Census. The assumptions and analysis procedures for the present study closely followed the approach of the previous (1993) study . A full ETE update study was not judged to be necessary for LaSalle Station. Comparison of the 1990 and 2000 Census data showed relatively small changes in population within the Emergency Planning Zone (EPZ), and no significant changes to the roadway network have occurred since 1993. This "partial" update study was therefore performed using data from the 1993 study to characterize the roadway network and the populations for special and transient facilities; only the population numbers for permanent and seasonal residents have been updated. The evacuation times are based on a detailed consideration of the EPZ roadway network and population distribution. The information in Table 4-1 presents representative evacuation times for daytime and nighttime scenarios under various weather conditions for the evacuation of various areas around the LaSalle Station, once a decision has been made to evacuate. The evacuation times noted include notification, mobilization, and travel time. These times are for the general population which include permanent population and special facilities (schools, nursing homes, hospitals, and recreational areas). Table 4-2 provides information on the scenario population distribution (by Subarea) that was used for this study. Table 4-3 provides a representation of the Subarea Locations in relation to the EPZ.

4.4 Protective Actions for Onsite Personnel

LaSalle Station has a siren system to warn personnel of emergency conditions. Upon hearing a continuous two (2) minute siren, or receiving notification by other means of communication, all personnel not having emergency assignments have been instructed to assemble in a predesignated assembly area. The onsite assembly area for LaSalle Station is the Service Building Trackway on 710' elevation of the South Service Building. Refer to Figure 4-2. Accountability of site personnel is accomplished by the Station Security force.

If a site evacuation of non-essential personnel is required, personnel will be released to their homes or relocated and monitored at a relocation center.

The designated relocation centers for LaSalle Station are:

- Mazon Relocation Center, Mazon, Illinois
- Dresden Station, Morris, Illinois

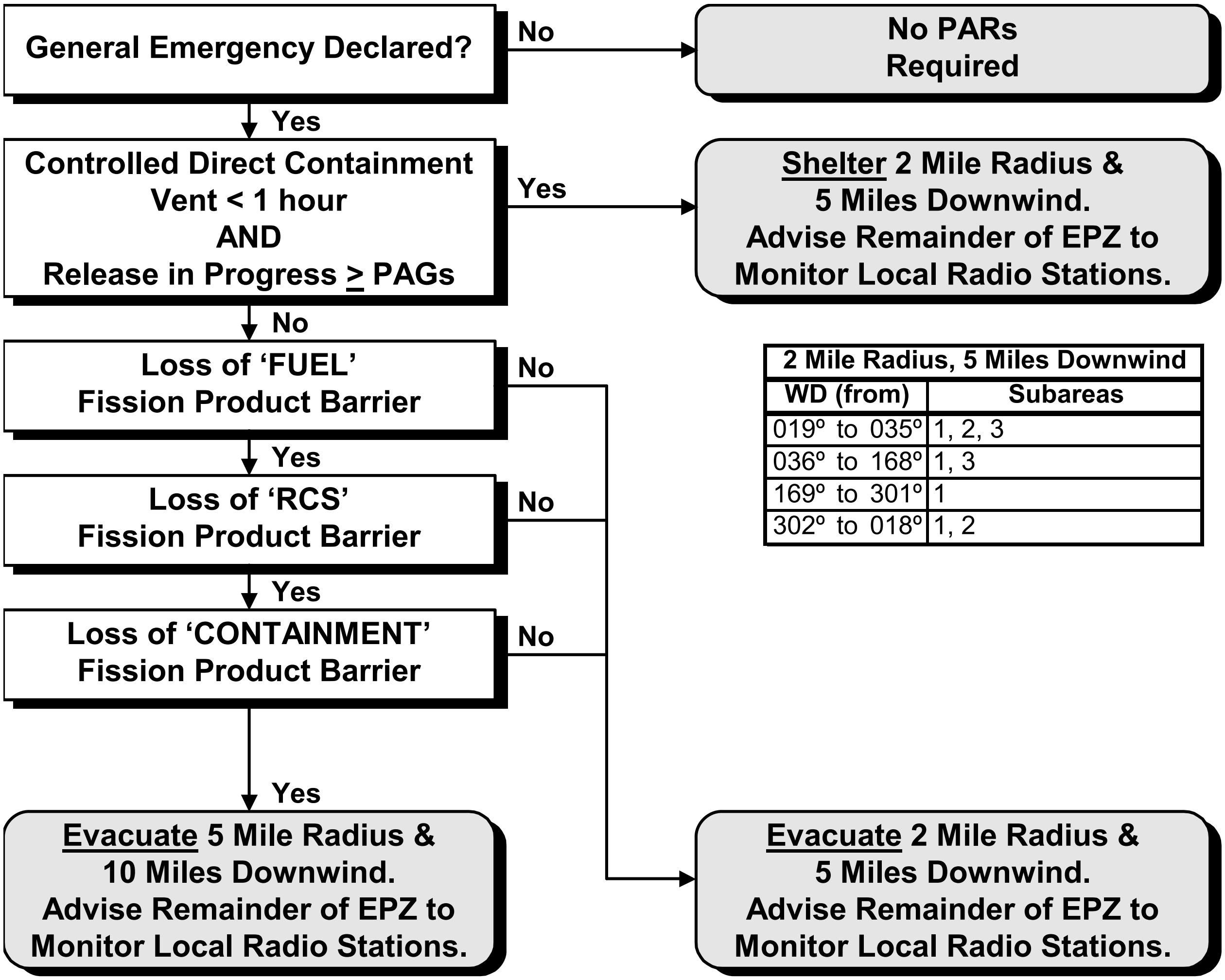
For evacuation routes, refer to EP-AA-113-F-20.

Traffic control for onsite areas will be the responsibility of the Station Security force. When a site evacuation is imminent, the Station Security force will post guards as necessary to assist in the evacuation.

Equipment and personnel would be available at the Mazon Relocation Center and Dresden Station for monitoring and decontamination of evacuated personnel. If major decontamination, follow-up or bioassay samples are necessary, those persons would be sent to Dresden and Braidwood Stations.

Other emergency measures are common to all nuclear stations and are thus discussed in the Emergency Plan.

Figure 4-1: LaSalle Station PAR Determination Flowchart



2 Mile Radius, 5 Miles Downwind	
WD (from)	Subareas
019° to 035°	1, 2, 3
036° to 168°	1, 3
169° to 301°	1
302° to 018°	1, 2

5 Mile Radius, 10 Miles Downwind	
WD (from)	Subareas
002° to 026°	1, 2, 3, 4, 5
027° to 056°	1, 2, 3, 4
057° to 076°	1, 2, 3, 4, 7
077° to 096°	1, 2, 3, 7
097° to 116°	1, 2, 3, 7, 8
117° to 121°	1, 2, 3, 8
122° to 149°	1, 2, 3, 8, 11
150° to 178°	1, 2, 3, 10, 11
179° to 197°	1, 2, 3, 10
198° to 218°	1, 2, 3, 6, 10
219° to 233°	1, 2, 3, 6, 9, 10
234° to 242°	1, 2, 3, 6, 9, 13
243° to 265°	1, 2, 3, 9, 13
266° to 281°	1, 2, 3, 13
282° to 316°	1, 2, 3, 13, 17
317° to 342°	1, 2, 3, 5, 17
343° to 001°	1, 2, 3, 5

2 Mile Radius, 5 Miles Downwind	
WD (from)	Subareas
019° to 035°	1, 2, 3
036° to 168°	1, 3
169° to 301°	1
302° to 018°	1, 2

Figure 4-2: LaSalle Onsite Assembly Areas and Emergency Response Facilities

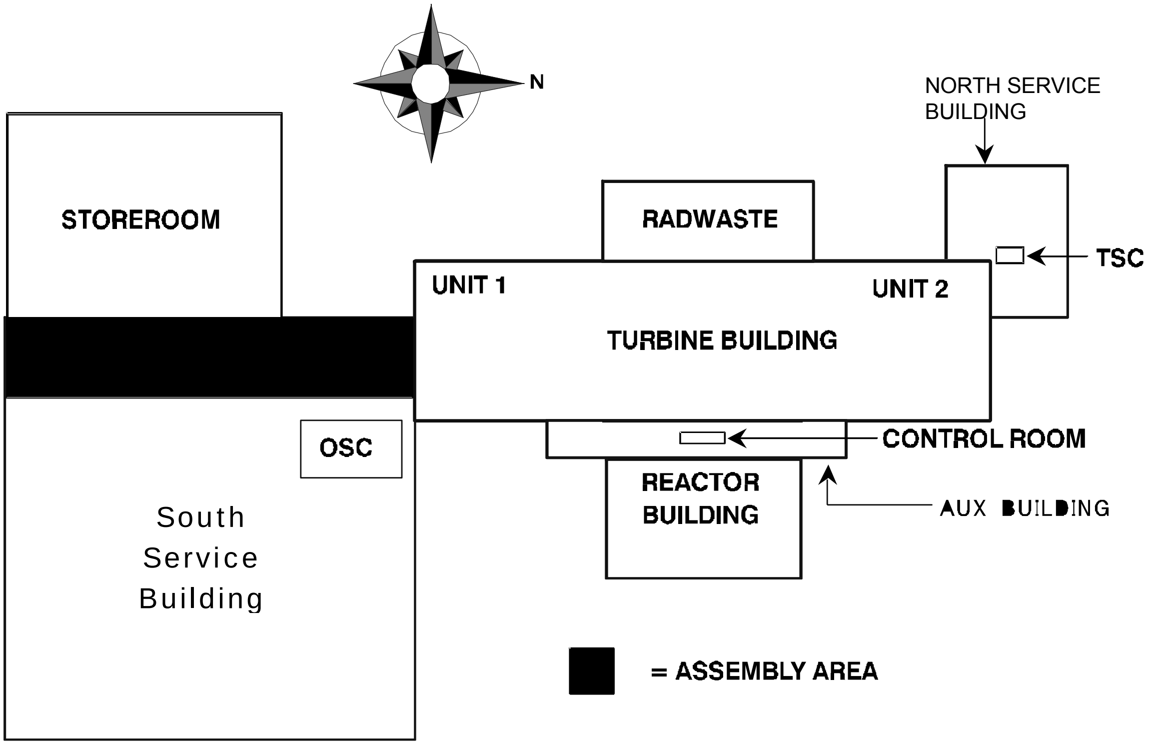


Table 4-1: LaSalle Evacuation Time Estimates (in minutes)

General Population Evacuation Times (minutes)

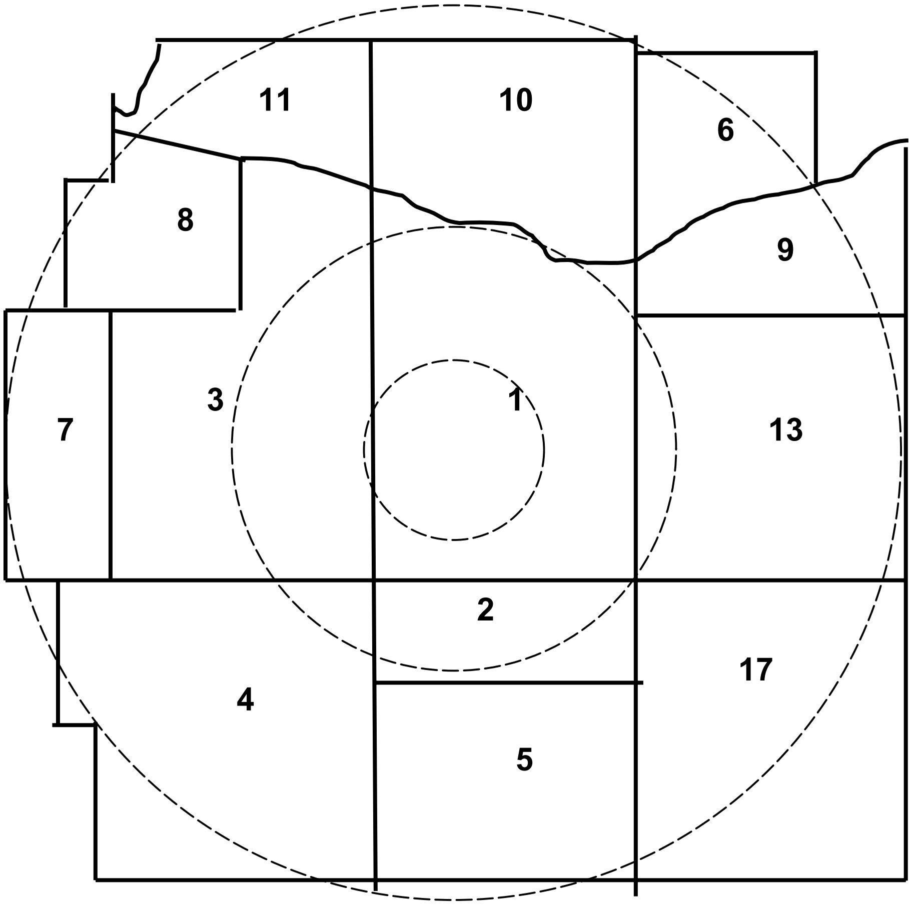
PAR Evacuation Zone		Summer Day		Summer Night		Winter Day		Winter Night	
Wind Direction	**Subareas**	Normal weather	Adverse weather	Normal weather	Adverse weather	Normal weather	Adverse weather	Normal weather	Adverse weather
2 mile Radius, 5 Miles Downwind									
019 to 035	1, 2, 3	160	190	80	80	160	190	80	80
036 to 168	1, 3	160	190	80	80	160	190	80	80
169 to 301	1	160	190	80	80	160	190	80	80
302 to 018	1, 2	160	190	80	80	160	190	80	80
5 mile Radius, 10 miles Downwind									
002 to 026	1, 2, 3, 4, 5	170	200	90	90	170	200	90	90
027 to 056	1, 2, 3, 4	170	200	90	90	170	200	90	90
057 to 076	1, 2, 3, 4, 7	170	200	90	100	170	200	90	90
077 to 096	1, 2, 3, 7	170	200	90	100	170	200	90	90
097 to 116	1, 2, 3, 7, 8	170	200	100	100	170	200	90	90
117 to 121	1, 2, 3, 8	170	200	100	100	170	200	90	90
122 to 149	1, 2, 3, 8, 11	185	215	110	120	185	215	105	115
150 to 178	1, 2, 3, 10, 11	185	215	110	120	185	215	105	115
179 to 197	1, 2, 3, 10	185	215	110	120	185	215	105	115
198 to 218	1, 2, 3, 6, 10	185	215	110	120	185	215	105	115
219 to 233	1, 2, 3, 6, 9, 10	185	215	110	120	185	215	105	115
234 to 242	1, 2, 3, 6, 9, 13	185	215	110	120	185	215	105	115
243 to 265	1, 2, 3, 9, 13	170	200	100	90	170	200	90	90
266 to 281	1, 2, 3, 13	170	200	90	90	170	200	90	90
282 to 316	1, 2, 3, 13, 17	170	200	90	90	170	200	90	90
317 to 342	1, 2, 3, 5, 17	160	190	80	90	160	190	80	90
343 to 001	1, 2, 3, 5	160	190	80	90	160	190	80	90
Entire EPZ		185	215	110	120	185	215	105	115

Table 4-2: LaSalle Scenario Population Distribution By Subarea

	Winter Day		Winter Night		Summer Day		Summer Night	
Sub-Area	Population	Vehicles	Population	Vehicles	Population	Vehicles	Population	Vehicles
1	1,240	538	986	382	4,674	1,579	2,236	727
2	67	26	67	26	67	26	67	26
3	704	269	704	269	781	321	781	321
4	3,447	1,351	3,363	1,284	3,393	1,294	3,393	1,294
5	709	235	571	218	585	221	571	218
6	159	61	159	61	165	63	165	63
7	1,108	303	665	254	665	254	665	254
8	766	275	662	246	874	280	770	251
9	387	221	286	120	387	221	286	120
10	10,205	4,032	8,081	3,264	10,904	4,466	10,431	4,135
11	7,012	2,939	5,907	2,404	6,930	3,075	6,315	2,580
Marseilles	5,808	2,325	4,853	1,940	5,493	2,330	5,228	2,105
13	638	244	638	244	638	244	638	244
17	314	120	314	120	314	120	314	120
EPZ	20,948	8,287	17,550	6,950	24,884	9,832	21,404	8,246

* City of Marseilles is included in both Sub-Area 10 and Sub-Area 11

Table 4-3: LaSalle Subarea Locations



Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities

Refer to Figure 4-2 for the location of the LaSalle Station Control Room, Technical Support Center (TSC), and Operations Support Center (OSC) within the Station's Protected Area boundary.

5.1.1 Station Control Room

The LaSalle Station Control Room is the initial onsite center of emergency control and is located on the 768' elevation of the Auxiliary Building.

5.1.2 Technical Support Center (TSC)

LaSalle Station has a designated TSC in the upper basement level (Elevation 694) of the North Service Building. Standard air sampling equipment is used to monitor air-borne radioactivity levels in the TSC. The TSC fully meets the requirements of Section H.1.b of the Emergency Plan.

5.1.3 Operational Support Center (OSC)

LaSalle Station has a designated Operational Support Center (OSC). The OSC is located on the ground floor of the South Service Building. The OSC conforms to the requirements of Section H.1.c of the Emergency Plan and is the location to which operations support personnel will report during an emergency and from which they will be dispatched for assignments in support of emergency operations.

5.2 Assessment Resources

5.2.1 Onsite Meteorological Monitoring Instrumentation

A 400 foot meteorological tower has been erected on the site on ground approximately final plant grade, 710 MSL. The tower is 180 feet from the nearest building which is approximately 30 feet tall. It is approximately 75 feet outside of and to the southeast of the protected area. The turbine building is approximately 134 feet tall, and the reactor building is approximately 185 feet tall.

The tower is instrumented at three levels. Wind speed and wind direction are measured at 33 feet, 200 feet and 375 feet. Ambient temperature is measured at the 33 feet level and differential temperatures referenced to 33 feet are measured at 200 feet and 375 feet. Precipitation is also measured nearby. The 375 feet level corresponds to the elevation of the possible point of airborne release.

The onsite meteorological monitoring program is covered in the contract specification and vendor procedures of the meteorological monitoring contractor. These data are used to generate wind roses and to provide estimates of airborne concentrations of gaseous effluents.

5.2.1.1 Instrumentation

The meteorological tower is instrumented with equipment that conforms with the system accuracy recommendations of Regulatory Guide 1.23 and ANSI/ANS 2.5 (1984). The equipment is placed on booms oriented normal to the general prevailing wind at the site. Equipment signals are brought to an instrument building with controlled environmental conditions. The building at the base of the tower houses the recording equipment, signal conditioners, etc., used to process and re-transmit the data to the end point users.

5.2.1.2 Meteorological Measurement Program During a Disaster

Cooperation between the corporate office and the meteorological contractor assures that a timely restoration of any outage can be made. Emergency field visits to the site are made as quickly as possible after detection of a failure.

Should a disaster of sufficient magnitude occur to destroy the tower structure, a contract is maintained to have a temporary tower erected within 72 hours, weather conditions permitting. Further, the meteorological contractor maintains two levels of sensors (wind speed, wind direction and temperature) in a state of readiness for use on the temporary tower.

Additionally, Exelon Nuclear's existing instrumentated towers at other nuclear sites provide a measurement network with multiple backup opportunities.

Meteorological data is available to the station Control Room, Technical Support Center, and Emergency Operations Facility for use in the Dose Assessment Computer Model for estimating the environmental impact of unplanned releases of radioactivity from the station.

5.2.2 Onsite Radiation Monitoring Equipment

Chapters 11 and 12 of the LaSalle Station UFSAR describe in detail the LaSalle Station radiation monitoring systems and equipment. The systems and equipment can be categorized into five (5) groups:

- A process radiological monitoring and sampling system;
- An effluent radiological monitoring and sampling system;
- An airborne radioactive monitoring system;
- An area radiation monitoring system; and
- Portable survey and counting equipment.

Some on-site equipment is particularly valuable for accident situations and is described in the following sections.

5.2.2.1 Radiological Noble Gas Effluent Monitoring

A General Atomic wide-range monitor is installed in the effluent stream which enters the LaSalle Station vent stack. A separate monitor is installed for the Standby Gas vent stack (contained inside the station vent stack). These wide-range monitors have a range of $1x10^{-7}$ uCi/cc to $1x10^{5}$ uCi/cc.

Instrument readings are available in the Control Room. The range of indication is 10^{1} to 10^{13} uCi/sec. Calibration factors for converting instrument responses to release rates are determined from energy response testing performed during calibration. The factors are then entered into the system microprocessor data base.

5.2.2.2 Radioiodine and Particulate Effluent Monitoring

Effluent sampling media are analyzed in the Station counting room by a Germanium isotopic analysis. The iodine cartridges are reverse-blown for at least ten minutes to reduce the level of entrapped noble gases or as otherwise directed by the Chemistry Supervisor. In addition, silver zeolite cartridges are to be used to further reduce the interference of noble gases.

5.2.2.3 High-Range Containment Radiation Monitors

The purpose of the containment atmosphere and gross gamma monitoring system is to provide the signals necessary to indicate and alarm high hydrogen concentration, high oxygen concentration, or high gross gamma radiation in the drywell following a loss-of-coolant-accident (LOCA).

The containment atmosphere monitoring subsystem monitors hydrogen and oxygen in the drywell resulting from radiolytic and chemical phenomena associated with an accident condition. The gross gamma monitoring subsystem, consisting of two high range (1 R/hr. to 10^{8} R/hr.) containment radiation detectors, monitors gamma radiation resulting from the gross release of fission products from the reactor fuel. Each subsystem has two redundant channels of instrumentation that are physically separated and electrically independent.

Each channel provides a local measurement and transmits the signals to the control room where a permanent record is made on seismically qualified recorders.

5.2.2.4 In-plant Iodine Instrumentation

LaSalle Station has the capability to sample and determine iodine concentrations in the plant using charcoal or silver zeolite cartridges and gamma ray spectroscopy. Portable monitors may be used to measure increasing levels of iodine during emergency conditions.

5.2.3 Onsite Process Monitors

Adequate monitoring capability exists to properly assess the plant status for all modes of operation. The operability of the postaccident instrumentation ensures information is available on selected plant parameters to monitor and assess important variables following an accident.

- Post-accident instrumentation is available to monitor:
- Reactor Vessel Pressure
- Reactor Vessel Water Level
- Suppression Chamber Water Level
- Suppression Chamber Water Temperature
- Suppression Chamber Air Temperature
- Drywell Pressure
- Drywell Temperature
- Containment Hydrogen Concentration
- Containment Gross Gamma Radiation

Station procedures have been developed which would aid personnel in recognizing inadequate core cooling using the above existing instrumentation.

5.2.4 Onsite Fire Detection Instrumentation

LaSalle Station has a fire protection system that is designed to quickly detect any fires; annunciating locally and in the Control Room. The fire detection system is designed to applicable National Fire Protection Association (NFPA) standards. The majority of the detectors consist of electrically supervised ionization smoke detectors. The system is normally powered from 120 VAC with automatic transfer to 125VDC on loss of power. In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire watches in affected areas is required.

A further description of the LaSalle Station fire protection system can be found in Chapter 9 of the LaSalle Station UFSAR.

5.2.5 Facilities and Equipment for Offsite Monitoring

Consult the station specific Offsite Dose Calculation Manual (ODCM) for the most current location for fixed continuous air samplers and TLD locations.

5.2.6 Site Hydrological Characteristics

The hydrological characteristics of the LaSalle Station vicinity are described in Section 2.4 of the LaSalle Station UFSAR. The LaSalle Station and the cooling lake cover an area of approximately 3,060 acres. The station is located approximately 5.0 miles south of the Illinois River. The cooling lake is approximately 2 miles south of the Illinois River at its closest point.

The terrain around the plant site is gently rolling, with ground surface elevations vary from 700 feet to 724 feet MSL, which is 217 feet above the normal pool elevation in the Illinois River. The plant grade and floor elevations are 710 feet and 710.5 feet MSL respectively. The plant floor is 188 feet above a postulated probable maximum flood (PMF) with coincident wind waves in the Illinois River. The station site may therefore be characterized as "floodproof" or " dry" regarding floods in the Illinois River. Safety-related structures at the plant site are similarly unaffected by wave run-up due to winds coincident with a postulated probable maximum water level in the cooling lake. The elevation of the perimeter road around the plant buildings (including all the safety-related structures) is 709 feet MSL or above.

The river screen house and the outfall structure, both non-safety-related structures, are the only plant facilities that are potentially affected by floods in the Illinois River. The river screen house is capable of withstanding a 100-year flood in the Illinois River.

5.2.6.1 Flood Design Considerations

In the event of a seismically induced dam failure, it is unlikely that the resulting flood stage would exceed the Illinois River PMF stage at the site. Breaching of the peripheral dikes of the cooling lake at the time of a postulated seismic event would cause the impounded water to discharge directly into local creeks that meet the Illinois River. Since the plant grade is set at elevation 710 feet MSL, and the plant floor is at elevation 710.5 feet MSL, there is no likelihood of flooding of the plant facilities due to this phenomenon.

Since cooling of the power plant condensers is accomplished by pumping from the cooling lake and not from the Illinois River directly, plant safety is not affected by postulated blockage of the Illinois River or by any other concurrent flooding condition.

Although ice formation takes place on all rivers in the Illinois River basin, flooding caused by ice jams is a rare event. Ice jam formation does not exist in the Illinois River near the site, since the river is approximately 800 feet wide and is kept navigable by dredging when required. The lake screen house is protected against icing in the lake by provision of warming lines near the screen house.

Makeup water is pumped from the Illinois River using three pumps with a total capacity of 90,000 gpm. The rate of pumping varies depending upon the plant operating load level and the weather conditions. It is designed to maintain a constant lake level and a total dissolved solids (TDS) level of less than 750 ppm in the blowdown. The minimum operating lake level is 697.75 feet MSL. Lake level is continuously monitored in the main control room of the power plant. Safety-related facilities at the plant site are unaffected by the probable maximum water level in the lake with coincident wind wave activity. In the event that the cooling lake level drops to an elevation of 690 feet MSL or lower, the nuclear reactors are shut down as described in Subsection 5.2.6.

Due to the considerable width of the Illinois River and the well-developed flood plain, there is little likelihood that rock falls, ice jams, or subsidence would completely divert the flow from the river screen house location.

5.2.6.2 Groundwater Use and Protection

The discussion of regional groundwater hydrology includes the hydrogeologic systems within a 25-mile radius circle centered at the LaSalle Station, Units 1 and 2. The discussion of site groundwater hydrology includes the hydrogeologic systems within the LSCS property lines.

Groundwater will be used to supply the water requirements for the following plant systems: makeup demineralizer; potable supply. Groundwater will be obtained from two deep wells in the Cambrian-Ordovician Aquifer. Each well is equipped with a deep well submersible pump with a rated capacity of 300 gpm. The water will be stored in a 350,000 - gallon, ground level tank prior to distribution to the demineralizer and domestic systems. Groundwater for public use within 10 miles of the site is obtained predominantly from wells in the Cambrian-Ordovician Aquifer.

An accidental spill of radioactive materials would have no effect on the public groundwater supplies. The principal aquifer in the area is overlain by approximately 350 feet of impervious till and underlying shales.

5.3 Protective Facilities and Equipment

The on-site assembly area for LaSalle Station is the Service Building Trackway on 710' elevation of the South Service Building as described in Section 6.4 of this annex. This area is suitable because:

1. It is an open area suitable for assembling large numbers of people in a short period of time;

2. It is relatively close to the Main Access Facility; and

3. It has a relatively low probability of being affected by a serious accident involving the NSSS.

The offsite relocation centers for LaSalle Station are discussed in Section 4 of this annex. All three centers are suitable because:

1. They are outside the LaSalle Station plume exposure pathway emergency planning zone; and

2. The relocation centers are owned by Exelon Nuclear, therefore, personnel, supplies and communication equipment are readily available.

5.4 First Aid and Medical Facilities

LaSalle Station has an inplant first aid/decontamination room on the ground floor of the Auxiliary Building near the station laboratory complex. This room is provided with a sink, a portable first aid table, a shower, and a supply cabinet.

First aid kits, stretchers, sinks, eyewashes, and emergency showers have been placed in strategic locations throughout the station.

Medical treatment given to injured persons at the station is of a "first aid" nature. When more professional care is needed, injured persons are transported to a local hospital or clinic. St. Mary's Hospital in Streator, Illinois is the LaSalle Station primary supporting medical facility for injured persons who are contaminated with radioactivity. Morris Hospital in Morris, Illinois is the LaSalle Station supporting Trauma Center for injured persons who are contaminated with radioactivity.

Provena St. Joseph Medical Center in Joliet, Illinois is the backup medical facility for evaluation and treatment of persons suffering from traumatic injury, medical illness, or radiation exposure and uptake.

Appendix 1: NUREG-0654 Cross-Reference

Annex Section	NUREG-0654	Annex Section	NUREG-0654
1.0	Part I, Section A	Figure 4-1	Part II, Section J.10.m
1.1	Part I, Section C	Figure 4-2	Part II, Section J.5
1.2	Part I, Section D	4.4	Part II, Section J.2 & 3
Figure 1-1	Part I, Section D	Table 4-1	Part II, Section J.8
		Table 4-2	Part II, Section J.10.b
2.0	Part II, Section A.4		
2.1	Part II, Section A.3	5.1	Part II, Section H.1 & G.3
		5.2.1	Part II, Section H.5.a & 8
3.0	Part II, Section D	5.2.2	Part II, Section H.5.b & I.2
		5.2.3	Part II, Section H.5.c
4.1	Part II, Section E.1 & J.7	5.2.4	Part II, Section H.5.d
4.2	Part II, Section I.2 & 3	5.2.5	Part II, Section H.6.b & 7
4.3	Part II, Section J.10.m	5.2.6	Part II, Section H.5.a & 6.a
4.3.1	Part II, Section E.6	5.3	Part II, Section J.1-5
4.3.2	Part II, Section J.8	5.4	Part II, Section L.1 & 2

Appendix 2: Station Letters of Agreement

1. LaSalle County Sheriff's Office - law enforcement
2. St. Mary's Hospital of Streator, Illinois - medical services
3. Morris Hospital in Morris, Illinois - medical services
4. Marseilles Rural Fire Department - fire protection
5. Seneca Fire Department - fire protection services
6. Seneca Ambulance - ambulance services

Attachment 6

EP-AA-1006

"Exelon Nuclear Standardized Radiological Emergency Plan Annex for Quad Cities Station"

Revision 25

Nuclear

EP-AA-1006
Revision 25

EXELON NUCLEAR

RADIOLOGICAL EMERGENCY PLAN ANNEX FOR QUAD CITIES STATION

Submitted: Kevin Appel Date: 10/10/07
Midwest Region Emergency Preparedness Manager

Authorized: Jim Meister Date: 10/12/07
Vice President – Operations Support

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APPENDIXES

REVISION HISTORY

Revision 0; 04/80	Revision 8; 09/94	Revision 8N; 01/98	Revision 21, 10/05
Revision 1; 07/80	Revision 8A; 01/95	Revision 8P; 07/98	Revision 22, 12/05
Revision 2; 04/81	Revision 8B; 03/95	Revision 9; 05/99	Revision 23, 04/06
Revision 3; 04/82	Revision 8C; 09/95	Revision 10; 01/00	Revision 24, 04/07
Revision 4; 04/83	Revision 8D; 12/93	Revision 11; 01/01	
Revision 5; 12/84	Revision 8E; 12/93	Revision 12; 10/01	
Revision 5A; 03/85	Revision 8F; 01/94	Revision 13; 10/01	
Revision 6; 03/86	Revision 8G; 04/94	Revision 14; 01/02	
Revision 7; 02/87	Revision 8H; 10/94	Revision 15; 07/02	
Revision 7A; 12/87	Revision 8I; 12/95	Revision 16; 09/02	
Revision 7B; 08/88	Revision 8J; 12/95	Revision 17; 06/03	
Revision 7C; 05/89	Revision 8K; 04/96	Revision 18; 08/03	
Revision 7D; 12/89	Revision 8L; 05/96	Revision 20, Canceled	

Section 1: Introduction

As required in the conditions set forth by the Nuclear Regulatory Commission (NRC) for the operating licenses for the Exelon Nuclear Stations, the management of Exelon recognizes its responsibility and authority to operate and maintain the nuclear power stations in such a manner as to provide for the safety of the general public.

The Exelon Emergency Preparedness Program consists of the Exelon Nuclear Standardized Emergency Plan (Emergency Plan), Station Annexes, emergency plan implementing procedures, and associated program administrative documents. The Emergency Plan outlines the basis for response actions that would be implemented in an emergency. Planning efforts common to all Exelon Nuclear stations are encompassed within the Emergency Plan.

This document serves as the Quad Cities Station Annex and contains information and guidance that is unique to the station. This includes Emergency Action Levels (EALs), and facility geography location for a full understanding and representation of the station's emergency response capabilities. The Station Annex is subject to the same review and audit requirements as the Emergency Plan.

1.1 Facility Description

The Quad Cities Station, Units 1 and 2, is located in Cordova Township of Rock Island County in northwestern Illinois. The station is located on the east bank of the Mississippi River three miles north of Cordova, Illinois. Cooling water for the plant is provided by the Mississippi River, with the water being returned to the river by diffuser pipes. The plant consists of two boiling water reactors (BWR), nuclear steam supply systems (NSSS), and turbine generators furnished by General Electric Company. The steam supply system is designed for a power output of 2957 MWt for each of the two units.

The Quad Cities Station area consists of approximately 126 acres (with a radius of about 1/4-mile about the Units 1/2 chimney) and is owned and controlled by Mid American Energy Company and Exelon Nuclear as tenants in common.

For more specific site location information, refer to the Updated Final Safety Analysis Report (UFSAR) for Quad Cities Station, Units 1 and 2.

1.2 Emergency Planning Zones

The Plume Exposure Emergency Planning Zone (EPZ) for Quad Cities Station is an area surrounding the station with a radius of about ten miles, (exact boundaries are determined by the States of Illinois and Iowa). Refer to Figure 1-1.

The Ingestion Pathway Emergency Planning Zone (EPZ) for Quad Cities Station is an area surrounding the Station with a radius of about 50 miles.

1.3 State of Iowa

Much of the Plume Exposure EPZ for the Quad Cities Station lies within the State of Iowa. The State of Iowa has developed an "Iowa Emergency Plan." This section provides a summary of the essential elements of the Iowa Emergency Plan, outlining the specific responsibilities of certain "key" Iowa State Agency players in a response operational mode. Basic descriptions for the Iowa State agencies responsible for actions in the event of a nuclear power station are as follows:

1.3.1 Iowa Emergency Management Division (IEMD)

IEMD coordinates all activities of State agencies and departments, all local governments, and the utility in support of emergency response activities. These activities are coordinated from the Iowa State EOC in Des Moines.

1.3.2 The Iowa Commissioner of Public Health

The Iowa Department of Public Health alerts the State Hygienic Lab when emergency action conditions are reported by a commercial nuclear power reactor, which impacts upon the public health and safety in Iowa, and when emergency team response has been determined to be necessary or imminent. They perform necessary calculations and evaluate the impact of existing and projected radioactivity releases in terms of public health risk. They translate the evaluation of existing and projected environmental contamination and resulting dose into terms of alternative protective actions. They recommend appropriate protective actions to the Governor's Office, IEMD and other State agencies as appropriate.

1.3.3 University Hygienic Lab (UHL)

The UHL, located in Iowa City, Iowa conducts and coordinates all field surveillance and monitoring activities directed toward measuring radiation exposure and radioactivity contamination in the environment resulting from an accident at a commercial nuclear power reactor; provides and coordinates laboratory support of all environmental sampling and radiological monitoring activities during a nuclear emergency; communicates all relevant data and protective action recommendations to the State Department of Public Health; provides radiological laboratory support for environmental samples analysis; and provides recommendations for decontamination of contaminated area.

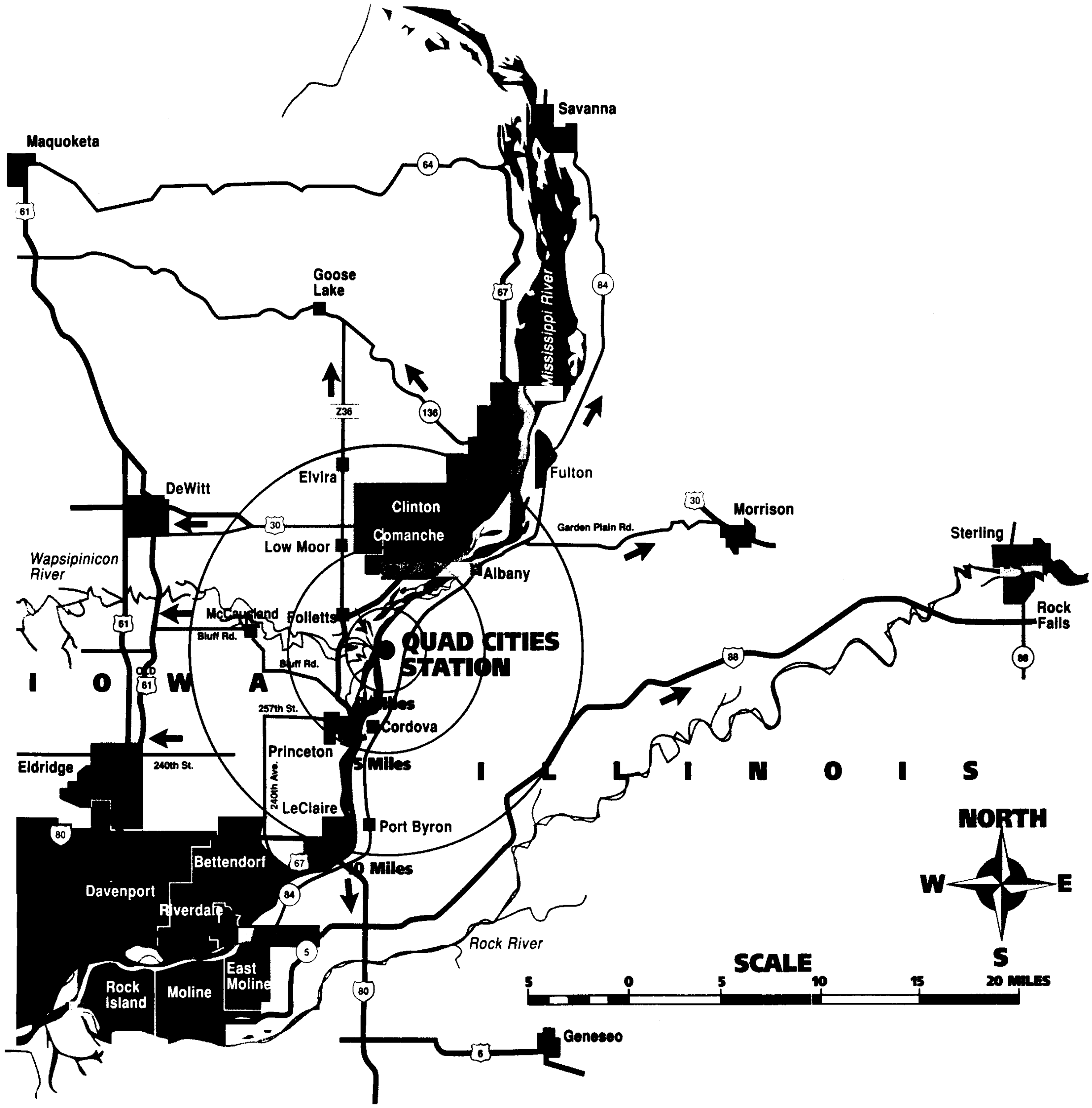
1.3.4 Clinton County

Clinton County will provide a coordinated local government response in conjunction with the State of Iowa, from the County Emergency Operations Center (EOC) in Clinton, IA.

1.3.5 Scott County

Scott County will provide a coordinated local government response in conjunction with the State of Iowa, from the County Emergency Operations Center (EOC) in Davenport, IA.

Figure 1-1: Quad Cities Station Location And 10 Mile EPZ



Section 2: Organizational Control of Emergencies

Initial response to any emergency is by the normal plant organization present at the site on a 24 hours per day basis as described in Section B.1 of the Emergency Plan.

Once an emergency is declared, the Emergency Response Organization (ERO) is activated as described in Section B.4 of the Emergency Plan.

2.1 Non-Exelon Nuclear Support Groups

Exelon Nuclear has contractual agreements with several companies whose services would be available in the event of a radiological emergency. These agencies and their available services are listed in Appendix 3 of the Emergency Plan.

Emergency response coordination with governmental agencies and other support organizations is discussed in Section A of the Emergency Plan.

Agreements exist on file at Quad Cities Station with several support agencies. These agencies and their support roles are listed in Appendix 2, Station Letters of Agreement.

Section 3: Classification of Emergencies

3.1 General

Section D of the Exelon Nuclear Standardized Emergency Plan divides the types of emergencies into four Emergency Classification Levels (ECLs). The first four are the UNUSUAL EVENT, ALERT, SITE AREA EMERGENCY, and GENERAL EMERGENCY. These ECLs are entered by meeting the Emergency Action Level (EAL) Threshold Values provided in this section of the Annex. 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. Depending on the severity of an event, prior to returning to a standard day-to-day organization, a state or phase called RECOVERY may be entered to provide dedicated resources and organization in support of restoration and communication activities following the termination of the emergency.

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 Guideline 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 acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

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

RECOVERY: Recovery can be considered as a phase of the emergency and is entered by meeting emergency termination criteria provided in EP-AA-111 Emergency Classification and Protective Action Recommendations.

An emergency is classified by assessing plant conditions and comparing abnormal conditions to Initiating Conditions and Threshold Values for each Emergency Action Level.

Individuals responsible for the classification of events will refer to the Initiating Condition and Threshold Values on the matrix of the appropriate station Standardized Emergency Plan Annex (this document). This matrix will contain Initiating Conditions, EAL Threshold Values, Mode Applicability Designators, appropriate EAL numbering system, and additional guidance necessary to classify events. It may be provided as a user aid.

The matrix is set up in four Recognition Categories. The first is designated as "R" and relates to Abnormal Radiological Conditions / Abnormal Radiological Effluent Releases. The second is designated as "F" and relates to Fission Product Barrier Degradation. The third is designated as "M" and relates to System Malfunctions. The fourth is designated as "H" and relates to Hazards and Other Conditions.

The matrix is designed to provide an evaluation of the Initiating Conditions from the worst conditions (General Emergencies) on the left to the relatively less severe conditions on the right (Unusual Events). Evaluating conditions from left to right will reduce the possibility that an event will be under classified. All Recognition Categories should be reviewed for applicability prior to classification.

The Initiating Conditions are coded with a two letter and one number code. The first letter is the Recognition Category designator, the second letter is the Classification Level, “U” for (NOTIFICATION OF) UNUSUAL EVENT, “A” for ALERT, “S” for SITE AREA EMERGENCY and “G” for GENERAL EMERGENCY. The EAL number is a sequential number for that Recognition Category series. All Initiating Conditions that are describing the severity of a common condition (series) will have the same number.

The EAL number may then be used to reference a corresponding page(s), which provides the basis information pertaining to the Initiating Condition:

- Threshold Value
- Mode Applicability
- Basis

Emergency Action Levels are the measurable, observable detailed conditions that must be met in order to classify the event. Classification is not to be made without referencing, comparing and satisfying the Threshold Values specified in the Emergency Action Levels.

A list of definitions is provided as part of this document for terms having specific meaning to the Emergency Action Levels. Site specific definitions are provided for terms with the intent to be used for a particular Initiating Condition/Threshold Value and may not be applicable to other uses of that term at other sites, the Emergency Plan or procedures.

References are also included to documents that were used to develop the EAL Threshold Values.

References to the Emergency Director means the person in Command and Control as defined in the Emergency Plan. Classification of emergencies is a non-delegable responsibility of Command and Control for the onsite facilities with responsibility assigned to the Shift Emergency Director (Control Room Shift Manager) or the Station Emergency Director (TSC). Classification of emergencies remains the responsibility of the applicable onsite facility even after Command and Control is transferred to the Corporate Emergency Director (EOF).

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

Indications used for monitoring and evaluation of plant conditions include the normally used instrumentation, backup or redundant instrumentation, and the use of other parameters that provide information that supports determination if an EAL threshold value has been reached. When an EAL refers to a specific instrument or indication that is determined to be inaccurate or unavailable, then alternate indications shall be used to monitor the specified condition.

During an event that results in changing parameters trending towards an EAL classification, and instrumentation that was available to monitor this parameter becomes unavailable or the parameter goes off scale, the parameter should be assumed to have been exceeded consistent with the trend and the classification made if there are no other direct or indirect means available to determine if the threshold has not been exceeded.

EALs are for unplanned events. A planned evolution involves 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. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL Threshold Value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

When two or more Emergency Action Levels are determined, declaration will be made on the highest classification level for the Unit. When both units are affected, the highest classification for the Station will be used for notification purposes and both units’ classification levels will be noted.

3.2 Mode Applicability

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

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 Fission Product Barrier Matrix EALs are applicable only to events that initiate in Hot Shutdown or higher.

If there is a change in 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.

3.3 Emergency Director Judgment

Emergency Director Judgment EALs are provided in the Hazards and Other Condition Affecting Plant Safety section and on the Fission Product Barrier Matrix. Both of the Emergency Director Judgment EALs have specific criteria for when they should be applied.

The Hazards Section Emergency Director Judgment EALs are intended to address unanticipated conditions which are not addressed explicitly by other EALs but warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under specific emergency classifications (UE, Alert, SAE or GE).

The FPB Matrix ED Judgment EALs are intended to include unanticipated conditions, which are not addressed explicitly by any of the other FPB threshold values, but warrant determination because conditions exist that fall under the broader definition for a significant Loss or Potential Loss of the barrier (equal to or greater than the defined FPB threshold values).

3.4 Fission Product Barrier Restoration

Fission Product Barriers (FPBs) are not treated the same as EAL threshold values. Conditions warranting declaration of the loss or potential loss of a Fission Product Barrier may occur resulting in a specific classification. The condition that caused the loss or potential loss declaration could be rectified as the result of Operator action, automatic actions, or designed plant response. Barriers will be considered re-established when there are direct verifiable indications (containment penetration or open valve has been isolated, coolant sample results, etc) that the barrier has been restored and is capable of mitigating future events.

The reestablishment of a fission product barrier does not alter or lower the existing classification. Entry into Termination/Recovery phase is still required for exiting the present classification. However the reestablishment of the barrier should be considered in determining future classifications should plant conditions or events change.

3.5 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 five or more persons violently protesting station operations or activities at the site.

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

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

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

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

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

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute 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, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

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.

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

NORMAL LEVELS: Normal levels can be considered as 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.

OPERATING MODES	REACTOR MODE SWITCH POSITION	TEMP
(1) Power Operation:	Run	N/A
(2) Startup:	Refuel [(a)] or Startup/Hot Standby	N/A
(3) Hot Shutdown [(a)]:	Shutdown	> 212° F
(4) Cold Shutdown [(a)]:	Shutdown	≤ 212° F
(5) Refueling [(b)]:	Shutdown or Refuel	N/A
(D) Defueled:	All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).	

[(a)] All reactor vessel head closure bolts fully tensioned.

[(b)] One or more reactor vessel head closure bolts less than fully tensioned.

Hot Matrix – applies in modes (1), (2), and (3)

Cold Matrix – applies in modes (4), (5), and (D)

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

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

SABOTAGE: A 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.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations > 10% Reactor Power change.

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 isolated from the Control Room.

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

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

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

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

Emergency Action Level Technical Basis Page Index

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs that exceeds or is expected to exceed **1.62 E+07 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs that exceeds or is expected to exceed **1.62 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on Radwaste effluent monitor 1/2-1799-01 or other monitor **> 200 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**. **OR** 2. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs is **> 8.34 E+05 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on Radwaste effluent monitor 1/2-1799-01 or other monitor **> 2 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**. **OR** 2. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs is **> 4.71 E+04 uCi/sec** for **≥ 60 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Abnormal Rad Levels		**Table R1** **Fuel Handling Incident Radiation Monitors** • 1(2) 1705-16A • 1(2) 1705-16B	**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1. **OR** 2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.	**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by: • Refueling Cavity water level < **417 in.** (Upper Wide Range normal) or **< 282 in**. (Upper Wide Range simulated signal). **OR** • Spent Fuel Pool water level **< 19 ft.** above the fuel (**– 4 ft.** indicated level). **OR** • Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal. **AND** b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1. **OR** 2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.
	Table R2 **Areas Requiring Continuous Occupancy** • Main Control Room (Unit 1 ARM Channel #22) • Central Alarm Station (by survey) • Secondary Alarm Station (by survey) • Radwaste Control Room (Unit 1 ARM Channel #27) • Main Access Facility (by survey)	**Table R3** **Areas Requiring Infrequent Access** • Refuel Floor • SBGT Floor • Reactor Building Third Floor • Reactor Building Second Floor • Reactor Building First Floor • Torus Area • HPCI Room • RCIC Room	**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions. **OR** 2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.	**RU3** Fuel clad degradation. 1 2 3 **EAL Threshold Values:** 1. Offgas system isolation due to VALID Offgas radiation monitor Hi-Hi trip. **OR** 2. Specific coolant activity **> 4.0 uCi/gm** Dose Equivalent I-131.

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

Fission Product Barrier Matrix **Hot Matrix**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
FG1 Loss of ANY Two Barriers AND Loss or Potential Loss of the third barrier. 1 2 3	**FS1** Loss or Potential Loss of ANY two barriers. 1 2 3	**FA1** ANY Loss or ANY Potential Loss of either Fuel Clad or RCS. 1 2 3	**FU1** ANY Loss or ANY Potential Loss of Containment. 1 2 3

Sub-Category	**FC - Fuel Clad**		**RC – Reactor Coolant System**		**CT - Containment**	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
1. RCS Activity →	Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.	None	None	None	None	None
2. RPV Water Level →	1. RPV level **< –166 in.** without at least one core spray loop **> 5050 gpm**. **OR** 2. RPV level **< –191 in.**	RPV level **< –142 in.** (TAF).	RPV level **< –142 in.** (TAF).	None	None	Plant conditions indicate that Primary Containment Flooding is required.
3. Drywell Pressure →	None	None	1. Drywell pressure **> 2.5 psig.** **AND** 2. Drywell pressure rise due to RCS leakage.	None	1. Rapid unexplained drop in Drywell pressure following initial pressure rise. **OR** 2. Drywell pressure response not consistent with LOCA conditions.	1. Drywell pressure **≥ 56 psig** and rising. **OR** 2. a. Drywell or torus hydrogen concentration **≥ 6%.** **AND** b. Drywell or torus oxygen concentration **≥ 5%.**
4. RCS Leakrate →	None	None	1. UNISOLABLE Main Steam Line (MSL) break as indicated by the failure of both MSIVs in ANY one line to close. **AND** 2. a. High MSL Flow AND High Steam Tunnel Temperature. **OR** b. Direct report of steam release.	1. RCS leakage **> 50 gpm** inside the drywell. **OR** 2. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> QGA 300, Maximum Normal** operating levels.		
5. Hi Cont/Drywell Rad →	Drywell radiation monitor reading **> Fuel Cladding Loss Threshold**, Table F1.	None	1. Drywell Radiation monitor reading **> 100 R/hr**. **AND** 2. Indications of RCS leakage into the Drywell.	None	None	Drywell radiation monitor reading **> Containment Potential Loss Threshold**, Table F2.
6. Breach/Bypass →			None	None	1. a. Failure of all isolation valves in any one line to close. **AND** b. Downstream pathway to the environment exists. **OR** 2. Intentional venting/purging of Primary Containment per EOPs or SAMGs due to accident conditions. **OR** 3. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> QGA 300, Maximum Safe** operating levels.	None
7. ED Judgment.→	Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Table F2 Drywell Radiation Thresholds

Time After Shutdown (hrs)	Containment Potential Loss (R/hr)
≤ 2	1.55 E+03
> 2 to 4	1.30 E+03
> 4 to 8	1.20 E+03
> 8 to 16	1.00 E+03
> 16 to 23	8.75 E+02
> 23	8.60 E+02

Table F1 Drywell Radiation Thresholds

Time After Shutdown (hrs)	Fuel Cladding Loss (R/hr)
≤ 2	6.65 E+02
> 2 to 4	5.90 E+02
> 4 to 8	5.05 E+02
> 8 to 16	4.25 E+02
> 16 to 23	3.85 E+02
> 23	3.75 E+02

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunctions				
AC Electrical Distribution	**MG1** Prolonged loss of all offsite power and prolonged loss of all onsite AC power to essential busses. 1 2 3 **EAL Threshold Values:** 1. Loss of power to Reserve Auxiliary Transformer TR-12 (TR-22) and Unit Auxiliary Transformer TR-11(TR-21). **AND** 2. Failure of Unit EDG 1(2), shared EDG 1/2, and SBO DG1(2) emergency diesel generators to supply power to unit ECCS busses. **AND** 3. a. Restoration of at least one unit ECCS bus within 1 hour is **not** likely. **OR** b. RPV level **cannot** be determined to be **> -142 in.** (TAF).	**MS1** Loss of all offsite power and loss of all onsite AC power to essential busses. 1 2 3 **EAL Threshold Values:** 1. Loss of power to Reserve Auxiliary Transformer TR-12 (TR-22) and Unit Auxiliary Transformer TR-11(TR-21). **AND** 2. Failure of Unit EDG 1(2), shared EDG 1/2, and SBO DG1(2) emergency diesel generators to supply power to unit ECCS busses. **AND** 3. Failure to restore power to at least one unit ECCS bus within **15 minutes** from the time of loss of both offsite and onsite AC power.	**MA1** AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout. 1 2 3 **EAL Threshold Values:** 1. AC power capability to unit ECCS busses reduced to only one of the following power sources for **> 15 minutes**: • Reserve Auxiliary Transformer TR-12 (TR-22) • Unit Auxiliary Transformer TR-11 (TR-21) • Unit Emergency Diesel Generator • Shared Emergency Diesel Generator • Unit crosstie breakers • SBO Diesel Generator **AND** 2. Any additional single power source failure will result in unit blackout.	**MU1** Loss of all offsite power to essential busses for greater than 15 minutes. 1 2 3 4 5 **EAL Threshold Values:** Loss of power to Reserve Auxiliary Transformer TR-12 (TR-22) **AND** Unit Auxiliary Transformer TR-11 (TR-21) for **> 15 minutes**.
RPS / Inadvertent Criticality	**MG3** Failure of the Reactor Protection System to complete an automatic scram and manual scram was NOT successful and there is indication of an extreme challenge to the ability to cool the core. 1 2 **EAL Threshold Values:** 1. Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**: a. Reactor power remains **> 5% APRM**. **OR** b. Torus temperature **> 110° F AND** boron injection required for reactivity control. **AND** 2. a. RPV level cannot be restored and maintained **> -166 in**. **OR** b. Heat Capacity Limit (QGA 200, Detail M) exceeded.	**MS3** Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded and manual scram was NOT successful. 1 2 **EAL Threshold Values:** Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**: 1. Reactor power remains **> 5% APRM**. **OR** 2. Torus temperature **> 110° F AND** boron injection required for reactivity control.	**MA3** Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded. 1 2 **EAL Threshold Values:** 1. A Reactor Protection System setpoint was exceeded. **AND** 2 Automatic scram did not reduce reactor power to **< IRM Range 7 and lowering**.	**MU3** Inadvertent criticality. 3 4 5 **EAL Threshold Values:** An UNPLANNED extended positive period observed on nuclear instrumentation.
DC Power		**MS4** Loss of all vital DC power. 1 2 3 **EAL Threshold Values:** Loss of all vital DC power based on **< 105 VDC** on 125 VDC battery busses #1 and #2 for **> 15 minutes**.		

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunctions				
Heat Sink		**MS5** Complete loss of heat removal capability. 1 2 3 **EAL Threshold Values:** Heat Capacity Limit (QGA 200, Detail M) exceeded.		
Annunciators		**MS6** Inability to monitor a SIGNIFICANT TRANSIENT in progress. 1 2 3 **EAL Threshold Values:** 1. Loss of most **(approximately 75%)** safety system annunciators (Table M2). **AND** 2. Indications needed to monitor safety functions (Table M3) are unavailable. **AND** 3. SIGNIFICANT TRANSIENT in progress (Table M4). **AND** 4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MA6** UNPLANNED Loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable. 1 2 3 **EAL Threshold Values:** 1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes.** **AND** 2. a. SIGNIFICANT TRANSIENT in progress (Table M4). **OR** b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.	**MU6** UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes. 1 2 3 **EAL Threshold Values:** 1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**. **OR** 2. UNPLANNED loss of most **(approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.
RCS Leakage				**MU7** RCS Leakage. 1 2 3 **EAL Threshold Values:** 1. Unidentified or pressure boundary leakage **> 10 gpm**. **OR** 2. Identified leakage **> 25 gpm**.

Table M2 - Control Room Panels	Table M3 - Safety Functions and Related Systems	Table M4 - Significant Transients
▪ 901(2)-3 ▪ 901(2)-5 ▪ 901(2)-8	• 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) • Fission Product Barriers	• Turbine trip • Reactor scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10%** Reactor Power change

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

<table>
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<tr><td>Communications</td><td></td><td></td><td>
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<tr><th colspan="3">Table M6 - Communications Capability</th></tr>
<tr><th>System</th><th>Onsite</th><th>Offsite</th></tr>
<tr><td>Plant Radio System</td><td>X</td><td></td></tr>
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<tr><td>Sound Power Phones</td><td>X</td><td></td></tr>
<tr><td>In-Plant Telephones</td><td>X</td><td></td></tr>
<tr><td>All Telephone Lines (commercial and microwave)</td><td></td><td>X</td></tr>
<tr><td>ENS</td><td></td><td>X</td></tr>
<tr><td>HPN</td><td></td><td>X</td></tr>
<tr><td>NARS</td><td></td><td>X</td></tr>
<tr><td>Cellular Phones</td><td></td><td>X</td></tr>
<tr><td>Satellite Phones</td><td></td><td>X</td></tr>
</table>
</td><td>MU10 UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5
<u>EAL Threshold Values:</u>
1. Loss of all Table M6 Onsite communications capability affecting the ability to perform routine operations.
OR
2. Loss of all Table M6 Offsite communications capability.</td></tr>
<tr><td>T. S. Time</td><td></td><td></td><td></td><td>MU11 Inability to reach required shutdown within Technical Specification limits. 1 2 3
<u>EAL Threshold Values:</u>
Plant is not brought to required operating mode within Technical Specifications LCO action statement time.</td></tr>
</table>

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, "Threat Assessment." **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 D **EAL Threshold Values:** 1. Control Room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per QOA 0010-05 in **< 30 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 D **EAL Threshold Values:** Entry into QOA 0010-05 for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** - Main Control Room - Reactor Building - Diesel Generator Rooms - 4 kV Switchgear Area - Battery Rooms - B-Train Control Room HVAC - Service Water Pumps - Turbine Building Cable Tunnel	**Table H3** **Internal Flooding Areas** - A RHR Room - B RHR Room - A Core Spray Room - B Core Spray Room - Torus Area - HPCI Area	**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by Strong Motion Seismograph output Alert **> 0.125 volts (0.10 g).** **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 100 mph** within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in any Table H3 area as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment. **OR** 6. a. High river water level **> 603 ft.** **OR** b. Low river water level **< 561 ft.**	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. Seismic event as indicated by any **TWO** of the following: • Earthquake felt in plant. • Seismic event confirmed by station seismic monitor procedure. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 100 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode. **OR** 6. River level transients potentially affecting safe operation of the plant: a. High river water level **> 594 ft.** **OR** b. Report of substantial reduction in river level by site personnel and confirmation by the Corps of Engineers that Lock and Dam # 14 has failed.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas		**Table H2** **Vital Areas** • Main Control Room • Reactor Building • Diesel Generator Rooms • 4 kV Switchgear Area • Battery Rooms • B-Train Control Room HVAC • Service Water Pumps • Turbine Building Cable Tunnel	**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 D **EAL Threshold Values:** 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.
ISFSI Events				**HU9** Damage to a loaded cask CONFINEMENT BOUNDARY. 1 2 3 4 5 D **EAL Threshold Values:** 1. Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY. **OR** 2. Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY. **OR** 3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask MPC CONFINEMENT BOUNDARY.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Radiological Effluents	**RG1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs that exceeds or is expected to exceed **1.62 E+07 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. **OR** b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**RS1** Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology. 1 2 3 4 5 D **EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs that exceeds or is expected to exceed **1.62 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid** **OR** 3. Field survey results at or beyond the site boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than 1 hour. **OR** b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**RA1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on Radwaste effluent monitor 1/2-1799-01 or other monitor **> 200 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**. **OR** 2. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs is **> 8.34 E+05 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.	**RU1** Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading on Radwaste effluent monitor 1/2-1799-01 or other monitor **> 2 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**. **OR** 2. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs is **> 4.71 E+04 uCi/sec** for **≥ 60 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate). **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Levels / Radiological Effluent				
Abnormal Rad Levels		**Table R1** **Fuel Handling Incident Radiation Monitors** • 1(2) 1705-16A • 1(2) 1705-16B	**RA2** Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1. **OR** 2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.	**RU2** Unexpected rise in plant radiation. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by: • Refueling Cavity water level < **417 in.** (Upper Wide Range normal) or **< 282 in**. (Upper Wide Range simulated signal). **OR** • Spent Fuel Pool water level **< 19 ft.** above the fuel (**– 4 ft.** indicated level). **OR** • Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal. **AND** b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R1. **OR** 2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.
	Table R2 **Areas Requiring Continuous Occupancy** • Main Control Room (Unit 1 ARM Channel #22) • Central Alarm Station (by survey) • Secondary Alarm Station (by survey) • Radwaste Control Room (Unit 1 ARM Channel #27) • Main Access Facility (by survey)	**Table R3** **Areas Requiring Infrequent Access** • Refuel Floor • SBGT Floor • Reactor Building Third Floor • Reactor Building Second Floor • Reactor Building First Floor • Torus Area • HPCI Room • RCIC Room	**RA3** Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions. **OR** 2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunctions				
AC Electrical Distribution				**MU1** Loss of all offsite power to essential busses for greater than 15 minutes. 1 2 3 4 5 **EAL Threshold Values:** Loss of power to Reserve Auxiliary Transformer TR-12(TR-22) **AND** Unit Auxiliary Transformer TR-11(TR-21) for **> 15 minutes**.
			MA2 Loss of all offsite power and loss of all onsite AC power to essential busses. 4 5 D **EAL Threshold Values:** 1. Loss of power to Reserve Auxiliary Transformer TR-12(TR-22) and Unit Auxiliary Transformer TR-11(TR-21). **AND** 2. Failure of Unit EDG 1(2), shared EDG 1/2, and SBO DG 1(2) emergency diesel generators to supply power to unit ECCS busses. **AND** 3. Failure to restore power to at least one unit ECCS bus within **15 minutes** from the time of loss of both offsite and onsite AC power.	
RPS				**MU3** Inadvertent criticality. 3 4 5 **EAL Threshold Values:** An UNPLANNED extended positive period observed on nuclear instrumentation.
DC Power				**MU4** UNPLANNED loss of required DC power for greater than 15 minutes. 4 5 **EAL Threshold Values:** 1. UNPLANNED loss of all required vital DC power based on **< 105 VDC** on 125 VDC battery busses #1 and #2. **AND** 2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.

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

<table>
<tr><th></th><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="5">System Malfunction</td></tr>
<tr>
<td>Heat Sink</td>
<td></td>
<td>
<table>
<tr><th colspan="3">Table M1 – RCS Reheat Duration Thresholds</th></tr>
<tr><th>RCS</th><th>Secondary Containment Closure</th><th>Duration</th></tr>
<tr><td>Intact</td><td>N/A</td><td>60 minutes*</td></tr>
<tr><td>Not Intact</td><td>Established</td><td>20 minutes*</td></tr>
<tr><td></td><td>Not Established</td><td>0 minutes</td></tr>
<tr><td colspan="3">*If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, then this EAL is <u>not</u> applicable.</td></tr>
</table>
</td>
<td>MA5 Inability to maintain plant in Cold Shutdown with irradiated fuel in the RPV. 4 5
<u>EAL Threshold Values:</u>
1. UNPLANNED loss of decay heat removal capability results in RCS temperature > 212° F for > Table M1 duration.
OR
2. UNPLANNED RPV pressure rise > 10 psig as a result of temperature rise due to loss of decay heat removal.</td>
<td>MU5 UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV. 4 5
<u>EAL Threshold Values:</u>
1. An UNPLANNED loss of decay heat removal capability results in RCS temperature > 212° F.
OR
2. Loss of all RCS temperature AND RPV level indication for > 15 minutes.</td>
</tr>
<tr>
<td>RCS Leakage / Inventory</td>
<td>MG8 Loss of RCS/RPV inventory affecting fuel clad integrity with Containment challenged with irradiated fuel in the RPV. 4 5
<u>EAL Threshold Values:</u>
1. Loss of RPV inventory per Table M5 indications.
AND
2. a. RPV level < - 142 in. (TAF) for > 30 minutes.
OR
b. RPV level unknown with indication of core uncovery for > 30 minutes as evidenced by one or more of the following:
• Fuel Handling ARM 1(2)-1705-16 A or B indicates > 3000 mR/hr. or off-scale high.
• Erratic Source Range Monitor indication.
AND
3. Containment is challenged as indicated by one or more of the following:
• Primary containment Hydrogen concentration ≥ 6%. and Oxygen concentration ≥ 5%.
• Drywell pressure ≥ 56 psig.
• Primary and Secondary CONTAINMENT CLOSURE not established.
• Any Secondary Containment radiation monitors > QGA 300, Maximum Safe operating level.</td>
<td>MS8 Loss of RCS/RPV inventory affecting core decay heat removal capability. 4
<u>EAL Threshold Values:</u>
1. <u>Without</u> Primary or Secondary CONTAINMENT CLOSURE established:
a. RPV level < - 65 in.
OR
b. RPV level unknown for > 30 minutes with a loss of RPV inventory per Table M5 indications.
OR
2. <u>With</u> Primary or Secondary CONTAINMENT CLOSURE established:
a. RPV level < - 142 in. (TAF).
OR
b. RPV level unknown for > 30 minutes with a loss of RPV inventory as evidenced by either of the following:
• Per Table M5 indications.
• Erratic Source Range Monitor indication.</td>
<td>MA8 Loss of RCS/RPV inventory with irradiated fuel in the RPV. 4 5
<u>EAL Threshold Values:</u>
1. Loss of RCS/RPV inventory as indicated by RPV level < - 59 in.
OR
2. a. Loss of RPV inventory per Table M5 indications.
AND
b. RCS/RPV level unknown for > 15 minutes.
<table>
<tr><th>Table M5 – Indications of RCS Leakage</th></tr>
<tr><td>• Unexplained floor or equipment sump level rise
• Unexplained Torus level rise
• Unexplained vessel make up rate rise
• Observation of leakage or inventory loss</td></tr>
</table>
</td>
<td>MU8 RCS leakage. 4
<u>EAL Threshold Values:</u>
RPV level cannot be restored and maintained > 0 in. (Low Level Scram Setpoint).</td>
</tr>
</table>

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

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

System Malfunction

RCS Leakage / Inventory

SITE AREA EMERGENCY

MS9 Loss of RPV inventory affecting core decay heat removal capability with irradiated fuel in the RPV. 5

EAL Threshold Values:

1. **Without** Secondary CONTAINMENT CLOSURE established:
    a. RPV level **< – 65 in.**
    OR
    b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
        - Fuel Handling ARM 1(2)-1705-16 A or B indicates **> 3000 mR/hr**. or off-scale high.
        - Erratic Source Range Monitor indication.

    OR
2. **With** Secondary CONTAINMENT CLOSURE established:
    a. RPV level **< – 142 in.** (TAF).
    OR
    b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
        - Fuel Handling ARM 1(2)-1705-16 A or B indicates **> 3000 mR/hr**. or off-scale high.
        - Erratic Source Range Monitor indication.

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Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise
• Unexplained Torus level rise
• Unexplained vessel make up rate rise
• Observation of leakage

UNUSUAL EVENT

MU9 UNPLANNED loss of RCS inventory with irradiated fuel in the RPV. 5

EAL Threshold Values:

1. UNPLANNED RPV level drop below the RPV flange for **≥ 15 minutes.**
    OR
2. a. Loss of RPV inventory per Table M5 indications.
    AND
    b. RPV level unknown.

Communications

ALERT

Table M6 - Communications Capability

System	Onsite	Offsite
Plant Radio System	X	
Plant Paging System	X	
Sound Power Phones	X	
In-Plant Telephones	X	
All Telephone Lines (commercial and microwave)		X
ENS		X
HPN		X
NARS		X
Cellular Phones		X
Satellite Phones		X

UNUSUAL EVENT

MU10 UNPLANNED loss of all onsite or offsite communications capabilities. 1 2 3 4 5

EAL Threshold Values:

1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations.
    OR
2. Loss of all Table M6 **Offsite** communications capability.

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

Hazards and Other Conditions Affecting Plant Safety

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Security	**HG1** Security event resulting in loss of physical control of the facility. 1 2 3 4 5 D **EAL Threshold Values:** A HOSTILE FORCE has taken control of: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1). **OR** 2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).	**HS1** Site attack. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.	**HA1** Notification of an airborne attack threat. 1 2 3 4 5 D **EAL Threshold Values:** A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.	**HU1** Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. A credible site-specific security threat notification as determined per SY-AA-101-132, “Threat Assessment”. **OR** 2. A validated notification from NRC providing information of an aircraft threat.
	Table H1 - Safety Functions and Related Systems • 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) • Fission Product Barriers		**HA2** Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA. 1 2 3 4 5 D **EAL Threshold Values:** A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.	
		HS3 Confirmed security event in a plant VITAL AREA 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.	**HA3** Confirmed security event in a plant PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.	**HU3** Confirmed security event which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 D **EAL Threshold Values:** Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.
C. R. Evacuation		**HS4** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 D **EAL Threshold Values:** 1. Control Room evacuation has been initiated. **AND** 2. Control of the plant **cannot** be established per QOA 0010-05 in **< 30 minutes**.	**HA4** Control Room evacuation has been initiated. 1 2 3 4 5 D **EAL Threshold Values:** Entry into QOA 0010-05 for Control Room evacuation.	

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Natural / Destructive Phenomena	**Table H2** **Vital Areas** • Main Control Room • Reactor Building • Diesel Generator Rooms • 4 kV Switchgear Area • Battery Rooms • B-Train Control Room HVAC • Service Water Pumps • Turbine Building Cable Tunnel	**Table H3** **Internal Flooding Areas** • A RHR Room • B RHR Room • A Core Spray Room • B Core Spray Room • Torus Area • HPCI Area	**HA5** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by Strong Motion Seismograph output Alert **> 0.125 volts (0.10 g).** **AND** b. Confirmed by **EITHER**: • Earthquake felt in plant. • National Earthquake Center. **OR** 2. Tornado or high winds **> 100 mph** within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems. **OR** 4. Turbine failure-generated missiles result in VISIBLE DAMAGE or penetration of any Table H2 area. **OR** 5. Uncontrolled flooding that results in **EITHER**: a. Degraded safety system performance in any Table H3 area as indicated in the Control Room. **OR** b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment. **OR** 6. a. High river water level **> 603 ft.** **OR** b. Low river water level **< 561 ft.**	**HU5** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 D **EAL Threshold Values:** 1. Seismic event as indicated by any **TWO** of the following: • Earthquake felt in plant. • Seismic event confirmed by station seismic monitor procedure. • National Earthquake Center. **OR** 2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 100 mph**, within PROTECTED AREA boundary. **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area. **OR** 4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** 5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode. **OR** 6. River level transients potentially affecting safe operation of the plant: a. High river water level **> 594 ft.** **OR** b. Report of substantial reduction in river level by site personnel and confirmation by the Corps of Engineers that Lock and Dam # 14 has failed.
Fire / Explosion			**HA6** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE or EXPLOSION in any Table H2 area. **AND** 2. a. Affected safety system parameter indications show degraded performance. **OR** b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.	**HU6** FIRE not extinguished within 15 minutes of detection, or EXPLOSION, within PROTECTED AREA boundary. 1 2 3 4 5 D **EAL Threshold Values:** 1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm. **OR** 3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

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

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards and Other Conditions Affecting Plant Safety				
Toxic / Flammable Gas		**Table H2** **Vital Areas** • Main Control Room • Reactor Building • Diesel Generator Rooms • 4 kV Switchgear Area • Battery Rooms • B-Train Control Room HVAC • Service Water Pumps • Turbine Building Cable Tunnel	**HA7** Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **OR** 2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).	**HU7** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 D **EAL Threshold Values:** 1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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.
Judgment	**HG8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an ALERT. 1 2 3 4 5 D **EAL Threshold Values:** Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU8** Other conditions existing which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. 1 2 3 4 5 D **EAL Threshold Values:** 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.
ISFSI Events				**HU9** Damage to a loaded cask CONFINEMENT BOUNDARY. 1 2 3 4 5 D **EAL Threshold Values:** 1. Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY. **OR** 2. Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY. **OR** 3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask MPC CONFINEMENT BOUNDARY.

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

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RG1

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs that exceeds or is expected to exceed **1.62 E+07 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 1000 mRem TEDE**

      OR

   b. **> 5000 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RG1 (cont)

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events that may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in a PAG dose being reached. Assumptions and DAPAR inputs are provided in calculation EP-EAL-0606.

The same value is used for ground level and elevated release points. An elevated release may not affect offsite areas as close to plant as ground level release; however, use of ground level values provides conservative estimates for exposure (cloud shine) to an overhead plume (EPA-400, section 5.6.1).

The sum of both units' monitors provides the total station release rates.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

Basis (cont):

Threshold #2 Basis:

The TEDE (1000 mRem) and the CDE Thyroid (5000 mRem) doses are set at the EPA PAG Limits.

The “site boundary” is defined by an approximately 800-meter (1/2-mile) radius around the plant. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release has been stopped and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AG1
2. EP-AA-112-500 Emergency Environmental Monitoring
3. Exelon DAPAR v. 3.1
4. EP-MW-110-200 Dose Assessment
5. EP-EAL-0606, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Quad Cities Station
6. QGA 400 Radioactivity Release Control

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RS1

Initiating Condition:

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the actual or projected duration of the release using actual meteorology.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results.

1. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs that exceeds or is expected to exceed **1.62 E+06 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

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

   a. **> 100 mRem TEDE**

      OR

   b. **> 500 mRem CDE Thyroid**

   OR

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

   a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour.

      OR

   b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RS1 (cont)

Basis:

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

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events that may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology. The effluent monitor readings have been determined with the DAPAR software program by calculating the monitor readings that would result in 10% of a PAG dose being reached. Assumptions and inputs are provided in EP-EAL-0606.

The same value is used for ground level and elevated release points. An elevated release may not affect offsite areas as close to plant as ground level release; however, use of ground level values provides conservative estimates for exposure (cloud shine) to an overhead plume (EPA-400, section 5.6.1).

The sum of both units' monitors provides the total station release rates.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of 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 EALs.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RS1 (cont)

Basis (cont):

Threshold #2 Basis:

The TEDE (100 mRem) and the CDE Thyroid (500 mRem) doses are set at 10% of the EPA PAG Limits.

The “site boundary” is defined by an approximately 800-meter (1/2-mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term “expected to continue for more than one hour” would not apply if:

- The release has been stopped and was less than one hour.

  OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AS1
2. EP-AA-112-500 Emergency Environmental Monitoring
3. Exelon DAPAR v. 3.1
4. EP-MW-110-200 Dose Assessment
5. EP-EAL-0606, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Quad Cities Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications for 15 minutes or longer.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading on Radwaste effluent monitor 1/2-1799-01 or other monitor **> 200 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 15 minutes**.

   OR

2. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs is **> 8.34 E+05 uCi/sec** for **≥ 15 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 200 times ODCM** Limit with a release duration of **≥ 15 minutes**.

Basis:

UNPLANNED: As used in this context, 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.

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

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

Basis (cont):

Threshold #1 Basis:

The threshold addresses radioactivity releases (liquid or gaseous) that for whatever reason cause effluent radiation monitor readings to exceed two hundred times the alarm setpoint established by the radioactive discharge permit. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the Offsite Dose Calculation Manual (ODCM) to warn of a release that is not in compliance with the Radiological Effluent Technical Specifications (RETS). Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

An elevated monitor reading while the effluent flow path is isolated is NOT considered to be a VALID reading.

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

The radwaste effluent monitor supplies signals to a one-pen recorder located in the radwaste building. An upscale trip or a downscale trip from this channel annunciates alarms both in the main control room and the radwaste building. The upscale trip provides a high radiation alarm and the downscale trip provides a separate monitor failure alarm, for both the control room and the RW control room. When an alarm is received on either the 912-5 (C-6 or F-6) panel in the Control Room or the 2212-4 (G-4) panel in the Radwaste Control Room, the operator actions are to stop the discharge. During a discharge, the Radwaste Control Panel is continuously occupied. The discharge to the river is stopped by shutting valve AO 1/2-2001-73.

Threshold #2 Basis:

QCNP incorporates 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 ODCM. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

This EAL addresses a potential or actual drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. However, the effluent monitor Alert value for gaseous effluents was reduced to a value one half way between the Unusual Event value and the Site Area Emergency Value to ensure sequential classifications. Calculation assumptions are provided in calc EP-EAL-0606. The sum of both gaseous effluent monitor readings provides a total station release rate because Unit 1 and Unit 2 discharge through the same monitors. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology as specified by the ODCM. The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA1 (cont)

Basis (cont):

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

Threshold #3 Basis:

Confirmed sample analyses in excess of two hundred times the site ODCM limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by increasing the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mRem/yr basis of the 10 CFR 20 limits for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be approximately 10 mRem/hr. The required release duration was reduced to 15 minutes in recognition of the increased severity.

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

Basis Reference(s):

1. NEI 99-01, Rev. 4 AA1
2. Sargent & Lundy calculation ATD-0224, Rev. 0, 1/12/93 and QC-16-88, Rev.2, 2/9/93
3. CY-QC-120-729 Liquid Effluent Monitor Alarm Setpoints
4. CY-QC-120-737 Radioactive Liquid Discharge Batch Analysis
5. CY-QC-110-602 Radwaste System Sampling
6. QOP 2000-24, Discharging to the River from the River Discharge Tank using the Waste Surge Pump
7. QOP 2000-25, Discharging to the River from the River Discharge Tank using the River Discharge Pump
8. CY-QC-120-729, Liquid Effluent Alarm Setpoints
9. QCOA 1700-02, High Radiation detected on Eberline Radiation Monitoring System
10. QCAN 912-5 C-6, Radwaste High Rad.
11. QCAN 901(2)-3-G-1, Liquid Process Rad. Monitor High Radiation
12. CY-QC-120-735, Main Chimney & Reactor Vent Noble Gas Release Rate Action Levels
13. QCOA 1700-01, Abnormal Chimney Radiation

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA1 (cont)

Basis Reference(s):

14. EP-EAL-0606, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Quad Cities Station
15. QGA 400 Radioactivity Release Control

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU1

Initiating Condition:

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications for 60 minutes or longer.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading on Radwaste effluent monitor 1/2-1799-01 or other monitor **> 2 times** the alarm setpoint established by a current radioactivity discharge permit for **≥ 60 minutes**.

   OR

2. The sum of VALID readings on the Rx Bldg Vent and Chimney SPINGs is **> 4.71 E+04 uCi/sec** for **≥ 60 minutes** (as determined from Control Room Panels or PPDS – Total Noble Gas Release Rate).

   OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

Basis:

UNPLANNED: As used in this context, 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.

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

The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU1 (cont)

Basis (cont):

Threshold #1 Basis:

The effluent release paths are monitored for radioactivity prior to the flow reaching the point where it would mix with the process flow to the environment. Prior to initiating batch releases, the discharge volume is sampled and analyzed for radioactivity. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate. Radiation monitor alarm setpoints are established to isolate the process flow at the point determined by the discharge permit. These limits are based on the Offsite Dose Calculation Manual ODCM.

An elevated monitor reading while the effluent flow path is isolated is NOT considered to be a VALID reading.

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

The radwaste effluent monitor supplies signals to a one-pen recorder located in the radwaste building. An upscale trip or a downscale trip from this channel annunciates alarms both in the main control room and the radwaste building. The upscale trip provides a high radiation alarm and the downscale trip provides a separate monitor failure alarm, for both the control room and the RW control room. When an alarm is received on either the 912-5 (C-6 or F-6) panel in the Control Room or the 2212-4 (G-4) panel in the Radwaste Control Room, the operator actions are to stop the discharge. During a discharge, the Radwaste Control Panel is continuously occupied. The discharge to the river is stopped by shutting valve AO 1/2-2001-73.

Threshold #2 Basis:

This EAL addresses a potential drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 2, regulatory commitments for an extended period of time. The sum of both gaseous effluent monitor readings provides a total station release rate because Unit 1 and Unit 2 discharge through the same monitors. The gaseous effluent value was determined using formulas, isotopic dose conversion factors and meteorology data as specified by the ODCM. Calculation EP-EAL-606 provides assumptions and calculation inputs.

The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

Basis (cont):

Threshold #3 Basis:

Confirmed sample analyses in excess of two times the site ODCM limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times ODCM for 30 minutes does not exceed this EAL. Grab samples are used to determine release concentrations or rates to confirm meter readings or when the effluent monitors are not in service. Maximum instantaneous release rate limits are calculated in accordance with ODCM. These are indicated on approved discharge permits.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AU1
2. Sargent & Lundy calculation ATD-0224, Rev. 0, 1/12/93 and QC-16-88, Rev. 2, 2/9/93
3. CY-QC-120-729 Liquid Effluent Monitor Alarm Setpoints
4. CY-QC-120-737 Radioactive Liquid Discharge Batch Analysis
5. CY-QC-110-602 Radwaste System Sampling
6. EP-EAL-0606, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values Quad Cities Station

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA2

Initiating Condition:

Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID reading **> 1000 mR/hr** on one or more of the radiation monitors in Table R1.

Table R1 - Fuel Handling Incident Radiation Monitors
• 1(2) 1705-16A • 1(2) 1705-16B

OR

2. Water level drop in the Reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.

Basis:

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

Threshold #1 Basis:

Uncovering spent fuel represents a substantial degradation of the level of safety of the plant and warrants an Alert classification. Time is available to take corrective actions. 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, no prompt fatalities are predicted and the risk of injury is low. 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 fuel uncovery begins. The value of 1000 mR/hr was conservatively chosen for classification purposes.

Radiation monitor readings are used to provide indication of fuel uncovery and/or fuel damage. High monitor readings associated with the transfer or relocation of a source, stored in or near the pool or readings responding to a planned evolution such as removal of the reactor head or equipment relocation are not classified under this threshold since the reading would not be indicative of fuel uncovery and/or fuel damage.

Dropping heavy loads onto the spent fuel can cause significant damage to the spent fuel and an Alert is also warranted under these conditions provided that the above radiation monitor threshold readings are reached.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA2 (cont)

Basis (cont):

Threshold #2 Basis:

When the RPV head is removed and the Upper Wide Range, LI-1(2)-263-101, instrument is calibrated to indicate levels as high as the refuel floor elevation, remote indication of Refueling Cavity water level is available in the Control Room. Once Spent Fuel Pool water level drops below the low level alarm setpoint (3 in. below normal level), further drops can be monitored only by visual observation unless the Spent Fuel Pool is in communication with the Refueling Cavity. Even so, uncovery of spent fuel seated in the Spent Fuel Pool storage racks cannot be monitored remotely because the bottom of the fuel transfer canal is above the elevation of the top of the storage racks. Any fuel that becomes uncovered while suspended from the refuel grapple may be indicated on the Upper Wide Range instrument but, without report of the vertical position of the grapple, fuel uncovery cannot be determined. Visual observation, therefore, provides the only viable mechanism of determining if spent fuel in the fuel pool or Refueling Cavity will be uncovered.

This EAL applies to irradiated fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AA2
2. QCOA 1900-01 Loss of Water Level in the Fuel Storage Pool or Reactor Cavity
3. QCAN 901(2)-3 B-1 Refuel Floor Hi Radiation
4. QCAN 901(2)-3 G-16/H-16 Fuel Pool Channel A/B Hi Radiation
5. QCIS 1700-07 Reactor Building Ventilation and Fuel Pool Radiation Monitoring Calibration and Functional Test

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU2

Initiating Condition:

Unexpected rise in plant radiation.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. a. VALID indication of uncontrolled water level drop in the reactor Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:

- Refueling Cavity water level **< 417 in.** (Upper Wide Range normal) or **< 282 in.** (Upper Wide Range simulated signal).

OR

- Spent Fuel Pool water level **< 19 ft.** above the fuel (**– 4 ft.** indicated level).

OR

- Report of visual observation of an uncontrolled drop in water level in the Refueling Cavity, Spent Fuel Pool or Fuel Transfer Canal.

AND

b. UNPLANNED VALID area radiation monitor reading rise on one or more radiation monitors in Table R1.

Table R1 - Fuel Handling Incident Radiation Monitors
• 1(2) 1705-16A • 1(2) 1705-16B

OR

2. UNPLANNED VALID Area Radiation Monitor reading rise by a factor of **1000** over NORMAL LEVELS.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU2 (cont)

Basis:

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

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

NORMAL LEVELS: Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Threshold #1 Basis:

During refueling when the RPV head is removed, Upper Wide Range, LI-1(2)-263-101, is recalibrated to indicate water level to the elevation of the refuel floor. A temporary reference leg signal is applied to the low side sensing line of LT 1(2)-263-61. The signal simulates one of two predetermined levels, based on the desired water level in the RPV or Refueling Cavity. These two elevations are:

- Elevation 680 ft. 2 in. el. – This is the elevation of the normal fill for the reference leg to LT-1(2)-263-61, and provides a direct readout of water level in the RPV from 42 inches below instrument zero in the RPV to approximately four feet above the vessel flange.
- Elevation 691 ft. 2 in. el. – This provides level monitoring in the range from the vessel flange to the level of the fuel pool for conditions when the Refueling cavity is flooded. When this reference level is used a correction factor of +135 inches must be added to the reading on Upper Wide Range and Process Computer point C-104(C-204) to obtain actual level.

The Refueling Cavity includes the fuel transfer canal. When the Refueling Cavity is flooded to normal level (689 ft. 6 in. el.), water level is approximately one foot below the refuel floor. Technical Specifications require Reactor Cavity water level be maintained at least 23 ft. above the top of the RPV flange. The threshold values correspond to 689 ft. 0 in. el.

Since no remote indication of Spent Fuel Pool water level exists, drops in Spent Fuel Pool water level can normally be detected only through visual observation.

Technical Specifications require the Spent Fuel Pool water level be maintained at least 19 ft. over the top of the irradiated fuel assemblies seated in the pool racks.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU2 (cont)

Basis (cont):

In addition, a local RPV pressure gage, as well as visual observation of level from the refueling floor, can be used to monitor water level when the RPV head is removed. Attachment A of QCOP 0201-13, Reactor Level Upper Wide Range Reference Leg Extension Use and Control, provides a cross-reference of indicated level to plant elevation.

Threshold #2 Basis:

Valid elevated area radiation levels usually have long lead times relative to the potential for radiological release beyond the site boundary, thus impact to public health and safety is very low.

This EAL addresses unplanned rise in radiation levels inside the plant. These radiation levels represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AU2
2. QCOP 0201-13 Reactor Level Upper Wide Range Reference Leg Extension Use and Control
3. Technical Specifications 3.7.8 Spent Fuel Storage Pool Water Level
4. Technical Specifications 3.9.6 Reactor Pressure Vessel (RPV) Water Level—Irradiated Fuel
5. QCAN 901(2)-4 B-24 FUEL POOL STORAGE HI/LO LEVEL
6. QCOA 1900-01 Loss of Water Level in the Fuel Storage Pool or Reactor Cavity

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA3

Initiating Condition:

Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. VALID radiation monitor or survey readings **> 15 mR/hr** in areas requiring continuous occupancy (Table R2) to maintain plant safety functions.

Table R2 – Areas Requiring Continuous Occupancy
• Main Control Room (Unit 1 ARM Channel #22)
• Central Alarm Station (by survey)
• Secondary Alarm Station (by survey)
• Radwaste Control Room (Unit 1 ARM Channel #27)
• Main Access Facility (by survey)

OR

2. VALID radiation monitor or survey readings **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.

Table R3 – Areas Requiring Infrequent Access
• Refuel Floor
• SBGT Floor
• Reactor Building Third Floor
• Reactor Building Second Floor
• Reactor Building First Floor
• Torus Area
• HPCI Room
• RCIC Room

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA3 (cont)

Basis:

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

Threshold #1 Basis:

This EAL addresses increased radiation levels that impede necessary access to operating stations requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. Areas requiring continuous occupancy include the Main Control Room, the central alarm station (CAS), the secondary security alarm station (SAS), the Radwaste Control Room, and the Main Access Facility. The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations.

The value of 15 mR/hr is derived from the General Design Criteria (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. A 30 day duration implies an event potentially more significant than an Alert.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EALs may be involved. For example, a dose rate of 15 mR/hr in the Main Control Room may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This EAL is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.).

Threshold #2 Basis:

This EAL addresses increased radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Typically areas requiring infrequent access to maintain plant safety functions include plant VITAL AREAS. Area radiation levels above 2000 mR/hr are indicative of radiation fields that may limit personnel access to equipment, the operation of which may be needed to assure adequate core cooling or shutdown the reactor.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RA3 (cont)

Basis (cont):

- The dose rate threshold selected is based on site administrative limits.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the rise in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved. For example, a dose rate of 2000 mR/hr may be a problem in itself. However, the rise may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.) or pre-existing radiation areas for which radiological controls already exist. The concern of this threshold is the unanticipated rise in radiation levels that results in unplanned restrictions to areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown.

Basis Reference(s):

1. NEI 99-01, Rev. 4 AA3
2. QCOP 1800-1 Operation of ARM Indicator/Trip Units
3. UFSAR Section 3.2
4. General Arrangement Drawings M-5, 6, 8 and 10

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU3

Initiating Condition:

Fuel clad degradation.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Offgas system isolation due to VALID Offgas radiation monitor Hi-Hi trip.

   OR

2. Specific coolant activity **> 4.0 uCi/gm** Dose Equivalent I-131.

Basis:

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

Threshold #1 Basis:

During unit operation, the steam jet air ejectors (SJAEs) remove all non-condensable gases from the main condenser including air in-leakage and disassociated products originating in the reactor and exhausts them to the offgas holdup volume. A rise in offgas activity could therefore indicate damage to the fuel cladding, a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

The gas from the main condenser normally includes relatively low levels of radioactivity. If radioactivity of the gas reaches the OFF GAS HIGH-HIGH RADIATION annunciator setpoint and the Offgas isolation timer is not reset, the Offgas system isolates (i.e. chimney isolation valve auto closes) after a fifteen-minute time delay. The fifteen-minute time delay is allotted for operator action to reduce the offgas radiation levels and exclude transient conditions.

The modifier “VALID” is appropriate because there are several conditions that may cause the monitor to alarm that are not related to fuel clad degradation and therefore should not result in the declaration of an Unusual Event.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENTS

RU3 (cont)

Basis (cont):

Threshold #2 Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding coolant Technical Specifications for iodine spiking. The specific iodine activity ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) accident is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits.

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 Reactor Coolant activity unless another cause is known, e.g., Reactor Coolant System chemical decontamination evolution (during shutdown) is ongoing with resulting high activity levels.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU4
2. Technical Specifications 3.4.6
3. Technical Specifications 3.7.6
4. QCAN 901(2)-3 C-2 OFF GAS HIGH-HIGH RADIATION

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FG1

Initiating Condition:

Loss of ANY two barriers AND Loss or Potential Loss of the third barrier.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the General Emergency classification level each barrier is weighted equally.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FS1

Initiating Condition:

Loss or Potential Loss of ANY two barriers.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Site Area Emergency classification level, each barrier is weighted equally.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FA1

Initiating Condition:

ANY Loss or ANY Potential Loss of either Fuel Clad or RCS.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

Fuel Cladding, RCS and Containment comprise the fission product barriers.

At the Alert classification level, 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FU1

Initiating Condition:

ANY Loss or ANY Potential Loss of Containment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Refer to Fission Product Barrier Loss and Potential Loss threshold values to determine barrier status.

Basis:

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 EAL 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 EAL FS1.

Basis Reference(s):

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

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC1 – Loss

Initiating Condition:

Primary coolant activity level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

<u>LOSS</u>

Coolant activity **> 300 uCi/gm** Dose Equivalent I-131.

Basis:

Loss Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

300 uCi/gm Dose Equivalent I-131 is well above that expected for iodine spikes and corresponds, generically, to about 2% to 5% fuel cladding damage. When reactor coolant activity reaches this level, significant clad damage has occurred and thus the Fuel Cladding barrier is considered lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. NF-AA-430, Failed Fuel Action Plan

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS
FC2 – Loss or Potential Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. RPV level **< – 166 in.** without at least one core spray loop **> 5050 gpm.**

   OR

2. RPV level **< – 191 in.**

POTENTIAL LOSS

RPV level **< – 142 in** (TAF).

Basis:

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

Loss Basis:

When primary containment flooding is required, all symptom-based EOPs (QGAs) are exited and the SAMGs are entered in order to restore and maintain cooling to the core and any core debris. Since it may not be possible to recover the core inside the RPV, flooding the primary containment to the elevation of the top of active fuel in the drywell may be required. EOPs/SAMGs require primary containment flooding when:

- QGA 100, RPV Control:
  - Cannot restore level above -166 in. and hold it there and neither Core Spray loop flow is at or above 5050 gpm.

    –166 in. is the Minimum Steam Cooling RPV Water Level (MSCRWL). At or above the MSCRWL, the covered portion of the core generates sufficient steam to prevent any cladding temperature in the uncovered part of the core from exceeding 1500° F (threshold temperature for fuel rod perforation).

    OR

  - Cannot restore level at or above -191 in. and hold it there.

    –191 in. is the top of the jet pump risers or ~2/3 core height. Water level held at or above this elevation with the specified design Core Spray loop flow satisfies the core cooling requirements of the LOCA event.
- QGA 101 RPV Control (ATWS), Failure to Scram: Cannot restore level above -166 in.
- QGA 500-4, RPV Flooding: Core damage is occurring.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC2 – Loss or Potential Loss

Basis (cont):

Potential Loss Basis:

Core submergence is the preferred method of core cooling and as such, the failure to re-establish RPV level above the top of active fuel for an extended period of time could lead to significant fuel damage.

An RPV level reading of -142 in. indicates RPV level is at the top of active fuel (TAF). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs (QGAs) specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. QGA 100 RPV Control
3. QGA 101 RPV Control (ATWS)
4. QGA 500-4 RPV Flooding

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC5 – Loss

Initiating Condition:

Drywell radiation monitoring.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Drywell radiation monitor reading **> Fuel Cladding Loss Threshold**, Table F1.

Table F1 – Drywell Radiation Thresholds

Time After Shutdown (hours)	Fuel Cladding Loss (R/hr)
≤ 2	6.65 E+02
> 2 to 4	5.90 E+02
> 4 to 8	5.05 E+02
> 8 to 16	4.25 E+02
> 16 to 23	3.85 E+02
> 23	3.75 E+02

Basis:

The drywell radiation monitor readings specified in Table F1 provide values that indicate the release of reactor coolant into the drywell with elevated activity indicative of fuel damage (~2%). The values are a function of time after shutdown and were derived using Core Damage Assessment Methodology (CDAM) with 2% clad damage, no drywell sprays in operation and a LOCA depressurized system. The reading is calculated assuming the instantaneous release and dispersal of the above reactor coolant noble gas and iodine inventory into the drywell atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations allowed within Technical Specifications (including iodine spiking) and are therefore indicative of fuel damage (approximately 2% - 5% cladding failure).

During at power (including ATWS) conditions the value listed for the "< 2 hours after shutdown" row is used as an indication of fuel damage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

FC7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.

Basis:

The Emergency Director judgment fuel cladding loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Fuel Cladding barrier. The inability to monitor fuel cladding integrity should also be considered as a factor in judging that the Fuel Cladding barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC2 – Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

RPV level **< –142 in**. (TAF).

Basis:

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

Loss Basis:

RPV level reading of -142 in. indicates RPV level is at the top of active fuel (TAF). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment barriers, and initiation of all ECCS. If RPV level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of any unplanned loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a loss of the RCS barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. QGA 100 RPV Control
3. QGA 500-4 RPV Flooding

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC3 – Loss

Initiating Condition:

Drywell pressure.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. Drywell pressure **> 2.5 psig**.

   AND

2. Drywell pressure rise due to RCS leakage.

Basis:

The drywell pressure value is the drywell high pressure ECCS initiation setpoint and is indicative of a Loss of Coolant Accident (LOCA) event that requires ECCS response. The drywell high pressure scram setpoint is an entry condition to QGA 100, RPV Control, and QGA 200, Primary Containment Control. Normal primary containment pressure control functions (e.g., operation of drywell cooling, Standby Gas Treatment system, etc.) are specified in QGA 200 in advance of less desirable but more effective functions (e.g., operation of drywell or torus sprays, etc.).

In the QCNS design basis, primary containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the increasing pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control primary containment vent/purge.

The second threshold focuses the fission product barrier loss threshold on a failure of the RCS instead of the non-LOCA malfunctions that may adversely affect primary containment pressure.

Therefore:

- Drywell pressure greater than 2.5 psig with corollary indications (drywell temperature, humidity) should therefore be considered a loss of RCS.
- Loss of drywell cooling that results in greater than 2.5 psig should not be considered a loss of RCS.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. QGA 100 RPV Control
3. QGA 200 Primary Containment Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC4 – Loss or Potential Loss

Initiating Condition:

RCS leak rate.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. UNISOLABLE Main Steam Line (MSL) break as indicated by the failure of both MSIVs in ANY one line to close.

   AND

2 a. High MSL Flow AND High Steam Tunnel Temperature.

   OR

   b. Direct report of steam release.

POTENTIAL LOSS

1. RCS leakage **> 50 gpm** inside the drywell.

   OR

2. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> QGA 300, Maximum Normal** operating levels.

Basis:

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Loss Basis:

High Steam Flow and High Steam Tunnel Temperature Annunciators are both indications of a Main Steam Line Break. Both of these parameters will cause a signal for closure of the MSIVs. Should both valves in any one line fail to isolate, this event would be considered a Loss of the RCS.

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside primary containment exists when flow is not prevented by downstream isolations. In the case of a failure of both Main Steam Isolation Valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required.

Direct report of steam release is meant to provide an alternate means of classification if the Hi Steam Flow Annunciator or the Hi Steam Tunnel Temperature Annunciator fails to operate and the observation of conditions indicates a Main Steam Line Break in the judgment of the Emergency Director. This is not meant to cause a declaration based on leaks such as valve packing leaks where the consequences offsite would be negligible.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC4 – Loss or Potential Loss (cont)

Basis (cont):

Potential Loss Threshold #1 Basis:

The potential loss of RCS based on leakage is set at a level indicative of a small breach of the RCS but which is well within the makeup capability of normal and emergency high-pressure systems. Core uncovery is not a significant concern for a 50 gpm leak; however, break propagation leading to significantly larger loss of inventory is possible. RCS leakage inside the drywell is normally determined by monitoring drywell equipment and floor drain sump pumpout rates. This method of monitoring leakage may be isolated as part of the drywell isolation, and thus may be unavailable. If primary system leak rate information is unavailable, other indicators of RCS leakage should be used. Inventory loss events, such as a stuck open SRV, should not be considered when referring to “RCS leakage” because they are not indications of a break, which could propagate.

Potential Loss Threshold #2 Basis:

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. The maximum normal values define this RCS threshold because it is the maximum normal operating value and signifies the onset of abnormal system operation. When parameters reach this level, equipment failure or misoperation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in QGA-300, Secondary Containment Control.

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the Reactor Enclosure since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the Reactor Enclosure, an unexpected rise in Feedwater flowrate, or unexpected Main Turbine Control Valve closure) may indicate that a primary system is discharging into the Reactor Building.

RC4 – Loss or Potential Loss (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. M-13 Main steam piping
3. UFSAR 5.2.5
4. QCOA 0201-01 Rev 016, Increasing Drywell Pressure
5. QOA 900-4 A-17 900-4 A-17 Annunciator
6. QCOS 1600-07 Reactor Coolant Leakage In The Drywell
7. QGA 300 Secondary Containment Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC5 – Loss

Initiating Condition:

Drywell Radiation Monitoring

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Drywell Radiation monitor reading **> 100 R/hr**.

   AND

2. Indications of RCS leakage into the Drywell.

Basis:

The drywell radiation monitor reading is a value that indicates a significant release of reactor coolant to the drywell. A reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the drywell atmosphere. Conservative estimates (high RCS uCi/cc) indicated that the readings from release of the normal RCS inventory would be ~ 100 R/hr. The reading is less than that specified for Fuel Cladding barrier Loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations.

Indication of a RCS leak into the drywell is added to qualify the radiation monitor indication to avoid declaring the loss of RCS barrier for situations where the radiation rise is not due to primary a system leak. For situations that involve failure of the Fuel Clad barrier alone, radiation monitor readings would rise due to shine and potentially giving a false indication of a loss of the RCS barrier. Therefore this EAL contains a qualifier to preclude over classification of the event if only fuel clad barrier failed.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Calc. EP-EAL-0611

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

RC7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.

POTENTIAL LOSS

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.

Basis:

The Emergency Director judgment RCS loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the RCS barrier. The inability to monitor RCS integrity should also be considered as a factor in judging that the RCS barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT2 – Potential Loss

Initiating Condition:

Reactor Vessel water level.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

POTENTIAL LOSS

Plant conditions indicate that Primary Containment Flooding is required.

Basis:

Potential Loss Basis:

When primary containment flooding is required, all EOPs (QGAs) are exited and the SAMGs are entered in order to restore and maintain cooling to the core and any core debris. Since it may not be possible to recover the core inside the RPV, flooding the primary containment to the elevation of the top of active fuel in the drywell may be required.

The EOP conditions requiring primary containment flooding represent imminent core melt sequences that, if not corrected, could lead to RPV failure and increased potential for containment failure.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. QGA 100 RPV Control
3. QGA 101 RPV Control (ATWS)
4. QGA 500-4 RPV Flooding

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT3 – Loss or Potential Loss

Initiating Condition:

Drywell pressure.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

1. Rapid unexplained drop in Drywell pressure following initial pressure rise.

   OR

2. Drywell pressure response not consistent with LOCA conditions.

POTENTIAL LOSS

1. Drywell pressure **≥ 56 psig** and rising.

   OR

2. a. Drywell or torus hydrogen concentration **≥ 6%.**

      AND

   b. Drywell or torus oxygen concentration **≥ 5%.**

Basis:

Loss Threshold #1 Basis:

Rapid unexplained loss of pressure (i.e., not attributable to drywell sprays, torus sprays or condensation effects) following an initial pressure rise indicates a loss of containment integrity.

Loss Threshold #2 Basis:

Drywell pressure should rise as a result of mass and energy release into the containment from a LOCA. Thus, drywell pressure response not consistent with LOCA conditions indicates a loss of containment integrity. This indicator relies on operator recognition of an unexpected response for the condition and therefore does not include a specific pressure value or trend. Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. For example, blowdown mass flowrate may be only 60-80% of the analyzed rate. The unexpected response is important because it is the indicator for a containment bypass condition.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT3 – Loss or Potential Loss (cont)

Basis (cont):

Potential Loss Threshold #1 Basis:

When the Primary Containment design pressure is challenged, primary containment venting is required even if offsite radioactivity release rate limits will be exceeded. This condition, if compounded by further plant degradation may challenge primary containment integrity and is, therefore, an appropriate threshold for potential loss of the Primary Containment barrier.

A Drywell pressure of 56 psig is based on the containment/drywell design pressure. If the containment design pressure is exceeded this represents a challenge to the containment structure because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a potential loss of the containment barrier even if a breach has NOT occurred.

Potential Loss Threshold #2 Basis:

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

Except for brief periods during plant startup and shutdown, oxygen concentration in the primary containment is maintained at insignificant levels by nitrogen inertion. The specified values for this potential loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen concentration) and readily recognizable because 6% hydrogen is above the hydrogen monitor alarm setpoint (2%) and the Primary Containment Control EOP entry condition. The minimum global deflagration hydrogen/oxygen concentrations (6% and 5%, respectively) require intentional primary containment venting, which is defined to be a barrier loss under Primary Containment barrier CT6.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. UFSAR Fig. 6.2-16a
3. UFSAR Section 15.6
4. UFSAR 6.2.1.1
5. QGA 200 Primary Containment Control
6. Quad Cities PSTG Section 5, Primary Containment Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT5 – Potential Loss

Initiating Condition:

Significant radioactive inventory in Containment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

POTENTIAL LOSS

Drywell radiation monitor reading **> Containment Potential Loss Threshold**, Table F2.

Table F2 – Drywell Radiation Thresholds

Time After Shutdown (hours)	Containment Potential Loss (R/hr)
≤ 2	1.55 E+03
> 2 to 4	1.30 E+03
> 4 to 8	1.20 E+03
> 8 to 16	1.00 E+03
> 16 to 23	8.75 E+02
> 23	8.60 E+02

Basis:

The drywell radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the Fuel Cladding barrier. NUREG-1228 “Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents” states that such readings do not exist when the amount of cladding damage is less than 20%. The values are a function of time after shutdown and were derived using Core Damage Assessment Methodology (CDAM) assuming 20% clad damage, no drywell sprays in operation and a LOCA depressurized system. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a significant failure into the reactor coolant has occurred.

During at power (including ATWS) conditions the value listed for the “< 2 hours after shutdown” row is used as an indication of fuel damage.

Regardless of whether the Primary Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Primary Containment barrier.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. Core Damage Assessment Methodology (CDAM version 1.1)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT6 – Loss

Initiating Condition:

Containment isolation failure or bypass.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

LOSS

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

   AND

   b. Downstream pathway to the environment exists.

   OR

2. Intentional venting/purging of Primary Containment per EOPs or SAMGs due to accident conditions.

   OR

3. UNISOLABLE primary system leakage outside drywell as indicated by Secondary Containment area temperatures or radiation levels **> QGA 300, Maximum Safe** operating levels.

Basis:

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

Threshold #1 Basis:

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway to the environment. The concern is the unisolable open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of primary containment integrity.

Failure of containment isolation valves to isolate with a downstream pathway to the environment is only a concern during an accident. If this condition exists during normal Power Operation, a Technical Specification Action Statement will address it. However, during accident conditions, this will represent a breach of Primary Containment.

Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include unisolable Main, HPCI or RCIC steamline breaks, unisolable RWCU system breaks, and unisloable containment atmosphere vent paths. Minor release paths such as instrument and sample lines are not considered under this threshold.

Examples of “downstream pathway to the Environment” could be through Turbine/Condenser, or direct release to the Turbine Enclosure or Reactor Enclosure.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS

CT6 – Loss (cont)

Basis (cont):

The breach is NOT isolable from the Control Room if an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation from the Control Room should be made prior to the accident classification. If operator actions from the Control Room are successful, then this IC is not applicable. Credit is NOT given for Operator actions taken in-plant (outside the Control Room) to isolate the leak.

This EAL is intended to cover containment isolation failures allowing a direct flow path to the environment such as failure of both MSIVs to close with open valves downstream to the turbine or to the condenser, even if these systems are not breached.

Threshold #2 Basis:

Intentional venting of the primary containment to the secondary containment and/or the environment per the EOPs/SAMG due to accident conditions is considered a loss of the Primary Containment barrier.

Threshold #3 Basis:

The presence of elevated general area temperatures and/or area radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. Temperatures and radiation levels beyond their maximum safe operating temperatures are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This EAL threshold addresses problematic discharges outside primary containment that may not originate from a high-energy line break.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2
2. QGA 200 Primary Containment Control
3. QGA 200-5 Hydrogen Control
4. QCOP 1600-13 Post-Accident Venting of the Primary Containment
5. QGA 300 Secondary Containment Control

RECOGNITION CATEGORY
FISSION PRODUCT BARRIERS
CT7 – Loss or Potential Loss

Initiating Condition:

Emergency Director judgment.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

<u>LOSS</u>

Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.

<u>POTENTIAL LOSS</u>

Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis:

The Emergency Director judgment Containment loss/potential loss addresses any event-specific factors that may be indicative of a loss or challenge to the Containment barrier. The inability to monitor Containment parameters should also be considered as a factor in judging that the Containment barrier may be considered lost or potentially lost.

Basis Reference(s):

1. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG1

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of power to Reserve Auxiliary Transformer TR-12(TR-22) and Unit Auxiliary Transformer TR-11(TR-21).

   AND

2. Failure of Unit EDG 1(2), shared EDG 1/2, and SBO DG1(2) emergency diesel generators to supply power to unit ECCS busses.

   AND

3. a. Restoration of at least one unit ECCS bus within 1 hour is **not** likely.

      OR

   b. RPV level **cannot** be determined to be **> – 142 in.** (TAF).

Basis:

Loss of all AC power to ECCS busses compromises the availability of all plant safety systems. Prolonged loss of all AC power may lead to loss of Fuel Cladding, RCS and Primary Containment barriers. The one-hour interval chosen to restore AC power to either unit ECCS bus is based on results of the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, “Station Blackout.”

The likelihood of restoring at least one ECCS bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

The ECCS busses 13-1(23-1) and 14-1(24-1) may be powered from any of the following onsite AC power sources:

- Unit emergency diesel generator – D/G1(2)
- Shared emergency diesel generator – D/G 1/2
- SBO Diesel Generator – SBO D/G 1(2)

Basis (cont):

Offsite AC power sources feed the ECCS busses through the Reserve Auxiliary Transformer TR-12(TR-22) and the Unit Auxiliary Transformer TR-11(TR-21). The ECCS busses of the affected unit can be powered from the unaffected unit through the crosstie breakers (13-1 to 23-1 or 14-1 to 24-1). The ECCS busses of an affected unit can also be backfed from the unaffected unit through the affected unit UAT. (Due to the time required to effect the backfeed, this source is likely only to be available when previously configured.) Unit crosstie and UAT backfeed are considered adequate sources of AC power when evaluating this EAL.

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

A reading of -142 in. below instrument zero indicates RPV level is at the top of active fuel (TAF). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops below TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG1 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SG1
2. UFSAR Figure 8.3-1
3. UFSAR Section 8.3
4. QCOA 6100-03 Loss of Offsite Power
5. QOP 6100-02 Restoring Reserve Auxiliary Transformer 12 To Service
6. QOP 6100-04 Restoring Reserve Auxiliary Transformer 22 To Service
7. QCOA 6100-04 Station Blackout
8. GE letter No. 92-38 from L.G. Knutson to Pat Donahue, dated April 7, 1992, “AC TURBINE LOADS SMALL TASK NO. QC107” (Station Blackout analysis)
9. QGA 100 RPV Control

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS1

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of power to Reserve Auxiliary Transformer TR-12(TR-22) and Unit Auxiliary Transformer TR-11(TR-21).

   AND

2. Failure of Unit EDG 1(2), shared EDG 1/2, and SBO DG1(2) emergency diesel generators to supply power to unit ECCS busses.

   AND

3. Failure to restore power to at least one unit ECCS bus within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power compromises all plant safety systems and represents failures of plant functions required for the protection of the public. The ECCS busses may be powered from any of the following onsite AC power sources:

- Unit emergency diesel generator – D/G 1(2)
- Shared emergency diesel generator – D/G 1/2
- SBO Diesel Generator – SBO D/G 1(2)

Offsite AC power sources feed the ECCS busses 13-1(23-1) and 14-1(24-1) through the Reserve Auxiliary Transformer TR-12(TR-22) and the Unit auxiliary Transformer TR-11(TR-21). The ECCS busses of the affected unit can be powered from the unaffected unit through the crosstie breakers (13-1 to 23-1 or 14-1 to 24-1). The ECCS busses of an affected unit can also be backfed from the unaffected unit through the affected unit UAT (due to the time required to effect the backfeed, this source is likely only to be available when previously configured). Unit crosstie and UAT backfeed are considered adequate sources of AC power when evaluating this EAL.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont):

Consideration should be given to available loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not available on the energized bus, the bus should not be considered available.

The fifteen-minute interval begins from the time of loss of both onsite and offsite AC power and was selected as a threshold to exclude transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS1
2. UFSAR Figure 8.3-1
3. UFSAR Section 8.3
4. QCOA 6100-03 Loss of Offsite Power
5. QOP 6100-02 Restoring Reserve Auxiliary Transformer 12 To Service
6. QOP 6100-04 Restoring Reserve Auxiliary Transformer 22 To Service
7. QCOA 6100-04 Station Blackout
8. GE letter No. 92-38 from L.G. Knutson to Pat Donahue, dated April 7, 1992, “AC TURBINE LOADS SMALL TASK NO. QC107” (Station Blackout analysis)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA1

Initiating Condition:

AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in unit blackout.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. AC power capability to unit ECCS busses reduced to only one of the following power sources for **> 15 minutes**:
    - Reserve Auxiliary Transformer TR-12(TR-22)
    - Unit Auxiliary Transformer TR-11(TR-21)
    - Unit Emergency Diesel Generator
    - Shared Emergency Diesel Generator
    - Unit crosstie breakers
    - SBO Diesel Generator

    AND

2. Any additional single power source failure will result in unit blackout.

Basis:

Capability: (pertaining to electrical power supplies) is equipment that is available to provide and maintain AC power at the required voltage and frequency for the required load.

The reduction of available reliable power sources to a condition in which any additional single failure will result in a Unit Blackout is a substantial degradation in the level of safety of the plant. A Unit Blackout is a loss of AC power to all unit ECCS busses. QCNP blackout coping duration is one hour.

The listed power supplies take into account sources that, if unavailable, establish single-failure vulnerability. This EAL allows for the use of the unit crosstie breakers if they are the only source of power to the affected unit. The Emergency Director must consider the use of the crosstie breakers and the consequent demand on the unaffected unit. The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA1 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA5
2. UFSAR Figure 8.3-1
3. UFSAR Section 8.3
4. QCOA 6100-03 Loss of Offsite Power
5. QOP 6100-02 Restoring Reserve Auxiliary Transformer 12 To Service
6. QOP 6100-04 Restoring Reserve Auxiliary Transformer 22 To Service
7. QCOA 6100-04 Station Blackout
8. GE letter No. 92-38 from L.G. Knutson to Pat Donahue, dated April 7, 1992, “AC TURBINE LOADS SMALL TASK NO. QC107” (Station Blackout analysis)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU1

Initiating Condition:

Loss of all offsite power to essential busses for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3, 4, 5

EAL Threshold Values:

Loss of power to Reserve Auxiliary Transformer TR-12(TR-22) **AND** Unit Auxiliary Transformer TR-11(TR-21) for **> 15 minutes**.

Basis:

The Essential busses are the safety-related, ECCS busses 13-1(23-1) and 14-1(24-1). The auxiliary power supply for Unit 1 is divided between the Unit Auxiliary Transformer (UAT) TR-11 connected to the Unit 1 generator and the Reserve Auxiliary Transformer (RAT) TR-12 connected to the 345-kV ring bus. The auxiliary power supply for Unit 2 is divided between UAT TR-21 connected to the Unit 2 generator and the Unit 2 RAT TR-22 connected to the 345-kV ring bus. The Unit 1 and 2 RATs are connected to the offsite network in the 345-kV switchyard. A 345-kV ring bus is interconnected with seven sources of power: Unit 1 and 2 outputs through their main transformers, and five 345-kV transmission lines.

Loss of offsite power causes a reactor scram and primary containment isolation. Emergency Diesel Generators (DG1, DG2 and DG1/2) should automatically start and be available to carry the essential loads for each affected unit. Balance of plant systems that would assist in plant operations (e.g., condensate pumps, etc.) may be unavailable due to the loss of power.

A loss of offsite AC power reduces the 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.

The intent of this EAL is to declare an Unusual Event when offsite power has been lost and the emergency diesel generators have successfully started and energized their respective ECCS busses. The fifteen-minute interval was selected as a threshold to exclude transient power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU1 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU1 & CU3
2. UFSAR Figure 8.3-1
3. UFSAR Section 8.3
4. QCOA 6100-03 Loss of Offsite Power
5. QOP 6100-02 Restoring Reserve Auxiliary Transformer 12 To Service
6. QOP 6100-04 Restoring Reserve Auxiliary Transformer 22 To Service
7. QCOA 6100-04 Station Blackout

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA2

Initiating Condition:

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability:

4, 5, D

EAL Threshold Values:

1. Loss of power to Reserve Auxiliary Transformer TR-12(TR-22) and Unit auxiliary Transformer TR-11(TR-21)

   AND

2. Failure of Unit EDG 1(2), Shared EDG 1/2, and SBO DG1(2) emergency diesel generators to supply power to unit ECCS busses.

   AND

3. Failure to restore power to at least one unit ECCS bus within **15 minutes** from the time of loss of both offsite and onsite AC power.

Basis:

The loss of all onsite and offsite AC power when in Cold Shutdown, Refueling or Defueled modes compromises safety systems required for decay heat removal and represents a substantial degradation of the level of safety of the plant. An Alert declaration (instead of a Site Area Emergency under EAL MS1) is appropriate in these modes because post-shutdown, decay heat energy levels offer more time to restore AC power to heat removal systems than the levels present when the reactor is in Power Operation, Startup or Hot Shutdown mode. Thus, the threat to the protection of the health and safety of the public is less severe.

The ECCS busses 13-1(23-1) and 14-1(24-1) may be powered from any of the following onsite AC power sources:

- Unit emergency diesel generator – D/G1(2)
- Shared emergency diesel generator – D/G 1/2
- SBO Diesel Generator – SBO D/G 1(2)

Offsite AC power sources feed the ECCS busses through the Reserve Auxiliary Transformer TR-12(TR-22) and the Unit auxiliary Transformer TR-11(TR-21). The ECCS busses of the affected unit can be powered from the unaffected unit through the crosstie breakers (13-1 to 23-1 or 14-1 to 24-1). The ECCS busses of an affected unit can also be backfed from the unaffected unit through the affected unit UAT. (Due to the time required to effect the backfeed, this source is likely only to be available when previously configured.) Unit crosstie and UAT backfeed are considered adequate sources of AC power when evaluating this EAL.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA2 (cont)

Basis (cont):

Consideration should be given to available loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of AC power to ECCS busses. Even though an ECCS bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not available on the energized bus, the bus should not be considered available.

The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA3
2. UFSAR Figure 8.3-1
3. UFSAR Section 8.3
4. QCOA 6100-03 Loss of Offsite Power
5. QOP 6100-02 Restoring Reserve Auxiliary Transformer 12 To Service
6. QOP 6100-04 Restoring Reserve Auxiliary Transformer 22 To Service
7. QCOA 6100-04 Station Blackout
8. GE letter No. 92-38 from L.G. Knutson to Pat Donahue, dated April 7, 1992, “ACTURBINE LOADS SMALL TASK NO. QC107” (Station Blackout analysis)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG3

Initiating Condition:

Failure of the Reactor Protection System to complete an automatic scram and manual scram was NOT successful and there is indication of an extreme challenge to the ability to cool the core.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**:
   a. Reactor power remains **> 5% APRM**.

   OR

   b. Torus temperature **> 110° F AND** boron injection required for reactivity control.

   AND

2. a. RPV level cannot be restored and maintained **> –166 in**.

   OR

   b. Heat Capacity Limit (QGA 200, Detail M) exceeded.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. The reactor power threshold (5%) is approximately equal to the APRM downscale setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 5% power. Classification at the General Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG3 (cont)

Basis (cont):

The torus water temperature criterion (110° F) is the Boron Injection Initiation Temperature (BIIT). The BIIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the torus heats the suppression pool to the Heat Capacity Limit (HCL). If torus temperature exceeds the BIIT, reactor power is heating the torus and the suppression pool cooling may be inadequate or incapable of performing its design function.

The second condition of this EAL indicates either:

- An extreme challenge to the ability to cool the core as indicated when RPV level cannot be restored and held above the Minimum Steam Cooling RPV Water Level, -166 in. or unknown. The Minimum Steam Cooling RPV Water Level (MSCRWL) is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500° F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core cooling could be a precursor of a core melt sequence.

  RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

- An extreme challenge to the primary containment as indicated when heat cannot be removed from the primary containment resulting in elevated torus temperature. The Heat Capacity Limit is the highest torus temperature from which a blowdown will not raise torus pressure above the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer from the RPV to the primary containment is within the capacity of the primary containment vent. When the PCPL is challenged, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded. The Heat Capacity Limit is a function of RPV pressure and torus temperature and is a measure of the maximum heat load that the primary containment can withstand. Plant parameters in excess of the Heat Capacity Limit could be a precursor of primary containment failure. The Heat Capacity Limit is given in Detail M in QGA 200, Primary Containment Control. Heat up of the torus to the Heat Capacity Limit signals the loss of functions required to maintain hot shutdown, including the ultimate heat sink. It also infers an RPV blowdown that could be caused by a loss of the RCS barrier. QGA 200 requires a blowdown per QGA 500-1 for either the inability to maintain temperature below the Heat Capacity Limit or for the inability to maintain the Torus level at > 11 ft. If compounded by further plant degradation, the event may challenge primary containment integrity. Torus level below 11 ft. in Modes 1, 2, or 3 means loss of pressure suppression, loss of the Torus as a heat sink and therefore, loss of heat removal capability.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis Reference(s):

1. NEI 99-01, Rev. 4 SG2
2. QGA 100 RPV Control
3. QGA 101 RPV Control (ATWS)
4. QGA 200 Primary Containment Control

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS3

Initiating Condition:

Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded and manual scram was NOT successful.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

Automatic scram, manual scram, and ARI were not successful from Reactor Console as indicated by **EITHER**:

1. Reactor power remains **> 5% APRM**.

   OR

2. Torus bulk temperature **> 110° F AND** boron injection required for reactivity control.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed The reactor power threshold (5%) is approximately equal to the APRM downscale setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 5% power.

The torus water temperature criterion is the Boron Injection Initiation Temperature (BIIT). The BIIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the torus heats the suppression pool to the Heat Capacity Limit (HCL).

Basis (cont):

If torus temperature exceeds the BIIT, reactor power is heating the torus and the suppression pool cooling may be inadequate or incapable of performing its design function.

Classification at the Site Area Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS2
2. QGA 100 RPV Control
3. QGA 101 RPV Control (ATWS)

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA3

Initiating Condition:

Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. A Reactor Protection System setpoint was exceeded.

   AND

2. Automatic scram did not reduce Reactor Power to **< IRM Range 7 and lowering**.

Basis:

This condition indicates a failure of the automatic reactor protection system to successfully scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. Site-specific indication of reactor shutdown is included as the criteria of whether the scram was successful when required. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue.

A successful scram has occurred when there is sufficient rod insertion to bring the reactor subcritical (< IRM Range 7 and lowering).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. For example, if reactor power is less than the lowered setpoint, then no automatic scram is initiated and this EAL is not applicable.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function.

The second condition of this EAL indicates a failure of the automatic RPS scram function to rapidly insert a sufficient number of control rods to achieve reactor shutdown. The CRD system backup scram valves and the Alternate Rod Insertion (ARI) system provide automatic, alternate methods of completing the scram function. These backups, however, insert control rods at a much slower rate than the automatic RPS scram function. For the purpose of emergency classification at the Alert level, reactor shutdown achieved by automatic backup scram valve operation and ARI initiation does not constitute a successful RPS automatic scram.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA3 (cont)

Basis (cont):

Following any automatic RPS scram signal QGA 101, RPV Control (ATWS), prescribes insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Alert.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded, or first-out annunciators), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA2
2. QGA 100 RPV Control
3. QGA 101 RPV Control (ATWS)
4. Technical Specifications Table 3.3.1.1-1
5. Technical Specification 3.3.1.3
6. Technical Specification Bases 3.3.1.1 and 3.3.1.3

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU3

Initiating Condition:

Inadvertent criticality.

Operating Mode Applicability:

3, 4, 5

EAL Threshold Values:

An UNPLANNED extended positive period observed on nuclear instrumentation.

Basis:

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

The term “extended” is used in order to allow exclusion of expected short-term positive periods from planned fuel bundle or control rod movements during core alteration. These short-term positive periods are the result of the rise in neutron population due to subcritical multiplication.

This EAL includes criticality events that occur in Cold Shutdown or Refueling modes (NUREG1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events as well as inadvertent criticalities occurring in Hot Shutdown mode. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification.

This condition can be identified using:

- SRM Channel 21, 22, 23 and 24 period meters on Panel 901(2)-5
- Amber short period lights on Panel 901(2)-5
- Annunciator 901(2)-5 E-5, SRM PERIOD SHORT (17 seconds).

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU8 & CU8
2. Technical Specifications B3.3.1
3. QCAN 901(2)-5 E-5 SRM Period Short

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS4

Initiating Condition:

Loss of all vital DC power.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Loss of all vital DC power based on **< 105 VDC** on 125 VDC battery busses #1 and #2 for **> 15 minutes**.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of Containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. Three station battery systems (250 VDC, 125 VDC and 24/48 VDC) are provided for each unit. A separate non-essential 250 VDC system is installed in Unit 1 and Unit 2 to provide service for non-safety-related loads. The 250-V “power” battery systems are provided to serve the larger loads such as DC motor-driven pumps, valves, etc. The 125 VDC “control” battery is provided to supply the power required for all DC control functions such as that required for control of 4160 VAC breakers, 480 VAC breakers, various control relays, annunciators, etc.

An alternate 125 VDC battery is provided to allow testing of the unit 125 VDC battery while both units remain at power. The alternate 125 VDC battery is also available upon the inoperability of the unit 125 VDC battery. Each 125 VDC bus receives power from either a 125 VDC battery or a 125 VDC battery charger.

The 125 VDC batteries are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit is 105 VDC.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS3
2. Technical Specifications 3.8.4 and B3.8.4
3. UFSAR Section 8.3.2
4. QOP 6900-02 125 VDC Electrical System
5. QCTS 0230-01 Unit One (Two) 125 VDC Service Test Normal or Alternate Battery

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU4

Initiating Condition:

UNPLANNED loss of required DC power for greater than 15 minutes.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. UNPLANNED loss of all required vital DC power based on **< 105 VDC** indication on 125 VDC battery busses #1 and #2.

   AND

2. Failure to restore power to at least one required DC bus within **15 minutes** from the time of loss.

Basis:

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

“Unplanned activities” is included in this EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, plants perform maintenance on a bus-related basis during shutdown periods.

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown, refueling or defueled operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

The intent of this EAL is to declare based on the loss of adequate voltage to both Division I and Division II busses on any unit. Failure of distribution busses on a given unit such that both Division I and Division II loads are lost satisfies this EAL.

Station batteries are provided as a final source of DC power for specific vital loads and control power. Three station battery systems (250 VDC, 125 VDC and 24/48 VDC) are provided for each unit. A separate non-essential 250 VDC system is installed in Unit 1 and Unit 2 to provide service for non-safety-related loads. The 250-V “power” battery systems are provided to serve the larger loads such as DC motor-driven pumps, valves, etc. The 125 VDC “control” battery is provided to supply the power required for all DC control functions such as that required for control of 4160 VAC breakers, 480 VAC breakers, various control relays, annunciators, etc.

An alternate 125 VDC battery is provided to allow testing of the unit 125 VDC battery while both units remain at power. The alternate 125 VDC battery is also available upon the inoperability of the unit 125 VDC battery. Each 125 VDC bus receives power from either a 125 VDC battery or a 125 VDC battery charger.

The 125 VDC batteries are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit is 105 VDC.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU7
2. Technical Specifications 3.8.4 and B3.8.4
3. UFSAR Section 8.3.2
4. QOP 6900-02 125 VDC Electrical System
5. QCTS 0230-01 Unit One (Two) 125 VDC Service Test Normal or Alternate Battery

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS5

Initiating Condition:

Complete loss of heat removal capability.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Heat Capacity Limit (QGA 200, Detail M) exceeded.

Basis:

Plant parameters associated with the Heat Capacity Limit (QGA 200 Detail M) are RPV pressure, torus level and torus temperature. The Heat Capacity Limit is the highest torus temperature from which a blowdown will not raise torus pressure above the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer from the RPV to the primary containment is within the capacity of the primary containment vent. When the PCPL is challenged, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded. If QGA actions to control torus temperature and RPV pressure below the Heat Capacity Limit are unsuccessful, RPV blowdown is required. The Heat Capacity Limit has been implemented as a single, bounding curve in the QGAs, valid for all torus levels at or below 17 ft.

Heat up of the torus to the Heat Capacity Limit signals the loss of functions required to maintain hot shutdown, including the ultimate heat sink. It also infers an RPV blowdown that could be caused by a loss of the RCS barrier. QGA 200 requires a blowdown per QGA 500-1 for either the inability to maintain temperature below the Heat Capacity Limit or for the inability to maintain the Torus level at > 11 ft. If compounded by further plant degradation, the event may challenge primary containment integrity. Torus level below 11 ft. in Modes 1, 2, or 3 means loss of pressure suppression, loss of the Torus as a heat sink and therefore, loss of heat removal capability.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS4
2. QGA 200 Primary Containment Control

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5

Initiating Condition:

Inability to maintain plant in Cold Shutdown with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 212° F** for > Table M1 duration.

Table M1 – RCS Reheat Duration Thresholds		
RCS	**Secondary Containment Closure**	**Duration**
Intact	N/A	60 minutes*
Not Intact	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. UNPLANNED RPV pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.

Basis:

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

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

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 main steam line nozzle plugs, etc.).

This EAL is based on concerns raised by Generic Letter 88-17, “Loss of Decay Heat Removal.” A number of phenomena such as decay heat removal system design and RPV level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5 (cont)

Basis (cont):

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

- TR 1(2)-260-11, 1(2)A and 1(2)B, Recirculation Loop Temperatures
- TR 1(2)-263-104, Rx Vessel Temperatures
- TR 1(2)-263-105, Rx Shell and Flange Temperatures

Threshold #1 Basis:

The first condition in Table M1 addresses complete loss of functions required for core cooling for greater than sixty minutes during Refueling and Cold Shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.). With secondary containment closure established, a low-pressure barrier to fission product release exists. In this threshold, containment status is of less importance than the status of RCS integrity because the RCS is intact and providing a high-pressure barrier to fission product release. The sixty-minute interval should allow sufficient time to restore cooling without a substantial degradation in plant safety. The asterisk highlights the note at the bottom of the table. The note indicates that the first threshold is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the sixty-minute interval.

The second condition in Table M1 addresses the complete loss of functions required for core cooling for greater than twenty minutes during Refueling and Cold Shutdown modes when secondary containment closure is established but RCS integrity is not established. RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.).

The allowed twenty-minute interval is 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 barrier to fission product release is established (i.e., secondary containment closure). The asterisk highlights the note at the bottom of the table. The note indicates that the second condition is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the twenty-minute interval.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5 (cont)

Basis (cont):

The third condition in Table M1 addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when secondary containment closure, and RCS integrity are not established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.). No delay time is allowed for this condition because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

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

Threshold #2 Basis:

The 10 psig pressure rise due to loss of decay heat removal infers an intact RCS with uncontrolled RPV temperature rise in excess of the Technical Specification cold shutdown limit (212° F) for which MA5 Threshold #1 would permit up to sixty minutes to restore RCS cooling before declaration of an Alert. This EAL therefore covers situations of high decay heat loads, in which the event should be declared without delay.

NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

RPV pressure is indicated by PI-1(2)-263-115 or computer point C109 (C209).

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA4
2. Technical Specifications Table 1.1-1
3. Technical Specifications 3.6.1.1
4. Technical Specifications 3.6.4.1
5. OU-AA-103 Shutdown Safety
6. QCOA 1000-02 Loss of Shutdown Cooling
7. QGA 100 RPV Control
8. QGA 100 RPV Control Detail A
9. QCGP 1-1 Normal Unit Startup
10. QCIS 0600-01 Unit One Division 1 Reactor Pressure 0 to 1200 psig Indication Calibration

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU5

Initiating Condition:

UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. An UNPLANNED loss of decay heat removal capability results in RCS temperature **> 212° F**.

   OR

2. Loss of all RCS temperature AND RPV level indication for **> 15 minutes**.

Basis:

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

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, 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. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the fuel cladding may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. 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 Reactor Vessel level so that escalation to the Alert under EAL MA5 will occur if required.

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in RCS/Reactor Vessel temperatures depending on the time since shutdown.

Basis (cont):

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 Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling modes, the second condition of this EAL would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication.

Reactor Vessel water level is normally monitored using the following instruments:

- Fuel Zone (-340 in. to +60 in.)
- Lower Wide Range (-344 in. to +66 in.)
- Medium Range (-60 in. to +60 in.)
- Upper Wide Range (-42 in. to +358 in.)
- Narrow Range (0 in. to +60 in.)

Detail A of QGA 100, Table C indicates when an instrument may be used for RPV level indication when EOPs are entered.

During shutdown conditions, Rx Water Level Upper Wide Range, LI-1(2)-263-101 located on panel 901(2)-4,is the primary instrument for monitoring RPV level. Procedures QCOP 0201-13 and 14 provide alternate level monitoring instrumentation when the normal, installed level instrumentation is unavailable for the desired level range or the head vent piping is removed and the RPV is fully vented to atmosphere. In order to make this instrument available during conditions when the RPV head piping is removed, a temporary reference leg signal is applied to the low side sensing line of LT 1(2)-263-61. The signal simulates one of two predetermined levels, based on the desired water level in the RPV or Refueling Cavity. These two elevations are:

- 680 ft. +/- 2 in. el. – This is the elevation of the normal fill for the reference leg to LT-1(2)-263-61, and provides a direct readout of water level from 42 inches below instrument zero in the RPV to approximately four feet above the vessel flange.
- 691 ft. +/- 2 in. el. – This provides level monitoring in the range from the vessel flange to the level of the fuel pool for conditions when the Refueling Cavity is flooded. When this reference level is used a correction factor of +135 inches must be added to the reading from Rx Water Level Upper Wide Range and Process Computer point C-104(C-204) to obtain actual level.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont):

In addition, a local RPV pressure gage as well as visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed. Attachment A of QCOP 0201-13, Reactor Level Upper Wide Range Reference Leg Extension Use and Control, provides a cross-reference of indicated level to plant elevation.

Several instruments and computer points are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212° F), such as:

- TR 1(2)-260-11, 1(2)A and 1(2)B, Recirculation Loop Temperatures
- TR-1(2)-263-104, Rx Vessel Temperatures
- TR 1(2)-263-105, Rx Shell and Flange Temperatures

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU4
2. Technical Specifications Table 1.1-1
3. QGA 100, RPV Control
4. QCOP 0201-02, Filling the Reactor Vessel and/or Reactor Cavity Using a Condensate Booster Pump via the Feedwater System
5. QCOP 0201-13, Reactor Vessel Upper Wide Range Reference Leg Extension Use and Control
6. QCOP 0201-14, Reactor Vessel Level Control Using a Local Pressure Gauge
7. QCOA 1000-02 Loss of Shutdown Cooling

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6

Initiating Condition:

Inability to monitor a SIGNIFICANT TRANSIENT in progress.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. Loss of most **(approximately 75%)** safety system annunciators (Table M2).

Table M2 – Control Room Panels
• 901(2)-3 • 901(2)-5 • 901(2)-8

AND

2. Indications needed to monitor safety functions (Table M3) are unavailable.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

AND

3. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Turbine trip • Reactor scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10 %** Reactor Power change

AND

4. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

Basis:

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6 (cont)

Basis (cont):

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations > 10% Reactor Power change.

Planned and unplanned actions are not differentiated since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not a factor.

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in a 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.

A Site Area Emergency exists if the Control Room staff cannot monitor safety functions needed for protection of the public. 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 parameters should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled and in a coolable geometry, remove heat from the core, and maintain the reactor coolant system and containment intact. These parameters are monitored and controlled in the symptom-based emergency operating procedures (QGAs).

Symptoms of a loss of annunciators can be:

- ALARM POTENTIAL FAILURE or ANNUNCIATOR DC POWER FAILURE alarms on one or more panels
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

QCOA 0900-01, Loss of Annunciators, provides instructions for restoring annunciators and, for a extended loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SS6
2. QGA 100 RPV Control
3. QGA 200 Primary Containment Control
4. QCOA 0900-01 Loss of Annunciators
5. QOP 9900-101 Process Computer
6. QOP 9900-102 Operation of Safety Parameter Display System

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) COMPENSATORY NON-ALARMING INDICATIONS are unavailable.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. a. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 901(2)-3 • 901(2)-5 • 901(2)-8

OR

b. UNPLANNED loss of most **(approximately 75%)** indications associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

AND

2. a. SIGNIFICANT TRANSIENT in progress (Table M4).

Table M4 - Significant Transients
• Turbine trip • Reactor scram • ECCS actuation • Recirc. Runback **> 25%** Reactor Power change • Thermal power oscillations **> 10 %** Reactor Power change

OR

b. COMPENSATORY NON-ALARMING INDICATIONS (computer points) are unavailable.

Basis:

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

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) Turbine Trip (2) Reactor Scram (3) ECCS Activation, (4) Recirc. Runback > 25% Reactor Power change, or (5) thermal power oscillations > 10% Reactor Power change

COMPENSATORY NON-ALARMING INDICATIONS: Process Computer, SPDS, and PPDS.

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in a 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the Reactor Control, ECCS, and Electrical panel annunciation or indication equipment. The availability of computer based indication equipment is considered.

Symptoms of a loss of annunciators can be:

- ALARM POTENTIAL FAILURE or ANNUNCIATOR DC POWER FAILURE alarms on one or more panels
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

QCOA 0900-01, Loss of Annunciators, provides instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The fifteen-minute interval offers time to recover from transient or momentary power losses.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SA4
2. QCOA 0900-01 Loss of Annunciators
3. QOP 9900-101 Process Computer
4. QOP 9900-102 Operation of Safety Parameter Display System

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU6

Initiating Condition:

UNPLANNED loss of most or all safety system annunciation or indication in the Control Room for greater than 15 minutes.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

1. UNPLANNED loss of most **(approximately 75%)** safety system annunciators (Table M2) for **> 15 minutes**.

Table M2 – Control Room Panels
• 901(2)-3 • 901(2)-5 • 901(2)-8

OR

2. UNPLANNED loss of most **(approximately 75%)** indicators associated with safety functions (Table M3) for **> 15 minutes**.

Table M3 – Safety Functions and Related Systems
• 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) • Fission Product Barriers

Basis:

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

For this EAL “most” is approximately 75% of the safety system annunciators or indicators being lost, resulting in a 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.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the Reactor Control, ECCS, and Electrical panel annunciation or indication equipment. The availability of computer based indication equipment is considered.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU6 (cont)

Basis (cont):

Symptoms of a loss of annunciators can be:

- ALARM POTENTIAL FAILURE or ANNUNCIATOR DC POWER FAILURE alarms on one or more panels
- Failure of annunciator test
- Loss of annunciator horn
- Loss of Sequence of Events Recorder monitor

QCOA 0900-01, Loss of Annunciators, provides instructions for restoring annunciators and, for a sustained loss of annunciators, increased plant monitoring at a frequency determined by the Unit Supervisor.

While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification.

The fifteen-minute interval offers time to recover from transient or momentary power losses.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU3
2. QCOA 0900-01 Loss of Annunciators
3. QOP 9900-101 Process Computer
4. QOP 9900-102 Operation of Safety Parameter Display System

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU7

Initiating Condition:

RCS Leakage.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

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

   OR

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

Basis:

The conditions of this EAL threshold may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. Only leakage inside the drywell qualifies toward exceeding Threshold #1. Various indications may be used to identify or verify potential leakage from the RCS into the drywell. They include drywell sump flow indications, drywell temperature and pressure changes, drywell air cooler cooling water differential temperature changes, and drywell atmosphere activity level changes. QCOS 1600-07 Reactor Coolant Leakage in the Drywell, provides direction for determining RCS leakage.

The 10 gpm value for unidentified leakage was selected because it is observable with normal Control Room measurement of sump pumpout rates. It is consistent with the Technical Specification threshold for leaks beyond which increased risk of crack propagation exists.

The 25 gpm value for identified leakage is set at a higher value because of the significance of identified leakage in comparison to unidentified or pressure boundary leakage.

No classification under this threshold is made for relief valve operation or leakage.

Both threshold values are observable on Control Room instrumentation and do not require a mass balance calculation.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU5
2. QCOS 1600-07 Reactor Coolant Leakage in the Drywell
3. Technical Specifications 3.4.4
4. UFSAR 5.2.5
5. QCOA 0201-01 Increasing Drywell Pressure

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8

Initiating Condition:

Loss of RCS/RPV inventory affecting fuel clad integrity with Containment challenged with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Torus level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

AND

2. a. RPV level **< –142 in**. (TAF) for **> 30 minutes**.

   OR

   b. RPV level unknown with indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following:

   - Fuel Handling ARM 1(2)-1705-16 A or B indicates **> 3000 mR/hr** or off-scale high.
   - Erratic Source Range Monitor indication.

AND

3. Containment is challenged as indicated by one or more of the following:

- Primary containment Hydrogen concentration ≥ **6%** and Oxygen concentration ≥ **5%.**
- Drywell pressure ≥ **56 psig.**
- Primary and Secondary CONTAINMENT CLOSURE not established.
- Any Secondary Containment radiation monitor **> QGA 300, Maximum Safe** operating level.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG8 (cont)

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

Threshold #1 and #2 Basis:

This EAL represents the inability to restore and maintain RPV level to above the top of active fuel, -142 in. (TAF). Fuel damage is probable if core uncovery is prolonged and submergence cannot be restored and maintained. Available decay heat will cause boiling and further drop RPV level.

This EAL is 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, (e.g., decay heat removal system design, etc.) can have a significant affect on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

When RPV level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated drywell radiation or unexplained rise in drywell floor or equipment drain sump pumpout rate. Detail A of QGA 100, Table C provides guidance on determining if RPV level can be monitored. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors (SRM CH 21, CH 22, CH 23, or CH 24) can be used as a tool for making such determinations.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The Fuel Handling Area Radiation Monitors ARM 1(2) 1705-16A or B indication of > 3000 mR/hr. is based on calculation EP-EAL-0501.

Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

Threshold #3 Basis:

Four conditions are associated with the challenge to containment integrity:

- With Primary or Secondary containment closure not established with prolonged core uncovery, the health and safety of the public may be threatened.
- When hydrogen and oxygen concentrations in primary containment reach or exceed the deflagration limits, imminent loss of the Primary Containment barrier exists. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont):

- The secondary containment area radiation level is the QGA Maximum Safe Operating level. The Maximum Safe Operating radiation level is based on the highest radiation level at which neither equipment necessary for the safe shutdown of the plant will fail nor personnel access necessary for the safe shutdown of the plant will be precluded.
- The primary containment design pressure (56 psig) is well in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CG1
2. QGA 100, RPV Control
3. Technical Specifications 3.3.1
4. Technical Specifications 3.6.1.1
5. Technical Specifications 3.6.4.1
6. QGA-200-5, Hydrogen Control
7. UFSAR 6.2.1.1
8. QGA 300 Secondary Containment Control
9. EP-EAL-0501, Estimation Of Radiation Monitor Readings Indicating Core Uncovery During Refueling

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS8

Initiating Condition:

Loss of RCS/RPV inventory affecting core decay heat removal capability.

Operating Mode Applicability:

4

EAL Threshold Values:

1. **Without** Primary or Secondary CONTAINMENT CLOSURE established:
    a. RPV level **< – 65 in.**

    OR

    b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Torus level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

OR

2. **With** Primary or Secondary CONTAINMENT CLOSURE established:
    a. RPV level **< –142 in.** (TAF).

    OR

    b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory as evidenced by either of the following:
    - Per Table M5 indications.
    - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS8 (cont)

Basis (cont):

Threshold #1 Basis:

Under the conditions specified by this threshold, continued drop in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV. If a low-pressure boundary to fission product release does not exist (i.e., primary or secondary containment closure is not established), the RPV level associated with this threshold is six inches below the low-pressure ECCS actuation setpoint (i.e., -59 in. - 6 in. = -65 in.). If primary or secondary containment closure is established, a low-pressure boundary to fission product release exists and RPV level can drop to the top of active fuel, -142 in. (TAF), before a Site Area Emergency declaration is required. 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 drop and potential core uncovery. In Cold Shutdown, the decay heat available to raise RPV temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding 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.

Threshold #2 Basis:

This threshold is 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, (e.g., decay heat removal system design, etc.) can have a significant impact on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

The thirty-minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

When RPV level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated drywell radiation or unexplained rise in drywell floor or equipment drain sump pumpout rate. Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage. Detail A of QGA 100, Table C provides guidance on determining if RPV level can be monitored. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors (SRM CH 21, CH 22, CH 23, or CH 24) can be used as a tool for making such determinations.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis Reference(s):

1. NEI 99-01, Rev. 4 CS1
2. Technical Specifications 3.3.5.1
3. Technical Specifications 3.6.1.1
4. Technical Specifications 3.6.4.1
5. QGA 100, RPV Control
6. Technical Specifications 3.3.1
7. Technical Specifications Table 3.3.3.1-1
8. Technical Specifications 3.3.5.1
9. QCOS 1600-07, Reactor Coolant Leakage in the Drywell
10. Technical Specifications 3.4.4
11. UFSAR 5.2.5
12. QCOA 0201-01, Increasing Drywell Pressure
13. QOA 900-4 A-17, Annuciator Response

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8

Initiating Condition:

Loss of RCS/RPV inventory with irradiated fuel in the RPV.

Operating Mode Applicability:

4, 5

EAL Threshold Values:

1. Loss of RCS/RPV inventory as indicated by RPV level **< – 59 in**.

OR

2. a. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise • Unexplained Torus level rise • Unexplained vessel make up rate rise • Observation of leakage or inventory loss

AND

b. RCS/RPV level unknown for **> 15 minutes.**

Basis:

This EAL serves as a precursor 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 drop and potential core uncovery.

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

The low-pressure ECCS actuation setpoint is 59 in. below RPV instrument zero. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

In Cold Shutdown mode, the decay heat available to raise RPV temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel cladding 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.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8 (cont)

Basis (cont):

In Cold Shutdown mode, the RCS will normally be intact and standard RPV inventory and RPV level monitoring means are available. In the Refueling mode, the RCS is not intact and RPV level and inventory are monitored by different means. In the Refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

RPV level is normally monitored using the following instruments:

- Fuel Zone (-340 in. to +60 in.)
- Lower wide Range (-344 in. to +66 in.)
- Medium Range (-60 in. to +60 in.)
- Upper Wide Range (-42 in. to +358 in.)
- Narrow Range (0 in. to +60 in.)

In the second condition of this EAL, all RPV level indication would be unavailable. Detail A of QGA 100, Table C provides guidance on determining if RPV level can be monitored. RPV inventory loss, therefore, must be detected by alternate means (i.e., drywell floor and equipment drain sump pumpout rates). Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency duration.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CA1 & CA2
2. Technical Specifications 3.3.5.1
3. QCOS 1600-07, Reactor Coolant Leakage in the Drywell
4. Technical Specifications 3.4.4
5. UFSAR 5.2.5
6. QCOA 0201-01, Increasing Drywell Pressure
7. QOA 900-4 A-17, Annuciator Response
8. QGA 100, RPV Control
9. QCOP 0201-02, Filling the Reactor Vessel and/or Reactor Cavity Using a Condensate Booster Pump via the Feedwater System
10. QCOP 0201-13, Reactor Vessel Upper Wide Range Reference Leg Extension Use and Control
11. QCOP 0201-14, Reactor Vessel Level Control Using a Local Pressure Gauge

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU8

Initiating Condition:

RCS leakage.

Operating Mode Applicability:

4

EAL Threshold Values:

RPV level **cannot** be restored and maintained **> 0 in. (Low Level Scram Setpoint).**

Basis:

The inability to restore and maintain level after reaching the RPS low level scram setpoint infers a degradation of the level of safety at the plant.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU1
2. Technical Specifications 3.3.5.1
3. Technical Specifications 3.4.4
4. UFSAR 5.2.5
5. QGA 100, RPV Control
6. QCOP 0201-02, Filling the Reactor Vessel and/or Reactor Cavity Using a Condensate Booster Pump via the Feedwater System
7. QCOP 0201-13, Reactor Vessel Upper Wide Range Reference Leg Extension Use and Control
8. QCOP 0201-14, Reactor Vessel Level Control Using a Local Pressure Gauge

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS9

Initiating Condition:

Loss of RPV inventory affecting core decay heat removal capability with irradiated fuel in the RPV.

Operating Mode Applicability:

5

EAL Threshold Values:

1. **Without** Secondary CONTAINMENT CLOSURE established:
    a. RPV level **< – 65 in.**

    OR

    b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
        - Fuel Handling ARM 1(2)-1705-16 A or B indicates **> 3000 mR/hr** or off-scale high.
        - Erratic Source Range Monitor indication.

OR

2. **With** Secondary CONTAINMENT CLOSURE established:
    a. RPV level **< – 142 in.** (TAF).

    OR

    b. RPV level unknown with Indication of core uncovery as evidenced by one or more of the following:
        - Fuel Handling ARM 1(2)-1705-16 A or B indicates **> 3000 mR/hr** or off-scale high.
        - Erratic Source Range Monitor indication.

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as defined by Technical Specifications.

Threshold #1 and #2 Basis:

Under the refueling conditions specified in this threshold, prolonged loss of the ability to monitor RPV level in conjunction with indirect indications of inventory loss infer a continued drop in RPV level and loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV.

Basis (cont):

RPV values are actual levels, not indicated levels. Therefore, they may need level compensation depending on conditions. Compensated values may be used in accordance with the SAMG program.

In the refueling mode, when RPV level indication is unavailable, the inventory loss must be detected by drywell floor and equipment drain sump pumpout rates or erratic Source Range Monitor indication. Detail A of QGA-100, Table C provides guidance on determining if RPV level can be monitored. Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to core shine, scattering and radiation bounce off of the solid surfaces in the area will result in readings on the Refuel floor Rad monitor 1(2)-1705-16 A or B > 3000 mR/hr. This threshold radiation value is based on calculations documented in EP-EAL-0501.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CS2
2. Technical Specifications 3.3.5.1
3. Technical Specifications 3.6.1.1
4. Technical Specifications 3.6.4.1
5. QGA 100, RPV Control
6. Technical Specifications 3.3.1
7. Technical Specifications Table 3.3.3.1-1
8. QCOS 1600-07, Reactor Coolant Leakage in the Drywell
9. Technical Specifications 3.4.4
10. UFSAR 5.2.5
11. QOA 900-4 A-17, Annunciator Response
12. EP-EAL-0501, Estimation Of Radiation Monitor Readings Indicating Core Uncovery During Refueling

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU9

Initiating Condition:

UNPLANNED loss of RCS inventory with irradiated fuel in the RPV.

Operating Mode Applicability:

5

EAL Threshold Values:

1. UNPLANNED RPV level drop below the RPV flange for **≥ 15 minutes.**

   OR

2. a. Loss of RPV inventory per Table M5 indications.

Table M5 – Indications of RCS Leakage
• Unexplained floor or equipment sump level rise
• Unexplained Torus level rise
• Unexplained vessel make up rate rise
• Observation of leakage or inventory loss

   AND

   b. RPV level unknown.

Basis:

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

Threshold #1 Basis:

The RPV flange is at 666 ft. 0 in. el. RPV level at this plant elevation is normally indicated by the Upper Wide Range, LI-1(2)-263-101, Rx Water Level Upper Wide Range, (-42 in. to +358 in.). When calibrated for normal plant operations, this instrument reads 191 in. at the RPV flange. With the RPV head removed, the instrument is calibrated to indicate reactor cavity water levels as high as the refuel floor. When calibrated for elevated indication, the instrument reads 56 in. at the RPV flange. Local pressure indicator as well as visual observation of water level in the refueling cavity and RPV is also used during refuel operations.

This threshold is applicable only in the Refueling mode and addresses loss of inventory to below the RPV flange during refueling operations. Refueling operations that drop RPV level below the RPV flange are carefully planned and procedurally controlled. An Unusual Event is appropriate because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant.

The fifteen-minute interval provides a reasonable time frame to restore level using one or more of the redundant means of refill that should be available. If RPV level cannot be restored in this interval, a more serious condition may exist.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

Basis (cont):

Threshold #2 Basis:

In the second condition of this threshold, all RPV level indication would be unavailable, Detail A of QGA 100, Table C provides guidance on determining if RPV level can be monitored. RPV inventory loss, therefore, must be detected by alternate means (i.e., drywell floor and equipment drain sump pumpout rates). Sump pumpout rate increases must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU2
2. UFSAR 5.2.5
3. QOA 900-4 A-17, Annuciator Response
4. QGA 100, RPV Control
5. QCOP 0201-02, Filling the Reactor Vessel and/or Reactor Cavity Using a Condensate Booster Pump via the Feedwater System
6. QCOP 0201-13, Reactor Vessel Upper Wide Range Reference Leg Extension Use and Control
7. QCOP 0201-14, Reactor Vessel Level Control Using a Local Pressure Gauge

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU10

Initiating Condition:

UNPLANNED loss of all onsite or offsite communications capabilities.

Operating Mode Applicability:

1, 2, 3, 4, 5

EAL Threshold Values:

1. Loss of all Table M6 **Onsite** communications capability affecting the ability to perform routine operations.

   OR

2. Loss of all Table M6 **Offsite** communications capability.

Table M6 - Communications Capability

System	Onsite	Offsite
Plant Radio System	X	
Plant Paging System	X	
Sound Power Phones	X	
In-Plant Telephones	X	
All Telephone Lines (commercial and microwave)		X
ENS		X
HPN		X
NARS		X
Cellular Phones		X
Satellite Phones		X

Basis:

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

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems with offsite authorities or personnel. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10CFR 50.72.

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

MU10 (cont)

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU6 & CU6
2. EP-MW-124-1001 Facilities Inventories and Equipment Tests
3. UFSAR Section 9.5.2

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU11

Initiating Condition:

Inability to reach required shutdown within Technical Specification limits.

Operating Mode Applicability:

1, 2, 3

EAL Threshold Values:

Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a prescribed shutdown mode when the Technical Specification configuration cannot be restored. The plant is within its safety envelope when being shut down 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 completion period elapses under Technical Specifications and is not related to how long a condition may have existed.

Basis Reference(s):

1. NEI 99-01, Rev. 4 SU2
2. QCNP Technical Specifications

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1

Initiating Condition:

Security event resulting in loss of physical control of the facility.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

A HOSTILE FORCE has taken control of:

1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions (Table H1).

Table H1 - Safety Functions and Related Systems
• 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) • Fission Product Barriers

OR

2. Spent Fuel Pool cooling systems if imminent fuel damage is likely (e.g., reactor fuel off-loaded in pool within 120 days).

Basis:

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.

Threshold #1 Basis

This threshold encompasses conditions under which a HOSTILE FORCE has taken physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions. As a result, equipment control cannot be transferred to and operated from another location.

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

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.

Threshold #2 Basis

This threshold addresses loss of physical control of spent fuel pool cooling systems if imminent fuel damage is likely because there is freshly off-loaded fuel in the pool. The condition “freshly off-loaded reactor fuel in pool” equates to fuel off-loaded within the last 120 days in NF-AA-310 Special Nuclear Material And Core Component Movement.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1 (cont)

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HG1
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NF-AA-310 Special Nuclear Material And Core Component Movement

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1

Initiating Condition:

Site attack.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert ICs (HA1 and HA2) in that a hostile force has progressed from the OWNER CONTROLLED AREA to the Protected Area.

Although Nuclear Power Plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for Offsite Response Organizations (ORO) 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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1 (cont)

Basis (cont):

This EAL is intended to address the potential for a very rapid progression of events due to a dedicated attack. It is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs. HOSTILE ACTION identified above encompasses various acts including:

- Air attack (LARGE AIRCRAFT impacting the protected area)
- Land-based attack (HOSTILE FORCE penetrating protected area)
- Waterborne attack (HOSTILE FORCE on water penetrating protected area)
- BOMBs breeching the protected area

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs.

This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for ORO to be notified and to activate in order to be better prepared to respond should protective actions become necessary. If not previously notified by NRC that the LARGE AIRCRAFT impact 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.

LARGE AIRCRAFT is meant to be an 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.

This EAL addresses the immediacy of a threat to impact site vital areas within a relatively short time. The fact that the site is under serious attack with minimal time available for additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HS4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition:

Notification of an airborne attack threat.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A validated notification from NRC of a LARGE AIRCRAFT attack threat **< 30 minutes** away.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

LARGE AIRCRAFT is meant to be an 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.

The intent of this EAL is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations (ORO) and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Alert. This EAL is met when a plant receives information regarding a LARGE AIRCRAFT attack threat from NRC and the LARGE AIRCRAFT is less than 30 minutes away from the plant.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from such an attack. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for OROs to be notified and encouraged to activate (if they do not normally) to be better prepared should it be necessary to consider further actions.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HA7
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

Initiating Condition:

Confirmed terrorism security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. A credible site-specific security threat notification as determined per SY-AA-101-132, “Threat Assessment”.

   OR

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

Basis:

Threshold #1 Basis

The intent of this threshold is to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat.

The determination of “credible” is made through use of information found in the Station Security Plan or SY-AA-101-132, “Threat Assessment” procedure.

Threshold #2 Basis

The intent of this threshold is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Unusual Event. This threshold is met when a plant receives information regarding an aircraft threat from NRC. Should the threat involve a LARGE AIRCRAFT (LARGE AIRCRAFT is meant to be an aircraft with the potential for causing significant damage to the plant), then escalation to Alert via HA1 would be appropriate if the LARGE AIRCRAFT is less than 30 minutes away from the plant. The status and size of the plane may be provided by NORAD through the NRC. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

Initiating Condition:

Notification of HOSTILE ACTION within the OWNER CONTROLLED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

A notification from the site Security Force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA.

Basis:

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

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

This EAL is intended to address the potential for a very rapid progression of events due to an attack including:

- Air attack (LARGE AIRCRAFT impacting the OCA)
- Land-based attack (HOSTILE FORCE progressing across licensee property or directing projectiles at the site)
- Waterborne attack (HOSTILE FORCE on water attempting forced entry or directing projectiles at the site)
- BOMBs

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2 (cont)

Basis (cont):

This EAL is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA. That initiating condition is adequately addressed by other EALs.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne terrorist attack such as that experienced on September 11, 2001, and the possibility for additional attacking aircraft. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for Offsite Response Organizations to be notified and to activate in order to be better prepared to respond should protective actions become necessary.

If not previously notified by NRC that the LARGE AIRCRAFT impact 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. LARGE AIRCRAFT is meant to be an 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.

This IC/EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time. The fact that the site is an identified attack candidate with minimal time available for further preparation requires a heightened state of readiness and implementation of protective measures that can be effective (onsite evacuation, dispersal or sheltering) before arrival or impact.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HA8
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01
5. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS3

Initiating Condition:

Confirmed security event in a plant VITAL AREA

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event in a plant VITAL AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC (HA3).

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a Station's security. Only those events that involve actual or likely major failures of plant functions needed for protection of the public need to be considered. The following events would not normally meet this requirement; (e.g., Failure by a Member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Station Security Plan.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HS1
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

Initiating Condition:

Confirmed security event in a plant PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event in a plant PROTECTED AREA as determined from Station Security Plan – Appendix C.

Basis:

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

This class of security events represents an escalated threat to plant safety above that contained in the Unusual Event.

Multi-unit stations with shared safety functions should further consider how this IC may affect more than one unit and how this may be a factor in escalating the emergency class.

The Station Security Plan identifies numerous events/conditions that constitute a threat/compromise to a station’s security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered. The following events would not normally meet this requirement; (e.g., failure by a member of the Security Force to carry out an assigned/required duty, internal disturbances, loss/compromise of safeguards materials or strike actions).

Reference is made to the Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HA4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU3

Initiating Condition:

Confirmed security event which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Value:

Notification by the Security Force of a security event as determined from Station Security Plan – Appendix C.

Basis:

Reference is made to Security Force because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan.

This threshold is based on Station Security Plan – Appendix C. 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.

Consideration should be given to the following types of events when evaluating an event against the criteria of the Station Security Plan: CIVIL DISTURBANCE, and STRIKE ACTION.

Basis Reference(s):

1. NRC Bulletin 2000-02 Emergency Preparedness and Response Actions for Security Based Events, HU4
2. SY-AA-101-132, Threat Assessment
3. Station Security Plan – Appendix C
4. NRC Safeguards Advisory 10/6/01

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS4

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Control Room evacuation has been initiated.

   AND

2. Control of the plant **cannot** be established per QOA 0010-05 in **< 30 minutes**.

Basis:

The 30 minute time period starts when either:

a. Control of the plant is no longer maintained in the Main Control Room

   OR

b. The last Operator has left the Main Control Room.

The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. The 30 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 determination of whether or not control is established outside of the Main Control Room is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the site-specific time for transfer that the licensee has control of the plant. Transfer of control to locations outside the Control Room is considered established when the Shift Manager has determined that the operators are capable of controlling reactivity, core cooling and heat sink functions.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HS2
2. QOA 0010-05 Plant Operation With The Control Room Inaccessible

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

Initiating Condition:

Control Room evacuation has been initiated.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Entry into QOA 0010-05 for Control Room evacuation.

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency operations centers are necessary. Procedure QOA 0010-05, Plant Operation With The Control Room Inaccessible specifies conditions under which Control Room evacuation may be necessary.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA5
2. QOA 0010-05, Plant Operation With The Control Room Inaccessible

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5

Initiating Condition:

Natural and destructive phenomena affecting the plant VITAL AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by Strong Motion Seismograph output Alert **> 0.125 volts (0.10 g)**.

   AND

   b. Confirmed by **EITHER**:

   - Earthquake felt in plant.
   - National Earthquake Center.

   OR

2. Tornado or high winds **> 100 mph** within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems.

   OR

3. Vehicle crash within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to any plant structures or equipment contained in a Table H2 area, or Control Room indication of degraded performance of those systems:

   OR

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE or penetration of any Table H2 area.

Table H2 – Vital Areas
• Main Control Room
• Reactor Building
• Diesel Generator Rooms
• 4 kV Switchgear Area
• Battery Rooms
• B-Train Control Room HVAC
• Service Water Pumps
• Turbine Building Cable Tunnel

OR

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5 (cont)

EAL Threshold Value(s) (cont):

5. Uncontrolled flooding that results in **EITHER**:

   a. Degraded safety system performance in any Table H3 area as indicated in the Control Room.

Table H3 – Internal Flooding Areas
• A RHR Room • B RHR Room • A Core Spray Room • B Core Spray Room • Torus Area • HPCI Area

   OR

   b. Industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.

OR

6. a. High river water level **> 603 ft.**

   OR

   b. Low river water level **< 561 ft.**

Basis:

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

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

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

Threshold #1 Basis:

This threshold addresses events that may have resulted in a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this threshold to assess the actual magnitude of the damage.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont):

This threshold is based on seismic ground acceleration in excess of 0.10 g for the UFSAR Operating Basis Earthquake (OBE). Seismic events of this magnitude are a factor of 4 greater that the Unusual Event threshold of EAL HU5 and can cause damage to plant safety functions.

A strong-motion seismograph is located on the basement floor of the Unit 1 turbine building (elevation 547 feet 0 inches) in the northeast corner of the condensate pump room. The unit is mounted directly onto the floor in a corner out of the way of the normal traffic pattern. By mounting the seismograph directly on the floor of the turbine building basement, an accurate recording of ground motion originated in the rock substructure which underlies the plant foundations will be obtained.

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

Threshold #2 Basis:

This threshold addresses events that may have resulted in a Table H2 area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The Alert classification is appropriate if visible damage is observed and relevant plant parameters indicate that the performance of safety systems in these areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

110 mph is the UFSAR design basis wind speed. Station structures are designed to withstand wind loads which may exist if sustained wind speeds reach or exceed 110 mph. Wind loads in excess of this magnitude can cause damage to safety functions. Since the wind speed instrument can only measure up to 100 mph, the threshold has been reduced to this value.

Threshold #3 Basis:

This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes, or impact of projectiles into a Table H2 area. This threshold addresses vehicle crashes that challenge the operability of systems necessary for safe shutdown of the plant. Table H2 areas include Category 1 structures and those Category 2 structures that contain Category 1 Systems and components.

The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Table H2 areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont):

Threshold #4 Basis:

This threshold addresses the threat to safety-related equipment imposed by missiles generated by main turbine rotating component failures. It is consistent with the definition of an ALERT in that, if missiles have damaged or penetrated areas containing safety-related equipment, the potential exists for substantial degradation of the level of safety of the plant.

Threshold #5 Basis:

This threshold addresses the effect of internal flooding that has resulted in degraded performance of safety systems or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant.

“Uncontrolled” as used in this threshold describes a condition where water is entering an area from an unplanned evolution. This flooding may have been caused by internal events such as component failures, equipment misalignment, and fire suppression system actuation or outage activity mishaps. Water entering an area, which resulted in degraded performance of safety systems within the area due to wetting or submergence, would meet the intent of this threshold. Minor leaks, such as valve packing or instrument line breaks would not constitute “'Uncontrolled Flooding'. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if indications of degraded system performance is available or a shock hazard is known to exist.

The Internal Flooding Areas listed in Table H3 include areas listed in QGA-300 containing systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

The Quad Cities Nuclear Power Station Unit 1 and 2 Internal Flooding Analysis Note Book provides insights concerning the areas listed in Table H3.

Threshold #6 Basis:

Evaluations for higher flood levels show that 603 ft. el., which envelops the Probable Maximum Flood (PMF) elevation, is the maximum flood elevation which can assure that the plant can be shut down and maintained in a safe condition. A river level of > 603 ft. el. represents the maximum possible flood stage.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont):

The station design is such that if Lock and Dam No. 14 were to fail, the water level would recede in the intake bay to the point where it would be separated from the river. As the water level recedes in the intake bay, circulating, service and fire diesel pumps would become inoperable, leaving only RHRSW and DGCW available to shutdown the units. Use of the ultimate heat sink to shutdown the reactors requires the operation of portable diesel pumps with a total capacity of 5100 gpm to reverse the normal flow of makeup water. Makeup water would be provided from the river through the discharge piping and return to the river across the log boom in the intake bay.

Portable pumps of sufficient capacity are available from a leasing facility. Following a dam failure, approximately 2 days would be available to position these pumps to take suction from the discharge flume and to discharge to the center bay (containing the RHRSW and DGCW suction piping) of the Cribhouse building.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA1
2. UFSAR Section 3.7.4
3. UFSAR Section 3.7.1
4. QCIS 0010-01 Strong Motion Accelerometer Recorder Operability Test
5. QCOA 0010-09 Earthquake
6. QCOP 0010-07 Seismic Event Retrieval
7. QCOA 0010-10 Tornado Watch/Warning, Severe Thunder Storm Warning or Severe Winds
8. UFSAR Section 3.3
9. Drawing B-01A Composite Site Plan
10. UFSAR Section 3.2
11. QGA 300 Secondary Containment Control
12. UFSAR Section 3.4.1.1
13. QCTP 0130-11 Internal Flood Protection Program
14. Drawing FL-1 Flood Barriers
15. Quad Cities Nuclear Power Station Unit 1 and 2 Internal Flooding Analysis Note Book, July 1993 Final Draft, prepared by Individual Plant Evaluation Partnership (IPEP)

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition:

Natural and destructive phenomena affecting the PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Seismic event as indicated by any **TWO** of the following:
   - Earthquake felt in plant.
   - Seismic event confirmed by station seismic monitor procedure.
   - National Earthquake Center.

   OR

2. Report by plant personnel of tornado striking or sustained (> 15 minutes) high winds **> 100 mph**, within PROTECTED AREA boundary.

   OR

3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting a Table H2 area.

Table H2 – Vital Areas
• Main Control Room • Reactor Building • Diesel Generator Rooms • 4 kV Switchgear Area • Battery Rooms • B-Train Control Room HVAC • Service Water Pumps • Turbine Building Cable Tunnel

   OR

4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

   OR

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont)

EAL Threshold Values:

5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode.

Table H3 – Internal Flooding Areas
• A RHR Room • B RHR Room • A Core Spray Room • B Core Spray Room • Torus Area • HPCI Area

OR

6. River level transients potentially affecting safe operation of the plant:
   a. High river water level **> 594 ft.**

   OR

   b. Report of substantial reduction in river level by site personnel and confirmation by the Corp. of Engineers that Dam # 14 has failed.

Basis:

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

Threshold #1 Basis:

A felt earthquake may be the first indication to Control Room personnel that a seismic event is occurring and warrants the emergency declaration because the seismograph does not provide a readout or alarm to the Control Room when actuated. The seismic monitor is located on the basement floor of the Unit 1 turbine building (547 ft. 0 in el.) in the northeast corner of the condensate pump room, which is a strong-motion seismograph that actuates at a sensed earthquake threshold of 0.025 g. Seismic events of this magnitude are 1/4 of the Alert event threshold (OBE) of EAL HA5 in which it is assumed the earthquake can cause damage to plant safety functions.

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. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01 g.”

Confirmation from the National Earthquake center shall not delay declaration in the presence of VALID confirming indications.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis (cont):

Threshold #2 Basis:

This threshold is based on the assumption that a tornado striking (touching down) or design force winds (> 110 mph) within the Protected Area boundary may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. Since the wind speed instrument can only measure up to 100 mph, the threshold has been reduced to this value.

The Protected Area boundary is within the security isolation zone and is defined in the QCNP Station Security Plan – Appendix C. Verification of a tornado is obtained by direct observation and reporting by station personnel. “Sustained” wind speeds exist for 15 minutes or longer. Wind speed is obtained from meteorological data in the Control Room.

Threshold #3 Basis:

In this context, a "vehicle crash" is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

Threshold #4 Basis:

This threshold is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this threshold to classify minor operational leakage.

Threshold #5 Basis:

“Uncontrolled” as used in this threshold describes a condition where water is entering the area from an unplanned evolution. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source if a potential to affect safety related equipment needed for the current operating mode exists.

This threshold addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, fire suppression system actuation or outage activity mishaps. Minor leaks, such as valve packing or instrument line breaks would not constitute “Uncontrolled Flooding.” The Internal Flooding Areas of concern for the Unusual Event declaration are those Table H3 areas that have the potential to affect safety related equipment needed for the current operating mode.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Basis (cont):

Threshold #6 Basis:

The Design Flood elevation is 594.5 ft. el. (rounded down to 594 ft. el. MSL). This initial design flood elevation is equal to the plant grade of 594.5 ft. el. and any mode of operation is, therefore, possible without additional protective measures. For a flood of any elevation from 594.5 ft. el. up to the surrounding ground elevation of 603 ft. el., the plant can and will be maintained in a safe condition by flooding the plant buildings to match the river elevation. A flood of this magnitude would provide sufficient time to enable shutdown procedures to take place and flooding of the structures to be initiated. Under this condition, no heat removal nor related electrical power supply systems are needed.

The station design is such that if Lock and Dam No. 14 were to fail, the water level would recede in the intake bay to the point where it would be separated from the river. As the water level recedes in the intake bay, circulating, service and fire diesel pumps would become inoperable, leaving only RHRSW and DGCW available to shutdown the units.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU1
2. UFSAR Section 3.7.4
3. UFSAR Section 3.7.1
4. QCIS 0010-01 Strong Motion Accelerometer Recorder Operability Test
5. QCOA 0010-09 Earthquake
6. QCOP 0010-07 Seismic Event Retrieval
7. QCOA 0010-10 Tornado Watch/Warning, Severe Thunder Storm Warning or Severe Winds
8. UFSAR Section 3.3
9. Drawing B-01A Composite Site Plan
10. UFSAR Section 3.2
11. QCTP 0130-11 Internal Flood Protection Program
12. Drawing FL-1 Flood Barriers
13. Quad Cities Nuclear Power Station Unit 1 and 2 Internal Flooding Analysis Note Book, July 1993 Final Draft, prepared by Individual Plant Evaluation Partnership (IPEP)

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6

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, D

EAL Threshold Values:

1. FIRE or EXPLOSION in any Table H2 area.

Table H2 – Vital Areas
• Main Control Room • Reactor Building • Diesel Generator Rooms • 4 kV Switchgear Area • Battery Rooms • B-Train Control Room HVAC • Service Water Pumps • Turbine Building Cable Tunnel

AND

2. a. Affected safety system parameter indications show degraded performance.

   OR

   b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.

Basis:

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

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

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6 (cont)

Basis (cont):

The areas listed in Table H2 house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Personnel access to these areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses FIRES and EXPLOSIONS that challenge the operability of systems necessary for safe shutdown of the plant.

The only FIRES and EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment required for safe shutdown. Visual observation of damage infers the ability to approach or enter the affected areas. Lacking the ability to adequately inspect the area for damage, the Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this EAL. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

A steam line break or steam EXPLOSION that damages permanent structures or equipment in one of these areas would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA2
2. UFSAR Section 3.2

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. FIRE in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

OR

2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

Table H2 – Vital Areas
• Main Control Room • Reactor Building • Diesel Generator Rooms • 4 kV Switchgear Area • Battery Rooms • B-Train Control Room HVAC • Service Water Pumps • Turbine Building Cable Tunnel

OR

3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

Basis:

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

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

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6 (cont)

Basis (cont):

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

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

Thresholds #1 and #2 Basis:

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins with a credible notification that a fire is occurring or indication of a valid fire detection system alarm. A verified alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm.

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 EAL HA7, Release of Toxic or Flammable Gases. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under EAL HA7.

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

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket, etc.). Such fires are excluded from consideration in this threshold since they have no safety consequence.

Threshold #3 Basis:

The only EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment in the PROTECTED AREA.

A steam line break or steam EXPLOSION that damages permanent structures or equipment in a PROTECTED AREA would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU2
2. UFSAR Section 3.2

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA7

Initiating Condition:

Release of toxic or flammable gases within or restricting access to a VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to listed areas) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to listed areas) in concentrations greater than LOWER FLAMMABILITY LIMIT (LFL).

Table H2 – Vital Areas
• Main Control Room • Reactor Building • Diesel Generator Rooms • 4 kV Switchgear Area • Battery Rooms • B-Train Control Room HVAC • Service Water Pumps • Turbine Building Cable Tunnel

Basis:

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

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Basis (cont)

Values for LFL for common gases at Quad Cities Station are:

- Acetylene – 2.2% (BOC Gasses MSDS)
- Propane – 2.2% (Air Liquide Safety Data Sheet)
- Hydrogen – 4% (BOC Gasses MSDS)

This EAL is based on toxic, asphyxiant, or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Toxic or flammable gases detected outside of these areas need not be considered for this EAL unless there is a spread of the gasses into one of these areas.

Threshold #1:

Declaration should not be delayed for confirmation from atmospheric testing if it is reasonable to conclude that the IDLH concentrations have been met (e.g., documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards).

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 this EAL. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under this EAL.

The first condition is met if measurement of toxic gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a Table H2 area. Non-Toxic Gases which displace oxygen (site examples; Halon or Nitrogen) to a life threatening level due to asphyxiation (oxygen deprivation) should also be considered for this EAL.

An Asphyxiant is a material 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.

Threshold #2:

The second condition is met when the flammable gas concentration in a Table H2 area exceeds the lower flammability limit. Flammable gases such as hydrogen and acetylene are routinely used to maintain plant systems (hydrogen – main generator cooling, reactor coolant chemistry control) or repair equipment/components (acetylene - welding). This condition addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a Table H2 area has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage or personnel injury.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA7 (cont)

Basis (cont)

Once it has been determined that an uncontrolled release of flammable gas is occurring, sampling must be done to determine if the gas concentration exceeds the lower flammability limit.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HA3
2. UFSAR Section 3.2

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU7

Initiating Condition:

Release of toxic or flammable gases deemed detrimental to normal operation of the plant.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Report or detection of toxic, asphyxiant, or flammable gases that have or could enter the site area boundary in amounts that can 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:

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.

This EAL is based on the existence of uncontrolled releases of toxic, asphyxiant, or flammable gas affecting plant operations or the health of plant personnel. The release may have originated within the Protected Area boundary, or it may have originated offsite and subsequently drifted inside the Protected Area boundary. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on normal plant operations.

It is intended that releases of toxic, asphyxiant, or flammable gases are of sufficient quantity and the release point of such gases is such that safe plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for safe plant operation. The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect safe plant operations or plant personnel safety.

An Asphyxiant is a material 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.

Basis Reference(s):

1. NEI 99-01, Rev. 4 HU3
2. UFSAR Section 3.2

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels (>1 Rem TEDE or > 5 Rem CDE Thyroid) outside the site boundary.

Basis Reference(s):

1. NEI 99-01, Rev 4 HG2
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

Basis Reference(s):

1. NEI 99-01, Rev 4 HS3
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA).

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

Basis Reference(s):

1. NEI 99-01, Rev 4 HA6
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)
3. EPA-400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU8

Initiating Condition:

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

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

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:

This EAL addresses unanticipated conditions not addressed elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency operating procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

Basis Reference(s):

1. NEI 99-01, Rev 4 HU5
2. Enhancement to Emergency Preparedness Programs for Hostile Action, May 2005 (Revised Nov. 18)

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU9

Initiating Condition:

Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability:

1, 2, 3, 4, 5, D

EAL Threshold Values:

1. Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY.

   OR

2. Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by damage to MPC CONFINEMENT BOUNDARY.

   OR

3. Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask MPC CONFINEMENT BOUNDARY.

Basis:

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

The CONFINEMENT BOUNDARY consists of the Multi-Purpose Canister (MPC) shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds. An Unusual Event in this EAL is based on a loaded fuel storage cask CONFINEMENT BOUNDARY being violated leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

Threshold #1 and #2

The results of the HI-STORM 100 and HI-STAR 100 Final Safety Analysis Reports (FSARs) were used to develop the list of natural phenomena events and accident conditions. These EALs address responses to a dropped cask, a tipped over cask, EXPLOSION, missile damage, fire damage or natural phenomena affecting a cask (e.g., seismic event, tornado, flood, etc.). The cask FSARs require that the cask and in some cases the MPC be inspected to determine if the cask or MPC may have been affected as a result of a natural phenomena event or accident condition. The inspections are performed to the extent practical to assess the potential damage to the cask and/or the confinement boundary. If it were determined during the assessment that the MPC CONFINEMENT BOUNDARY was damaged such that boundary integrity is in question, declaration under this EAL would be necessitated.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU9 (cont)

Basis:

Threshold #3

Any condition not detailed as an EAL threshold value, which, in the judgment of the Emergency Director, is a potential degradation in the level of safety of the ISFSI. Emergency Director judgment is to be based on known conditions and the expected response to mitigating activities within a short time period.

Basis Reference(s):

1. NEI 99-01, Rev. 4 E-HU1
2. HI-STORM 100 FSAR Rev. 3
3. HI-STAR 100 FSAR Rev. 1

Section 4: Emergency Measures

4.1 Notification of the Emergency Organization

Standard NARS notifications for the Quad Cities Station are made to the State of Illinois Emergency Management Agency (IEMA), the State of Iowa Emergency Management Division (IEMD), Scott County Sheriff's Office, and Clinton County Sheriff's Office. At the Quad Cities Generating Station, if a General Emergency is the initiating event, the Emergency Director is responsible for notifying the following additional Illinois, Iowa and local agencies:

- Rock Island Communications Center
- Whiteside County Sheriff

4.2 Assessment Actions

Throughout each emergency situation, continuing assessment will occur. Assessment actions at Quad Cities Station may include an evaluation of plant conditions; in-plant, onsite, and initial offsite radiological measurements; and initial estimates of offsite doses. Core damage information is used to refine dose assessments and confirm or extend initial protective action recommendations. Quad Cities Station utilizes NEDC-33045P-A, Revision 0, (2001) as the basis for the methodology for post-accident core damage assessment. This methodology utilizes real-time plant indications. In addition, Quad Cities Station may use samples of plant fluids and atmospheres as inputs to the CDAM (Core Damage Assessment Methodology) program for core damage estimation.

4.3 Protective Actions for the Offsite Public

To aid Control Room personnel during a rapidly developing emergency situation, Figure 4.1: "Protective Action Recommendation (PAR) Determination Flowchart for Quad Cities Station" has been developed based on Section J.10.m of the Emergency Plan.

4.3.1 Alert and Notification System (ANS) Sirens

This ANS consists of a permanently installed outdoor notification system within a ten-mile radius around the station. The ten-mile radius around the station is primarily an agricultural area with a population density below 2000 persons per square mile. The ANS, as installed, consists of mechanical and electronic sirens that will cover this entire area with a minimum sound level of 60 db. Additionally, the ANS will cover the heavily populated areas within ten-mile radius around the station with a minimum sound level of 70 db to ensure complete coverage.

4.3.2 Evacuation Time Estimates

The evacuation time estimates were developed per the requirements of NUREG-0654, and to support the Illinois Plan For Radiological Accidents (IPRA) - Quad Cities Volume IV and the Clinton County / Scott County Radiological Emergency Response Plan (RERP). The purpose of the evacuation time estimates is to assess postulated evacuation times for the Quad Cities Station Emergency Planning Zone (EPZ).

The evacuation time estimate data was updated per a study performed by Earth Tech. Inc. documented in their report dated June, 2005 entitled “Update of Evacuation Time Estimates for the Plume Exposure Pathway Emergency Planning Zone for Quad Cities Nuclear Generating Station.”

An updated set of evacuation time estimates (ETEs) for Quad Cities Nuclear Generating Station has been developed, using population data from the 2000 Census. The assumptions and analysis procedures for the present study closely followed the approach of the previous (1994) study . A full ETE update study was not judged to be necessary for Quad Cities Station. Comparison of the 1990 and 2000 Census data showed relatively small changes in population within the Emergency Planning Zone (EPZ), and no significant changes to the roadway network have occurred since 1994 . This “partial” update study was therefore performed using data from the 1994 study to characterize the roadway network and the populations for special and transient facilities; only the population numbers for permanent and seasonal residents have been updated.

The evacuation times are based on a detailed consideration of the EPZ roadway network and population distribution. The information in Table 4-1 presents representative evacuation times for daytime and nighttime scenarios under various weather conditions for the evacuation of various areas around the Quad Cities Station, once a decision has been made to evacuate. The evacuation times noted include notification, mobilization, and travel time. These times are for the general population which include permanent population and special facilities (schools, nursing homes, hospitals, and recreational areas). Table 4-2 provides information on the scenario population distribution (by Subarea) that was used for this study. Table 4-3 provides a representation of the Subarea Locations in relation to the EPZ.

4.4 Protective Actions for Onsite Personnel

Quad Cities Station has a siren system to warn personnel of emergency conditions. Upon hearing a continuous two (2) minute siren, all personnel not having emergency assignments have been instructed to assemble in predesignated assembly areas. Refer to Figure 4-2.

If a site evacuation of non-essential personnel is required, personnel will be released to their homes or relocated and monitored at one of the following designated relocation centers for Quad Cities:

- Morrison Relocation Center, Morrison, Illinois
- Byron Station, Byron Illinois

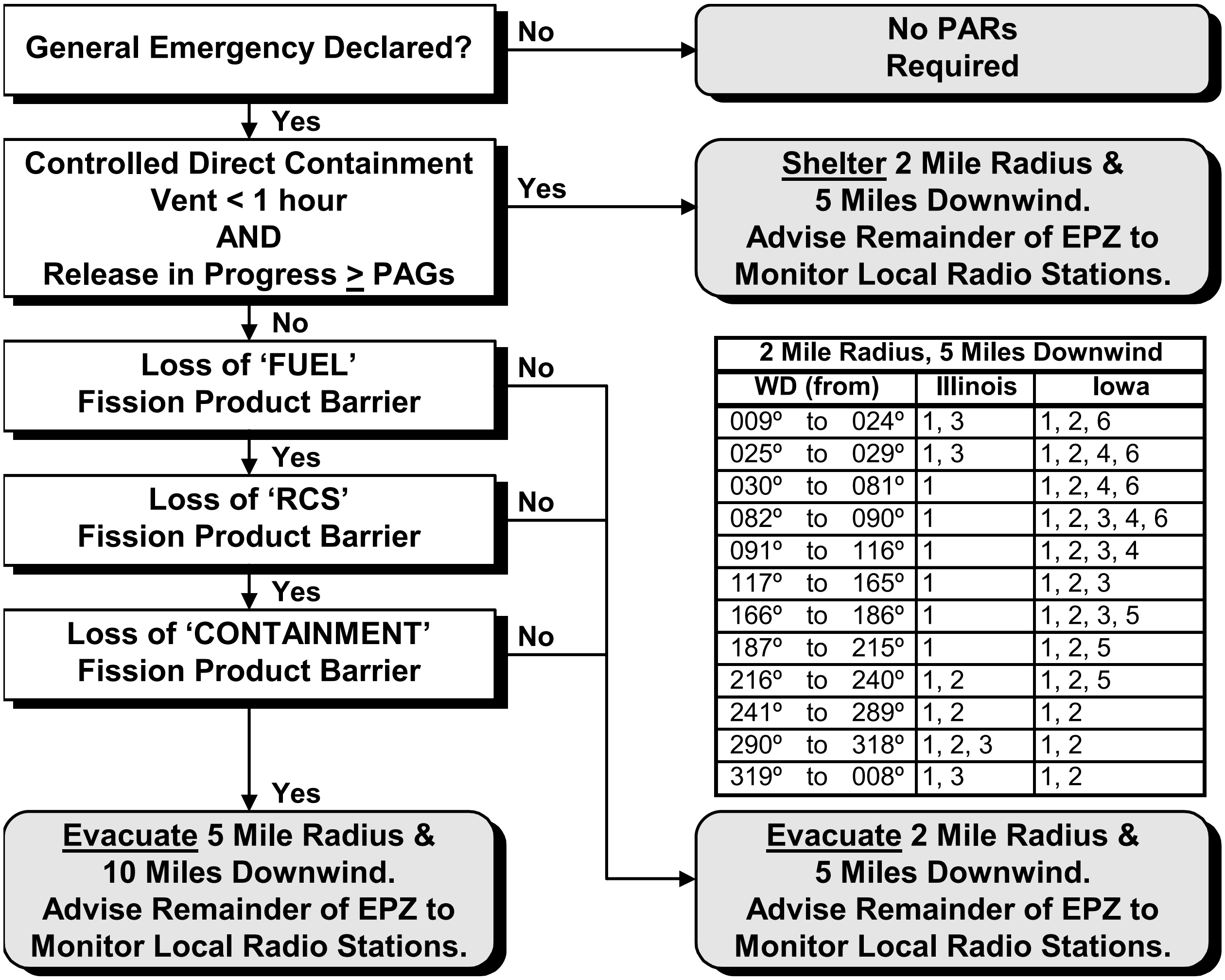
For evacuation routes, refer to EP-AA-113-F-21.

Traffic control for onsite areas will be handled by the Quad Cities Station security force, if necessary.

When a site evacuation is imminent, the TSC Security Coordinator notifies by phone or dispatches a security guard to notify those personnel in buildings outside the protected area (Training Building, Warehouse, Wastewater Treatment Plant, etc.). These personnel are evacuated using the prescribed route to the designated relocation center. Personnel in the warehouses, sewage treatment plant, wastewater treatment plant, and training building will assemble at their present location and await further instructions (e. g. evacuation).

Equipment and personnel would be available at the Morrison Relocation Center and Byron Station for monitoring and decontamination of evacuated personnel. If major decontamination and follow-up or bioassay samples are necessary, those persons would be sent to Byron Station.

Figure 4-1: PAR Determination Flowchart



2 Mile Radius, 5 Miles Downwind		
WD (from)	**Illinois**	**Iowa**
009° to 024°	1, 3	1, 2, 6
025° to 029°	1, 3	1, 2, 4, 6
030° to 081°	1	1, 2, 4, 6
082° to 090°	1	1, 2, 3, 4, 6
091° to 116°	1	1, 2, 3, 4
117° to 165°	1	1, 2, 3
166° to 186°	1	1, 2, 3, 5
187° to 215°	1	1, 2, 5
216° to 240°	1, 2	1, 2, 5
241° to 289°	1, 2	1, 2
290° to 318°	1, 2, 3	1, 2
319° to 008°	1, 3	1, 2

5 Mile Radius, 10 Miles Downwind		
WD (from)	**Illinois**	**Iowa**
002° to 025°	1, 2, 3, 6	1, 2, 3, 4, 5, 6, 12
026° to 031°	1, 2, 3	1, 2, 3, 4, 5, 6, 12
032° to 049°	1, 2, 3	1, 2, 3, 4, 5, 6, 10, 12
050° to 058°	1, 2, 3	1, 2, 3, 4, 5, 6, 8, 10, 12
059° to 075°	1, 2, 3	1, 2, 3, 4, 5, 6, 8, 10
076° to 087°	1, 2, 3	1, 2, 3, 4, 5, 6, 8
088° to 106°	1, 2, 3	1, 2, 3, 4, 5, 6, 7, 8
107° to 116°	1, 2, 3	1, 2, 3, 4, 5, 6, 7
117° to 146°	1, 2, 3	1, 2, 3, 4, 5, 6, 7, 9
147° to 169°	1, 2, 3	1, 2, 3, 4, 5, 6, 9
170° to 186°	1, 2, 3	1, 2, 3, 4, 5, 6, 9, 11
187° to 215°	1, 2, 3	1, 2, 3, 4, 5, 6, 11
216° to 229°	1, 2, 3, 4	1, 2, 3, 4, 5, 6, 11
230° to 239°	1, 2, 3, 4	1, 2, 3, 4, 5, 6
240° to 267°	1, 2, 3, 4, 5	1, 2, 3, 4, 5, 6
268° to 296°	1, 2, 3, 5	1, 2, 3, 4, 5, 6
297° to 318°	1, 2, 3, 5, 6	1, 2, 3, 4, 5, 6
319° to 001°	1, 2, 3, 6	1, 2, 3, 4, 5, 6

2 Mile Radius, 5 Miles Downwind		
WD (from)	**Illinois**	**Iowa**
009° to 024°	1, 3	1, 2, 6
025° to 029°	1, 3	1, 2, 4, 6
030° to 081°	1	1, 2, 4, 6
082° to 090°	1	1, 2, 3, 4, 6
091° to 116°	1	1, 2, 3, 4
117° to 165°	1	1, 2, 3
166° to 186°	1	1, 2, 3, 5
187° to 215°	1	1, 2, 5
216° to 240°	1, 2	1, 2, 5
241° to 289°	1, 2	1, 2
290° to 318°	1, 2, 3	1, 2
319° to 008°	1, 3	1, 2

Figure 4-2: Predesignated Assembly Areas

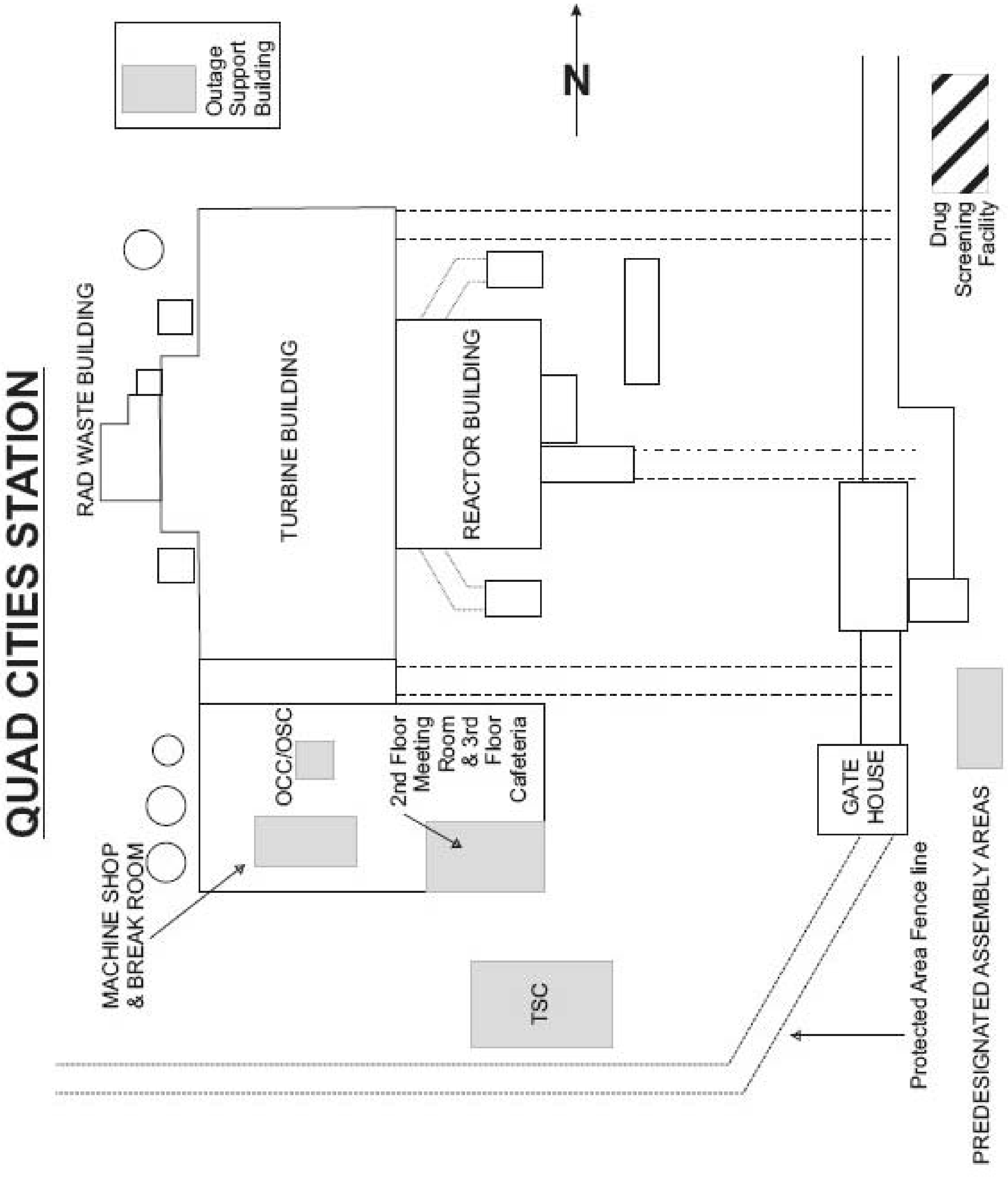


Table 4-1: Evacuation Time Estimates (in minutes)

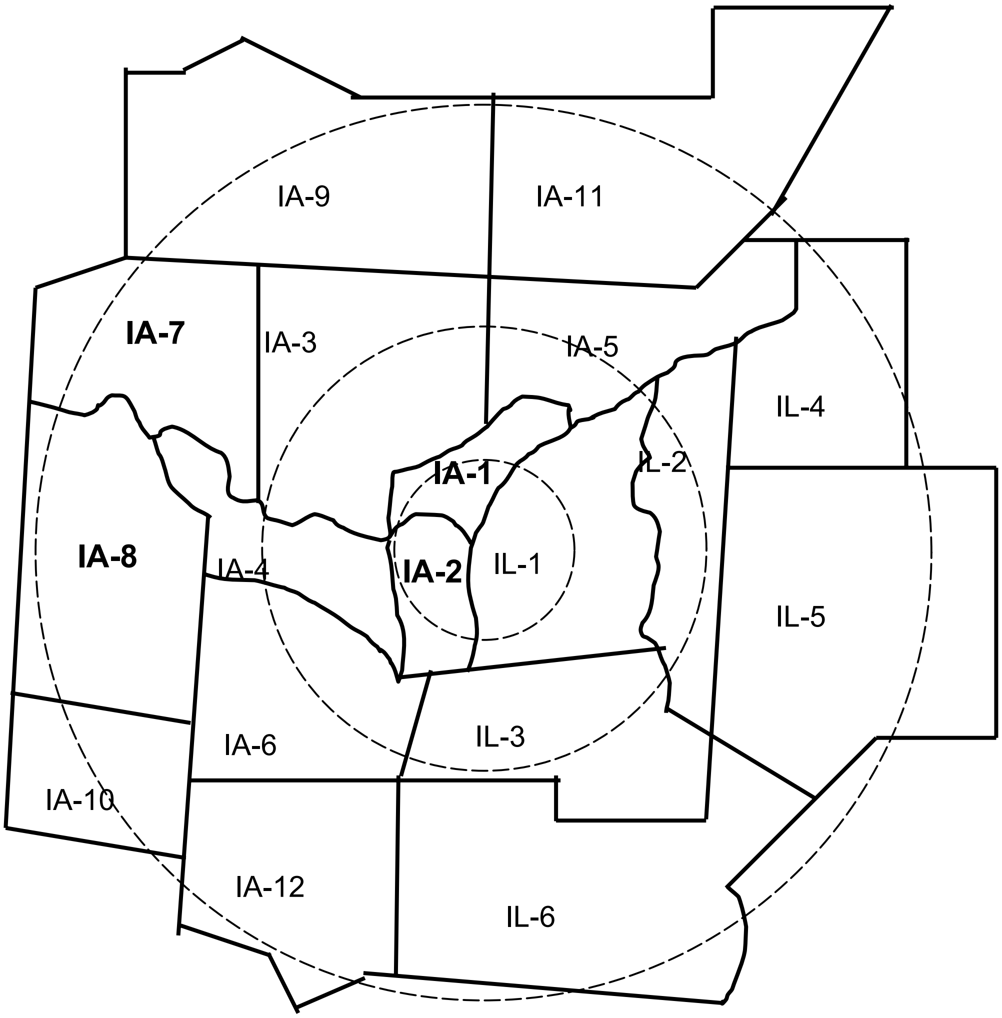
GENERAL POPULATION EVACUATION TIMES (MINUTES)

PAR Evacuation Zone										
	IL	**IA**	**Summer Day**		**Summer Night**		**Winter Day**		**Winter Night**	
Wind Direction	**Subareas**	**Sub-areas**	Normal weather	Adverse weather	Normal weather	Adverse weather	Normal weather	Adverse weather	Normal weather	Adverse weather
2 mile Radius, 5 Miles Downwind										
009 to 024	1, 3	1, 2, 6	150	180	90	90	150	180	90	90
025 to 029	1, 3	1, 2, 4, 6	150	180	90	90	150	180	90	90
030 to 081	1	1, 2, 4, 6	150	180	90	90	150	180	90	90
082 to 090	1	1, 2, 3, 4, 6	150	180	90	90	150	180	90	90
091 to 116	1	1, 2, 3, 4	150	180	90	90	150	180	90	90
117 to 165	1	1, 2, 3	140	180	80	90	140	180	80	80
166 to 186	1	1, 2, 3, 5	160	195	90	100	160	195	90	90
187 to 215	1	1, 2, 5	160	195	90	100	160	195	90	90
216 to 240	1, 2	1, 2, 5	160	195	90	100	160	195	90	90
241 to 289	1, 2	1, 2	140	180	80	90	140	180	80	80
290 to 318	1, 2, 3	1, 2	140	180	80	90	140	180	80	80
319 to 008	1, 3	1, 2	140	180	80	90	140	180	80	80
5 mile Radius, 10 miles Downwind										
002 to 025	1-3, 6	1-6, 12	180	210	100	110	180	210	100	110
026 to 031	1-3	1-6, 12	180	210	100	100	180	210	100	100
032 to 049	1-3	1-6, 10, 12	180	210	100	110	180	210	100	100
050 to 058	1-3	1-6, 8, 10, 12	180	210	100	110	180	210	100	100
059 to 075	1-3	1-6, 8, 10	180	210	100	110	180	210	100	100
076 to 087	1-3	1-6, 8	180	210	100	100	180	210	100	100
088 to 106	1-3	1-6, 7, 8	180	210	100	100	180	210	100	100
107 to 116	1-3	1-6, 7	170	210	100	100	170	210	100	100
117 to 146	1-3	1-6, 7, 9	170	210	100	100	170	210	100	100
147 to 169	1-3	1-6, 9	170	210	100	100	170	210	100	100
170 to 186	1-3	1-6, 9, 11	224	280	195	240	225	275	180	225
187 to 215	1-3	1-6, 11	224	280	195	240	225	275	180	225
216 to 229	1-3, 4	1-6, 11	224	280	195	240	225	275	180	225
230 to 239	1-3, 4	1-6	170	210	100	100	170	210	100	100
240 to 267	1-3,4,5	1-6	180	225	110	110	170	225	110	110
268 to 296	1-3, 5	1-6	180	225	110	110	170	225	110	110
297 to 318	1-3,5,6	1-6	180	225	110	110	180	225	110	110
319 to 001	1-3, 6	1-6	180	210	100	110	180	210	100	110
Entire EPZ			224	280	195	240	195	240	180	225

Table 4-2: Quad Cities Scenario Population Distribution By Subarea

	Winter Day		Winter Night		Summer Day		Summer Night	
Subarea	Population	Vehicles	Population	Vehicles	Population	Vehicles	Population	Vehicles
IL-1	578	374	336	167	806	523	497	249
IL-2	1,342	460	1,060	403	1,119	442	1,087	412
IL-3	1,007	398	982	373	1,027	405	1,002	380
IL-4	614	252	584	222	629	249	611	231
IL-5	434	165	434	165	434	165	434	165
IL-6	4,509	1,506	2,827	1,075	3,782	1,567	3,419	1,274
IL total	8,484	3,156	6,223	2,406	7,797	3,350	7,050	2,710
IA-1	113	42	92	40	341	138	174	77
IA-2	26	17	15	6	66	24	15	6
IA-3	637	246	637	246	637	246	637	246
IA-4	448	170	448	170	449	171	449	171
IA-5	9,140	4,937	6,026	2,894	8,252	4,931	6,197	2,951
IA-6	1,640	550	1,344	511	1,376	533	1,359	516
IA-7	434	165	434	165	434	165	434	165
IA-8	547	208	547	208	548	208	548	208
IA-9	690	196	476	182	476	182	476	182
IA-10	369	140	369	140	370	141	370	141
IA-11	39,272	16,275	31,794	14,209	35,681	16,548	33,944	14,873
IA-12	5,327	1,695	4,072	1,548	4,553	1,760	4,262	1,629
IA total	58,643	24,643	46,254	20,321	53,183	25,047	48,865	21,165
EPZ	67,127	27,798	52,477	22,726	60,980	28,398	55,915	23,876

Table 4-3: Quad Cities Subarea Locations



Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities

Refer to Figure 5-1 for the location of the Quad Cities Station Control Room, Technical Support Center (TSC), and Operations Support Center (OSC) within the Station's Protected Area boundary.

5.1.1 Station Control Room

The Quad Cities Station Control Room shall be the initial onsite center of emergency control. The Control Room is located on the 620-foot elevation of the Service Building.

5.1.2 Technical Support Center (TSC)

Quad Cities Station has established a Technical Support Center (TSC) in a building located south of the Service Building. The TSC fully meets the requirements of Section H.1.b of the Emergency Plan.

5.1.3 Operational Support Center (OSC)

Quad Cities Station has designated an Operational Support Center. The OSC is located on the ground floor in the Service Building in a space designated as the Outage Control Center. The OSC conforms to the requirements of Section H.1.c of the Emergency Plan, and is the location to which operations support personnel will report during an emergency and from which they will be dispatched for assignments in support of emergency operations.

5.2 Assessment Resources

5.2.1 Onsite Meteorological Monitoring Instrumentation

The meteorological tower, located 1623 meters SSE of the plant, is 300 ft. high and is instrumented at three levels. The 33 ft., 196 ft. and 296 ft. levels correspond to the elevations of the possible points of airborne effluent release. Wind speed and wind direction are measured at all three elevations. Ambient temperature is measured at 33 ft. and differential temperatures referenced to 33 ft. are measured at 196 ft. and 296 ft. Precipitation is measured nearby.

The onsite meteorological monitoring program is covered in the contractor specification and vendor procedures of the meteorological monitoring contractor. These data are used to generate wind roses and to provide estimates of airborne concentrations of gaseous effluents.

5.2.1.1 Instrumentation

The meteorological tower is instrumented with equipment that conforms with the recommendations of Regulatory Guide 1.23 and ANSI/ANS 2.5 (1984). The equipment is placed on booms oriented into the general prevailing wind at the site. Equipment signals are brought to the process computers and to an instrument building with controlled environmental conditions. The building at the base of the tower houses the recording equipment, signal conditioners, etc., used to process and re-transmit the data to the end point users.

5.2.1.2 Meteorological Measurement Program During a Disaster

Cooperation between the corporate office and the meteorological contractor assures that a timely restoration of any outage can be made. Emergency field visits to the site are made as quickly as possible after detection of a failure.

Should a disaster of sufficient magnitude occur to destroy the tower structure, a contract is maintained to have a temporary tower erected within 72 hours, weather conditions permitting. Further, the meteorological contractor maintains two levels of sensors (wind speed, wind direction and temperature) in a state of readiness for use on the temporary tower.

Additionally, Exelon Nuclear's existing instrumented towers at other nuclear sites provide a high density measurement network with multiple backup opportunities.

Meteorological data are available to the station Control Room, Technical Support Center, and Emergency Operations Facility for use in the Dose Assessment Computer Model for estimating the environmental impact of unplanned releases of radioactivity from the station.

5.2.2 Onsite Radiation Monitoring Equipment

Sections 2.7, 7.6 and 9.5 of the UFSAR for Quad Cities Station, Unit 1 and 2, describe in detail the radiation monitoring systems and equipment. The modified off-gas treatment system is described in Section 9.2 of the UFSAR. In addition to the dedicated systems described here, chemistry and health physics personnel are trained and equipped to perform radiological monitoring and sampling.

The radiation monitoring systems and equipment can be categorized into four (4) groups:

5.2.2.1 Radiological Noble Gas Effluent Monitoring: A wide-range monitoring system is installed in the effluent stream in the main chimney and in the effluent stream of the reactor building vent stack. Methods for converting instrument readings to release rates have been developed and are incorporated into Station procedures.

5.2.2.2 Radioiodine and Particulate Effluent Monitoring: Effluent sampling media are analyzed in the Station counting room using a GeLi isotopic system.

5.2.2.3 High-Range Containment Radiation Monitors: Two high range containment radiation monitors are installed on each of Quad Cities Station's units. The range of these monitors is from 1 R/hr to 108 R/hr.

5.2.2.4 In-plant Iodine Instrumentation: Quad Cities Station has the capability to sample and determine iodine concentrations in the plant using Silver Zeolite or charcoal cartridges and gamma ray spectroscopy. Monitors may be used to measure increasing levels of iodine during emergency conditions (e.g. a portable gamma ray spectroscopy system).

5.2.3 Onsite Process Monitors

Adequate monitoring capability exists to properly assess the plant status for all modes of operation. The operability of the post-accident instrumentation ensures information is available on selected plant parameters to monitor and assess important variables following an accident. Instrumentation is available to monitor the parameters and ranges given in Technical Specifications.

Station procedures have been developed which would aid personnel in recognizing inadequate core cooling using applicable instrumentation.

5.2.4 Onsite Fire Detection Instrumentation

Quad Cities Station has a fire protection system that is designed to quickly detect any fires, annunciate locally and in the Control Room, and initiate the appropriate automatic action.

The station fire protection system is described in the Fire Hazards Analysis Report. The detection instrument minimum requirements and further system description are contained in QCAP 1500-1 (Administrative Requirements for Fire Protection). In the event that a portion of the fire detection instrumentation is inoperable, contingency actions are taken as defined in the above.

5.2.5 Facilities and Equipment for Offsite Monitoring

Consult Chapter 11 of the station specific Offsite Dose Calculation Manual (ODCM) for the most current location for fixed continuous air samplers and TLD locations.

5.2.6 Site Hydrological Characteristics

Assessments covering the hydrological aspects of the site (i.e., effects of the Mississippi River) are made as follows:

a. Onsite: River level gauge located in the intake bay.

b. Offsite: The U.S. Army Corps. of Engineers will provide information regarding river levels and other conditions of importance. (Flood information can be obtained from the U.S. National Weather Service.)

5.3 Protective Facilities and Equipment

The onsite assembly areas for Quad Cities Station are shown in Figure 4-2. These areas are suitable because:

1) They are large open areas suitable for assembling a large number of people in a short time;
2) They are relatively close to the Security Gatehouse; and
3) They have a low probability of being affected by a serious accident involving the Nuclear Steam Supply System (NSSS).

The offsite evacuation assembly areas for Quad Cities Station are discussed in Section 4.4 of this annex. These areas are suitable because they are easily accessible. The relocation routes to these facilities would be determined by the actual wind direction at the time of evacuation.

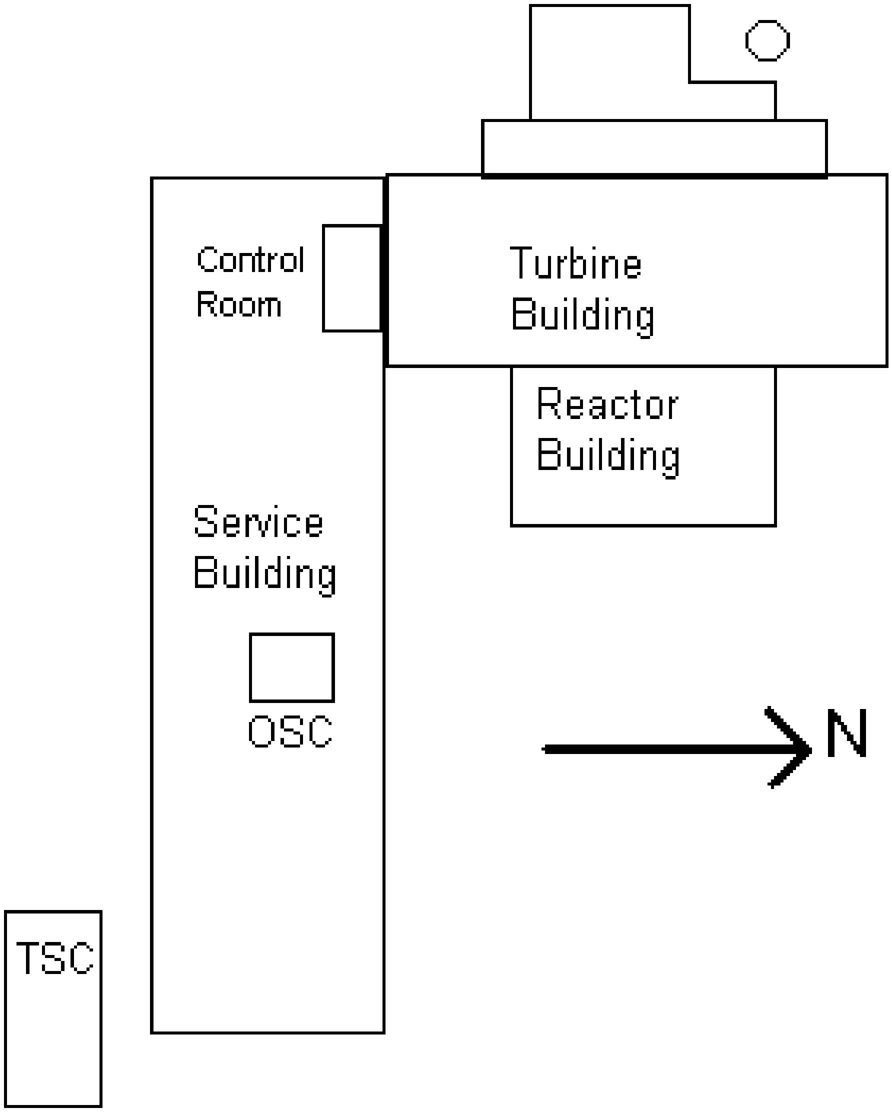
5.4 First Aid and Medical Facilities

Quad Cities Station has a decontamination/first aid room on the ground floor of the Service Building near the entrance to the plant. This room is provided with a sink, showers, and supply cabinet.

First aid kits, stretchers, sinks, eyewashes, and emergency showers have been placed in strategic locations throughout the station.

Medical treatment given to injured persons at the station is of a "first aid" nature. When more professional care is needed, injured persons are transported to a local hospital or clinic. Genesis Medical Center Illini Campus in Silvis, Illinois, is the Quad Cities Station primary supporting medical facility. Trinity Medical Center West Campus in Rock Island, Illinois is the backup medical facility for evaluation and treatment of persons suffering from traumatic injury, medical illness, or radiation exposure and uptake.

Figure QCA 5-1: Location Of Onsite Area Emergency Response Facilities



Appendix 1: NUREG-0654 Cross-Reference

Annex Section	NUREG-0654		Annex Section	NUREG-0654
1.0	Part I, Section A		5.1	Part II, Section H.1 & G.3
1.1	Part I, Section C		5.2.1	Part II, Section H.5.a & 8
1.2	Part I, Section D		5.2.2	Part II, Section H.5.b & I.2
1.3	Part II, Section A.1		5.2.3	Part II, Section H.5.c
Figure 1-1	Part I, Section D		5.2.4	Part II, Section H.5.d
			5.2.5	Part II, Section H.6.b & 7
2.0	Part II, Section A.4		5.2.6	Part II, Section H.5.a & 6.a
2.1	Part II, Section A.3		5.3	Part II, Section J.1-5
			5.4	Part II, Section L.1 & 2
3.0	Part II, Section D		Figure 5-1	Part II, Section H.1
4.1	Part II, Section E.1 & J.7			
4.2	Part II, Section I.2 & 3			
4.3	Part II, Section J.10.m			
4.3.1	Part II, Section E.6			
4.3.2	Part II, Section J.8			
4.4	Part II, Section J.1-5			
Figure 4-1	Part II, Section J.10.m			
Figure 4-2	Part II, Section J.5			
4.4	Part II, Section J.2 & 3			
Table 4-1	Part II, Section J.8			
Table 4-2	Part II, Section J.10.b			

Appendix 2: Station Letters of Agreement

1. The Illinois State Police - law enforcement
2. The Rock Island County Sheriff's Office -law enforcement
3. Genesis Medical Center Illini Campus in Silvis, Illinois - medical treatment and ambulance services
4. Trinity Medical Center West Campus – medical treatment.
5. Cordova Fire Department - fire protection

Attachment 7

EP-AA-110-301

"Core Damage Assessment (BWR) "

Revision 4

CORE DAMAGE ASSESSMENT (BWR)

1. PURPOSE

1.1. This procedure provides emergency response personnel with the methodology to estimate the degree of possible core damage at Exelon Nuclear's Boiling Water Reactor (BWR) stations. Refer to EP-AA-110-302 for methodology to estimate potential core damage for a Pressurized Water Reactor (PWR).

1.2. This Core Damage Assessment process is designed to assist in estimating core damage after an accident with potential clad or core damage conditions, and is intended to provide an acceptable alternative to existing station core damage assessment models and methods utilized by Reactor Engineering to assist in the following:

- Determining if the fuel barriers are breached to evaluate the appropriate Emergency Action Level (EAL) classification.
- Providing input on core configuration (coolable or uncoolable) for prioritization of mitigating activities.
- Determining the potential quantity and isotopic mix of a radiological release to project offsite doses.
- Predicting the radiation protection actions that should be considered for long term recovery activities.
- Satisfying inquiries from local and federal government agencies and provide evidence that the utility knows the plant conditions.

1.3. Core damage may be assessed by:

- Evaluating the drywell radiation levels (and confirmed by evaluating the extent of time the core was uncovered),
- Concentration of certain isotopes in a reactor coolant analysis, or
- Concentration of hydrogen in the primary containment.
- History of Core Cooling

2. TERMS AND DEFINITIONS

2.1. **BWR** – Boiling Water Reactor

2.2. **Cladding** – The outer coating (usually zirconium alloy), which covers the nuclear fuel elements to prevent corrosion of the fuel and the release of fission products into the coolant.

2.3. **Containment Type** –

- Clinton (Mark III)
- Dresden (Mark I)
- LaSalle (Mark II)
- Limerick (Mark II): 764 assemblies

  Cont. Volume (384,570 ft^3) = Suppression Pool (149,380 ft^3) + Drywell (235, 190 ft^3)
- Peach Bottom (Mark I): 764 assemblies

  Cont. Volume (303,600 ft^3) = Suppression Pool (127,800 ft^3) + Drywell (175, 800 ft^3)
- Quad Cities (Mark I)

2.4. **Core Release Fraction** – The fraction of each isotope in the core inventory that is assumed to be released from the core under given core conditions.

2.5. **Core Uncovery Time** – For BWRs this is the period of time when reactor water level is less than that required for minimum steam cooling, or about $\geq$ 20% of the core active fuel is uncovered.

2.6. **Cladding Failure**

1. Also referred to as “Cladding Oxidation”, “Gap Release” or “Clad Rupture” in other documents.
2. 100% clad failure refers to the rupture of 100% of the fuel rods in the core. This would result in all fission products contained in the gap space being released to the reactor coolant system.

2.7. **Equilibrium** – Conditions associated with evaluation of different volumes of liquid or gas that contain concentrations of radioactive materials or hydrogen, when these concentrations are approximately the same. This is normally an extended period of time following accident initiation.

2.8. **Fission Products** – The nuclei (fission fragments) formed by the fission of heavy elements or by subsequent radioactive decay of the fission fragments.

2.9. **Fuel Melt**

1. Referred to as “Core Melt,” “In-Vessel Melt” or “Over-temperature” damage in reference documents.
2. 100% fuel melt refers to high temperatures in the fuel pellets in 100% of the fuel rods in the core. This would result in all the fission products contained in the fuel pellet matrix being released to the reactor coolant system.

2.10. **Gap** – The space inside a reactor fuel rod that exists between the fuel pellet and the fuel rod cladding.

2.11. **Gap Release** – The release into containment of fission products in the fuel pin gap.

2.12. **In-Vessel Core Melt** – A condition during a reactor accident in which some of the cladding or reactor fuel melts as a result of overheating the fuel and remains inside the reactor vessel.

2.13. **In-Vessel Core Melt Release** – A release into containment from the reactor vessel, which assumes the entire core has melted, releasing a representative mixture of radioisotopes.

2.14. **Minimum Steam Cooling RPV Water Level (MSCRWL)** – The lowest RP water level at which the covered portion of the reactor core will generate sufficient steam to maintain the hottest clad temperature below 1500ºF.

2.15. **Minimum Zero-Injection RPV Water Level (MZIRWL)** – The lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to maintain the hottest clad temperature below 1800ºF, assuming no injection into the RPV.

2.16. **Shutdown** – As defined by station emergency operating procedures.

2.17. **Slump** – Relocation of molten reactor core during an accident.

2.18. **Source Term** – The amount and isotopic composition of material released or the release rate, used in modeling releases of material to the environment.

2.19. **Spiked Coolant** – Reactor coolant containing increased concentrations of non-noble isotopes, sometimes seen with rapid shutdown or depressurization of primary system.

2.20. **Spiked Coolant Release** – The release into containment of 100 times the non-noble gas fission products found in the coolant.

2.21. **Subcritical** – The reactor condition when the number of neutrons released by the fission is not sufficient to achieve a self-sustaining nuclear chain reaction. Defined under station emergency operating procedures.

2.22. **Suppression Chamber** – May also be referred to as Wetwell or Torus. The Large steel pressure vessel containing a large volume of water that acts as a heat sink for the Drywell.

2.23. **TID** – Total Isotopic Distribution

2.24. **Vessel Melt-Through**

1. Referred to as “Ex-Vessel Melt” or “Melt Release” in reference documents.
2. Core debris is relocated to the primary containment building after the reactor pressure vessel has failed.

3. RESPONSIBILITIES

3.1. The TSC Core/Thermal Hydraulic Engineer shall serve as the Core Damage Assessment Methodology (CDAM) Evaluator.

3.2. The TSC Radiation Controls Engineer shall coordinate radiological and chemistry information with the Core/Thermal Hydraulic Engineer in support of core damage assessment.

3.3. The TSC Technical Manager shall coordinate core damage assessment activities.

4. MAIN BODY

4.1. **REFER** to Attachment 1, BWR CDAM User Guide for instructions on use of the Core Damage Assessment Methodology (CDAM) Software Program.

5. DOCUMENTATION

5.1. A Summary Form and method specific reports are generated by the BWR CDAM Software for use in documenting the results of the assessment.

6. REFERENCES

6.1. NEDO-22214, Procedures for the Determination of the Extent of Core Damage Under Accident Conditions

6.2. NEDC-33045P, Rev 0 (July 2001), Methods of Estimating Core Damage in BWRs

6.3. WCAP-14696 (July 1996) Westinghouse Owners Group Core Damage Assessment Guidance.

6.4. WCAP-14696-A (November 1999), Westinghouse Owners Group Core Damage Assessment Guidance.

6.5. NUREG-1228, “Source Term Estimation During Incident Response to Severe Nuclear Power Accidents”

6.6. Station Commitments

6.6.1. Peach Bottom

CM-1 T04511 (Attachment 1, 5.6)

6.6.2. Limerick Bottom

CM-2 T04512 (Attachment 1, 5.6)

7. ATTACHMENTS

7.1. Attachment 1, BWR CDAM User Guide

Attachment 1
BWR CDAM User Guide
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1. OVERVIEW

1.1. As a windows based application designed in Microsoft Access, BWR CDAM, uses many standard user interfaces. Instructions are not provided in basic computer operations in the windows® environment. The user must be familiar with these to efficiently operate the program.

1.2. It is also assumed user is familiar with basic reactor physics and core damage fundamentals. Emergency Response Organization training will provide an overview of core damage assessment methodologies.

1.3. The program should be used by qualified personnel as a tool to estimate the type and amount of core damage.

2. DETERMINE APPROPRIATE AND AVAILABLE ASSESSMENT METHODS

Mid-West Region Stations

REFER to EP-MW-110-1001 for a listing of appropriate plant parameter points to be used following a LOCA.

2.1. The magnitude and type of event, transport mechanism and time after shutdown will be influencing factors on the method(s) utilized to determine the extent of core damage. Damage estimates can be developed using one or more methods as they become available or applicable.

2.1.1. Indications Of Core Damage

1. The primary indicators of core damage that are available during the early phases of an event:
   - Drywell/Containment Radiation Monitor Readings
   - Drywell/Containment Hydrogen Readings
2. Auxiliary indicators that are used to confirm and better define the possible type of damage are:
   - Reactor Pressure Vessel Level Indication System readings
   - Estimation of maximum temperature reached within the core
   - Estimated core uncovery time
   - Abnormal Source Range Monitor readings

3. Long Term Indicators (once liquid or gaseous samples can be safely obtained) are:
    - Isotopic Ratios
    - Presence of high levels of rare isotopes
    - Quantity of isotopes present in samples

2.1.2. **SELECT** the assessment method(s) most appropriate for the existing conditions. Methods available for assisting in the determination of the extent of core damage include the following:

Method	Use	Comment
Containment Radiation Monitor	Early Indication of Core Damage	Uncertainties due to variables in release of fission products from RCS and effects of containment sprays.
Core Conditions	Indication of onset of Core Damage	May not be reliable during later phases of core overheating due to changes in core geometry.
RPV Level	Indication of Core Uncovery	Indicates possible damage not useful in estimating the quantity of damage.
Source Range Monitor	Indication of Core Uncovery	Loss of water level leads to decrease in the monitor's measured radiation field.
Containment Hydrogen Monitor	Early Indication of Core Damage	Significant uncertainties due to variable Hydrogen generation in core and in release of Hydrogen from RCS and effects of containment sprays.
RCS Samples and Containment Sump and Atmosphere Samples	Late Indication of Core Damage —-Suppression Pool Samples provide indication of Rx Vessel Failure	Very large uncertainties until all systems have reached equilibrium. Useful in planning long term recovery.

3. <u>START UP THE CDAM PROGRAM</u>

3.1. **ACCESS** the application by one of the following:

3.1.1. **OPEN** the BWR CDAM desktop icon on applicable computers.

1. **START** the BWR CDAM program for the plant that has declared an emergency. Programs are labeled BWR CDAM.

2. **SELECT** the appropriate icon or run from the 'start bar' and type in the file path and name as follows **<u>C:\CDAM\BWR CDAM.MDB</u>**

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3.1.2. **If** the assigned Core Damage Assessment Computer cannot access the application or the CDAM program will not run, **then** install BWR CDAM on any computer from CDs or Disks located in the TSC or the EOF Library.

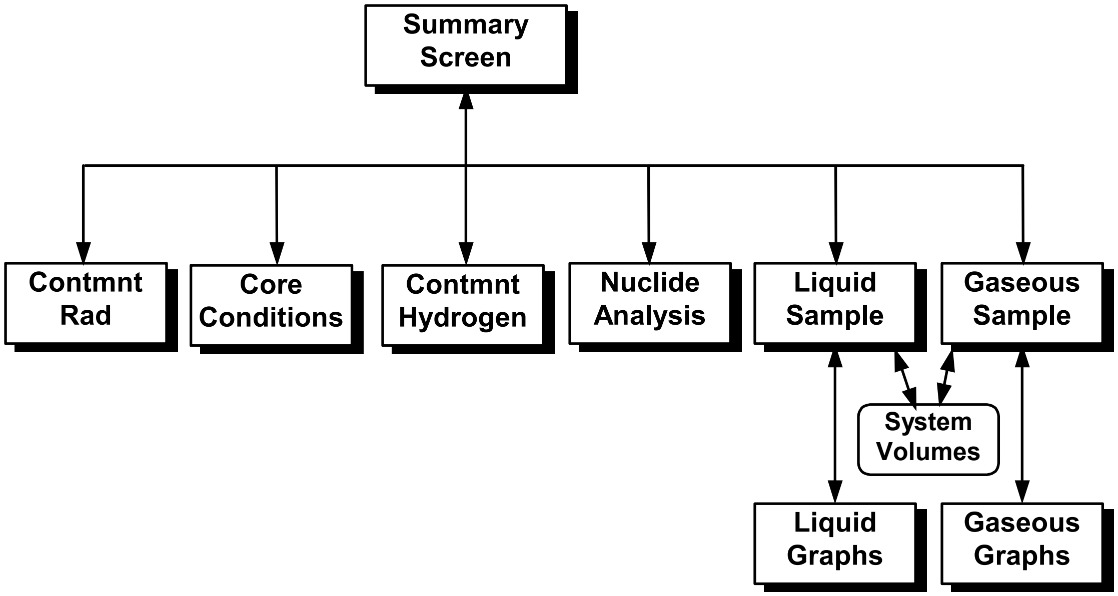
1. **INSTALL** CDAM by copying appropriate file to computer's hard drive.
2. **UPDATE** the "properties" of the file by deselecting write protection.

4. SELECTION AND PERFORMANCE OF ASSESSMENT

4.1. **SELECT** the assessment method(s) most appropriate for the existing conditions. Methods available for assisting in the determination of the extent of core damage include the following:

- Containment Radiation Analysis - (Section 5.2)
- Core Conditions Analysis (Cooling History) - (Section 5.3)
- Containment Hydrogen Analysis - (Section 5.4)
- Nuclide Analyses (Ratios and Abnormal Isotopes) - (Section 5.5)
- Liquid Samples Analysis - (Section 5.6)
- Gaseous Samples Analysis - (Section 5.7)

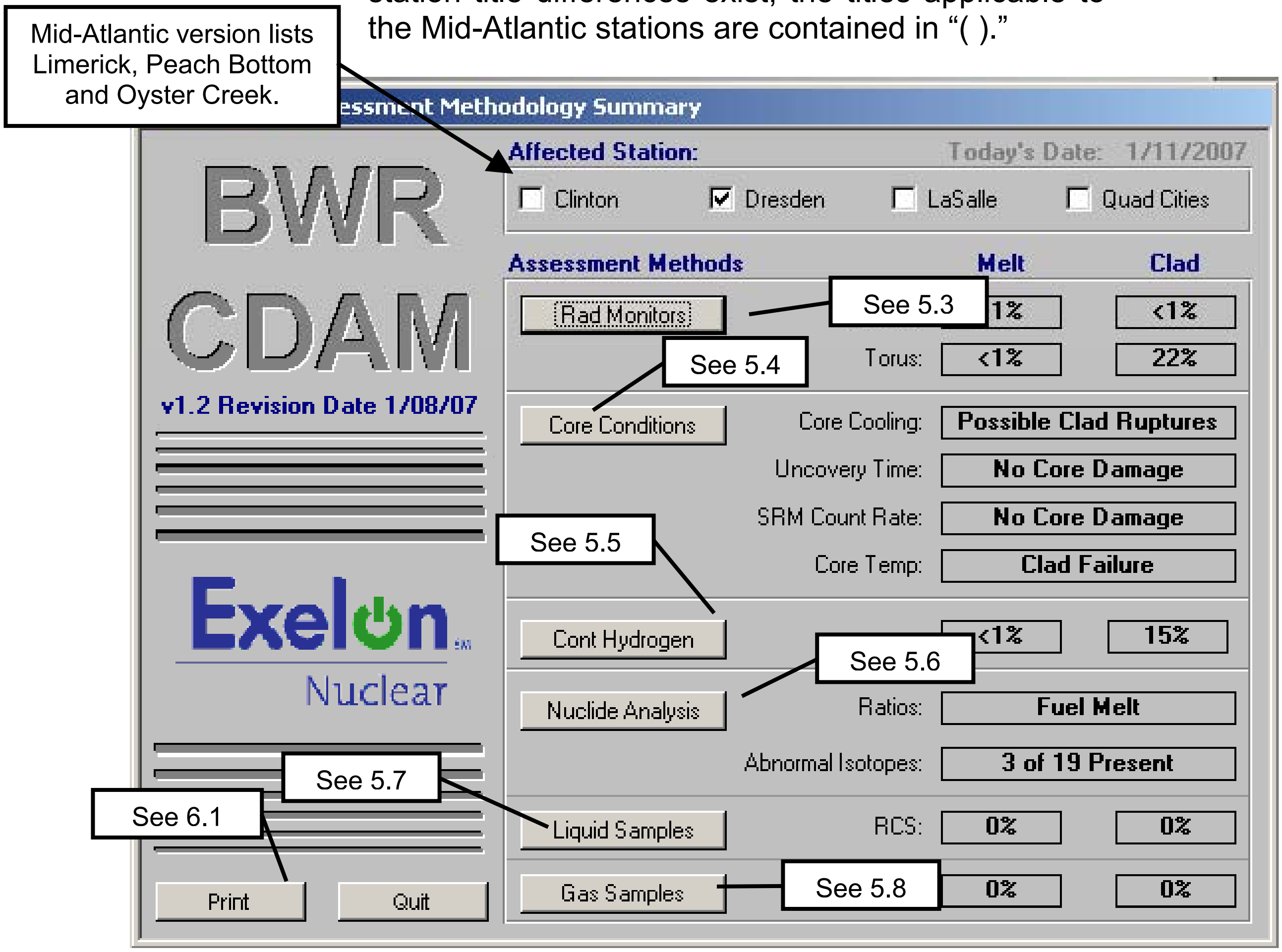
Basic Program Flow Diagram



5. <u>PROGRAM SCREENS AND INPUTS</u>

5.1. When the program is started the following screen appears:

NOTE: The value boxes are empty when the program is originally launched. The examples below may deviate from the CDAM displays during use due to different software versions in use in the Mid-Atlantic and Midwest regions. The display differences do **<u>not</u>** impact the functionality of the program. Where station title differences exist, the titles applicable to the Mid-Atlantic stations are contained in "( )."



Mid-Atlantic version lists Limerick, Peach Bottom and Oyster Creek.

essment Methodology Summary

v1.2 Revision Date 1/08/07

Affected Station: ☐ Clinton ☑ Dresden ☐ LaSalle ☐ Quad Cities

Today's Date: 1/11/2007

Assessment Methods		Melt	Clad
Rad Monitors		1%	<1%
	Torus:	<1%	22%
Core Conditions	Core Cooling:	Possible Clad Ruptures	
	Uncovery Time:	No Core Damage	
	SRM Count Rate:	No Core Damage	
	Core Temp:	Clad Failure	
Cont Hydrogen		<1%	15%
Nuclide Analysis	Ratios:	Fuel Melt	
	Abnormal Isotopes:	3 of 19 Present	
Liquid Samples	RCS:	0%	0%
Gas Samples		0%	0%

See 5.3
See 5.4
See 5.5
See 5.6
See 5.7
See 6.1
See 5.8

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CAUTION

Selecting an "Affected Station" resets all inputs to default values.

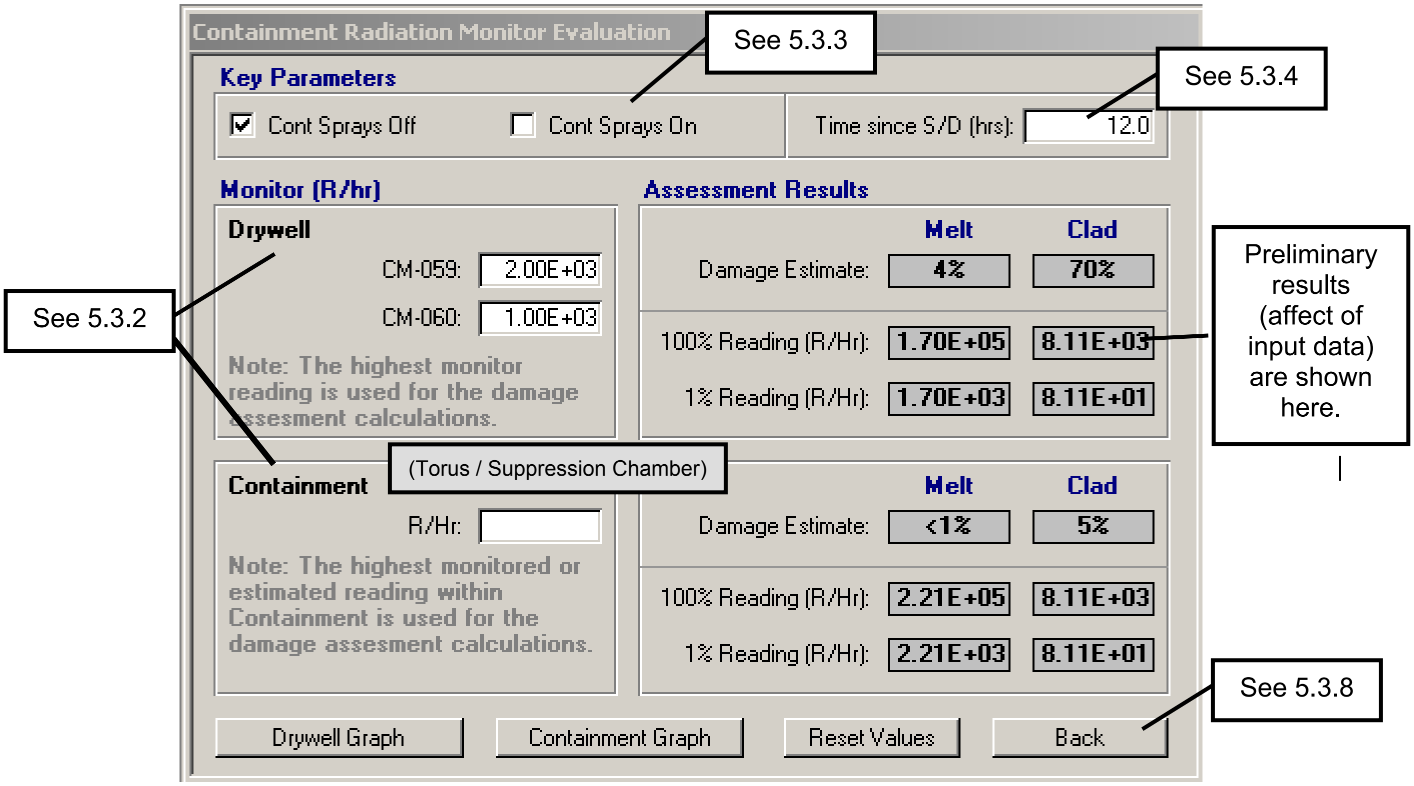
5.2. **SELECT** the Affected Station before other “Assessment Methods.”

CAUTION

Pressing the **“Quit”** button exits the program. When the program is closed all data is reset. Program saves no information to disk; printed reports serve as record of core damage assessments.

5.3. Drywell/Containment Radiation Monitor Method

5.3.1. **PRESS** the “Cont Rad Monitors” button on the Summary Screen to open the following form:



Containment Radiation Monitor Evaluation

Key Parameters

☑ Cont Sprays Off ☐ Cont Sprays On

Time since S/D (hrs): 12.0

Monitor (R/hr)

Drywell

CM-059: 2.00E+03

CM-060: 1.00E+03

Note: The highest monitor reading is used for the damage assesment calculations.

Containment

R/Hr:

Note: The highest monitored or estimated reading within Containment is used for the damage assesment calculations.

Assessment Results

	Melt	Clad
Damage Estimate:	4%	70%
100% Reading (R/Hr):	1.70E+05	8.11E+03
1% Reading (R/Hr):	1.70E+03	8.11E+01

	Melt	Clad
Damage Estimate:	<1%	5%
100% Reading (R/Hr):	2.21E+05	8.11E+03
1% Reading (R/Hr):	2.21E+03	8.11E+01

NOTE: Program allows entry from 2 high range monitors for Drywell location and 1 for Torus or Containment / Suppression Chamber, however a reading may be entered from any monitor or measurement taken external to suppression chamber, which accurately indicated containment radiation levels. If two entries are made only the highest is used.

5.3.2. **ENTER** the highest Drywell radiation monitor reading that occurred in these boxes

1. **If** Drywell radiation monitor readings are not available, **then** enter the containment / Suppression Chamber radiation monitor reading.

5.3.3. **SELECT** Drywell/Containment Spray status:

1. Choose “Cont Sprays On” when sprays are operated for the majority of the time after the following events:

The expected start of Core Damage

AND

A release of the core isotopic inventory into the Drywell / Containment atmosphere has occurred.

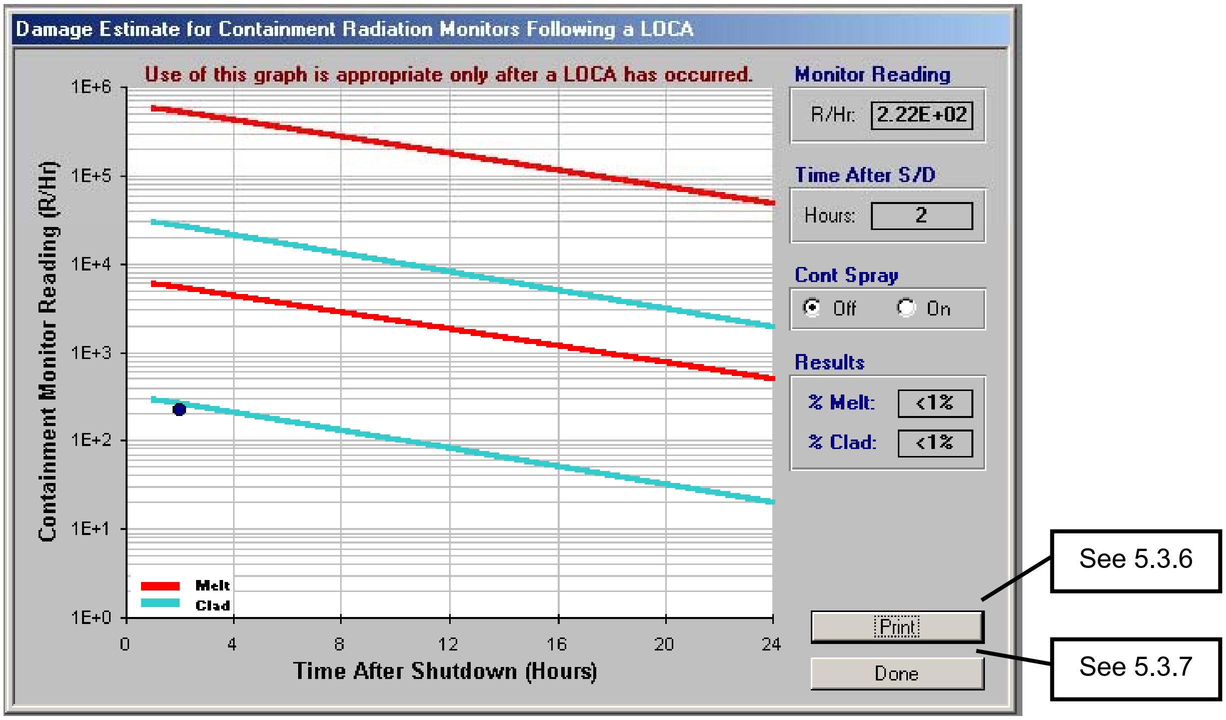
2. **If** the Drywell/Containment Spray system was **not** operated for the majority of the time after the start of Core Damage and the release of the isotopic inventory into containment **then** choose "Cont. Spray Off."

5.3.4. **ENTER** the time after reactor shutdown, which corresponds the time the containment radiation reading was taken. Value must be between 1 hour and 24 hours after shutdown, which corresponds to the time period in which this method is considered effective.

NOTE: Pressing "Reset" button resets values on this form only.

5.3.5. **PRESS** "Containment Graph" or "Supp Chamber Graph" button to display a screen similar to the following:

(See example display on next page.)



Damage Estimate for Containment Radiation Monitors Following a LOCA

Use of this graph is appropriate only after a LOCA has occurred.

Monitor Reading

R/Hr: 2.22E+02

Time After S/D

Hours: 2

Cont Spray

● Off ○ On

Results

% Melt: <1%

% Clad: <1%

See 5.3.6

See 5.3.7

NOTE: Graph shows high and low containment radiation levels which correspond to 100% Melt or Clad or 1% Melt or Clad damage. A dot shows the last containment radiation level entered into the program for assessment.

5.3.6. **PRESS** the “Print” button to print a report of containment radiation method inputs and best estimate of damage.

5.3.7. **PRESS** the “Done” button to return to the Containment Radiation Monitor Evaluation Screen.

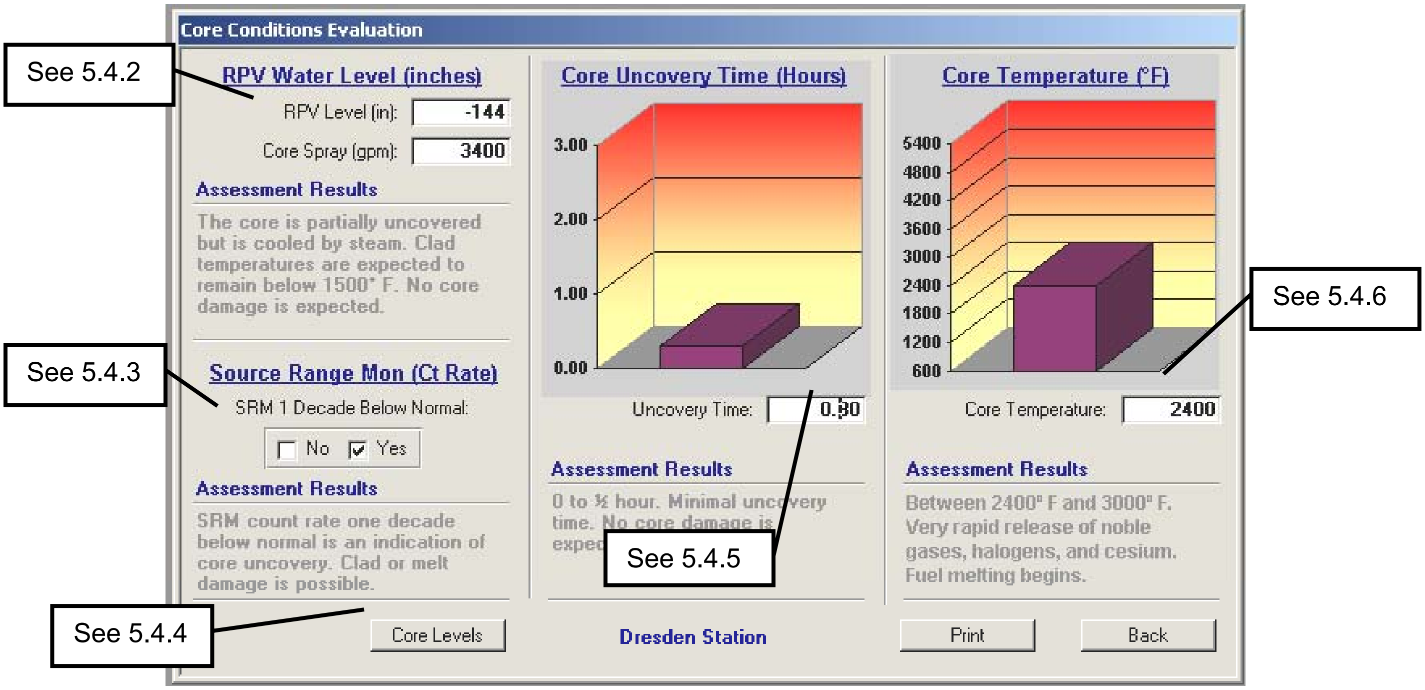
5.3.8. **PRESS** the **“**BACK**”** button to return to the Summary Screen.

5.4. Core Conditions Methods

NOTE: Each of these four methods is an independent assessment method.

5.4.1. **PRESS** the “Core Conditions” button on the Summary Screen to open the following form:

(See example form on next page.)



Core Conditions Evaluation

RPV Water Level (inches)

RPV Level (in): -144

Core Spray (gpm): 3400

Assessment Results

The core is partially uncovered but is cooled by steam. Clad temperatures are expected to remain below 1500° F. No core damage is expected.

Source Range Mon (Ct Rate)

SRM 1 Decade Below Normal:

☐ No ☑ Yes

Assessment Results

SRM count rate one decade below normal is an indication of core uncovery. Clad or melt damage is possible.

Core Uncovery Time (Hours)

Uncovery Time: 0.30

Assessment Results

0 to ½ hour. Minimal uncovery time. No core damage is expec

Core Temperature (°F)

Core Temperature: 2400

Assessment Results

Between 2400° F and 3000° F. Very rapid release of noble gases, halogens, and cesium. Fuel melting begins.

Dresden Station

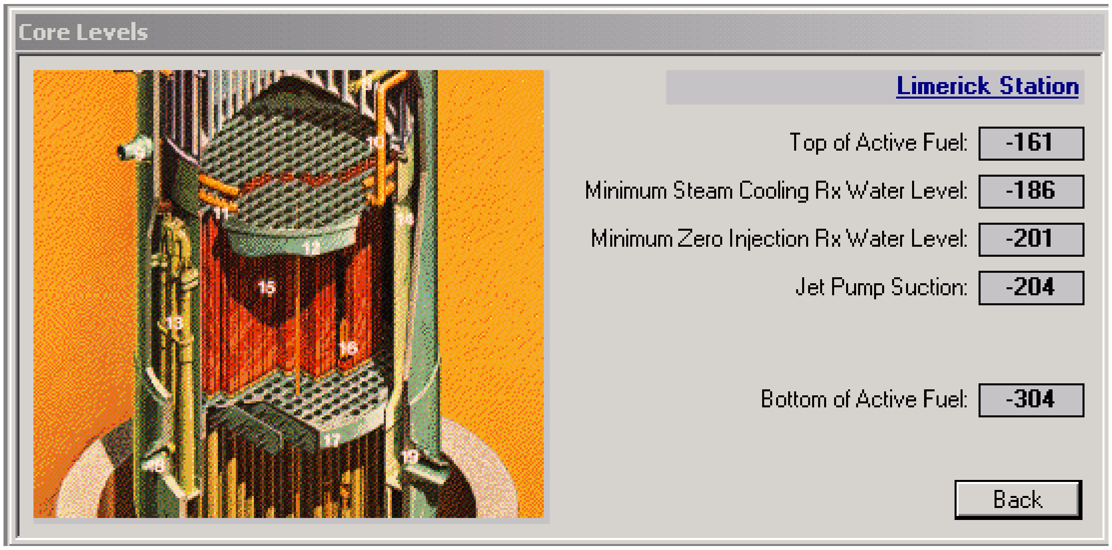
5.4.2. Under Reactor Pressure Vessel (RPV) Water Level **ENTER** the lowest recorded (or estimated) RPV level (range 0 to –350 inches) and core spray flow at time of lowest reading

NOTE: Steps 5.4.3 through 5.4.6 are based on inputs from Reactor Operators, TSC Staff and other engineering personnel (including outside sources such as General Electric personnel).

5.4.3. Under Source Range Monitor **REVIEW** plant parameter history and if the SRM (Wide Range Monitor at Peach Bottom) had indications of a reading 1 decade below those expected check “Yes.”

5.4.4. **PRESS** the “Core Levels” button to view information regarding water levels associated with the Station reactor and vessel level indications.

(See example form on next page.)



Top of Active Fuel: -161
Minimum Steam Cooling Rx Water Level: -186
Minimum Zero Injection Rx Water Level: -201
Jet Pump Suction: -204
Bottom of Active Fuel: -304

5.4.5. **ENTER** the estimated time the reactor core (20% of top of active core) was uncovered without steam (level below the Minimum Steam Cooling Rx Water Level) or spray cooling reactor core.

5.4.6. **ENTER** the estimated highest temperature reached in the reactor core.

5.4.7. **PRESS** the “Print” button to print a report of inputs and results of core temperature methods of core damage assessment.

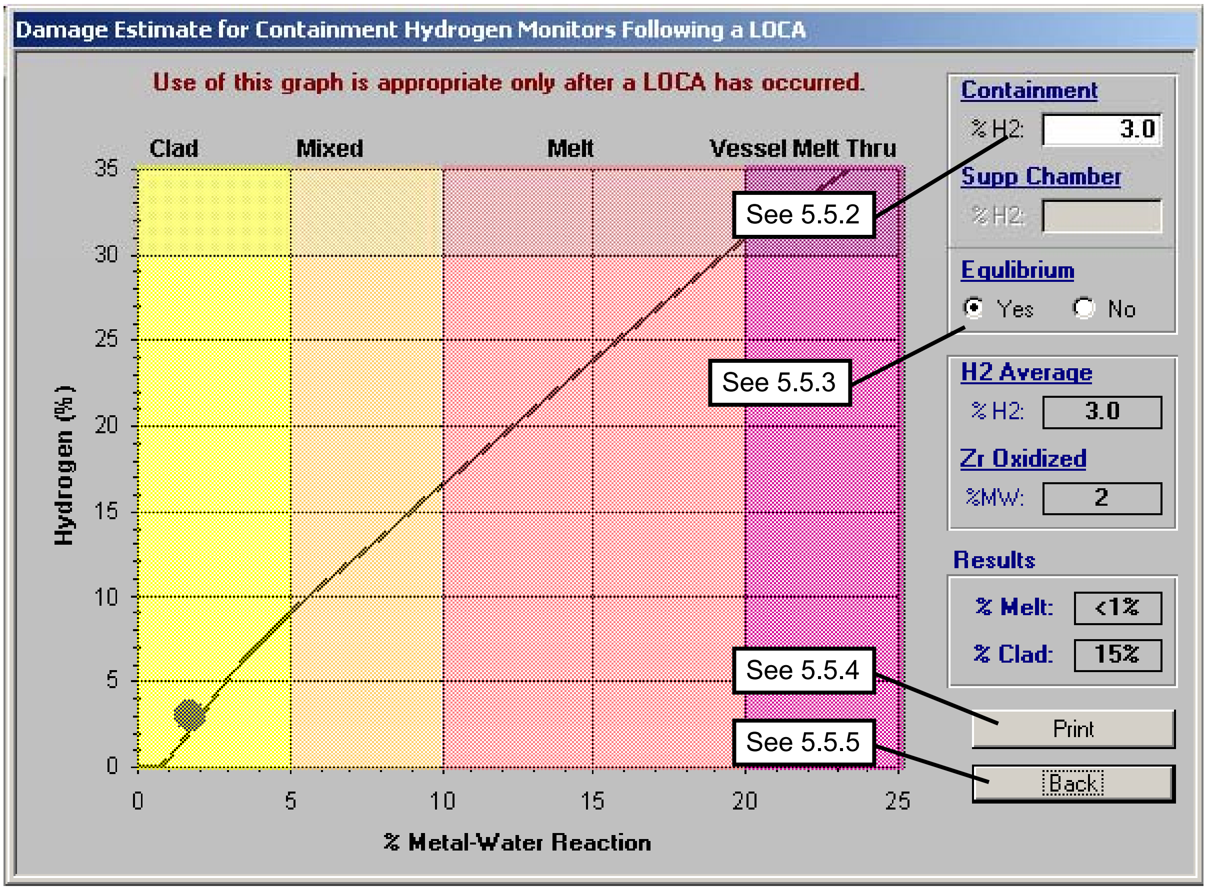
5.4.8. **PRESS** the “Back” button to return to the Summary Screen.

5.5. Containment Hydrogen Evaluations

CAUTION

This CDAM assumes no ignitor operation. Ignitor use limits containment hydrogen concentration affecting the reliability of this method.

5.5.1. **PRESS** the “Cont Hydrogen” button on the Summary Screen to open the following form:



Damage Estimate for Containment Hydrogen Monitors Following a LOCA

Use of this graph is appropriate only after a LOCA has occurred.

Containment
%H2: 3.0

Supp Chamber
%H2:

Equlibrium
● Yes ○ No

H2 Average
%H2: 3.0

Zr Oxidized
%MW: 2

Results
% Melt: <1%
% Clad: 15%

See 5.5.2
See 5.5.3
See 5.5.4
See 5.5.5

5.5.2. **ENTER** highest Drywell and/or Suppression Chamber hydrogen level measured.

NOTE: Suppression Chamber reading can only be entered if user selects “no” under Equilibrium in step 5.5.3 below.

5.5.3. **SELECT** the applicable System Equilibrium status based on the following:

NOTE: Atmospheres reach equilibrium through mechanical mixing i.e. fans circulating air between locations for several hours. Equilibrium conditions may be indentified when tempterature indications from different areas are approaching similar readings.

1. **If** Containment and Suppression Chamber monitors read the same or atmospheres are assumed equalized, **then SELECT** “Yes” for equilibrium.

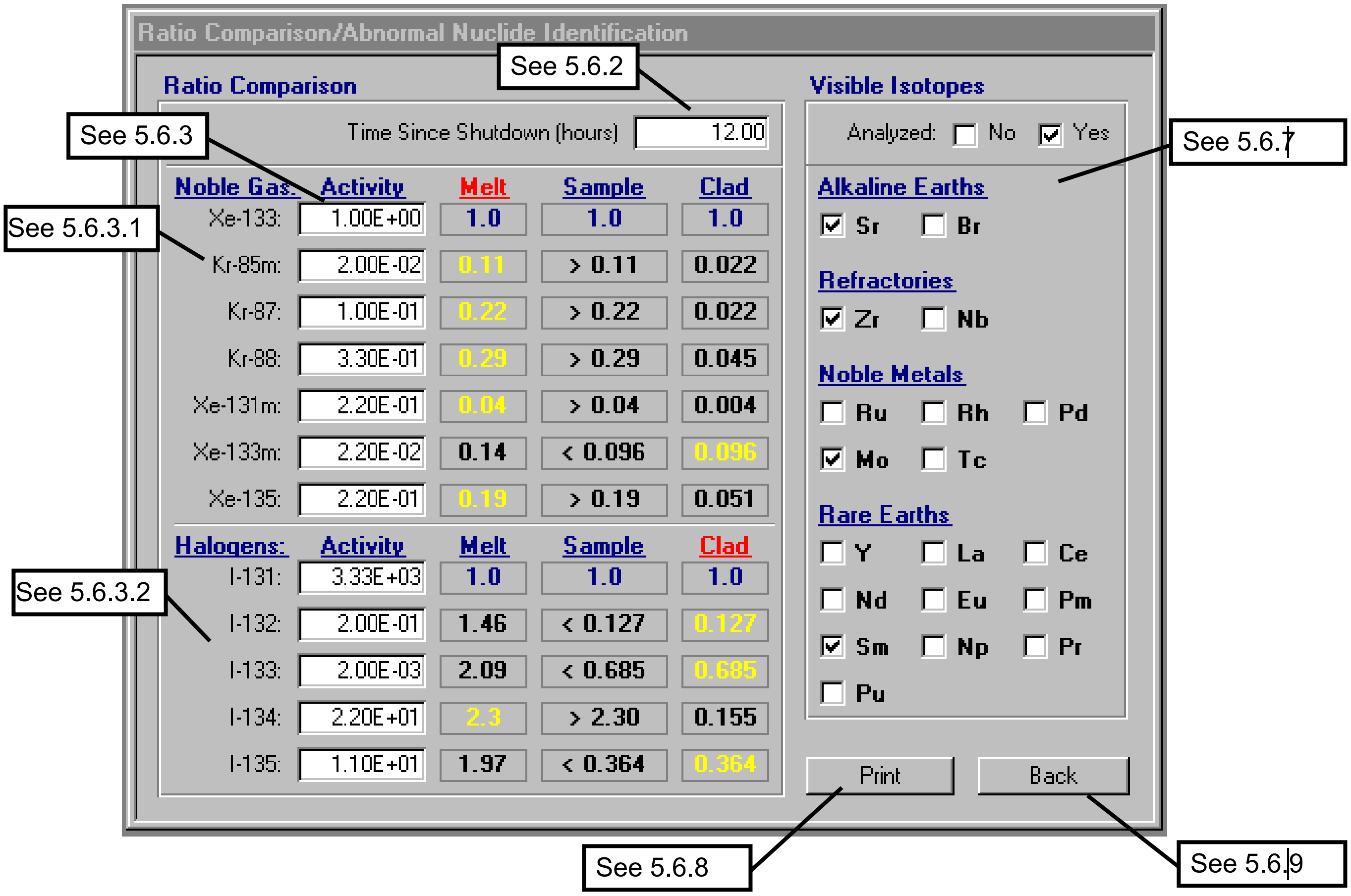
2. **If** containment and suppression chamber atmospheres are not in equilibrium **then SELECT** "No" for equilibrium.

5.5.4. **PRESS** the "Print" button to print a report of inputs and results of core level methods of core damage assessment.

5.5.5. **PRESS** the "Back" button to return to the Summary Screen.

5.6. Nuclide Analysis **(CM-1, CM-2)**

5.6.1. **PRESS** the “Nuclide Analysis” button on the Summary Screen to open the following form:



Ratio Comparison

Time Since Shutdown (hours) 12.00

Noble Gas:	Activity	Melt	Sample	Clad
Xe-133:	1.00E+00	1.0	1.0	1.0
Kr-85m:	2.00E-02	0.11	> 0.11	0.022
Kr-87:	1.00E-01	0.22	> 0.22	0.022
Kr-88:	3.30E-01	0.29	> 0.29	0.045
Xe-131m:	2.20E-01	0.04	> 0.04	0.004
Xe-133m:	2.20E-02	0.14	< 0.096	0.096
Xe-135:	2.20E-01	0.19	> 0.19	0.051

Halogens:	Activity	Melt	Sample	Clad
I-131:	3.33E+03	1.0	1.0	1.0
I-132:	2.00E-01	1.46	< 0.127	0.127
I-133:	2.00E-03	2.09	< 0.685	0.685
I-134:	2.20E+01	2.3	> 2.30	0.155
I-135:	1.10E+01	1.97	< 0.364	0.364

Visible Isotopes

Analyzed: ☐ No ☑ Yes

Alkaline Earths
☑ Sr ☐ Br

Refractories
☑ Zr ☐ Nb

Noble Metals
☐ Ru ☐ Rh ☐ Pd
☑ Mo ☐ Tc

Rare Earths
☐ Y ☐ La ☐ Ce
☐ Nd ☐ Eu ☐ Pm
☑ Sm ☐ Np ☐ Pr
☐ Pu

5.6.2. **ENTER** the time since reactor shutdown (time between shutdown and sample being drawn).

5.6.3. **ENTER** isotopic sample results in uCi/cc. Sample results are to be decay corrected back to time after shutdown that the sample was drawn.

1. Noble Gases are ratioed to Xe-133
2. Halogens are ratioed to I-131

5.6.4. **If** the ratios evaluated above are greater than predicted melt ratio, **then** melt damage is predicted

5.6.5. **If** the ratios evaluated above are less than clad ratio, **then** clad damage is predicted.

5.6.6. If the ratio is between the melt and clad ratio, the program determines damage type using following logic:

$$\frac{\text{Ratio}_{\text{melt}}}{\text{Ratio}_{\text{sample}}} < \frac{\text{Ratio}_{\text{sample}}}{\text{Ratio}_{\text{clad}}}$$

- **IF** the relationship is true, **THEN**: ....Fuel Melt
- **IF** the relationship is not true, **THEN**: ....Clad Failure

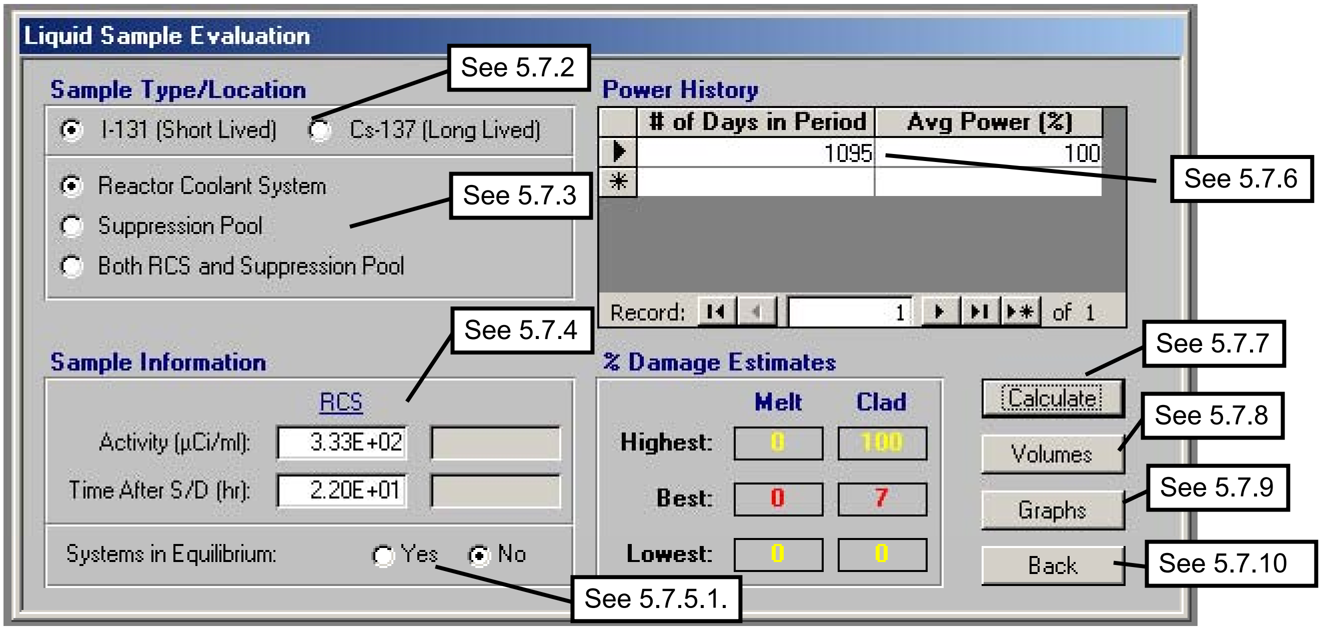
5.6.7. **If** abnormal levels of rare isotopes are present **then** check “Yes” **and** check which isotopes are present.

5.6.8. **PRESS** the “Print” button to print a report of inputs and results of core level methods of core damage assessment.

5.6.9. **PRESS** the “Back” button to return to the Summary Screen.

5.7. Liquid Samples

5.7.1. **PRESS** the “Liquid Samples” button on the Summary Screen to open the following form:



Liquid Sample Evaluation

Sample Type/Location

● I-131 (Short Lived) ○ Cs-137 (Long Lived)

● Reactor Coolant System
○ Suppression Pool
○ Both RCS and Suppression Pool

Sample Information

	RCS	
Activity (µCi/ml):	3.33E+02	
Time After S/D (hr):	2.20E+01	

Systems in Equilibrium: ○ Yes ● No

Power History

	# of Days in Period	Avg Power (%)
▶	1095	100
*		

Record: 1 of 1

% Damage Estimates

	Melt	Clad
Highest:	0	100
Best:	0	7
Lowest:	0	0

5.7.2. **SELECT** appropriate isotope.

5.7.3. **SELECT** sample location.

1. **If** samples are available from both locations, **then** select both.

5.7.4. **ENTER** Sample Information:

1. Activity is isotopic sample results in uCi/cc (uCi/ml). Sample results are to be decay corrected back to time after shutdown that the sample was drawn

2. Time After S/D (reactor shutdown) is the time between shutdown and sample being drawn.

5.7.5. **SELECT** the appropriate System Equilibrium status:

NOTE: Systems reach equilibrium through mechanical mixing i.e. pumps circulating water between locations for several hours. Equilibrium conditions may be indentified when tempterature indications from different systems are approaching similar readings.

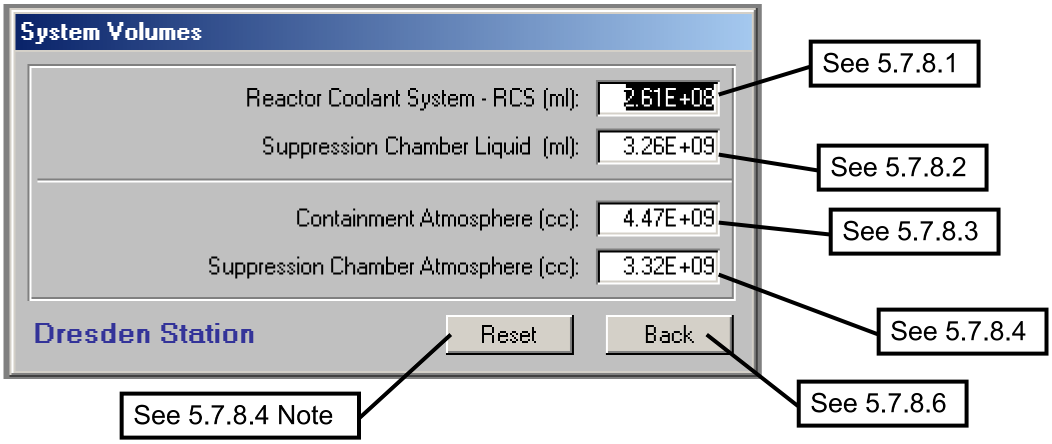
1. **If** sample was taken from only one location and systems are in equilibrium, **then** check "yes" for "Systems in Equilibrium," **otherwise** check "no."

5.7.6. **ENTER** power history (past to present, i.e. oldest steady state history as record number) of core since last refueling. Shutdown times are entered as the number of days with Ave Power (%) set at 0.

1. For short-lived isotopes, **EXTEND** Power History at least 30 days.
2. For long-lived isotopes, **EXTEND** power history at least 100 days, <u>however the power history for the extent of the cycle is preferred.</u>
3. **LIMIT** variations in steady state power to ± 20% within each operational period entered.

5.7.7. Once all data has been entered, **PRESS** the "Calculate" button to display the % Damage Estimates.

5.7.8. **PRESS** the "Volumes" button to display the follow screen:



Reactor Coolant System - RCS (ml): 2.61E+08
Suppression Chamber Liquid (ml): 3.26E+09
Containment Atmosphere (cc): 4.47E+09
Suppression Chamber Atmosphere (cc): 3.32E+09

See 5.7.8.1
See 5.7.8.2
See 5.7.8.3
See 5.7.8.4
See 5.7.8.4 Note
See 5.7.8.6

1. Program enters default RCS volume, which the user may change based on RPV Level Readings at time of sample.

2. Program enters default Suppression Chamber volume, which the user may change based on readings at time of sample.

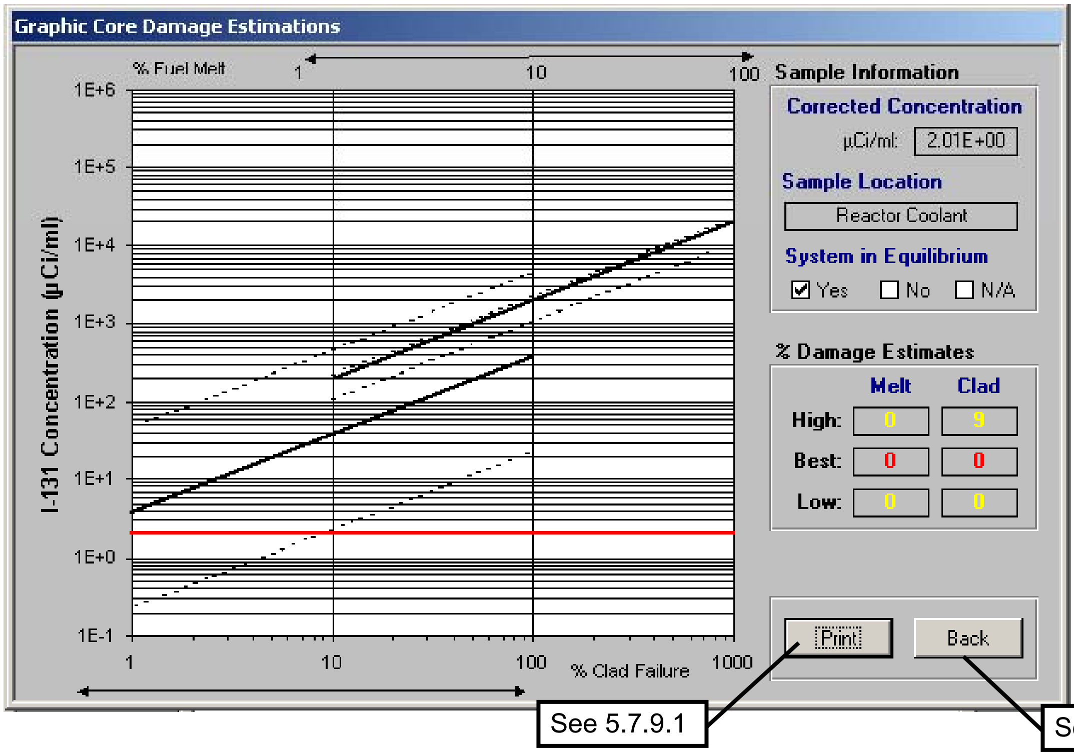
3. Program enters default Containment free air volume which user may change based on conditions at time of sample. Unless there has been significant flooding of drywell this value will not change.

4. Program enters default Suppression Chamber free air volume which user may change based on conditions at time of sample. If there has been a significant increase or decrease in the water level in the Suppression Pool or Torus then the free air volume will change.

NOTE: Pressing the “Reset” button will reset all volumes to default values.

5. **PRESS** the “Back” button to return to the Liquid or Gaseous screen, which user used to call volume form.

5.7.9. **PRESS** the “Graph” button to display the following screen:



µCi/ml: 2.01E+00

Reactor Coolant

☑ Yes ☐ No ☐ N/A

	Melt	Clad
High:	0	9
Best:	0	0
Low:	0	0

See 5.7.9.1

See 5.7.9.2

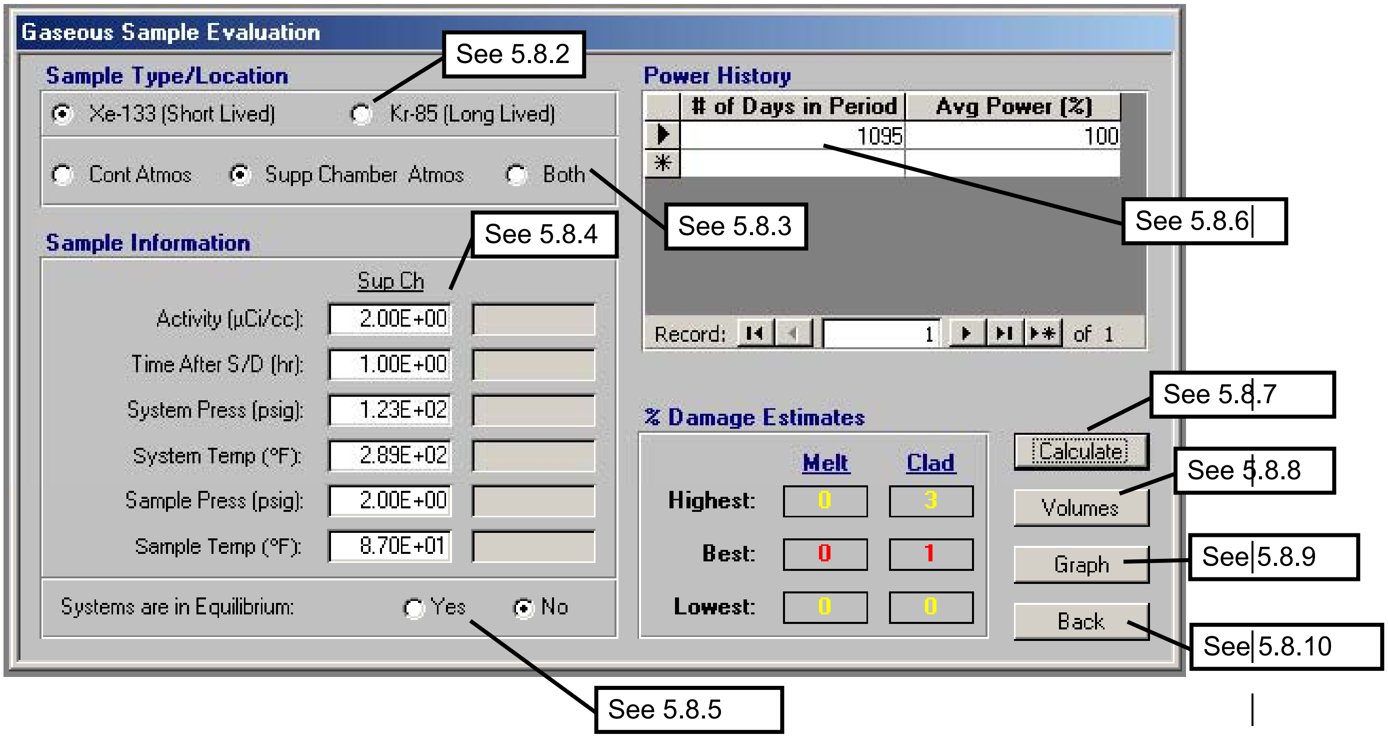
NOTE: Graph shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating entered corrected sample concentration.**PRESS** the "Print" button to print a graph and summary of inputs.

2. **PRESS** the "Back" button to go back to liquid or gaseous form which called this form.

5.7.10. **PRESS** the "Back" button to return to the Summary Screen.

5.8. Gaseous Samples

5.8.1. **PRESS** the "Gas Samples" button on the Summary Screen to open the following form:



Gaseous Sample Evaluation

Sample Type/Location

● Xe-133 (Short Lived) ○ Kr-85 (Long Lived)

○ Cont Atmos ● Supp Chamber Atmos ○ Both

Sample Information

	Sup Ch	
Activity (µCi/cc):	2.00E+00	
Time After S/D (hr):	1.00E+00	
System Press (psig):	1.23E+02	
System Temp (°F):	2.89E+02	
Sample Press (psig):	2.00E+00	
Sample Temp (°F):	8.70E+01	

Systems are in Equilibrium: ○ Yes ● No

Power History

	# of Days in Period	Avg Power (%)
▶	1095	100
*		

Record: 1 of 1

% Damage Estimates

	Melt	Clad
Highest:	0	3
Best:	0	1
Lowest:	0	0

See 5.8.2
See 5.8.3
See 5.8.4
See 5.8.5
See 5.8.6
See 5.8.7
See 5.8.8
See 5.8.9
See 5.8.10

5.8.2. **SELECT** appropriate isotope.

5.8.3. **SELECT** and sample location.

1. **If** samples are available form both locations, **then SELECT** "Both" option.

5.8.4. **ENTER** Sample Information:

1. **ENTER** sample activity for selected isotope in uCi/cc (uCi/ml). Sample results are to be decay corrected back to time after shutdown that the sample was drawn
2. **ENTER** Time After S/D that sample was taken.
3. **ENTER** the pressure and temperature of the system sampled
4. **ENTER** the end pressure and temperature of sample.

5.8.5. **SELECT** the appropriate System Equilibrium status:

NOTE: Atmospheres reach equilibrium through mechanical mixing i.e. fans circulating air between locations for several hours. Equilibrium conditions may be indentified when temperature indications from different locations are approaching similar readings.

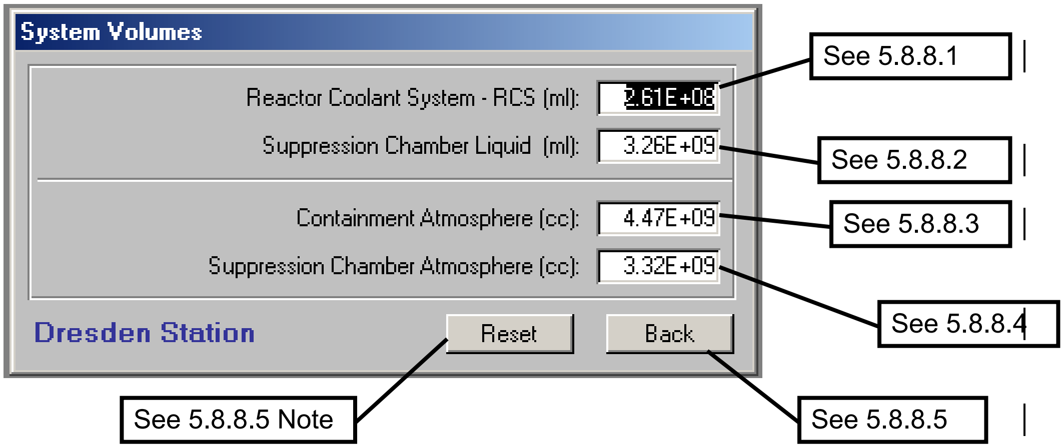
1. **If** sample was taken from only one location and atmospheres are in equilibrium, **then** check “yes” for “Systems in Equilibrium,” **otherwise** check “no.”

5.8.6. **ENTER** power history (past to present, i.e. oldest steady state history as record number 1) of core since last refueling. Shutdown times are entered as the number of days with Avg Power (%) set at 0.

1. For short-lived isotopes, **EXTEND** Power History at least 30 days.
2. For long-lived isotopes, **EXTEND** power history at least 100 days, <u>however the power history for the extent of the cycle is preferred.</u>
3. **LIMIT** variations in steady state power to ± 20% within each operational period entered.

5.8.7. Once all data has been entered **PRESS** the “Calculate” button to display the % Damage Estimates.

5.8.8. **PRESS** the “Volumes” button to display the following screen (same as 5.7.8):



Reactor Coolant System - RCS (ml): 2.61E+08
Suppression Chamber Liquid (ml): 3.26E+09
Containment Atmosphere (cc): 4.47E+09
Suppression Chamber Atmosphere (cc): 3.32E+09

See 5.8.8.1
See 5.8.8.2
See 5.8.8.3
See 5.8.8.4
See 5.8.8.5 Note
See 5.8.8.5

1. Program enters default RCS volume, which the user may change based on RPV Level Readings at time of sample.

2. Program enters default Suppression Chamber volume, which the user may change based on readings at time of sample.

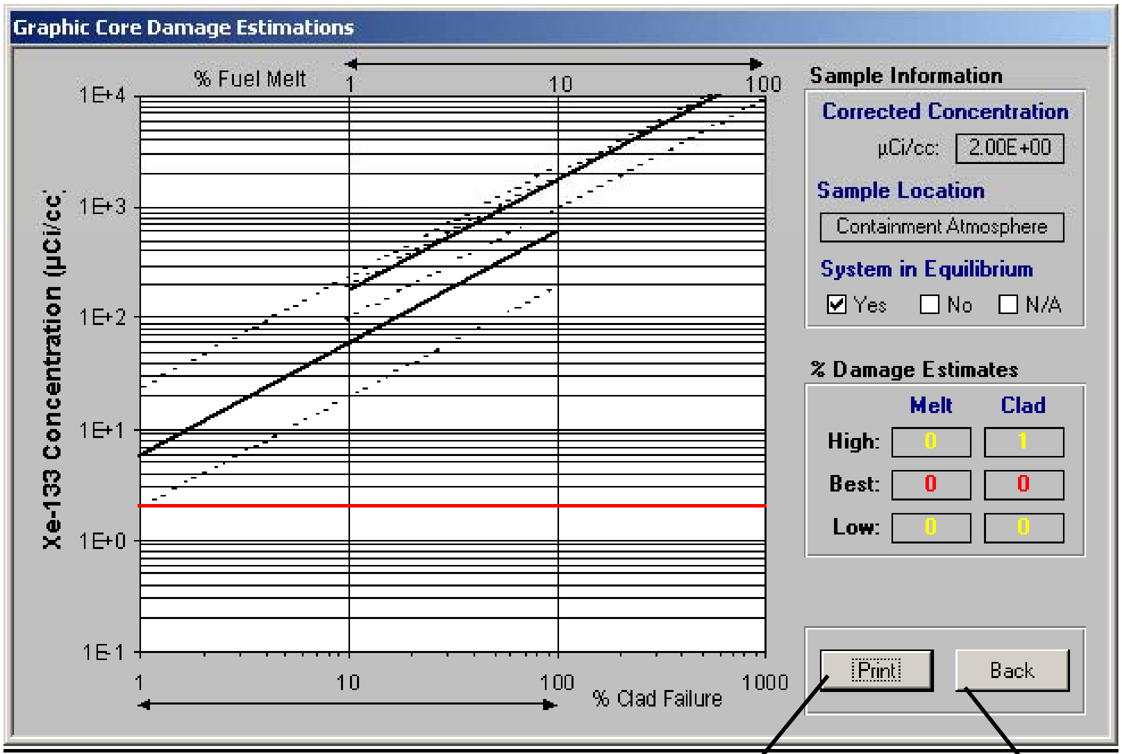
3. Program enters default Containment free air volume which user may change based on conditions at time of sample. Unless there has been significant flooding of drywell this value will not change.

4. Program enters default Suppression Chamber free air volume which user may change based on conditions at time of sample. If there has been a significant increase or decrease in the water level in the Suppression Pool or Torus then the free air volume will change.

NOTE: Pressing the “Reset” button will reset all volumes to default values.

5. **PRESS** the “Back” button to return to the Liquid or Gaseous screen, which user used to call volume form.

5.8.9. **PRESS** the “Graph” button to display the following screen:



Sample Information

Corrected Concentration

µCi/cc: 2.00E+00

Sample Location

Containment Atmosphere

System in Equilibrium

☑ Yes ☐ No ☐ N/A

% Damage Estimates

	Melt	Clad
High:	0	1
Best:	0	0
Low:	0	0

See 5.8.9.1

See 5.8.9.2

NOTE: Graph shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating corrected concentration.

1. **PRESS** the “Print” button to print a graph and summary of inputs.

2. **PRESS** the “Back” button to go back to liquid or gaseous form which called this form.

5.8.10. **PRESS** the “Back” button to return to the Summary Screen.

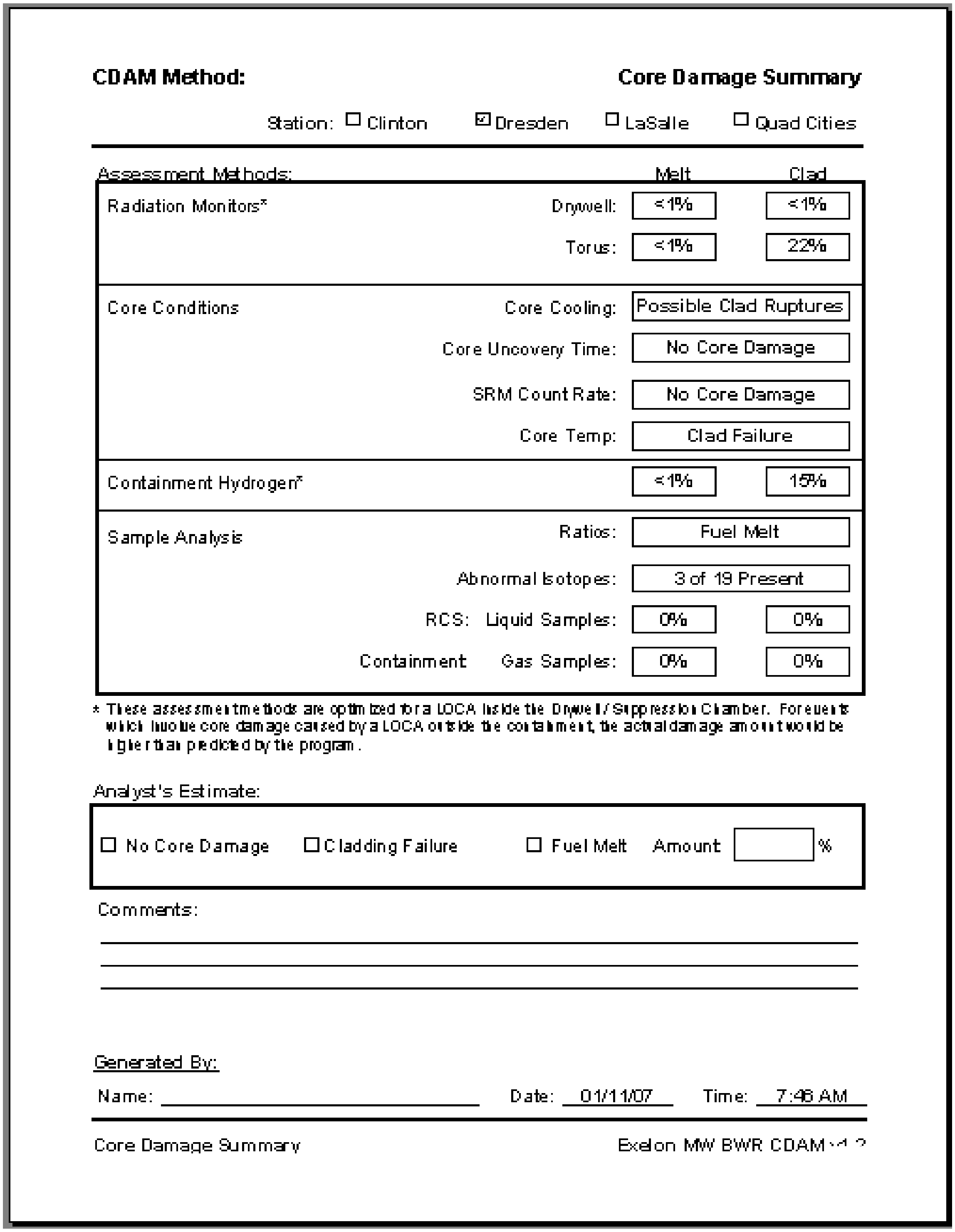
6. CORE DAMAGE SUMMARY REPORT

6.1. Once the program user enters data for all available assessment methods and the program calculates damage based on inputs, **SELECT** the "Print" button to print a summary of all methods used.

6.2. The values presented in the Assessment Methods section of the summary report show that they are in percent (%). Containment Hydrogen values are also in percent (but do not show the % symbol).

Attachment 1
BWR CDAM User Guide
Page 23 of 24

(Sample report on next page.)



CDAM Method: **Core Damage Summary**

Station: ☐ Clinton ☒ Dresden ☐ LaSalle ☐ Quad Cities

Assessment Methods:

		Melt	Clad
Radiation Monitors*	Drywell:	<1%	<1%
	Torus:	<1%	22%
Core Conditions	Core Cooling:	Possible Clad Ruptures	
	Core Uncovery Time:	No Core Damage	
	SRM Count Rate:	No Core Damage	
	Core Temp:	Clad Failure	
Containment Hydrogen*		<1%	15%
Sample Analysis	Ratios:	Fuel Melt	
	Abnormal Isotopes:	3 of 19 Present	
	RCS: Liquid Samples:	0%	0%
	Containment Gas Samples:	0%	0%

* These assessment methods are optimized for a LOCA inside the Drywell / Suppression Chamber. For events which involve core damage caused by a LOCA outside the containment, the actual damage amount would be higher than predicted by the program.

Analyst's Estimate:

☐ No Core Damage ☐ Cladding Failure ☐ Fuel Melt Amount ____ %

Comments:

Generated By:

Name: ______________________ Date: 01/11/07 Time: 7:46 AM

Core Damage Summary Exelon MW BWR CDAM v1.2

6.3. The Individual tasked with assessing core damage shall then **ANALYZE** the report to determine best estimate of type and amount of damage.

7. QUITING, OR EXITING, THE PROGRAM

NOTE: When the program is closed all data is reset.

CAUTION

Program saves no information to disk; printed reports serve as record of core damage assessments.

7.1. **PRESS** the "Quit" button on the Summary Screen exits the program.

Attachment 8

EP-AA-110-302

"Core Damage Assessment (PWR) "

Revision 2

CORE DAMAGE ASSESSMENT (PWR)

1. PURPOSE

1.1 This Core Damage Assessment process is designed to assist in estimating core damage after an accident with potential clad or core damage conditions. This is done to assist in:

1.1.1 Determining if the fuel barriers are breached to evaluate the appropriate Emergency Action Level (EAL) classification.

1.1.2 Providing input on core configuration (coolable or uncoolable) for prioritization of mitigating activities.

1.1.3 Determining the potential quantity and isotopic mix of a radiological release to project offsite doses.

1.1.4 Predicting the radiation protection actions that should be considered for long term recovery activities.

1.1.5 Satisfying inquiries from local and federal government agencies and provide evidence that the utility knows the plant conditions.

2. TERMS AND DEFINITIONS

2.1 Core Damage - a term used to qualify and quantify the core state and amount of damage

2.2 Cladding Failure:

1. Also referred to as “Cladding Oxidation”, “Gap Release” or “Clad Rupture” in other documents.
2. 100% clad failure refers to the rupture of 100% of the fuel rods in the core. This would result in all fission products contained in the gap space being released to the reactor coolant system.

2.3 Fuel Melt:

1. Referred to as “Core Melt” “In-Vessel Melt” or “Over-temperature” damage in reference documents.
2. 100% fuel melt refers to high temperatures in the fuel pellets in 100% of the fuel rods in the core. This would result in all the fission products

contained in the fuel pellet matrix being released to the reactor coolant system.

2.4 Vessel Melt-Through:

1. Referred to as “Ex-Vessel Melt” or “Melt Release” in reference documents.
2. Core debris is relocated to the containment building where the reactor pressure vessel has failed.

3. RESPONSIBILITIES

3.1 The *TSC Technical Manager* shall coordinate core damage assessment activities.

3.2 The *TSC Core/Thermal Hydraulic Engineer* shall serve as the Core Damage Assessment Methodology (CDAM) Evaluator.

3.3 The *TSC Radiation Controls Engineer* shall coordinate radiological and chemistry information with the Core/Thermal Hydraulic Engineer in support of core damage assessment.

4. MAIN BODY

4.1 Select the appropriate attachment for the station experiencing the potential clad or core damage condition and implement the prescribed steps.

- **REFER** to Attachment 1 for Braidwood or Byron Station CDAM User’s Guide

OR

- **REFER** to Attachment 2 for TMI CDAM User’s Guide

5. DOCUMENTATION

5.1 A Summary Form is generated by the PWR CDAM Software for use in documenting the results of the assessment.

5.2 Refer to Attachment 1 or 2, Section 6

6. REFERENCES

6.1 Westinghouse Owner's Group Post Accident Core Damage Methodology, Revision 2, November, 1984.

6.2 Westinghouse Owner's Group Core Damage Assessment Guidance (WCAP-14696-A, Rev. 1).

6.3 Braidwood Commitment - #20-84-074

7. ATTACHMENTS

7.1 Attachment 1, Braidwood or Byron CDAM Users Guide

7.2 Attachment 2, TMI CDAM Users Guide

ATTACHMENT 1

BRAIDWOOD OR BYRON CDAM USER'S GUIDE

1. OVERVIEW

1.1 As a Windows based application designed in Access, PWR CDAM, uses many standard user interfaces. Instructions are not provided in basic computer operations in the Windows® environment. The user must be familiar with these to efficiently operate the program.

1.2 It is also assumed user is familiar with basic reactor physics and core damage fundamentals. Emergency Response Organization training will provide an overview of core damage assessment methodologies.

1.3 The program should be used by qualified personnel as a tool to estimate the type and amount of core damage.

2. DETERMINE APPROPRIATE AND AVAILABLE ASSESSMENT METHODS

2.1.1 The magnitude and type of event, transport mechanism and time after shutdown will be influencing factors on the method(s) utilized to determine the extent of core damage. Damage estimates can be developed using one or more methods as they become available or applicable.

1. Indications of Core Damage
    - A. The primary indicators of core damage that are available during the early phases of an event:
        - 1. Containment Radiation Monitor Readings
        - 2. Core Exit Thermocouple Readings
2. Auxiliary indicators that are used to confirm and better define the possible type of damage are:
    - A. Estimation of maximum temperature reached within the core
    - B. Reactor Coolant System Hot Leg Temperature
    - C. Estimated core uncovery time
    - D. Reactor Vessel Level Indication System readings
    - E. Abnormal Source Range Monitor readings

ATTACHMENT 1

BRAIDWOOD OR BYRON CDAM USER'S GUIDE

F. Containment Hydrogen Readings

3. Long Term Indicators (once liquid or gaseous samples can be safely obtained) are:

A. Isotopic Ratios

B. Presence of high levels of rare isotopes

C. Quantity of isotopes present in samples

2.1.2 Choose the assessment method(s) most appropriate for the existing conditions. Methods available for assisting in the determination of the extent of core damage include the following:

Method	Use	Comment
Core Exit Thermocouples	Indication of onset of Core Damage	Limited due to range of instruments. Not reliable during later phases of core overheating due to changes in core geometry.
RVLIS	Indication of Core Uncovery	Indicates possible damage not useful in estimating the quantity of damage.
Source Range Monitor	Indication of Core Uncovery	Loss of water level leads to increase in neutron detection (fast flux leakage).
Hot Leg RTDs	Indication of Core Uncovery	Only measures bulk flow through core. Hot spots in core may not be detected by exit thermocouples.
Containment Radiation Monitor	Early Indication of Core Damage	Uncertainties due to variables in release of fission products from RCS and effects of containment sprays.
Containment Hydrogen Monitor	Early Indication of Core Damage	Significant uncertainties due to variable Hydrogen generation in core and in release of Hydrogen from RCS and effects of containment sprays.
RCS Samples and Containment Sump and Atmosphere Samples	Late Indication of Core Damage —Sump Samples provide indication of Rx Vessel Failure	Very large uncertainties until all systems have reached equilibrium. Useful in planning long term recovery.

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BRAIDWOOD OR BYRON CDAM USER'S GUIDE

3. START UP

3.1.1 The application is accessed by one of the following:

1. Open the PWR CDAM desktop icon on applicable dose assessment computers.
    - Start the PWR CDAM program for the plant that has declared an emergency.
    - Programs are labeled PWR CDAM.
2. Select RUN from the 'Start Bar' and type in the file path and name as follows:
    - C:\CDAM\PWR CDAM.MDB

3.1.2 **IF** the assigned Core Damage Assessment Computer cannot access the application or the CDAM program will not run, **THEN** Install PWR CDAM on any computer from CDs or Disks located in the TSC or the EOF Library. CDAM is installed by copying appropriate file to computer's hard drive.

4. SELECTION AND PERFORMANCE OF ASSESSMENT

4.1 Choose the assessment method(s) most appropriate for the existing conditions. Methods available for assisting in the determination of the extent of core damage include the following:

- Containment Radiation Analysis - (Section 5.2)
- Core Temperature Analyses - (Section 5.3)
- Core Water Level Analyses - (Section 5.4)
- Containment Hydrogen Analysis - (Section 5.5)
- Nuclide Analyses (Ratios and Abnormal Isotopes) - (Section 5.6)
- Liquid Samples Analysis - (Section 5.7)
- Gaseous Samples Analysis - (Section 5.8)

ATTACHMENT 1

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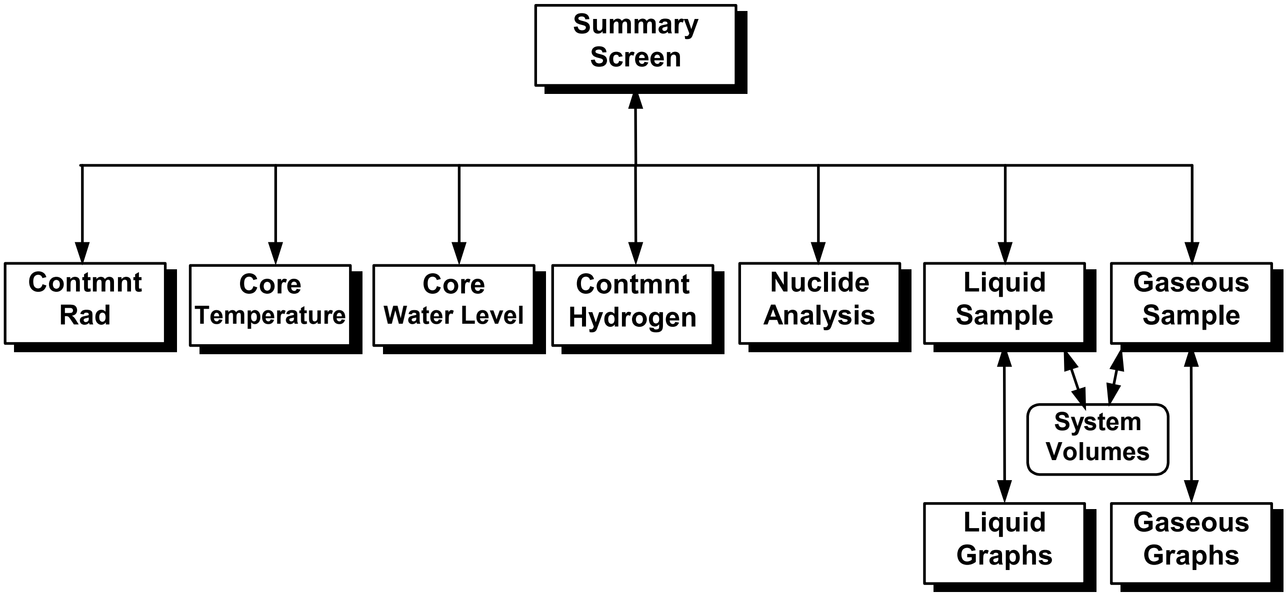
4.2 Basic Program Flow Diagram



5. PROGRAM SCREENS AND INPUTS

5.1 Main Screen – Summary Page

5.1.1 When program is started the following screen appears: (boxes are empty when program is originally launched.)

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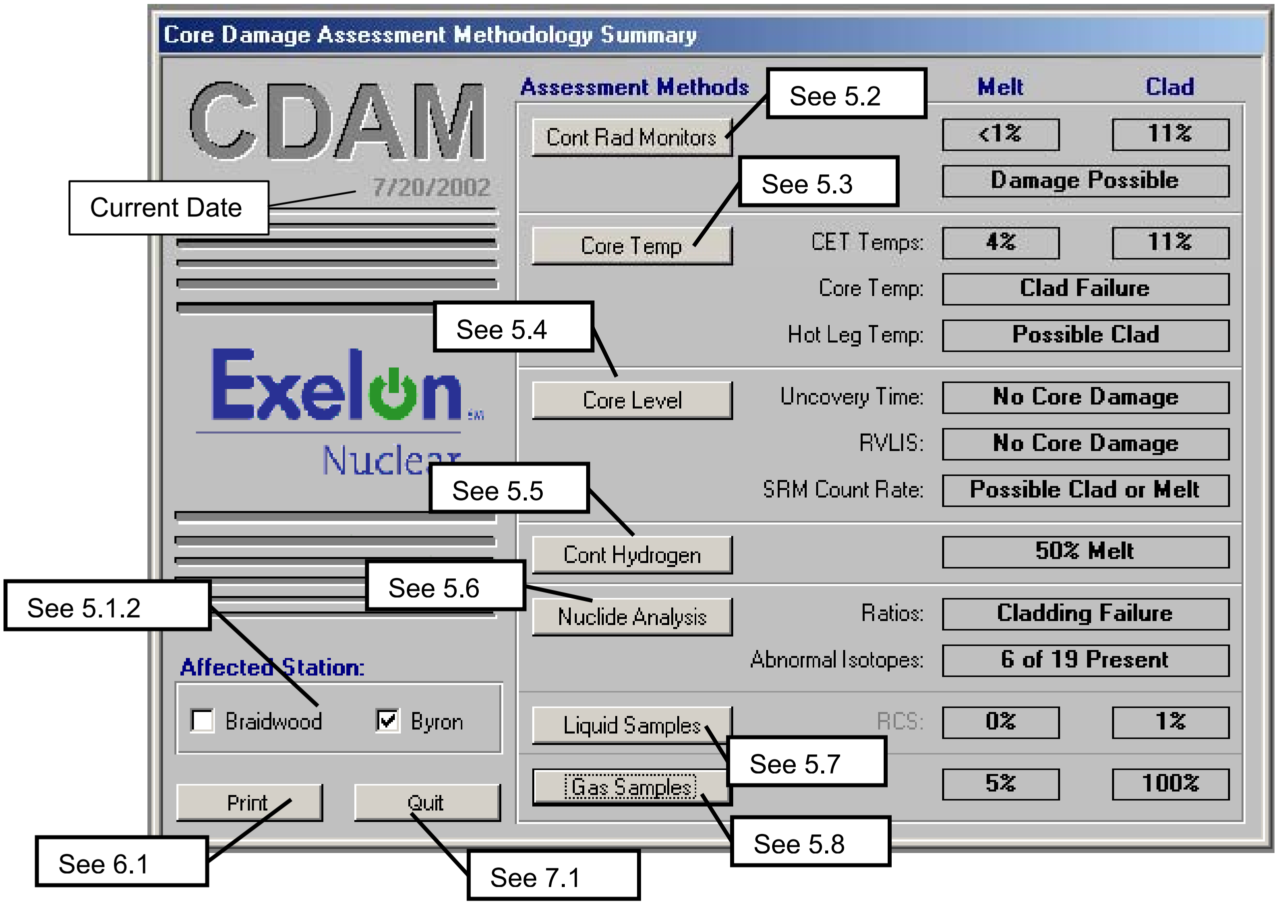
BRAIDWOOD OR BYRON CDAM USER'S GUIDE



Core Damage Assessment Methodology Summary

CDAM 7/20/2002

Assessment Methods		Melt	Clad
Cont Rad Monitors		<1%	11%
		Damage Possible	
Core Temp	CET Temps:	4%	11%
	Core Temp:	Clad Failure	
	Hot Leg Temp:	Possible Clad	
Core Level	Uncovery Time:	No Core Damage	
	RVLIS:	No Core Damage	
	SRM Count Rate:	Possible Clad or Melt	
Cont Hydrogen		50% Melt	
Nuclide Analysis	Ratios:	Cladding Failure	
	Abnormal Isotopes:	6 of 19 Present	
Liquid Samples	RCS:	0%	1%
Gas Samples		5%	100%

Affected Station:
☐ Braidwood ☑ Byron

CAUTION

Selecting an "Affected Station" resets all inputs to default values.

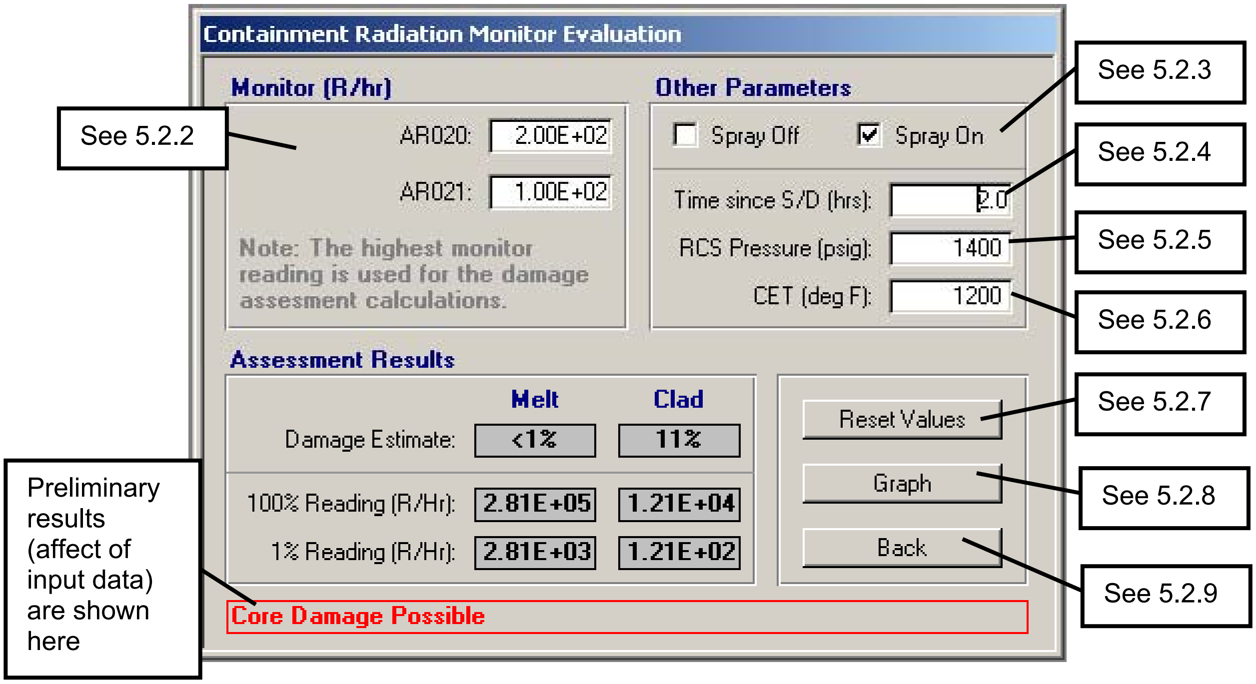
5.1.2 **SELECT** the Affected Station before other "Assessment Methods".

CAUTION

Pressing the **"Quit"** button exits the program. When the program is closed all data is reset. Program saves no information to disk; printed reports serve as record of core damage assessments.

5.2 Containment Radiation Monitor Method

5.2.1 Pressing **"Cont Rad Monitors"** button opens the following form:



Containment Radiation Monitor Evaluation

Monitor (R/hr)

AR020: 2.00E+02

AR021: 1.00E+02

Note: The highest monitor reading is used for the damage assesment calculations.

Other Parameters

☐ Spray Off ☑ Spray On

Time since S/D (hrs): 2.0

RCS Pressure (psig): 1400

CET (deg F): 1200

Assessment Results

	Melt	Clad
Damage Estimate:	<1%	11%
100% Reading (R/Hr):	2.81E+05	1.21E+04
1% Reading (R/Hr):	2.81E+03	1.21E+02

Core Damage Possible

See 5.2.2

See 5.2.3

See 5.2.4

See 5.2.5

See 5.2.6

See 5.2.7

See 5.2.8

See 5.2.9

Preliminary results (affect of input data) are shown here

5.2.2 Highest containment radiation monitor reading which occurred is entered in these boxes. Program only lists containment high range monitors, however a reading may be entered from any monitor which accurately showed containment radiation levels. If two entries are made only the highest is used.

5.2.3 Containment Spray

1. Choose "Spray On" when sprays are operated for the majority of the time after the following events:

   The expected start of Core Damage

   AND

   A release of the core isotopic inventory into the Containment atmosphere has occurred.

2. **If** the Containment Spray system was **not** operated for the majority of the time after the start of Core Damage and the release of the isotopic inventory into containment **then** choose "Spray Off."

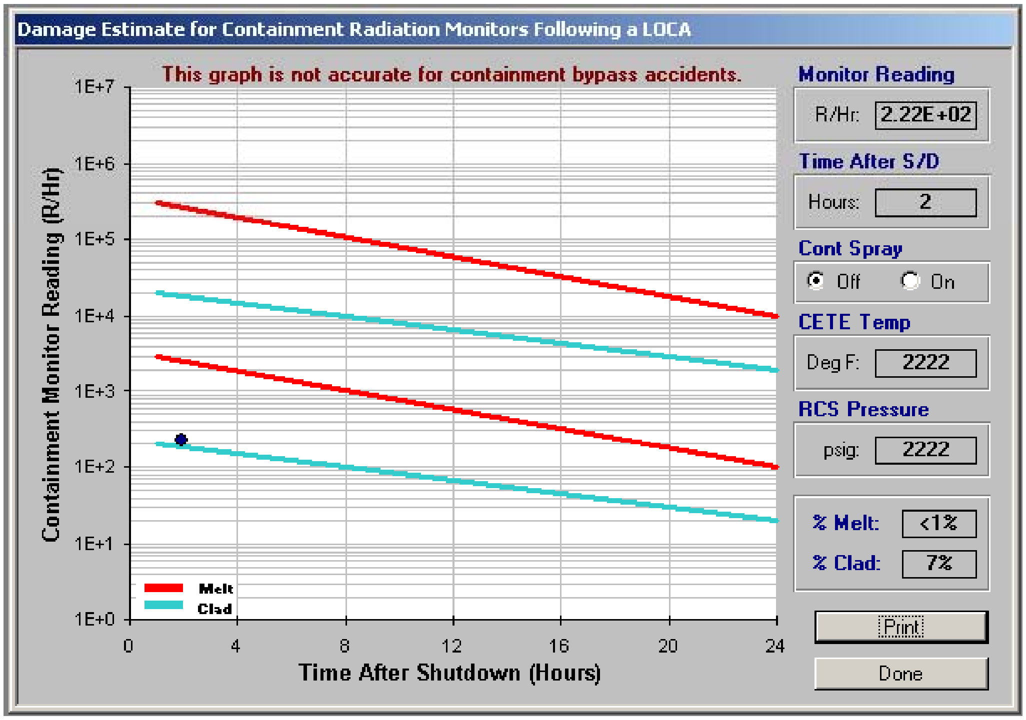
5.2.4 Enter the time after reactor shutdown, which corresponds the time the containment radiation reading was taken. Value must be between 1 hour and 24 hours after shutdown, which corresponds to the time period in which this method is considered effective.

5.2.5 Enter the estimated Reactor Coolant System pressure at the time when core damage occurred (usually same time as high CET temperatures were observed).

5.2.6 Enter the highest Core Exit Thermocouple reading observed during the event.

5.2.7 Pressing **"Reset"** button resets values on this form only.

5.2.8 Pressing **"Graph"** button displays the follow screen.



Damage Estimate for Containment Radiation Monitors Following a LOCA

This graph is not accurate for containment bypass accidents.

Monitor Reading
R/Hr: 2.22E+02

Time After S/D
Hours: 2

Cont Spray
● Off ○ On

CETE Temp
Deg F: 2222

RCS Pressure
psig: 2222

% Melt: <1%
% Clad: 7%

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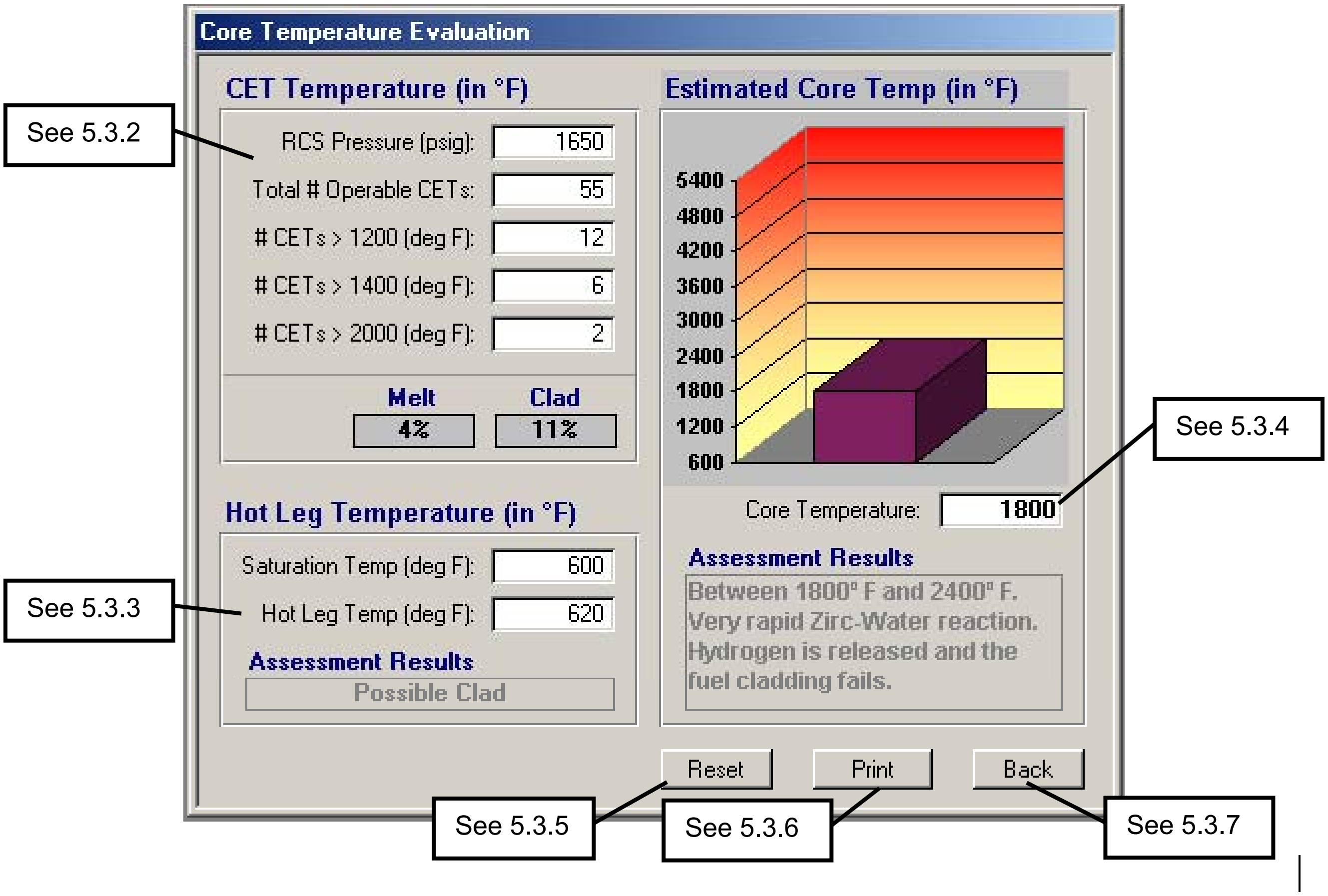
1. Graph shows high and low containment radiation levels which correspond to 100% Melt or Clad or 1% Melt or Clad damage. A dot shows the last containment radiation level entered into the program for assessment.

2. Pressing "Print Button" will print report of containment radiation method inputs and best estimate of damage.

5.2.9 Pressing **"Back"** button takes the user back to the summary screen.

5.3 Core Temperature Methods

5.3.1 Pressing **"Core Temp"** button opens the following form:



Core Temperature Evaluation

CET Temperature (in °F)

RCS Pressure (psig): 1650
Total # Operable CETs: 55
CETs > 1200 (deg F): 12
CETs > 1400 (deg F): 6
CETs > 2000 (deg F): 2

Melt	Clad
4%	11%

Hot Leg Temperature (in °F)

Saturation Temp (deg F): 600
Hot Leg Temp (deg F): 620

Assessment Results
Possible Clad

Estimated Core Temp (in °F)

Core Temperature: 1800

Assessment Results
Between 1800° F and 2400° F. Very rapid Zirc-Water reaction. Hydrogen is released and the fuel cladding fails.

5.3.2 Core Exit Thermocouples (CETs)

1. Enter the Reactor Coolant System pressure at the time the CETs readings were taken.

2. Normally there are 65 operating CETs, however user should enter the number that was operating when temperature readings were taken.

3. Enter number of CETs that exceeded the listed temperatures. Program will not allow user to enter a higher number of CETs than the temperature box above it. (i.e. if only 5 CETs exceeded 1200 °F there can not be 6 exceeding 1400 °F).

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5.3.3 Reactor Coolant System Hot Leg temperature.

1. Enter saturation temperature for RCS pressure at time of highest RCS Hot Leg temperature. Value must be looked up in steam tables. Value is limited to 650 °F, which corresponds to max system pressure.
2. Enter highest Hot Leg temperature observed during expected time of core damage.

5.3.4 Based on inputs from Reactor Operators, TSC Staff and other engineering personnel (including outside sources such as Westinghouse personnel) enter the estimated highest temperature reached in the reactor core.

5.3.5 Pressing **"Reset"** button resets values on this form only.

5.3.6 Pressing **"Print"** button prints report of inputs and results of core temperature methods of core damage assessment.

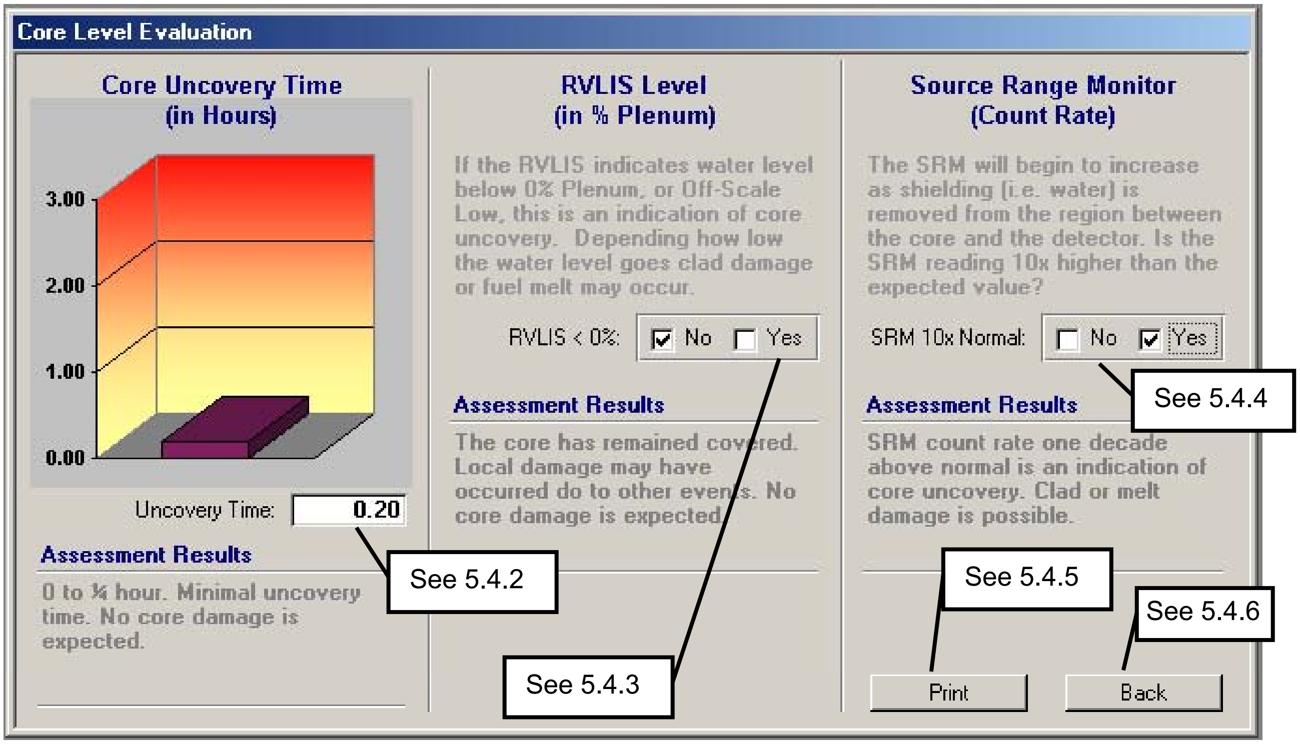
5.3.7 Pressing **"Back"** button takes the user back to the summary screen.

ATTACHMENT 1

BRAIDWOOD OR BYRON CDAM USER'S GUIDE

5.4 Core Level Evaluations

5.4.1 Pressing **"Core Level"** button opens the following form:



Core Uncovery Time (in Hours)

Uncovery Time: 0.20

Assessment Results

0 to ¼ hour. Minimal uncovery time. No core damage is expected.

RVLIS Level (in % Plenum)

If the RVLIS indicates water level below 0% Plenum, or Off-Scale Low, this is an indication of core uncovery. Depending how low the water level goes clad damage or fuel melt may occur.

RVLIS < 0%: ☑ No ☐ Yes

Assessment Results

The core has remained covered. Local damage may have occurred do to other events. No core damage is expected.

Source Range Monitor (Count Rate)

The SRM will begin to increase as shielding (i.e. water) is removed from the region between the core and the detector. Is the SRM reading 10x higher than the expected value?

SRM 10x Normal: ☐ No ☑ Yes

Assessment Results

SRM count rate one decade above normal is an indication of core uncovery. Clad or melt damage is possible.

See 5.4.2

See 5.4.3

See 5.4.4

See 5.4.5

See 5.4.6

5.4.2 Enter estimated time portions of the reactor core was uncovered.

5.4.3 Enter if the Reactor Vessel Level Indication System (RVLIS) was off-scale low or indicated below 0% Plenum.

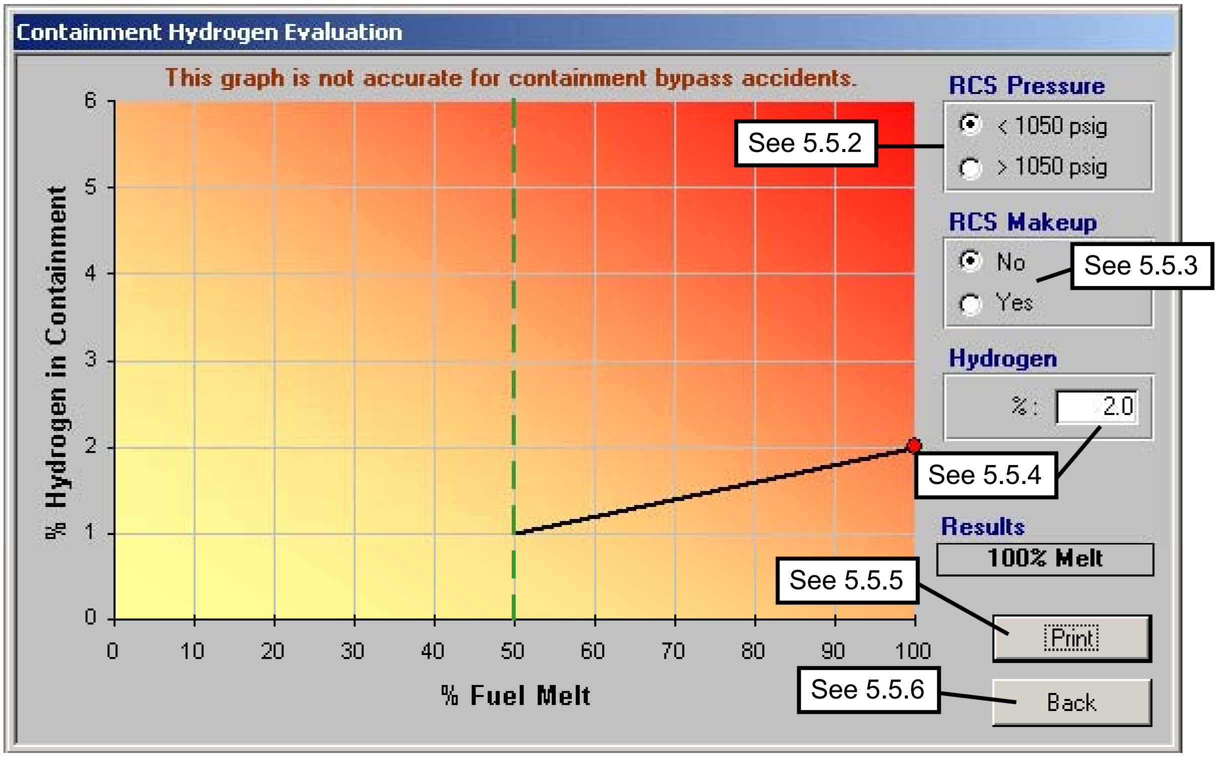
5.4.4 Check if the Source Range Monitoring system indicated abnormally high readings during the event (i.e., 1 decade above normal reading).

5.4.5 Pressing **"Print"** button prints report of inputs and results of core level methods of core damage assessment.

5.4.6 Pressing **"Back"** button takes the user back to the summary screen.

5.5 Containment Hydrogen Evaluations

5.5.1 Pressing **"Cont Hydrogen"** button opens the following form:



Containment Hydrogen Evaluation

This graph is not accurate for containment bypass accidents.

RCS Pressure
● < 1050 psig
○ > 1050 psig

RCS Makeup
● No
○ Yes

Hydrogen
%: 2.0

Results
100% Melt

5.5.2 Choose the estimated Reactor Coolant System (RCS) pressure at the time core damage was occurring.

5.5.3 RCS Makeup:

1. Choose **"No"** if no or little water was added to the RCS system during the time period core damage was occurring.
2. Choose **"Yes"** if water was added to the RCS during the time core damage was occurring or prior to the time a Large Leak occurred from the RCS into the containment structure.

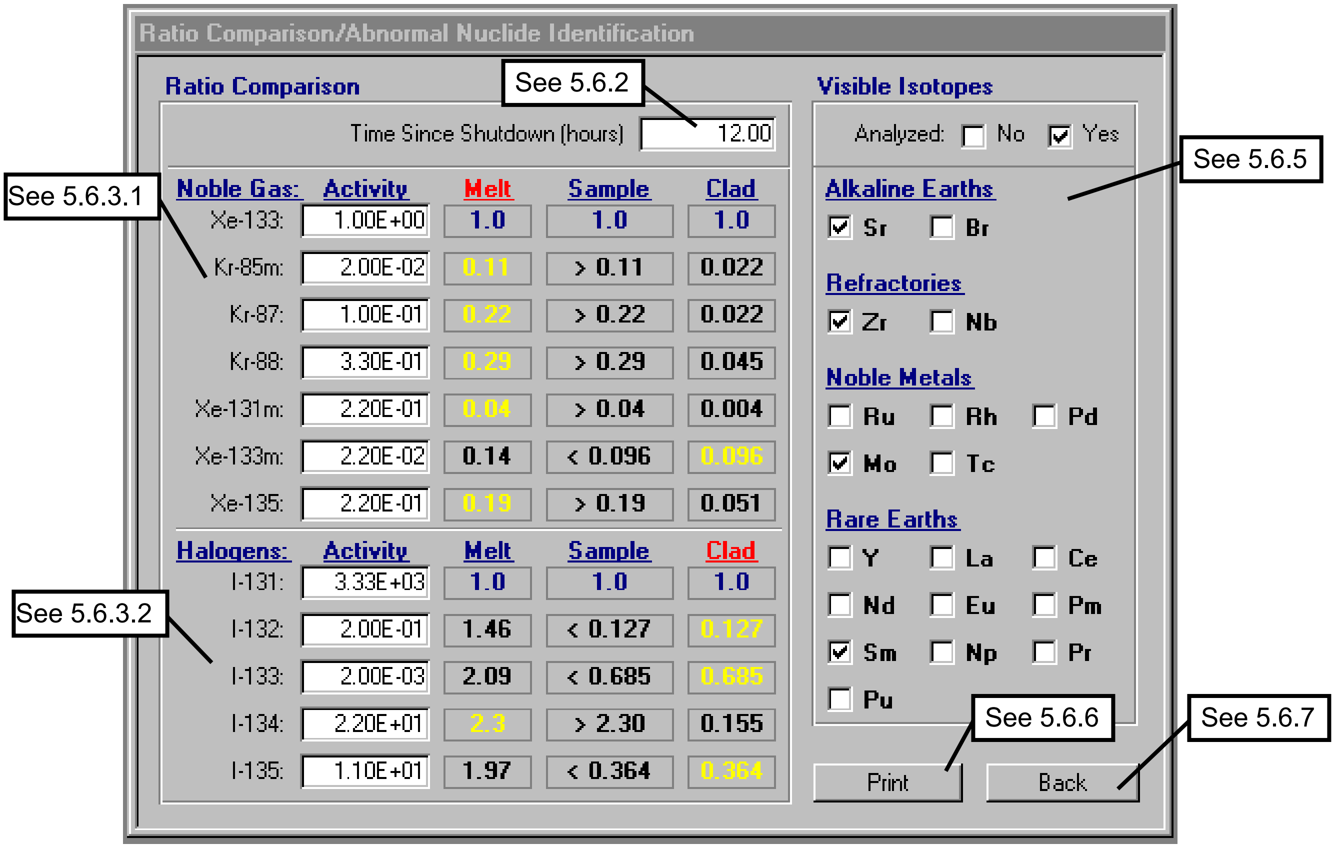
5.5.4 Enter highest containment hydrogen level measured. H_2 monitoring equipment is only accurate within a ± 1 % range so no damage is reported until level reaches at least 1 %. Range of instrument is 0 – 30 %.

5.5.5 Pressing **"Print"** button prints report of inputs and results of Containment Hydrogen methods of core damage assessment.

5.5.6 Pressing **"Back"** button takes the user back to the summary screen.

5.6 Nuclide Analysis

5.6.1 Pressing **"Nuclide Analysis"** button opens the following form:



Ratio Comparison/Abnormal Nuclide Identification

Ratio Comparison

Time Since Shutdown (hours): 12.00

Noble Gas:	Activity	Melt	Sample	Clad
Xe-133:	1.00E+00	1.0	1.0	1.0
Kr-85m:	2.00E-02	0.11	> 0.11	0.022
Kr-87:	1.00E-01	0.22	> 0.22	0.022
Kr-88:	3.30E-01	0.29	> 0.29	0.045
Xe-131m:	2.20E-01	0.04	> 0.04	0.004
Xe-133m:	2.20E-02	0.14	< 0.096	0.096
Xe-135:	2.20E-01	0.19	> 0.19	0.051

Halogens:	Activity	Melt	Sample	Clad
I-131:	3.33E+03	1.0	1.0	1.0
I-132:	2.00E-01	1.46	< 0.127	0.127
I-133:	2.00E-03	2.09	< 0.685	0.685
I-134:	2.20E+01	2.3	> 2.30	0.155
I-135:	1.10E+01	1.97	< 0.364	0.364

Visible Isotopes

Analyzed: ☐ No ☑ Yes

Alkaline Earths
☑ Sr ☐ Br

Refractories
☑ Zr ☐ Nb

Noble Metals
☐ Ru ☐ Rh ☐ Pd
☑ Mo ☐ Tc

Rare Earths
☐ Y ☐ La ☐ Ce
☐ Nd ☐ Eu ☐ Pm
☑ Sm ☐ Np ☐ Pr
☐ Pu

See 5.6.2
See 5.6.3.1
See 5.6.3.2
See 5.6.5
See 5.6.6
See 5.6.7

5.6.2 Enter the time since reactor shutdown when the sample was taken.

5.6.3 **ENTER** isotopic sample results in uCi/cc. Sample results are to be decay corrected back to time after shutdown that the sample was drawn.

1. Noble Gases are ratioed to Xe-133
2. Halogens are ratioed to I-131

5.6.4 If the ratio is greater than predicted melt ratio, melt damage is predicted. If the ratio is less than clad ratio, clad damage is predicted. If the ratio is between the melt and clad ratio, the program determines damage type using following logic:

$$\frac{\textbf{Ratio}_{\textbf{melt}}}{\textbf{Ratio}_{\textbf{sample}}} < \frac{\textbf{Ratio}_{\textbf{sample}}}{\textbf{Ratio}_{\textbf{clad}}}$$

- **IF** the relationship is true, **THEN**: ....Fuel Melt
- **IF** the relationship is not true, **THEN**: ....Clad Failure

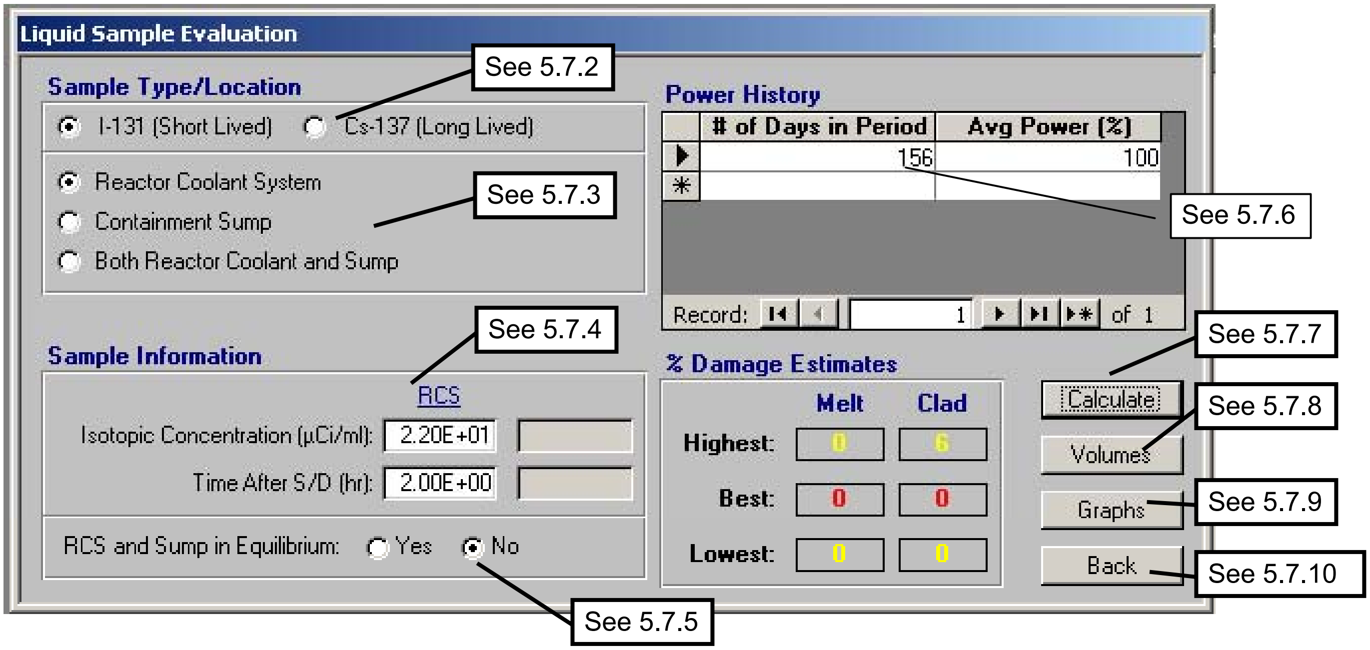
5.6.5 **IF** abnormal levels of rare isotopes are present **THEN** check yes AND check which isotopes are present.

5.6.6 Pressing **"Print"** button prints report of inputs and results of Nuclide Analysis methods of core damage assessment.

5.6.7 Pressing **"Back"** button takes the user back to the summary screen.

5.7 <u>Liquid Samples</u>

5.7.1 Pressing **"Liquid Samples"** button opens the following form:



Sample Type/Location

● I-131 (Short Lived) ○ Cs-137 (Long Lived)

● Reactor Coolant System
○ Containment Sump
○ Both Reactor Coolant and Sump

Sample Information

RCS

Isotopic Concentration (µCi/ml): 2.20E+01

Time After S/D (hr): 2.00E+00

RCS and Sump in Equilibrium: ○ Yes ● No

Power History

# of Days in Period	Avg Power (%)
156	100

% Damage Estimates

	Melt	Clad
Highest:	0	6
Best:	0	0
Lowest:	0	0

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5.7.2 Select appropriate isotope.

NOTE: Before you can choose "Containment Sump" or "Both Reactor Coolant and Sump" a volume entry must be made for sump volume, refer to step 5.7.7.

5.7.3 Select sample location. If samples are available from both locations select both.

5.7.4 Enter sample concentrations(s) in µCi/ml and Time After S/D that samples were taken.

5.7.5 **SELECT** the appropriate System Equilibrium status:

NOTE: Systems reach equilibrium through mechanical mixing i.e. pumps circulating water between systems for several hours. Equilibrium conditions may be identified when temperature indications from different systems are approaching similar readings.

1. **If** sample was taken from only one location and systems are in equilibrium, **then** check "yes" for "RCS and Sump in Equilibrium," **otherwise** check "no."

5.7.6 Enter power history of core since last refueling. Shutdown times are entered as the number of days with Ave Power (%) set at 0.

1. For short-lived isotopes power history should extend at least 30 days.
2. For long-lived isotopes power history should extend at least 100 days, however the power history for the extent of the cycle is preferred.
3. Variations in steady state power should be limited to ± 20% within each operational period entered.

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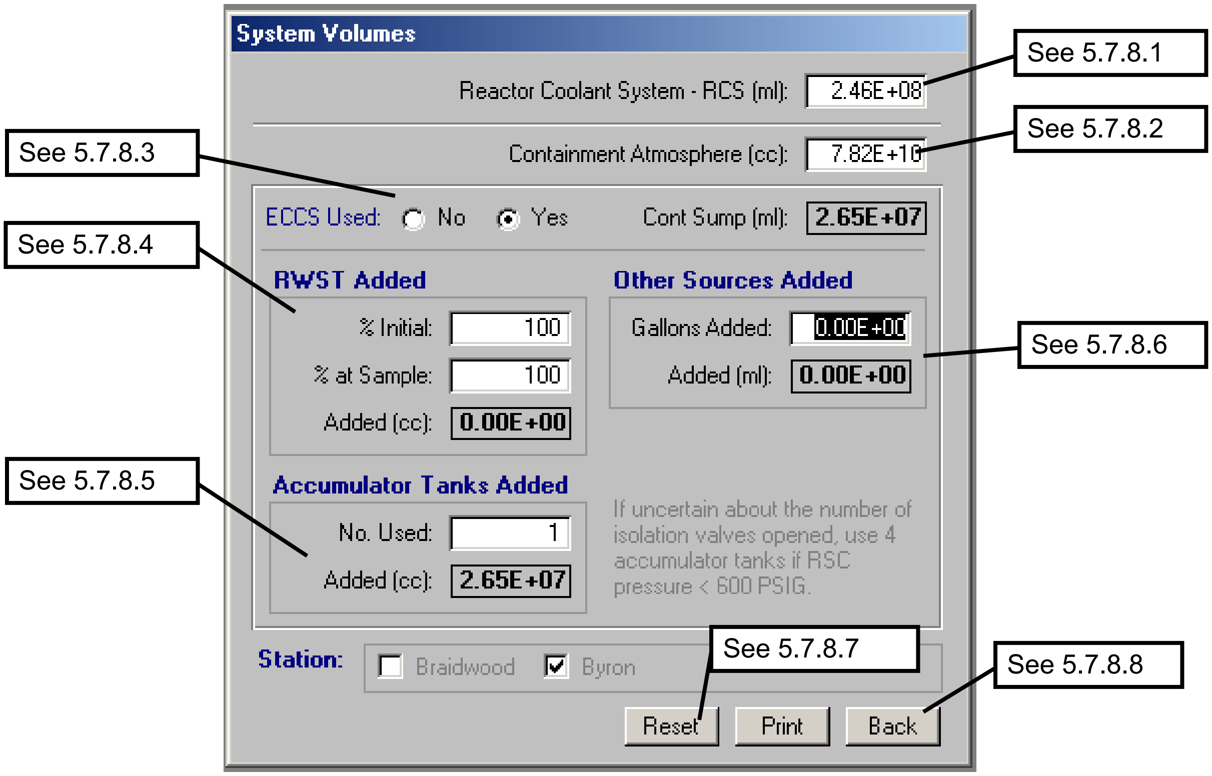
5.7.7 Once all data has been entered pressing the "Calculate" button will display the % Damage Estimates.

5.7.8 Pressing "Volumes" button displays the following screen:



System Volumes

Reactor Coolant System - RCS (ml): 2.46E+08

Containment Atmosphere (cc): 7.82E+10

ECCS Used: ○ No ● Yes

Cont Sump (ml): 2.65E+07

RWST Added

% Initial: 100

% at Sample: 100

Added (cc): 0.00E+00

Other Sources Added

Gallons Added: 0.00E+00

Added (ml): 0.00E+00

Accumulator Tanks Added

No. Used: 1

Added (cc): 2.65E+07

If uncertain about the number of isolation valves opened, use 4 accumulator tanks if RSC pressure < 600 PSIG.

Station: ☐ Braidwood ☑ Byron

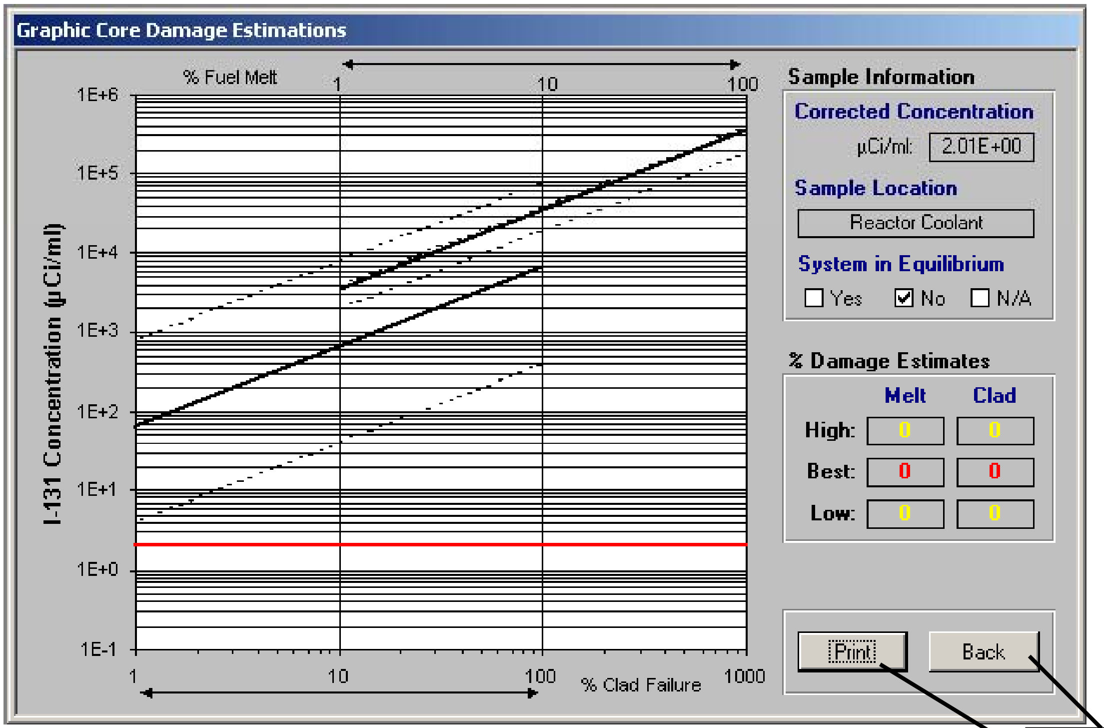
1. Program enters default RCS volume, which the user may change based on RVLIS Readings and Pressurizer level at time of sample.
2. Program enters default Containment free air volume which user may change based on containment sump level at time of sample.
3. Program assumes Containment Sump volume is 0 unless there has been an activation of the Emergency Core Cooling System (ECCS). Checking yes allows user to estimate Containment Sump volume.
4. The change in level of the Refueling Water Storage Tank (RWST) determines amount of water in Containment Sump from this source.
5. The number of Accumulators that have injected into the RCS determines amount of water in Containment Sump from this source.

6. User may enter other sources of water added during an event. (such as fire main, secondary water, potable water, etc.).

7. Pressing "Reset" button resets all volumes to default values.

8. Pressing **"Back"** button takes the user back to the Liquid or Gaseous screen, which user used to call volume form.

5.7.9 Pressing **"Graph"** button displays the following screen:



Sample Information

Corrected Concentration

µCi/ml: 2.01E+00

Sample Location

Reactor Coolant

System in Equilibrium

☐ Yes ☑ No ☐ N/A

% Damage Estimates

	Melt	Clad
High:	0	0
Best:	0	0
Low:	0	0

See 5.7.9.2

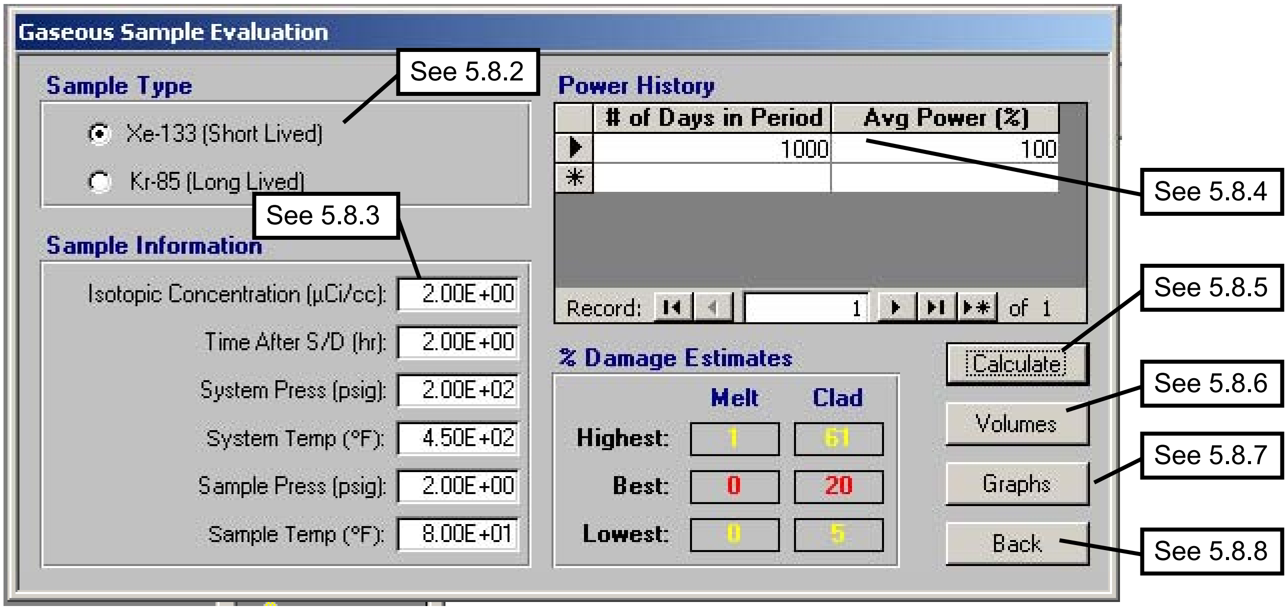
1. Graph shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating corrected sample concentration.

2. User can select "**Print**" button to print graph and summary of inputs or press "Back" button to go back to liquid or gaseous form which called this form.

5.7.10 While in the liquid sample evaluation form, pressing **"Back"** button takes the user back to the summary screen.

5.8 Gaseous Samples

5.8.1 Pressing **"Gas Samples"** button opens the following form:



Gaseous Sample Evaluation

Sample Type

● Xe-133 (Short Lived)

○ Kr-85 (Long Lived)

Sample Information

Isotopic Concentration (µCi/cc): 2.00E+00
Time After S/D (hr): 2.00E+00
System Press (psig): 2.00E+02
System Temp (°F): 4.50E+02
Sample Press (psig): 2.00E+00
Sample Temp (°F): 8.00E+01

Power History

# of Days in Period	Avg Power (%)
1000	100

Record: 1 of 1

% Damage Estimates

	Melt	Clad
Highest:	1	61
Best:	0	20
Lowest:	0	5

5.8.2 **Select** appropriate isotope.

5.8.3 **Enter** Sample Information:

1. Enter sample Isotopic Concentration(s) in µCi/ml for selected isotope.
2. Enter Time After S/D that sample was taken.
3. Enter the pressure and temperature of the system sampled
4. Enter the end pressure and temperature of sample.

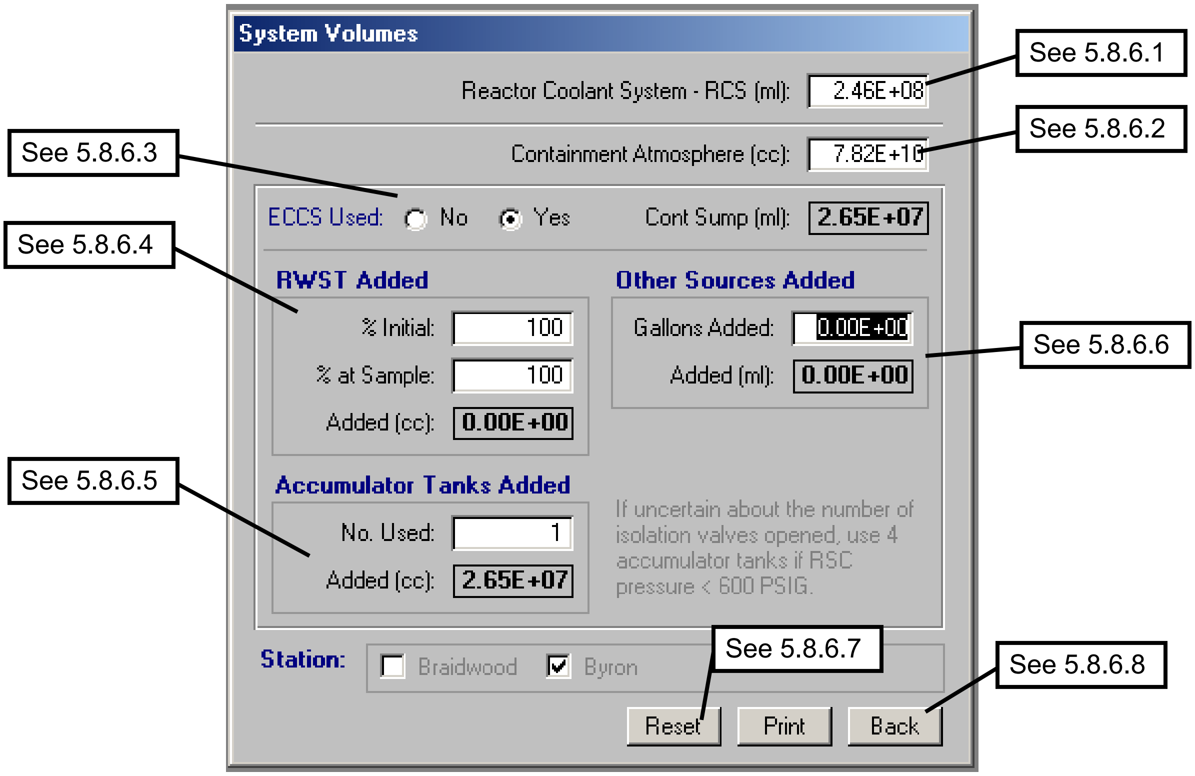
5.8.4 **Enter** power history of core since last refueling. Shutdown times are entered as the number of days with Ave Power (%) set at 0.

1. For short-lived isotopes power history should extend at least 30 days.
2. For long-lived isotopes power history should extend at least 100 days, however the power history for the extent of the cycle is preferred.

3. Variations in steady state power should be limited to ± 20% within each operational period entered.

5.8.5 Once all data has been entered pressing the "Calculate" button will display the % Damage Estimates.

5.8.6 Pressing "Volumes" button displays the following screen (Same as 5.7.7):



Reactor Coolant System - RCS (ml): 2.46E+08

Containment Atmosphere (cc): 7.82E+10

ECCS Used: ○ No ● Yes

Cont Sump (ml): 2.65E+07

RWST Added

% Initial: 100

% at Sample: 100

Added (cc): 0.00E+00

Other Sources Added

Gallons Added: 0.00E+00

Added (ml): 0.00E+00

Accumulator Tanks Added

No. Used: 1

Added (cc): 2.65E+07

If uncertain about the number of isolation valves opened, use 4 accumulator tanks if RSC pressure < 600 PSIG.

Station: ☐ Braidwood ☑ Byron

1. Program enters default RCS volume, which the user may change based on RVLIS Readings and Pressurizer level at time of sample.

2. Program enters default Containment free air volume which user may change based on containment sump level at time of sample.

3. Program assumes Containment Sump volume is 0 unless there has been an activation of the Emergency Core Cooling System (ECCS). Checking yes allows user to estimate Containment Sump volume.

4. The change in level of the Refueling Water Storage Tank (RWST) determines amount of water in Containment Sump from this source.

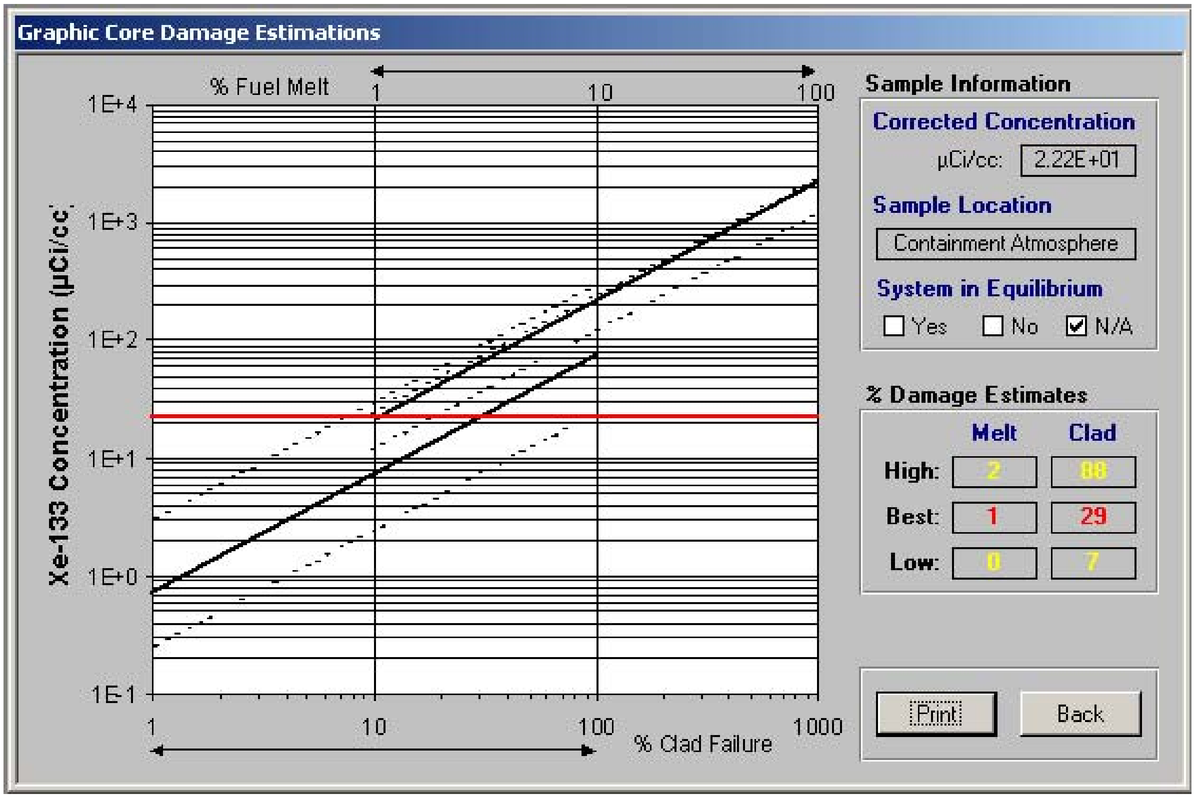
5. The number of Accumulators that have injected into the RCS determines amount of water in Containment Sump from this source.

6. User may enter other sources of water added during an event. (such as fire main, secondary water, potable water, etc.).

7. Pressing "Reset" button resets all volumes to default values.

8. Pressing **"Back"** button takes the user back to the Liquid or Gaseous screen, which user used to call volume form.

5.8.7 Pressing **"Graph"** button displays the following screen:



Sample Information

Corrected Concentration

µCi/cc: 2.22E+01

Sample Location

Containment Atmosphere

System in Equilibrium

☐ Yes ☐ No ☑ N/A

% Damage Estimates

	Melt	Clad
High:	2	88
Best:	1	29
Low:	0	7

1. Graph shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating corrected concentration.

2. User can select "Print" button to print graph and summary of inputs or press "Back" button to go back to liquid or gaseous form which called this form.

5.8.8 Pressing **"Back"** button takes the user back to the summary screen.

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6. CORE DAMAGE SUMMARY REPORT

6.1 Once the program user enters data for all available assessment methods and the program calculates damage based on inputs, **SELECT** the **"Print"** button to print a summary of all methods used.

6.1.1 A sample report is shown on the next page.

6.1.2 Individual tasked with assessing core damage shall then analyze report to determine best estimate of type and amount of damage.

~~**Degree of Degradation**~~	~~**Minor (<10%)**~~	~~**Intermediate (10% to 50%)**~~	~~**Major (>50)**~~
~~**No Core Damage**~~	~~1~~	~~1~~	~~1~~
~~**Cladding Failure**~~	~~2~~	~~3~~	~~4~~
~~**Fuel Overheat**~~	~~5~~	~~6~~	~~7~~
~~**Fuel Melt**~~	~~8~~	~~9~~	~~10~~

7. QUITING, OR EXITING, THE PROGRAM

7.1 Pressing the "Quit" button on the Summary Screen exits the program.

7.1.1 When the program is closed all data is reset.

7.1.2 Program saves no information to disk; printed reports serve as record of core damage assessments.

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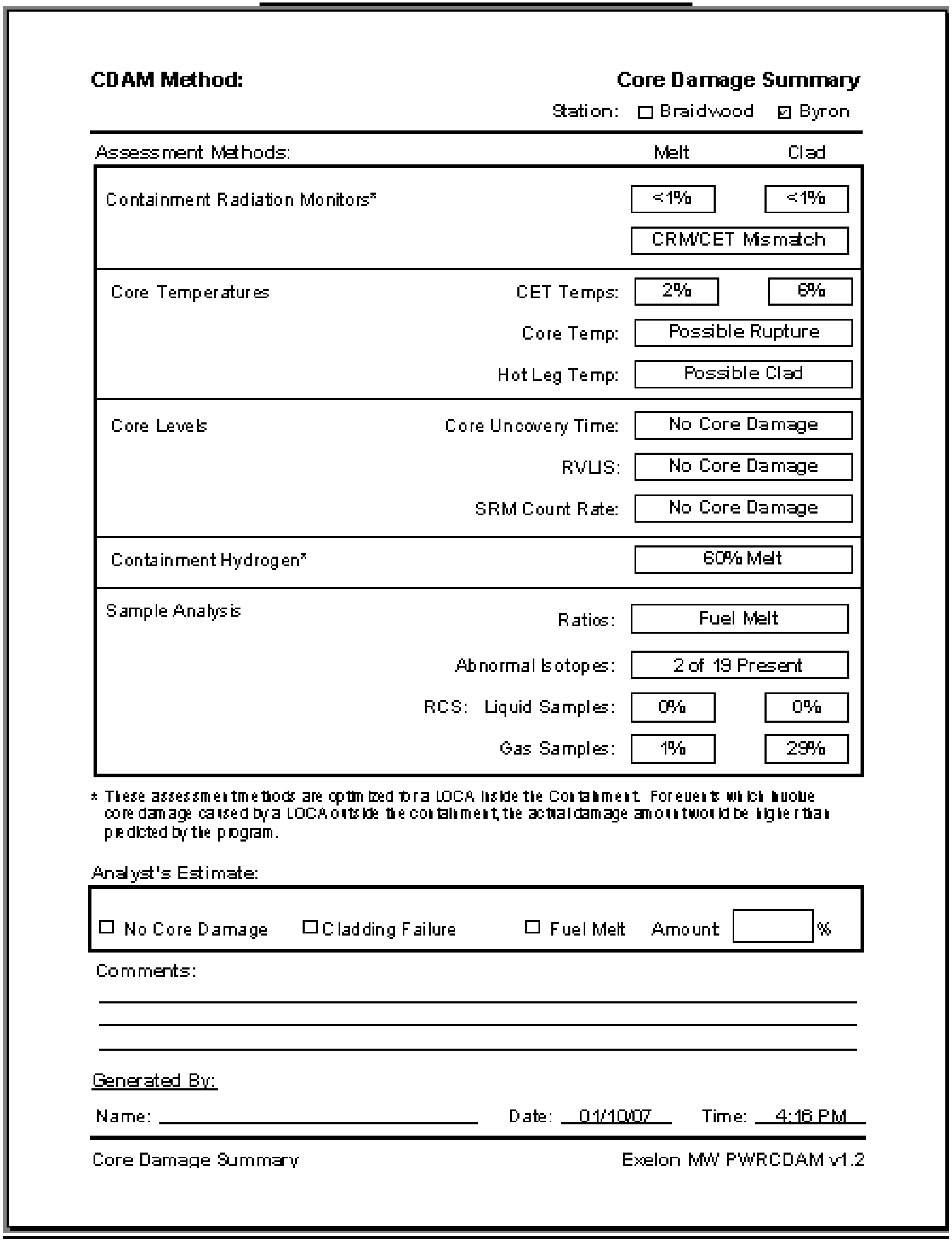
SAMPLE SUMMARY REPORT



CDAM Method: **Core Damage Summary**

Station: ☐ Braidwood ☒ Byron

Assessment Methods:		Melt	Clad
Containment Radiation Monitors*		<1%	<1%
		CRM/CET Mismatch	
Core Temperatures	CET Temps:	2%	6%
	Core Temp:	Possible Rupture	
	Hot Leg Temp:	Possible Clad	
Core Levels	Core Uncovery Time:	No Core Damage	
	RVLIS:	No Core Damage	
	SRM Count Rate:	No Core Damage	
Containment Hydrogen*		60% Melt	
Sample Analysis	Ratios:	Fuel Melt	
	Abnormal Isotopes:	2 of 19 Present	
	RCS: Liquid Samples:	0%	0%
	Gas Samples:	1%	29%

* These assessment methods are optimized for a LOCA inside the Containment. For events which involve core damage caused by a LOCA outside the containment, the actual damage amount would be higher than predicted by the program.

Analyst's Estimate:

☐ No Core Damage ☐ Cladding Failure ☐ Fuel Melt Amount ______ %

Comments:

Generated By:

Name: ____________________ Date: 01/10/07 Time: 4:16 PM

Core Damage Summary Exelon MW PWRCDAM v1.2

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1. OVERVIEW

1.1 As a Windows based application designed in Access, PWR CDAM, uses many standard user interfaces. Instructions are not provided in basic computer operations in the Windows® environment. The user must be familiar with these to efficiently operate the program.

1.2 It is also assumed user is familiar with basic reactor physics and core damage fundamentals. Emergency Response Organization training will provide an overview of core damage assessment methodologies.

1.3 The program should be used by qualified personnel as a tool to estimate the type and amount of core damage.

2. DETERMINE APPROPRIATE AND AVAILABLE ASSESSMENT METHODS

2.1.1 The magnitude and type of event, transport mechanism and time after shutdown will be influencing factors on the method(s) utilized to determine the extent of core damage. Damage estimates can be developed using one or more methods as they become available or applicable.

1. Indications of Core Damage

   The primary indicators of core damage that are available during the early phases of an event:

   A. Containment Radiation Monitor Readings

   B. Core Exit Thermocouple Readings

2. Auxiliary indicators that are used to confirm and better define the possible type of damage are:

   A. Estimation of maximum temperature reached within the core

   B. Reactor Coolant System Hot Leg Temperature

   C. Estimated core uncovery time

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D. Reactor Vessel Level Indication System readings

E. Abnormal Source Range Monitor readings

F. Containment Hydrogen Readings

3. Long Term Indicators (once liquid or gaseous samples can be safely obtained) are:

A. Isotopic Ratios

B. Presence of high levels of rare isotopes

C. Quantity of isotopes present in samples

2.1.2 Choose the assessment method(s) most appropriate for the existing conditions. Methods available for assisting in the determination of the extent of core damage include the following:

Method	Use	Comment
Core Exit Thermocouples	Indication of onset of Core Damage	Limited due to range of instruments. Not reliable during later phases of core overheating due to changes in core geometry.
RCITS	Indication of Core Uncovery	Indicates possible damage not useful in estimating the quantity of damage.
Source Range Monitor	Indication of Core Uncovery	Loss of water level leads to increase in neutron detection (fast flux leakage).
Hot Leg RTDs	Indication of Core Uncovery	Only measures bulk flow through core. Hot spots in core may not be detected by RTD thermocouples.
Containment Radiation Monitor	Early Indication of Core Damage	Uncertainties due to variables in release of fission products from RCS and effects of containment sprays.
Containment Hydrogen Monitor	Early Indication of Core Damage	Significant uncertainties due to variable Hydrogen generation in core and in release of Hydrogen from RCS and effects of containment sprays.
RCS Samples and Containment Sump and Atmosphere Samples	Late Indication of Core Damage —Sump Samples provide indication of Rx Vessel Failure	Very large uncertainties until all systems have reached equilibrium. Useful in planning long term recovery.

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3. START UP

3.1.1 The application is accessed by one of the following:

1. Open the PWR CDAM desktop icon on applicable dose assessment computers.
    - A. Start the PWR CDAM program for the plant that has declared an emergency.
    - B. Programs are labeled PWR CDAM.
2. Select RUN from the 'Start Bar' and type in the file path and name as follows:
    - C:\CDAM\PWR CDAM.MDB

3.1.2 **IF** the assigned Core Damage Assessment Computer cannot access the application or the CDAM program will not run, **THEN** Install PWR CDAM on any computer from CDs or Disks located in the TSC or the EOF Library. CDAM is installed by copying appropriate file to computer's hard drive.

4. SELECTION AND PERFORMANCE OF ASSESSMENT

4.1 Choose the assessment method(s) most appropriate for the existing conditions. Methods available for assisting in the determination of the extent of core damage include the following:

1. Containment Radiation Analysis - (Section 5.2)
2. Core Temperature Analyses - (Section 5.3)
3. Core Water Level Analyses - (Section 5.4)
4. Containment Hydrogen Analysis - (Section 5.5)
5. Nuclide Analyses (Ratios and Abnormal Isotopes) - (Section 5.6)
6. Liquid Samples Analysis - (Section 5.7)
7. Gaseous Samples Analysis - (Section 5.8)

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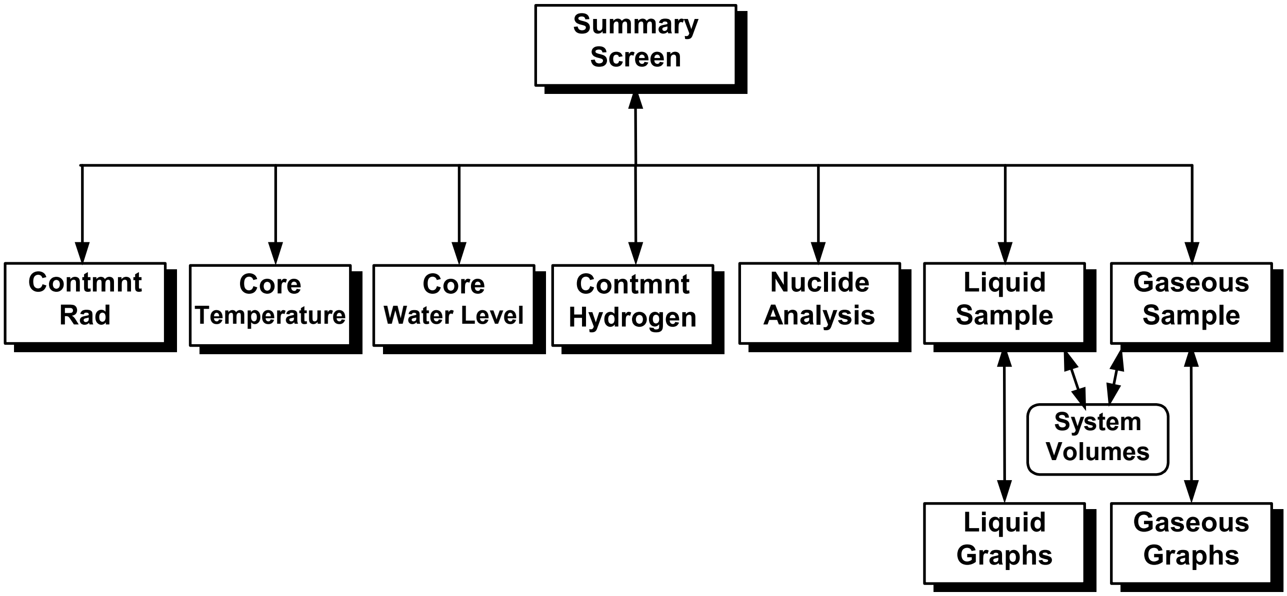
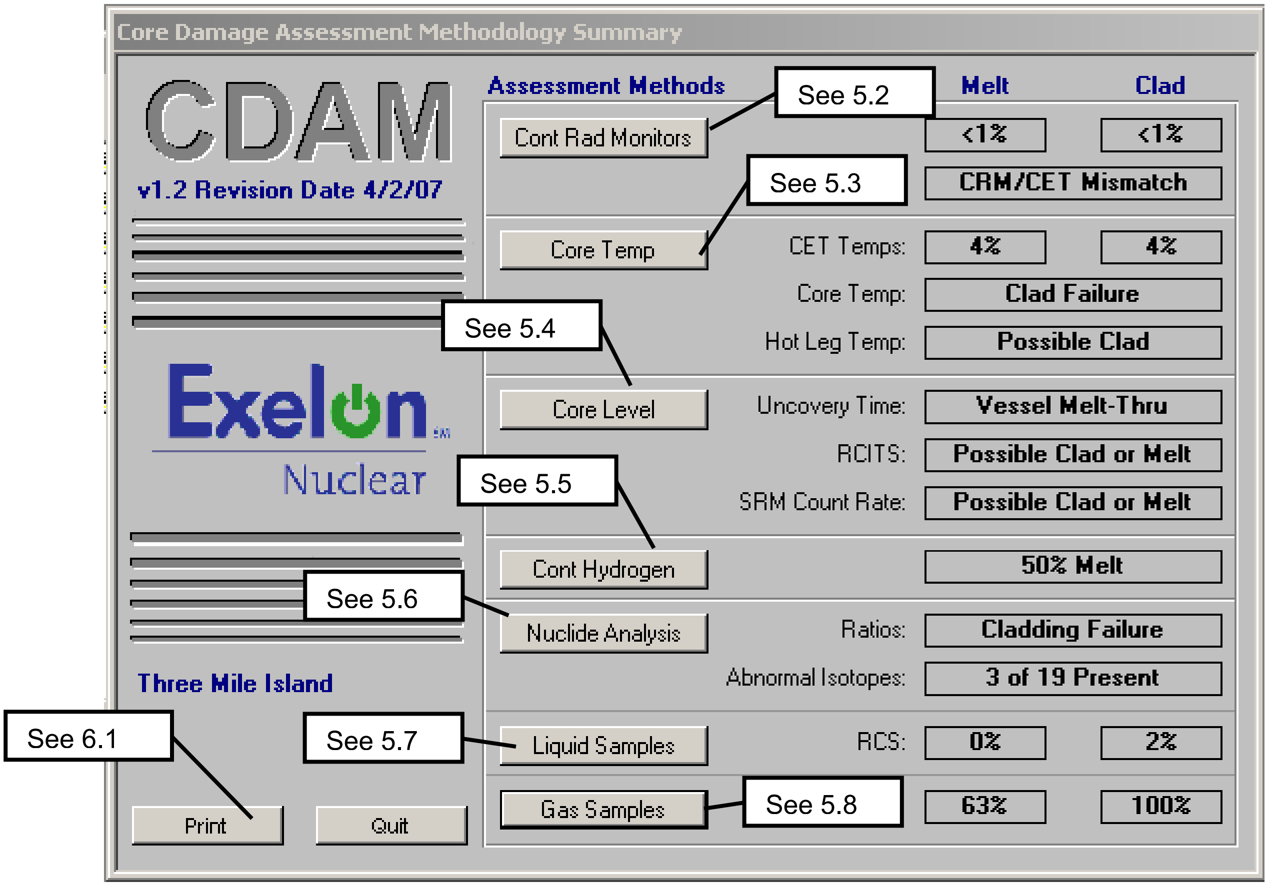
4.2 Basic Program Flow Diagram



5. PROGRAM SCREENS AND INPUTS

5.1 Main Screen – Summary Page

5.1.1 When program is started the following screen appears: (boxes are empty when program is originally launched.)



Core Damage Assessment Methodology Summary

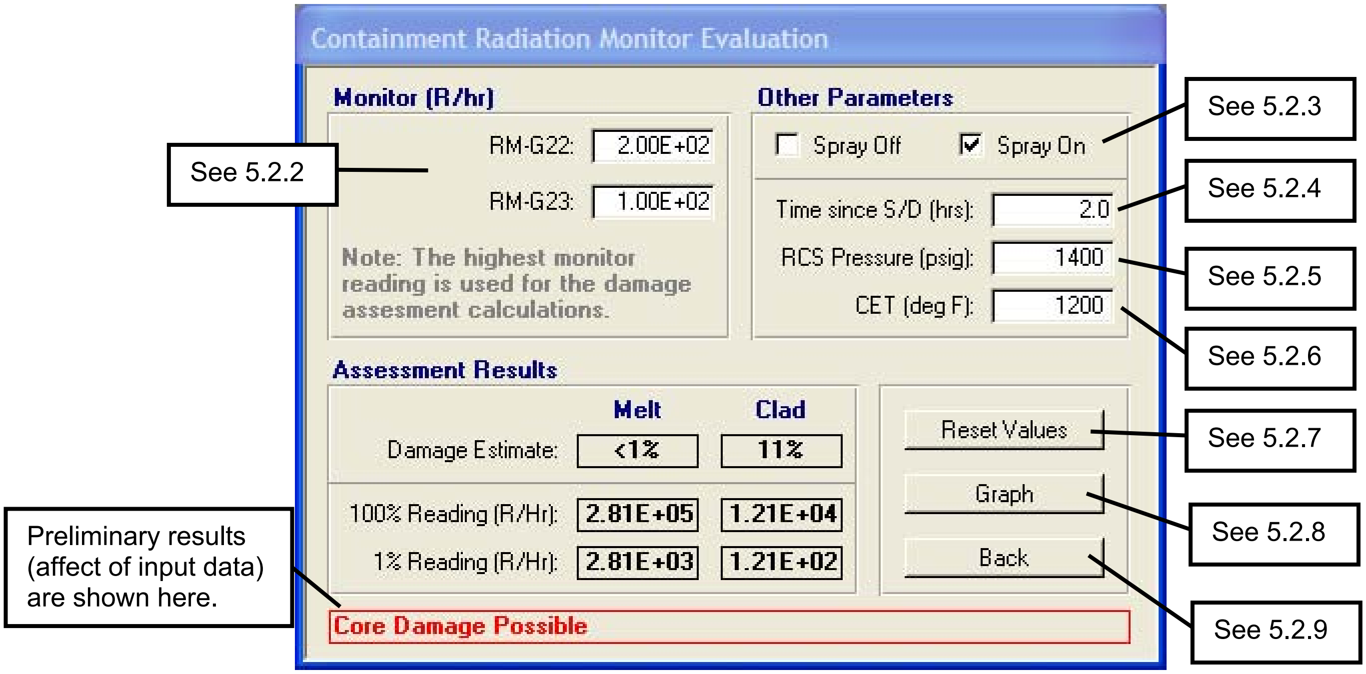
Assessment Methods		Melt	Clad
Cont Rad Monitors		<1%	<1%
		CRM/CET Mismatch	
Core Temp	CET Temps:	4%	4%
	Core Temp:	Clad Failure	
	Hot Leg Temp:	Possible Clad	
Core Level	Uncovery Time:	Vessel Melt-Thru	
	RCITS:	Possible Clad or Melt	
	SRM Count Rate:	Possible Clad or Melt	
Cont Hydrogen		50% Melt	
Nuclide Analysis	Ratios:	Cladding Failure	
	Abnormal Isotopes:	3 of 19 Present	
Liquid Samples	RCS:	0%	2%
Gas Samples		63%	100%

CAUTION

Pressing the **"Quit"** button exits the program. When the program is closed all data is reset. Program saves no information to disk; printed reports serve as record of core damage assessments.

5.2 Containment Radiation Monitor Method

5.2.1 Pressing **"Cont Rad Monitors"** button opens the following form:



Containment Radiation Monitor Evaluation

Monitor (R/hr)

RM-G22: 2.00E+02

RM-G23: 1.00E+02

Note: The highest monitor reading is used for the damage assesment calculations.

Other Parameters

☐ Spray Off ☑ Spray On

Time since S/D (hrs): 2.0

RCS Pressure (psig): 1400

CET (deg F): 1200

Assessment Results

	Melt	Clad
Damage Estimate:	<1%	11%
100% Reading (R/Hr):	2.81E+05	1.21E+04
1% Reading (R/Hr):	2.81E+03	1.21E+02

Core Damage Possible

See 5.2.2

See 5.2.3

See 5.2.4

See 5.2.5

See 5.2.6

See 5.2.7

See 5.2.8

See 5.2.9

Preliminary results (affect of input data) are shown here.

5.2.2 Highest containment radiation monitor reading which occurred is entered in these boxes. Program only lists containment high range monitors, however a reading may be entered from any monitor which accurately showed containment radiation levels. If two entries are made only the highest is used.

5.2.3 Containment Spray

1. Choose “Spray On” when sprays are operated after the following events:

   The expected start of Core Damage

   AND

   A release of the core isotopic inventory into the Containment atmosphere has occurred.

2. **If** the Containment Spray system was **not** operated after the start of Core Damage and the release of the isotopic inventory into containment **then** choose “Spray Off.”

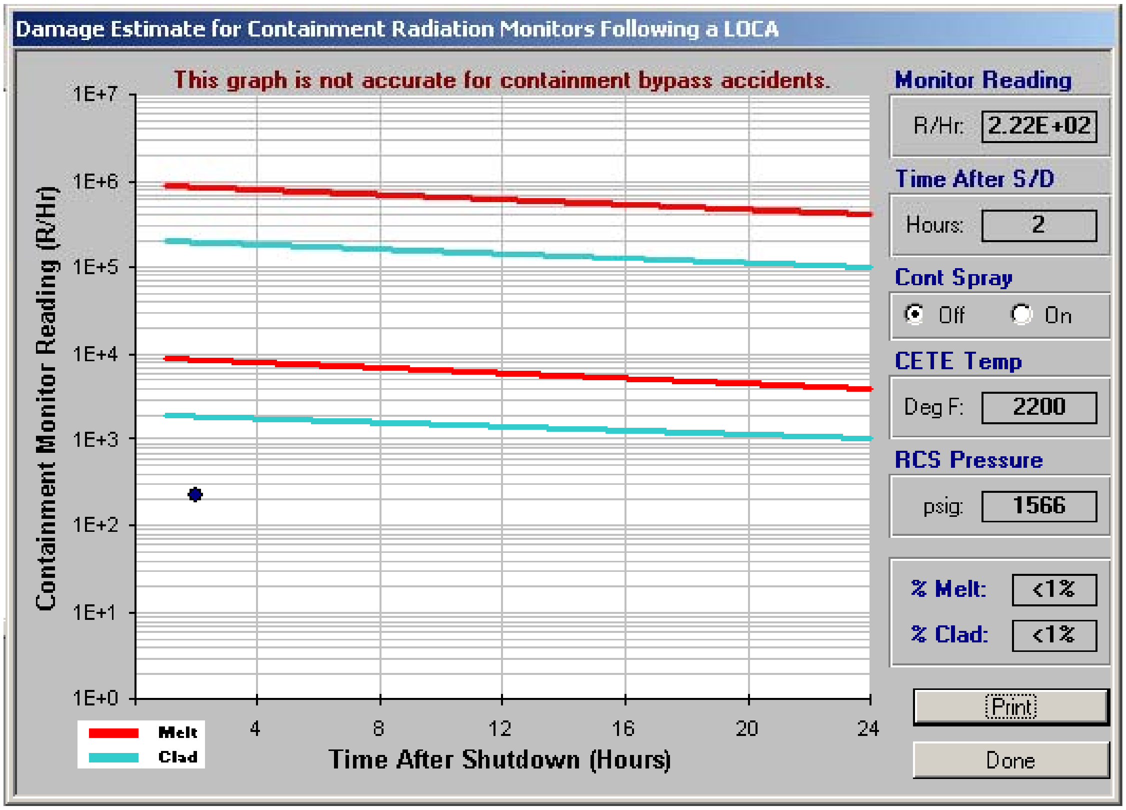
5.2.4 Enter the time after reactor shutdown, which corresponds to the time the containment radiation reading was taken. Value must be between 1 hour and 24 hours after shutdown, which corresponds to the time period in which this method is considered effective.

5.2.5 Enter the estimated Reactor Coolant System pressure at the time when core damage occurred (usually same time as high CET temperatures were observed).

5.2.6 Enter the highest Core Exit Thermocouple reading observed during the event.

5.2.7 Pressing **"Reset"** button resets values on this form only.

5.2.8 Pressing **"Graph"** button displays the follow screen.



Damage Estimate for Containment Radiation Monitors Following a LOCA

This graph is not accurate for containment bypass accidents.

Monitor Reading
R/Hr: 2.22E+02

Time After S/D
Hours: 2

Cont Spray
● Off ○ On

CETE Temp
Deg F: 2200

RCS Pressure
psig: 1566

% Melt: <1%
% Clad: <1%

1. Graph shows high and low containment radiation levels which correspond to 100% Melt or Clad or 1% Melt or Clad damage. A dot shows the last containment radiation level entered into the program for assessment.

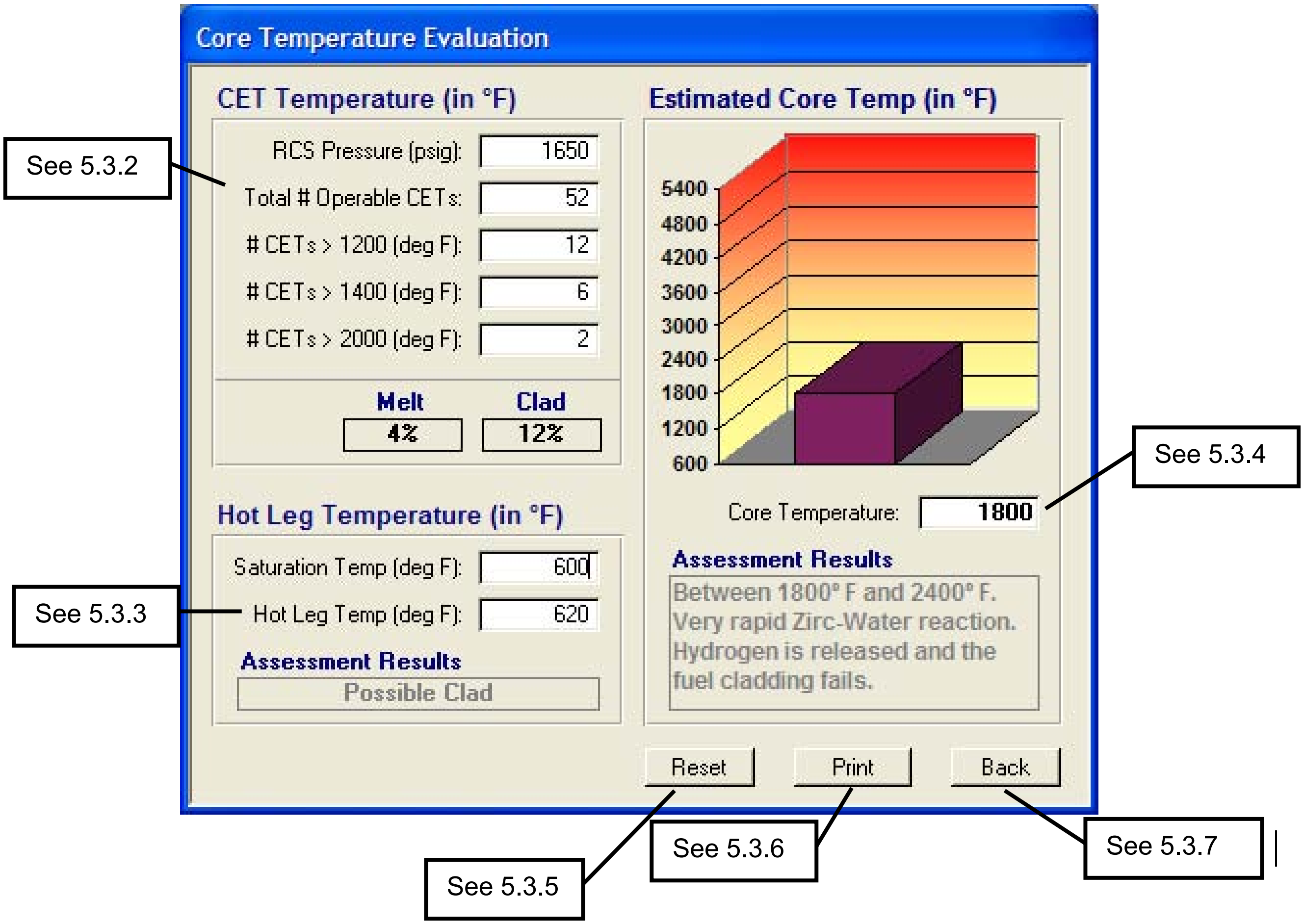
2. Pressing "Print Button" will print report of containment radiation method inputs and best estimate of damage.

3. Pressing "Done" returns the user back to the Evaluation screen

5.2.9 Pressing **"Back"** button takes the user back to the summary screen.

5.3 Core Temperature Methods

5.3.1 Pressing **"Core Temp"** button opens the following form:



Core Temperature Evaluation

CET Temperature (in °F)

RCS Pressure (psig): 1650
Total # Operable CETs: 52
CETs > 1200 (deg F): 12
CETs > 1400 (deg F): 6
CETs > 2000 (deg F): 2

Melt 4%
Clad 12%

Hot Leg Temperature (in °F)

Saturation Temp (deg F): 600
Hot Leg Temp (deg F): 620

Assessment Results
Possible Clad

Estimated Core Temp (in °F)

Core Temperature: 1800

Assessment Results
Between 1800° F and 2400° F. Very rapid Zirc-Water reaction. Hydrogen is released and the fuel cladding fails.

5.3.2 Core Exit Thermocouples (CETs)

1. Enter the Reactor Coolant System pressure at the time the CETs readings were taken.

2. Normally there are 50 operating CETs, however user should enter the number that was operating when temperature readings were taken.

3. Enter number of CETs that exceeded the listed temperatures. Program will not allow user to enter a higher number of CETs than the temperature box above it. (i.e. if only 5 CETs exceeded 1200 °F there can not be 6 exceeding 1400 °F).

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5.3.3 Reactor Coolant System Hot Leg temperature.

1. Enter saturation temperature for RCS pressure at time of highest RCS Hot Leg temperature. Value must be looked up in steam tables. Value is limited to 650 °F, which corresponds to max system pressure.
2. Enter highest Hot Leg temperature observed during expected time of core damage.

5.3.4 Based on inputs from Reactor Operators, TSC Staff and other engineering personnel (including outside sources such as AREVA personnel) enter the estimated highest temperature reached in the reactor core.

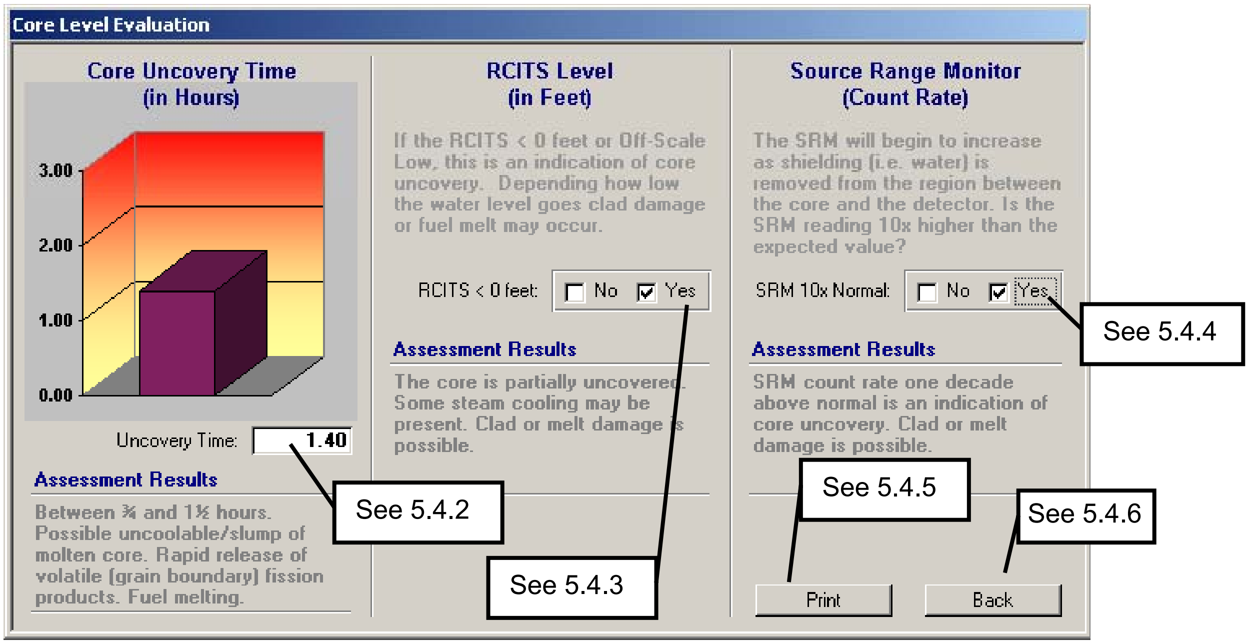
5.3.5 Pressing **"Reset"** button resets values on this form only.

5.3.6 Pressing **"Print"** button prints report of inputs and results of core temperature methods of core damage assessment.

5.3.7 Pressing **"Back"** button takes the user back to the summary screen.

5.4 Core Level Evaluations

5.4.1 Pressing **"Core Level"** button opens the following form:



Core Level Evaluation

Core Uncovery Time (in Hours)

Uncovery Time: 1.40

Assessment Results

Between ¾ and 1½ hours. Possible uncoolable/slump of molten core. Rapid release of volatile (grain boundary) fission products. Fuel melting.

RCITS Level (in Feet)

If the RCITS < 0 feet or Off-Scale Low, this is an indication of core uncovery. Depending how low the water level goes clad damage or fuel melt may occur.

RCITS < 0 feet: ☐ No ☑ Yes

Assessment Results

The core is partially uncovered. Some steam cooling may be present. Clad or melt damage is possible.

Source Range Monitor (Count Rate)

The SRM will begin to increase as shielding (i.e. water) is removed from the region between the core and the detector. Is the SRM reading 10x higher than the expected value?

SRM 10x Normal: ☐ No ☑ Yes

Assessment Results

SRM count rate one decade above normal is an indication of core uncovery. Clad or melt damage is possible.

ATTACHMENT 2

TMI CDAM USERS GUIDE

5.4.2 Enter estimated time portions of the reactor core was uncovered.

5.4.3 Enter if the Reactor Coolant Inventory Tracking System (RCITS) was off-scale low or indicated < 0 feet.

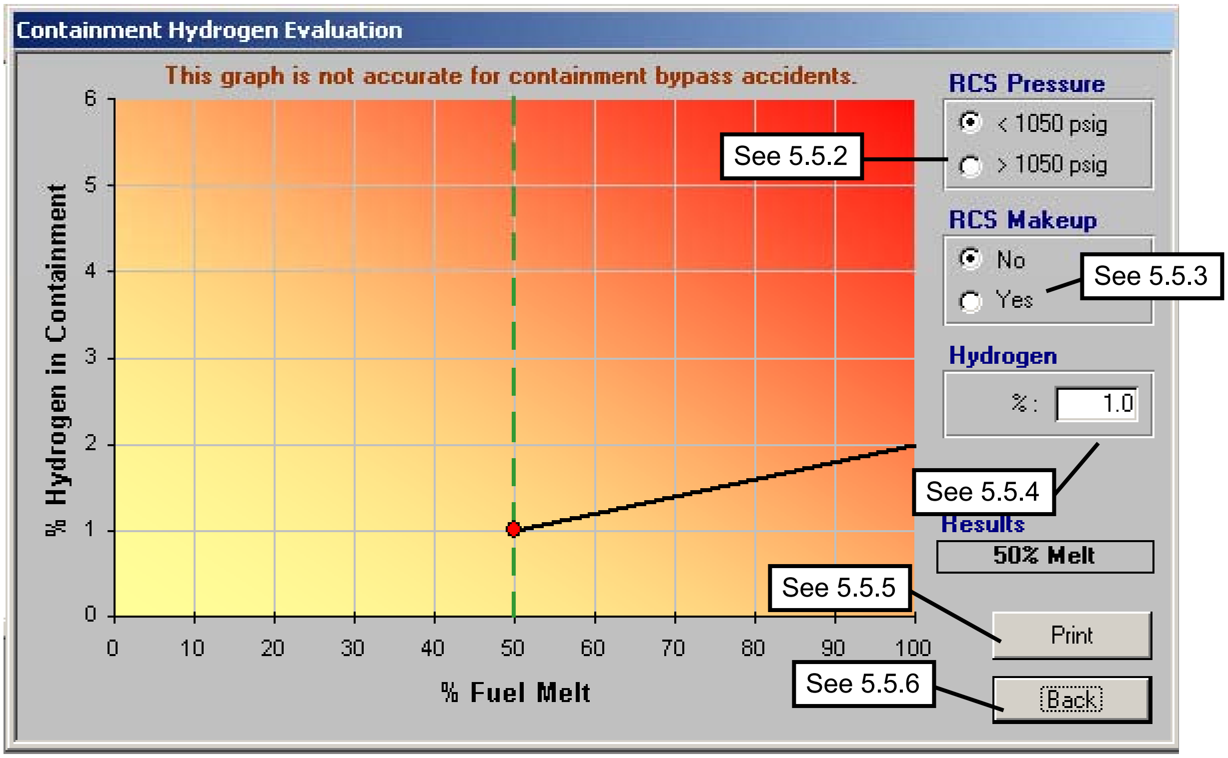
5.4.4 Check if the Source Range Monitoring system indicated abnormally high readings during the event (i.e., 1 decade above normal reading, indicating erratic performance).

5.4.5 Pressing **"Print"** button prints report of inputs and results of core level methods of core damage assessment.

5.4.6 Pressing **"Back"** button takes the user back to the summary screen.

5.5 Containment Hydrogen Evaluations

5.5.1 Pressing **"Cont Hydrogen"** button opens the following form:



Containment Hydrogen Evaluation

This graph is not accurate for containment bypass accidents.

RCS Pressure
● < 1050 psig
○ > 1050 psig

RCS Makeup
● No
○ Yes

Hydrogen
%: 1.0

Results
50% Melt

5.5.2 Choose the estimated Reactor Coolant System (RCS) pressure at the time core damage was occurring.

ATTACHMENT 2

TMI CDAM USERS GUIDE

5.5.3 RCS Makeup:

1. Choose **“No”** if no or little water was added to the RCS system during the time period core damage was occurring.
2. Choose **“Yes”** if water was added to the RCS during the time core damage was occurring or prior to the time a Large Leak occurred from the RCS into the containment structure.

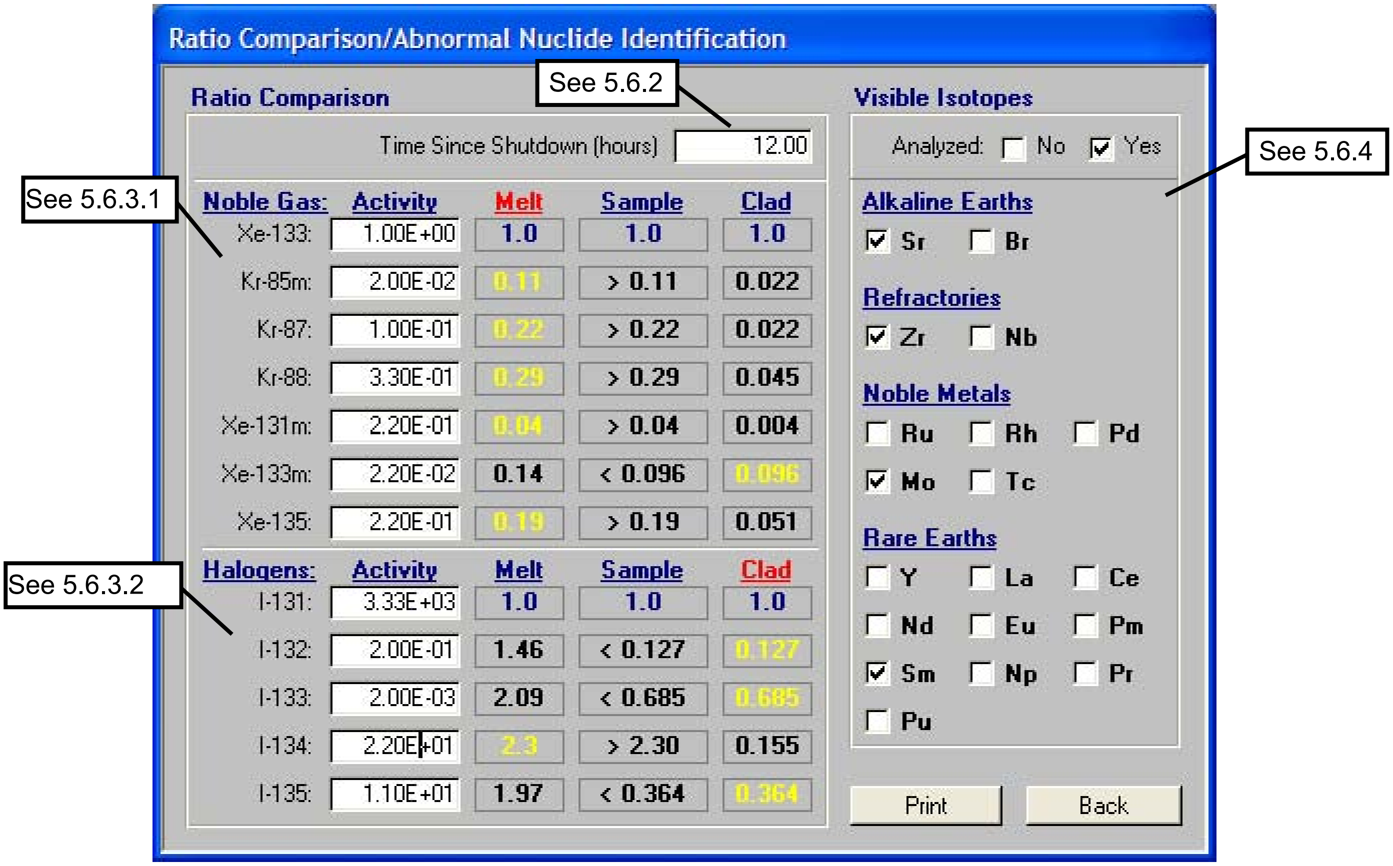
5.5.4 Enter highest containment hydrogen level measured. H_2 monitoring equipment is only accurate within a ± 1 % range so no damage is reported until level reaches at least 1 %. Range of instrument is 0 – 30 %.

5.5.5 Pressing **“Print”** button prints report of inputs and results of core level methods of core damage assessment.

5.5.6 Pressing **“Back”** button takes the user back to the summary screen.

5.6 Nuclide Analysis

5.6.1 Pressing **“Nuclide Analysis”** button opens the following form:



See 5.6.2

Time Since Shutdown (hours) 12.00

See 5.6.3.1

Noble Gas:	Activity	Melt	Sample	Clad
Xe-133:	1.00E+00	1.0	1.0	1.0
Kr-85m:	2.00E-02	0.11	> 0.11	0.022
Kr-87:	1.00E-01	0.22	> 0.22	0.022
Kr-88:	3.30E-01	0.29	> 0.29	0.045
Xe-131m:	2.20E-01	0.04	> 0.04	0.004
Xe-133m:	2.20E-02	0.14	< 0.096	0.096
Xe-135:	2.20E-01	0.19	> 0.19	0.051

See 5.6.3.2

Halogens:	Activity	Melt	Sample	Clad
I-131:	3.33E+03	1.0	1.0	1.0
I-132:	2.00E-01	1.46	< 0.127	0.127
I-133:	2.00E-03	2.09	< 0.685	0.685
I-134:	2.20E+01	2.3	> 2.30	0.155
I-135:	1.10E+01	1.97	< 0.364	0.364

See 5.6.4

Analyzed: ☐ No ☑ Yes

Alkaline Earths
☑ Sr ☐ Br

Refractories
☑ Zr ☐ Nb

Noble Metals
☐ Ru ☐ Rh ☐ Pd
☑ Mo ☐ Tc

Rare Earths
☐ Y ☐ La ☐ Ce
☐ Nd ☐ Eu ☐ Pm
☑ Sm ☐ Np ☐ Pr
☐ Pu

5.6.2 Enter the time since reactor shutdown when the sample was taken.

5.6.3 **ENTER** isotopic sample results in uCi/cc. Sample results are to be decay corrected back to time after shutdown that the sample was drawn.

1. Noble Gases are ratioed to Xe-133
2. Halogens are ratioed to I-131

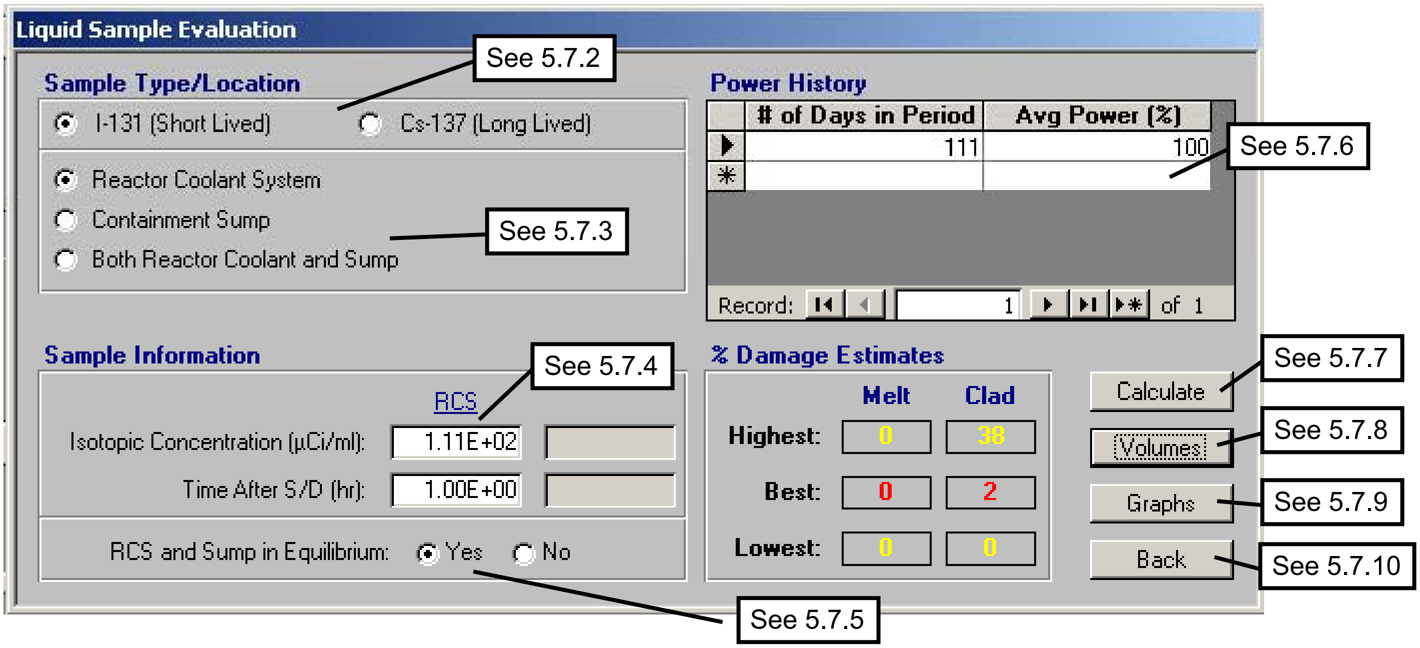
5.6.4 **IF** abnormal levels of rare isotopes are present **THEN** check yes AND check which isotopes are present.

5.6.5 Pressing **“Print”** button prints report of inputs and results of Nuclide Analysis methods of core damage assessment.

5.6.6 Pressing **"Back"** button takes the user back to the summary screen.

5.7 Liquid Samples

5.7.1 Pressing **"Liquid Samples"** button opens the following form:



Liquid Sample Evaluation

Sample Type/Location

● I-131 (Short Lived) ○ Cs-137 (Long Lived)

● Reactor Coolant System
○ Containment Sump
○ Both Reactor Coolant and Sump

Power History

# of Days in Period	Avg Power (%)
111	100

Record: 1 of 1

Sample Information

	RCS	
Isotopic Concentration (µCi/ml):	1.11E+02	
Time After S/D (hr):	1.00E+00	

RCS and Sump in Equilibrium: ● Yes ○ No

% Damage Estimates

	Melt	Clad
Highest:	0	38
Best:	0	2
Lowest:	0	0

5.7.2 Select appropriate isotope.

NOTE: A volume entry must be made for sump volume, Before you can choose "Containment Sump" or "Both Reactor Coolant and Sump." Refer to Step 5.7.7.

5.7.3 Select sample location. If samples are available from both locations select both.

5.7.4 Enter sample Isotopic Concentrations(s) in µCi/ml for selected isotope.

5.7.5 **SELECT** the appropriate System Equilibrium status:

NOTE: Systems reach equilibrium through mechanical mixing i.e. pumps circulating water between systems for several hours. Equilibrium conditions may be identified when temperature indications from different systems are approaching similar readings.

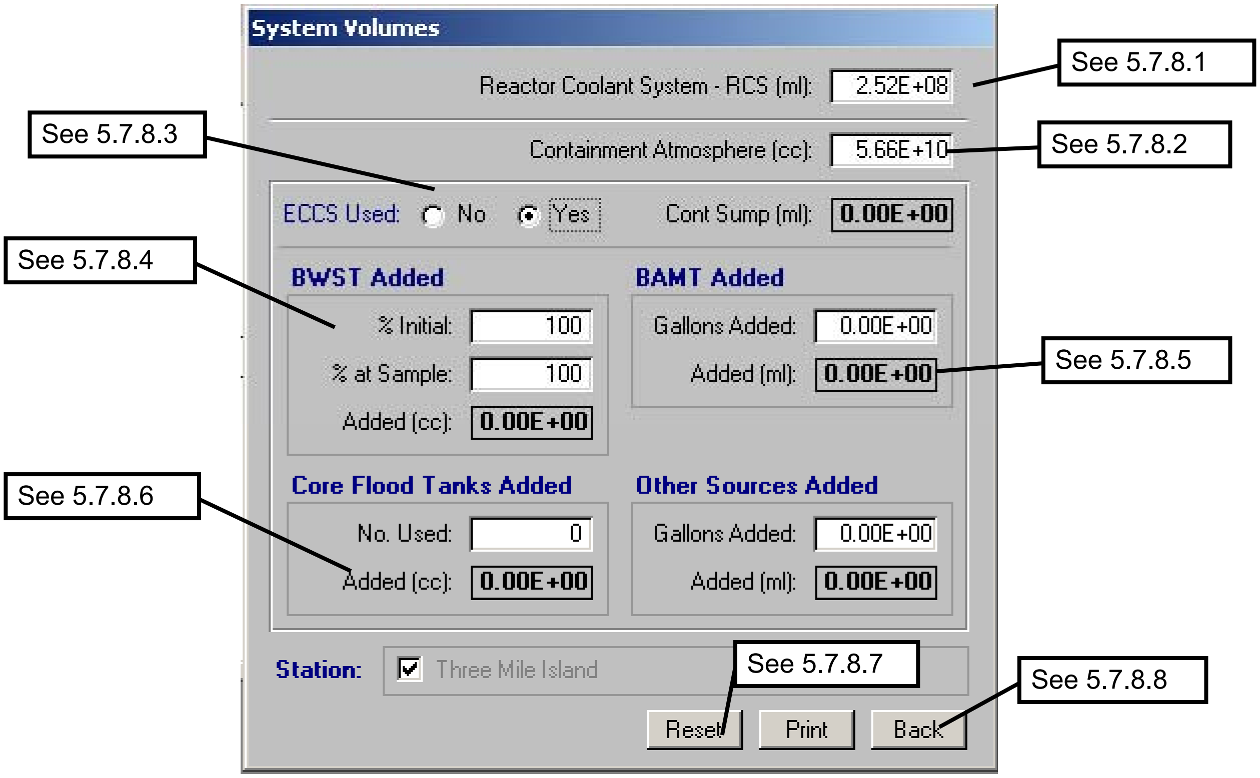
1. **If** sample was taken from only one location and systems are in equilibrium, **then** check “yes” for “RCS and Sump in Equilibrium,” **otherwise** check “no.”

5.7.6 Enter power history of core since last refueling. Shutdown times are entered as the number of days with Ave Power (%) set at 0.

1. For short-lived isotopes power history should extend at least 30 days.
2. For long-lived isotopes power history should extend at least 100 days, however the power history for the extent of the cycle is preferred.
3. Variations in steady state power should be limited to ± 20% within each operational period entered.

5.7.7 Once all data has been entered pressing the “Calculate” button will display the % Damage Estimates.

5.7.8 Pressing “Volumes” button displays the following screen:



System Volumes

Reactor Coolant System - RCS (ml): 2.52E+08

Containment Atmosphere (cc): 5.66E+10

ECCS Used: ○ No ● Yes

Cont Sump (ml): 0.00E+00

BWST Added

% Initial: 100

% at Sample: 100

Added (cc): 0.00E+00

BAMT Added

Gallons Added: 0.00E+00

Added (ml): 0.00E+00

Core Flood Tanks Added

No. Used: 0

Added (cc): 0.00E+00

Other Sources Added

Gallons Added: 0.00E+00

Added (ml): 0.00E+00

Station: ☑ Three Mile Island

ATTACHMENT 2

TMI CDAM USERS GUIDE

1. Program enters default RCS volume, which the user may change based on RCITS Readings and Pressurizer level at time of sample.

2. Program enters default Containment free air volume which user may change based on containment sump level at time of sample.

3. Program assumes Containment Sump volume is 0 unless there has been an activation of the Emergency Core Cooling System (ECCS). Checking yes allows user to estimate Containment Sump volume

4. The change in level of the Borated Water Storage Tank (BWST) determines amount of water in Containment Sump from this source.

5. The change in level of the Boric Acid Mix Tank (BAMT) determines amount of water in Containment Sump from this source.

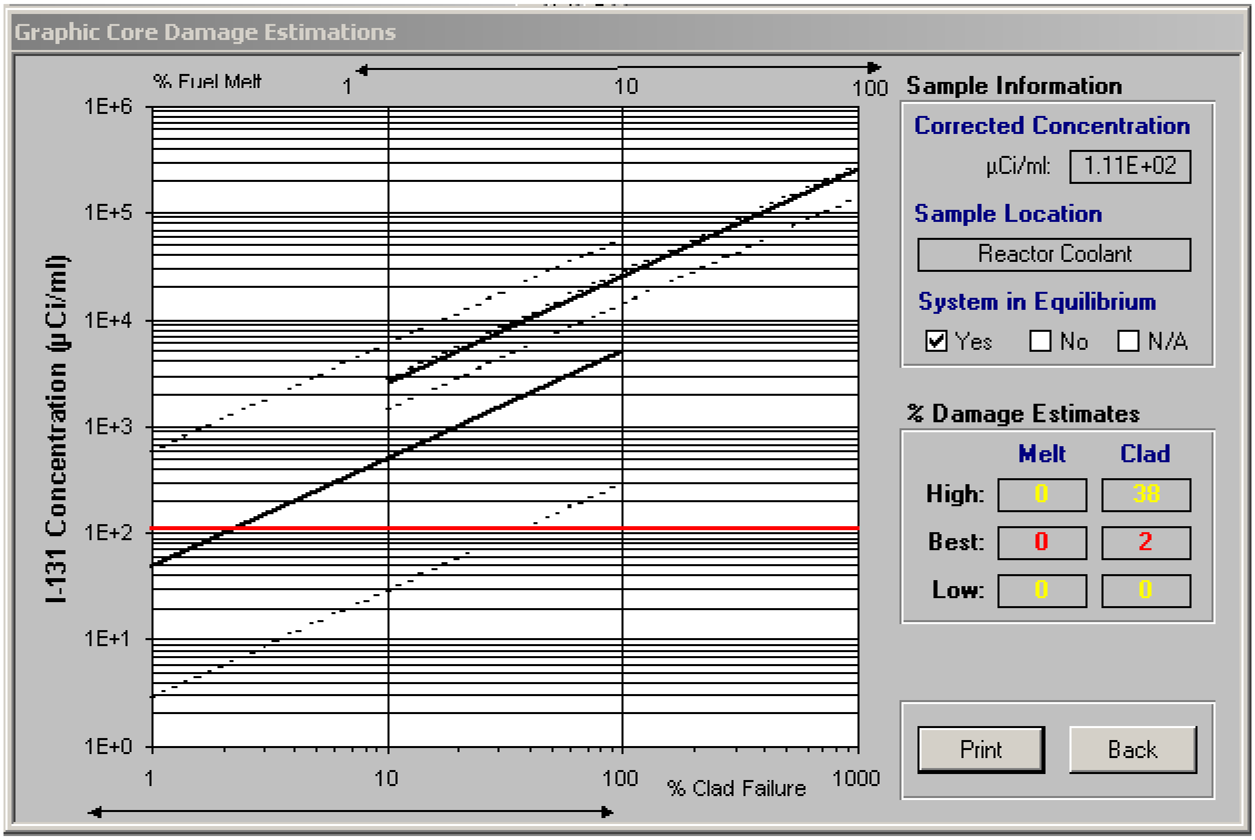
6. The number of Core Flood Tanks that have injected into the RCS determines amount of water in Containment Sump from this source.

NOTE: User may enter other sources of water added during an event. (such as fire main, secondary water, potable water, etc.).

7. Pressing "**Reset**" button resets all volumes to default values.

8. Pressing **"Back"** button takes the user back to the Liquid or Gaseous screen, which user used to call volume form.

5.7.9 Pressing **"Graph"** button displays the following screen:



Sample Information

Corrected Concentration

µCi/ml: 1.11E+02

Sample Location

Reactor Coolant

System in Equilibrium

☑ Yes ☐ No ☐ N/A

% Damage Estimates

	Melt	Clad
High:	0	38
Best:	0	2
Low:	0	0

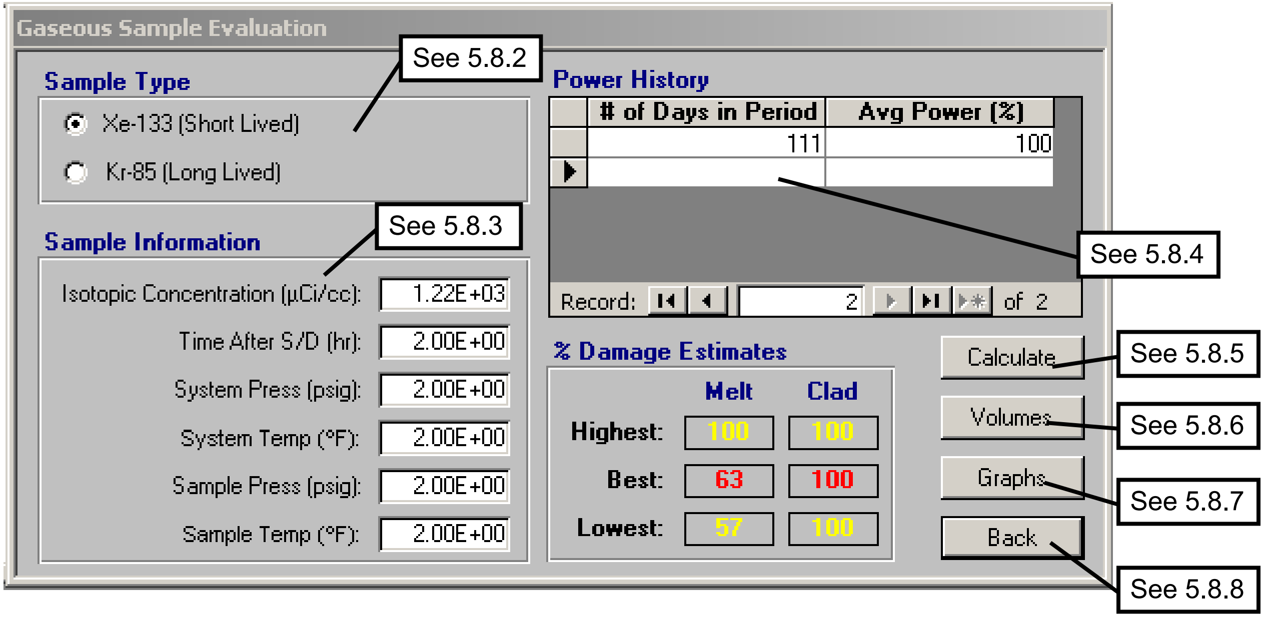
1. Graph shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating entered corrected sample concentration.

2. User can select "Print" button to print graph and summary of inputs or press "Back" button to go back to liquid or gaseous form which called this form.

5.7.10 Pressing **"Back"** button takes the user back to the summary screen.

5.8 <u>Gaseous Samples</u>

5.8.1 Pressing **"Gas Samples"** button opens the following form:



Gaseous Sample Evaluation

Sample Type

● Xe-133 (Short Lived)
○ Kr-85 (Long Lived)

Sample Information

Isotopic Concentration (µCi/cc): 1.22E+03
Time After S/D (hr): 2.00E+00
System Press (psig): 2.00E+00
System Temp (°F): 2.00E+00
Sample Press (psig): 2.00E+00
Sample Temp (°F): 2.00E+00

Power History

# of Days in Period	Avg Power (%)
111	100

Record: 2 of 2

% Damage Estimates

	Melt	Clad
Highest:	100	100
Best:	63	100
Lowest:	57	100

5.8.2 **Select** appropriate isotope.

5.8.3 **Enter** Sample Information:

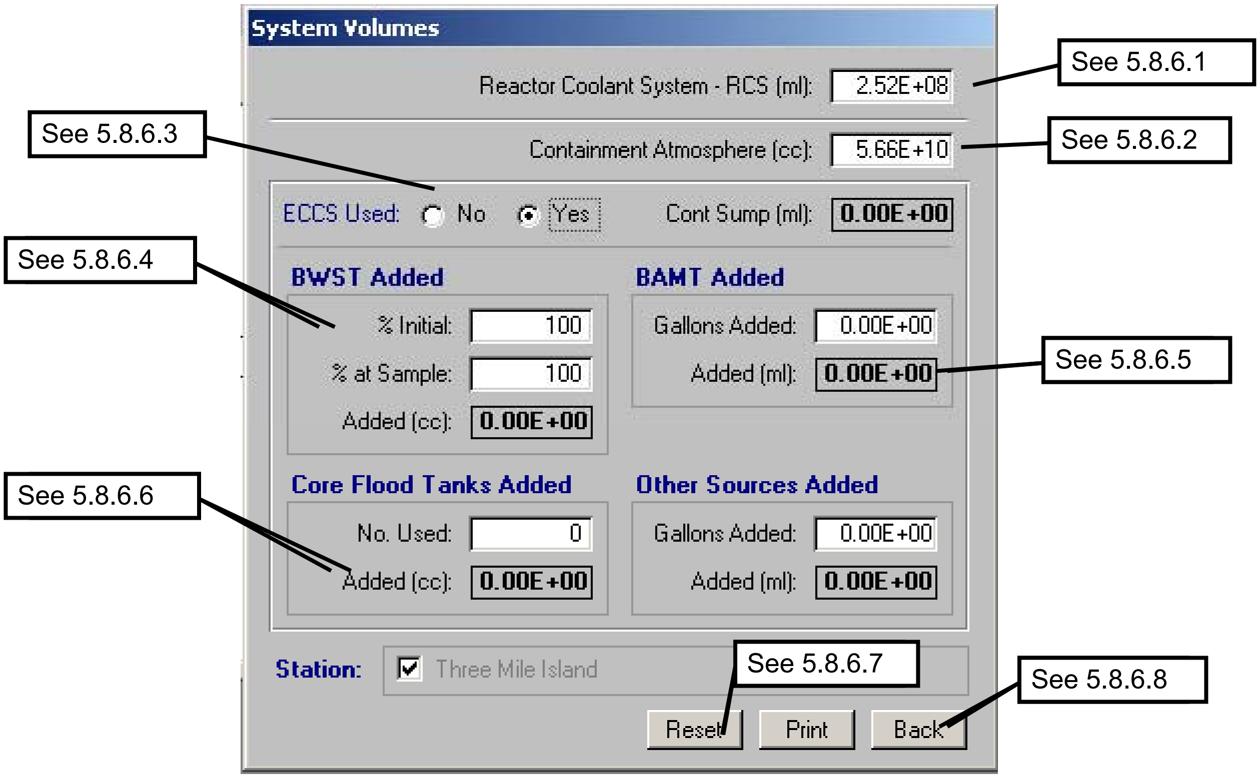
1. Enter sample Isotopic Concentrations(s) in µCi/ml for selected isotope.
2. Enter Time After S/D that sample was taken.
3. Enter the pressure and temperature of the system sampled
4. Enter the end pressure and temperature of sample.

5.8.4 **Enter** power history of core since last refueling. Shutdown times are entered as the number of days with Ave Power (%) set at 0.

1. For short-lived isotopes power history should extend at least 30 days.
2. For long-lived isotopes power history should extend at least 100 days, however the power history for the extent of the cycle is preferred.
3. Variations in steady state power should be limited to ± 20% within each operational period entered.

5.8.5 Once all data has been entered pressing the “Calculate” button will display the % Damage Estimates.

5.8.6 Pressing “Volumes” button displays the following screen (Same as 5.7.7):



System Volumes

Reactor Coolant System - RCS (ml): 2.52E+08

Containment Atmosphere (cc): 5.66E+10

ECCS Used: ○ No ● Yes

Cont Sump (ml): 0.00E+00

BWST Added

% Initial: 100

% at Sample: 100

Added (cc): 0.00E+00

BAMT Added

Gallons Added: 0.00E+00

Added (ml): 0.00E+00

Core Flood Tanks Added

No. Used: 0

Added (cc): 0.00E+00

Other Sources Added

Gallons Added: 0.00E+00

Added (ml): 0.00E+00

Station: ☑ Three Mile Island

See 5.8.6.1

See 5.8.6.2

See 5.8.6.3

See 5.8.6.4

See 5.8.6.5

See 5.8.6.6

See 5.8.6.7

See 5.8.6.8

1. Program enters default RCS volume, which the user may change based on RCITS Readings and Pressurizer level at time of sample.
2. Program enters default Containment free air volume which user may change based on containment sump level at time of sample.
3. Program assumes Containment Sump volume is 0 unless there has been an activation of the Emergency Core Cooling System (ECCS). Checking yes allows user to estimate Containment Sump volume.
4. The change in level of the Borated Water Storage Tank (BWST) determines amount of water in Containment Sump from this source.
5. The change in level of the Boric Acid Mix Tank (BAMT) determines amount of water in Containment Sump from this source.

ATTACHMENT 2

TMI CDAM USERS GUIDE

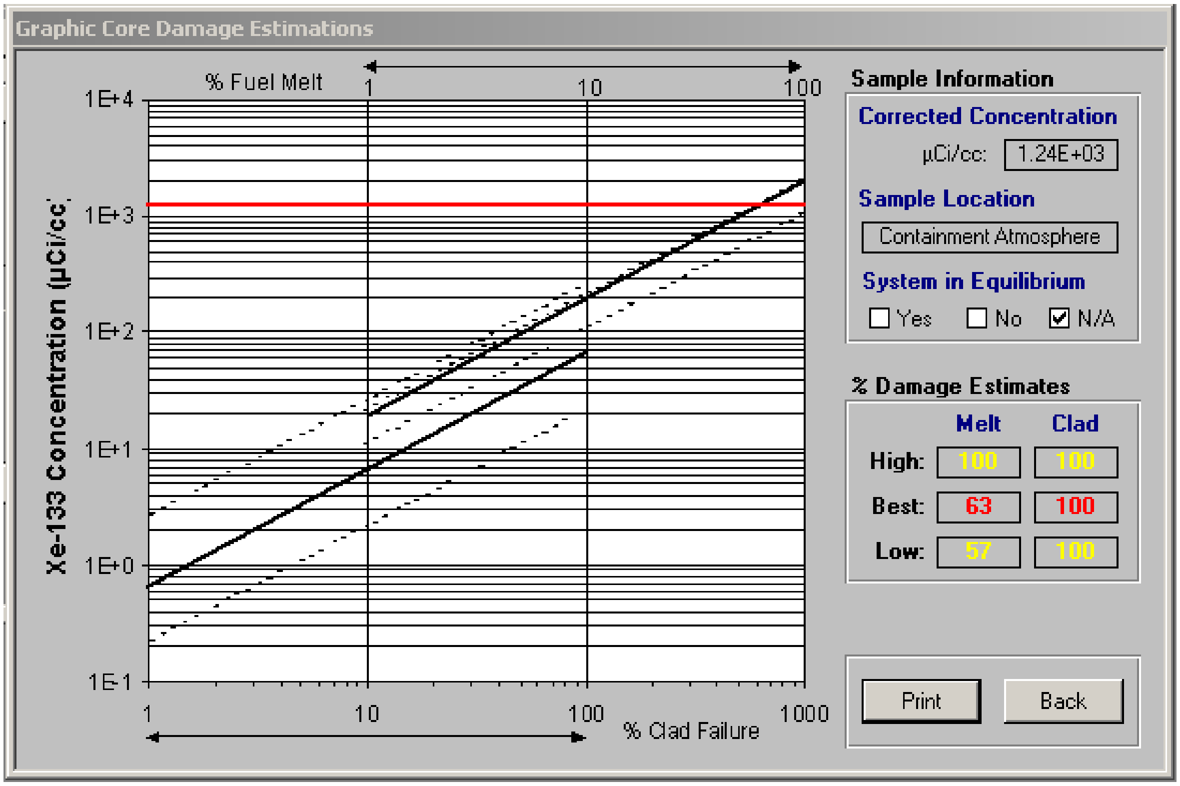
6. The number of Core Flood Tanks that have injected into the RCS determines amount of water in Containment Sump from this source.

NOTE: User may enter other sources of water added during an event. (such as fire main, secondary water, potable water, etc.).

7. Pressing “**Reset**” button resets all volumes to default values.

8. Pressing **“Back”** button takes the user back to the Liquid or Gaseous screen, which user used to call volume form.

5.8.7 Pressing **“Graph”** button displays the following screen:



Sample Information

Corrected Concentration

µCi/cc: 1.24E+03

Sample Location

Containment Atmosphere

System in Equilibrium

☐ Yes ☐ No ☑ N/A

% Damage Estimates

	Melt	Clad
High:	100	100
Best:	63	100
Low:	57	100

1. Graph shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating corrected concentration.

2. User can select “Print” button to print graph and summary of inputs or press “Back” button to go back to liquid or gaseous form which called this form.

5.8.8 Pressing **“Back”** button takes the user back to the summary screen.

ATTACHMENT 2

TMI CDAM USERS GUIDE

6. CORE DAMAGE SUMMARY REPORT

6.1 Once the program user enters data for all available assessment methods and the program calculates damage based on inputs, **SELECT** the **“Print”** button to print a summary of all methods used

6.1.1 A sample report is shown on the next page.

6.1.2 Individual tasked with assessing core damage shall then analyze report to determine best estimate of type and amount of damage.

7. QUITING, OR EXITING, THE PROGRAM

7.1 Pressing the “Quit” button on the Summary Screen exits the program.

7.1.1 When the program is closed all data is reset.

7.1.2 Program saves no information to disk; printed reports serve as record of core damage assessments.

ATTACHMENT 2

TMI CDAM USERS GUIDE

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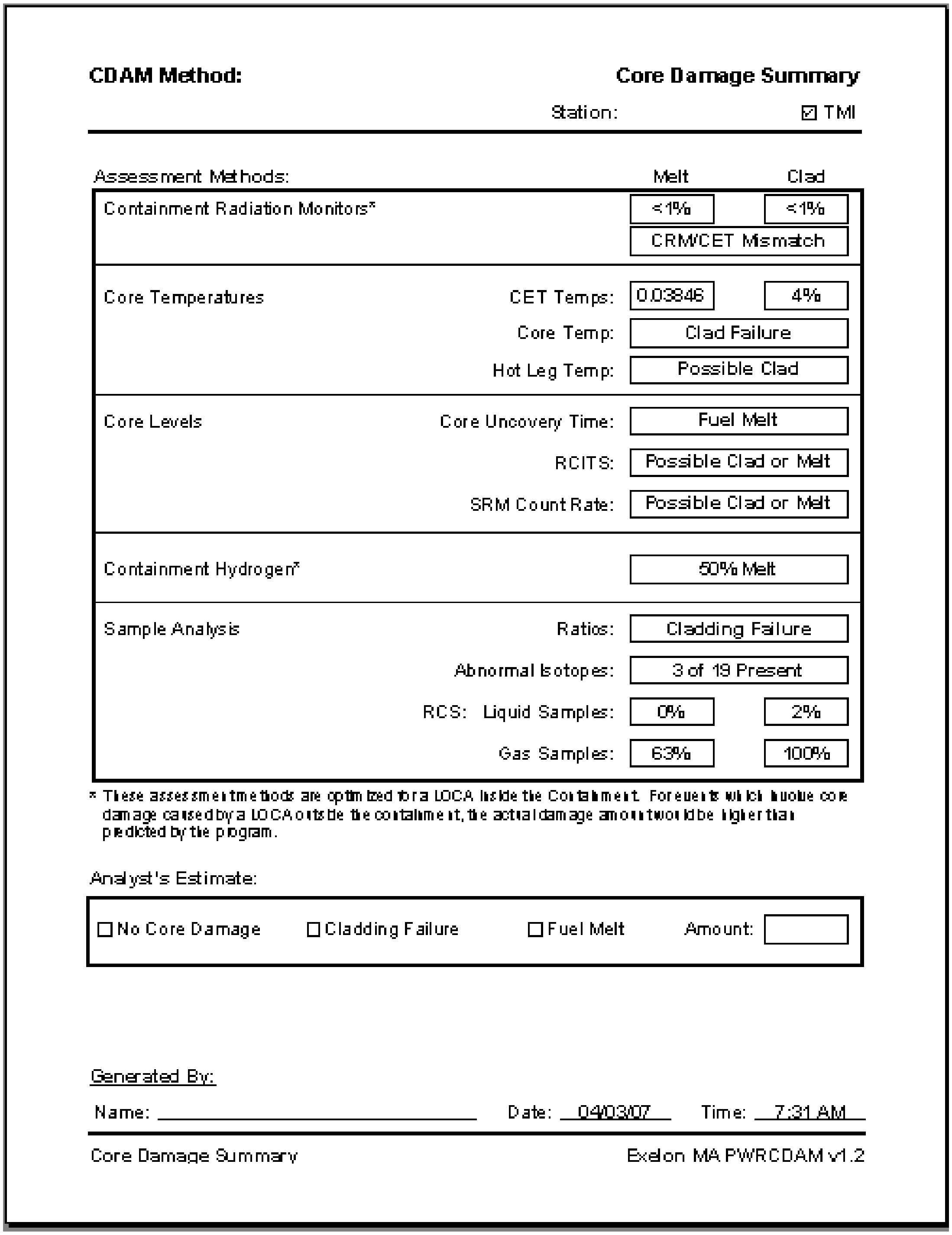
Sample Summary Report



CDAM Method: **Core Damage Summary**

Station: ☑ TMI

Assessment Methods:

		Melt	Clad
Containment Radiation Monitors*		<1%	<1%
		CRM/CET Mismatch	
Core Temperatures	CET Temps:	0.03846	4%
	Core Temp:	Clad Failure	
	Hot Leg Temp:	Possible Clad	
Core Levels	Core Uncovery Time:	Fuel Melt	
	RCITS:	Possible Clad or Melt	
	SRM Count Rate:	Possible Clad or Melt	
Containment Hydrogen*		50% Melt	
Sample Analysis	Ratios:	Cladding Failure	
	Abnormal Isotopes:	3 of 19 Present	
	RCS: Liquid Samples:	0%	2%
	Gas Samples:	63%	100%

* These assessment methods are optimized for a LOCA inside the Containment. For events which involve core damage caused by a LOCA outside the containment, the actual damage amount would be higher than predicted by the program.

Analyst's Estimate:

☐ No Core Damage ☐ Cladding Failure ☐ Fuel Melt Amount: ______

Generated By:

Name: ______________________ Date: 04/03/07 Time: 7:31 AM

Core Damage Summary Exelon MA PWRCDAM v1.2

Attachment 9

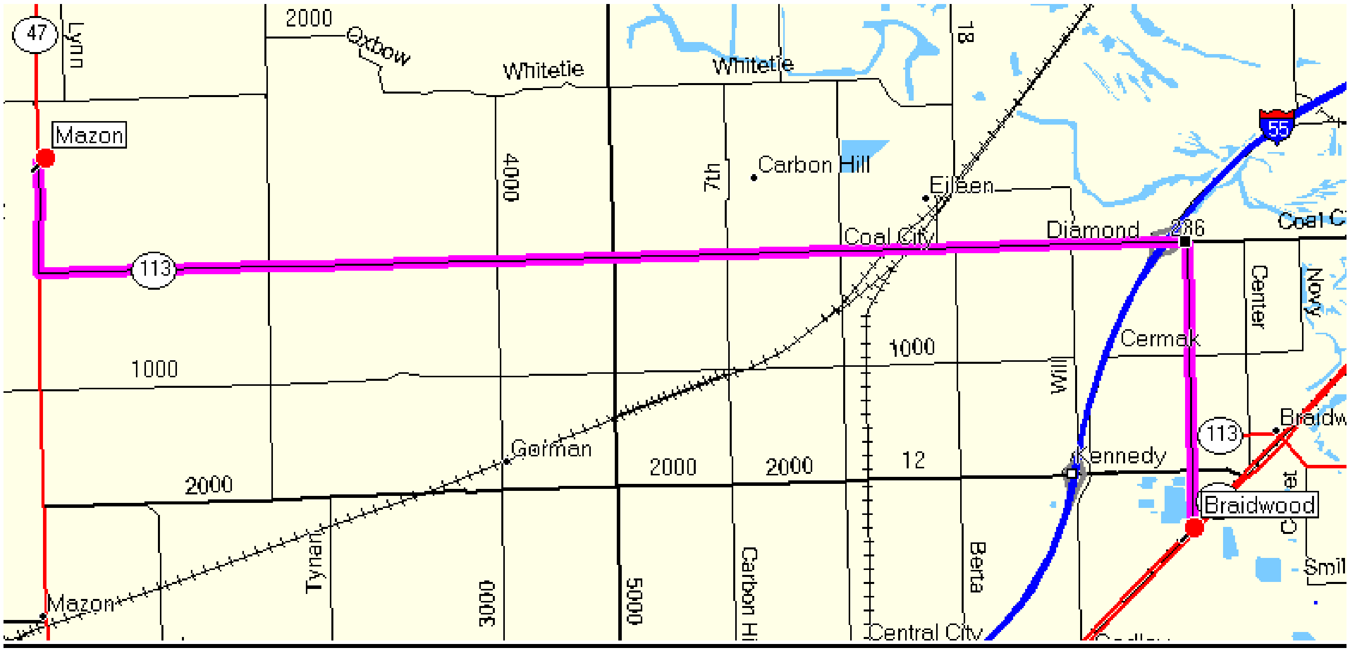
EP-AA-113-F-17

"Braidwood Assembly, Accountability and Evacuation Guidelines"

Revision D

BRAIDWOOD ASSEMBLY, ACCOUNTABILITY AND EVACUATION GUIDELINES

BRAIDWOOD STATION EVACUATION ROUTE TO MAZON RELOCATION CENTER



Travel Direction from Braidwood Station to the Mazon Relocation Center

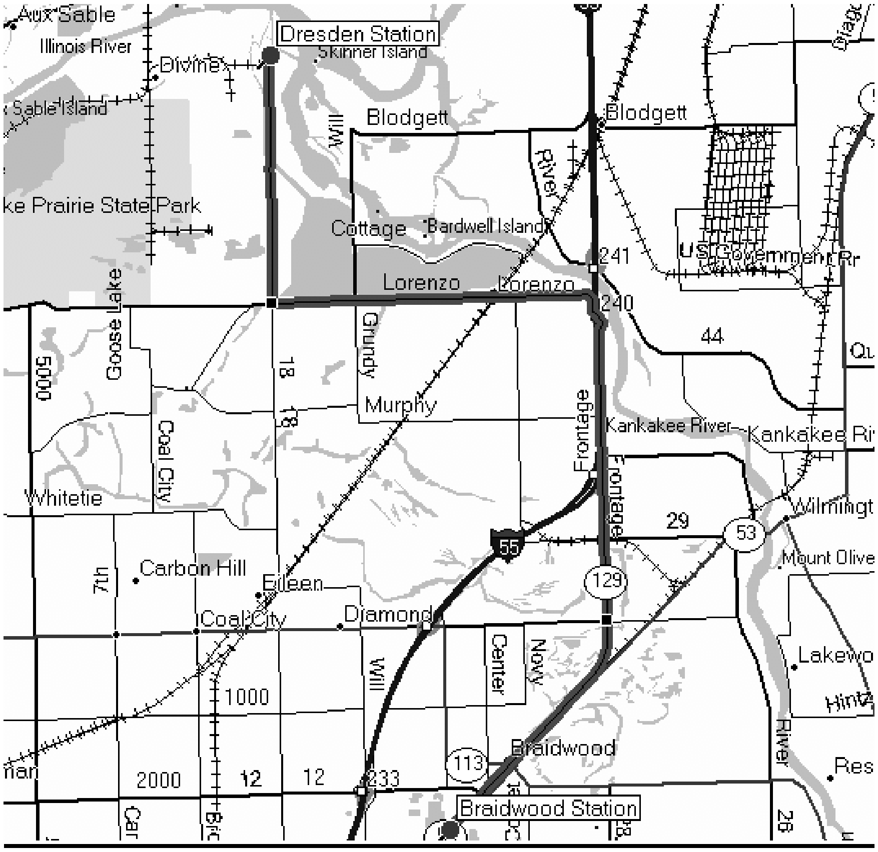
1. Take S. Division St. North (~2.5 miles)
2. Turn Left on Route 113 (~10 miles)
3. Turn Right on IL-47 to the Mazon Relocation Center (~1 miles)

When driving to the Relocation Center:

- Keep windows up
- Turn off air-conditioning/heater or place on re-circulation mode
- Leave keys in the car upon arrival at the Relocation Center
- Ensure Contamination is not spread
- Upon arrival follow Security/RP directions

BRAIDWOOD ASSEMBLY, ACCOUNTABILITY AND EVACUATION GUIDELINES

BRAIDWOOD STATION EVACUATION ROUTE to Dresden Station



Travel Direction from Braidwood Station to Dresden Station

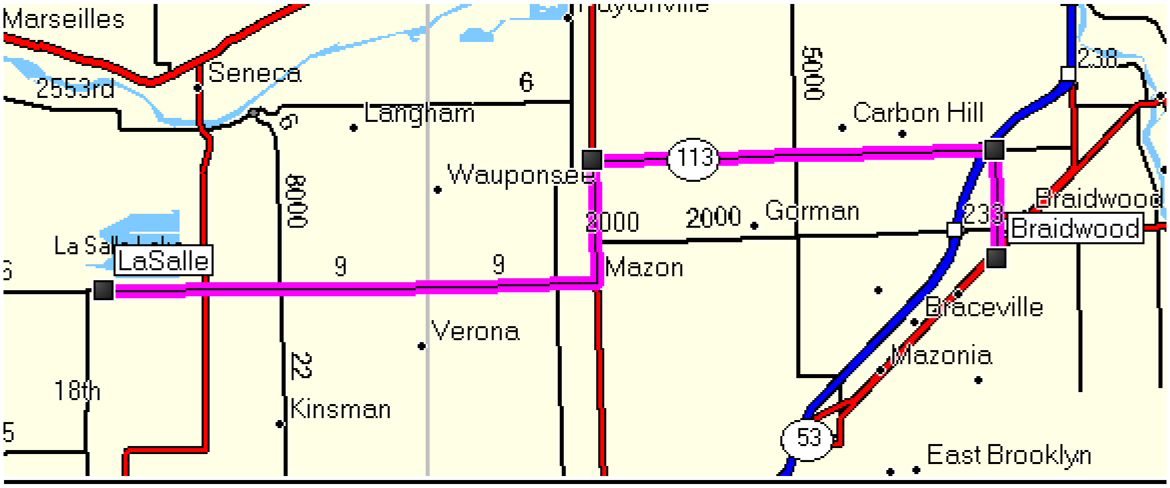
1. Take IL-129 North to I-55 (~5 miles)
2. Get on I-55 North to Lorenzo Road Exit (~2 miles)
3. Proceed West on Lorenzo Rd to Dresden Rd (~4 miles)
4. Turn Right on Dresden Rd to Dresden Station (~3 miles)

When driving to the Relocation Center:

- Keep windows up
- Turn off air-conditioning/heater or place on re-circulation mode
- Leave keys in the car upon arrival at the Relocation Center
- Ensure Contamination is not spread
- Upon arrival follow Security/RP directions

BRAIDWOOD ASSEMBLY, ACCOUNTABILITY AND EVACUATION GUIDELINES

BRAIDWOOD STATION EVACUATION ROUTE to LaSalle Station



Travel Direction from Braidwood Station to LaSalle Station

1. Take S. Division St. North to Route 113 (~2.5 miles)
2. Turn Left on Route 113 to IL 47(~10 miles)
3. Turn Left on IL 47 to Grand Ridge Road (~3 miles)
4. Turn right on Grand Ridge Road to LaSalle (~12 miles)

When driving to the Relocation Center:

- Keep windows up
- Turn off air-conditioning/heater or place on re-circulation mode
- Leave keys in the car upon arrival at the Relocation Center
- Ensure Contamination is not spread
- Upon arrival follow Security/RP directions

BRAIDWOOD ASSEMBLY, ACCOUNTABILITY AND EVACUATION GUIDELINES

Potentially Occupied Areas Outside The Protected Area

Sheet 1 of 1

Locations	Type of Occupant	Area Checked
New Training building	Site employees /Janitorial Services	☐
Old Training building	Security/Contractors/ Janitorial Services	☐
Chlorine Addition Building	Site Employees	☐
Warehouses (Total of 3)	Site Employees	☐
Warehouse /Mtce Bldg.	Site employees/Streator Garage Employees	☐
Switchyard	ComEd Employees	☐
Parking lot	Contractors	☐
Security checkpoint	Security	☐
Environs sampling locations	Environs monitoring contractor	☐
Various	Delivery people	☐
Met Tower	M&T, Site personnel	☐

Attachment 10

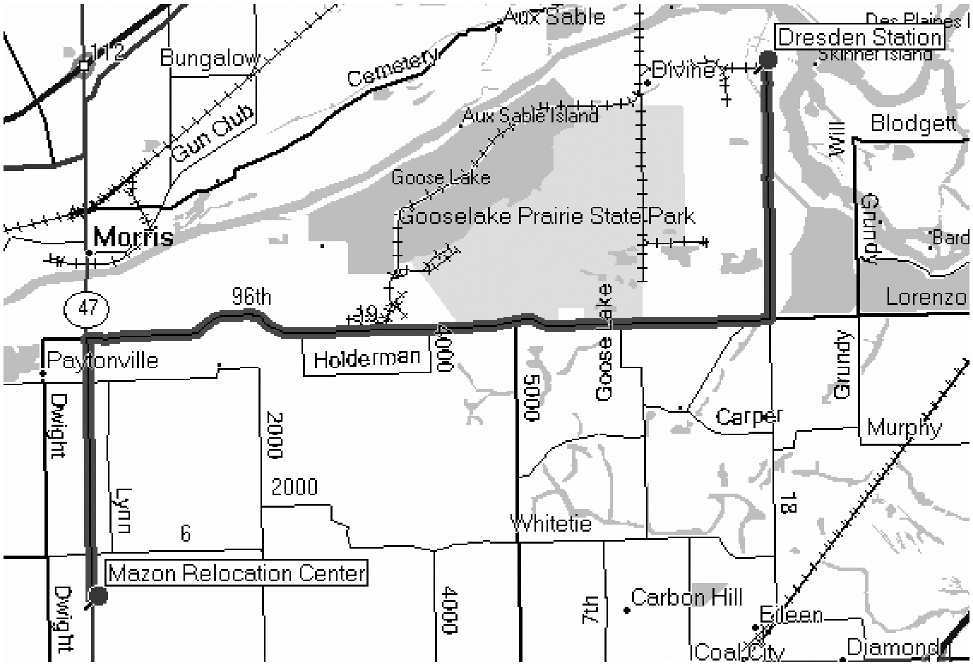
EP-AA-113-F-19

"Dresden Assembly, Accountability and Evacuation Guidelines"

Revision C

DRESDEN ASSEMBLY, ACCOUNTABILITY AND EVACUATION GUIDELINES

Evacuation Routes – Mazon Relocation Center



Travel Directions from Dresden Station to the Mazon Relocation Center

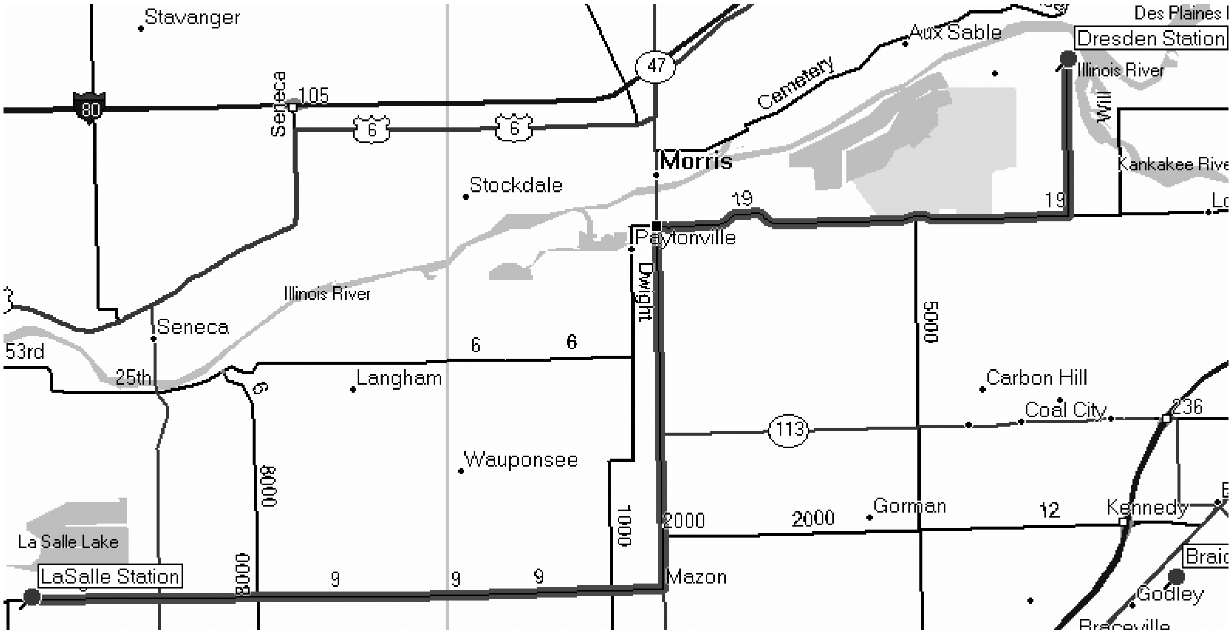
1. Take Dresden Road South to Pine Bluff Road (~3 miles)
2. Turn Right on Pine Bluff Road to IL-47 (~8 miles)
3. Turn Left on IL-47 to Mazon Relocation Center (~3 miles)

When driving to the Relocation Center:

- Keep windows up
- Turn off air-conditioning/heater or place on re-circulation mode
- Leave keys in the car upon arrival at the Relocation Center
- Ensure Contamination is not spread
- Upon arrival follow Security/RP directions

DRESDEN ASSEMBLY, ACCOUNTABILITY AND EVACUATION GUIDELINES

Evacuation Routes – LaSalle Station



Travel Directions from Dresden Station to LaSalle Station

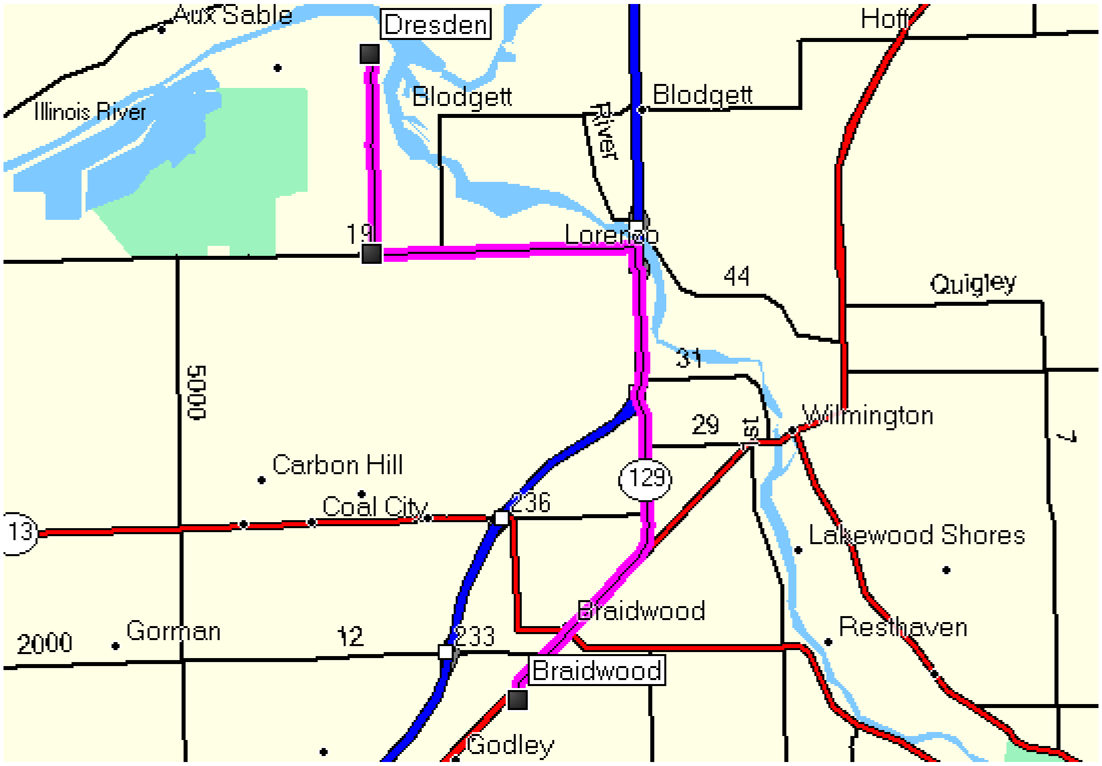
1. Take Dresden Road South to Pine Bluff Road (~3 miles)
2. Turn Right on Pine Bluff Road to IL-47 (~8 miles)
3. Turn Left on IL-47 to Grand Ridge Road (~7 miles)
4. Turn Right on Grand Ridge Road to LaSalle Station (~12.5 miles)

When driving to the Relocation Center:

- Keep windows up
- Turn off air-conditioning/heater or place on re-circulation mode
- Leave keys in the car upon arrival at the Relocation Center
- Ensure Contamination is not spread
- Upon arrival follow Security/RP directions

DRESDEN ASSEMBLY, ACCOUNTABILITY AND EVACUATION GUIDELINES

Evacuation Routes – Braidwood Station



Travel Directions from Dresden Station to Braidwood Station

1. Take Dresden Road South to Pine Bluff Road (~3 miles)
2. Turn Left on Pine Bluff Road to I-55 South (~4 miles)
3. Take I-55 South and Exit on Hwy 129 (~4.5 miles)
4. Take Hwy 129 to Division St. (~5 miles)
5. Turn Left on Division St to Braidwood Station

When driving to the Relocation Center:

- Keep windows up
- Turn off air-conditioning/heater or place on re-circulation mode
- Leave keys in the car upon arrival at the Relocation Center
- Ensure Contamination is not spread
- Upon arrival follow Security/RP directions

DRESDEN ASSEMBLY, ACCOUNTABILITY AND EVACUATION GUIDELINES

Potentially Occupied Areas Outside the Protected Area

Locations	Type of Occupant	Area Checked
Training building	Site Employees / Food Service Workers	☐
Training annex	Site Employees	☐
Maintenance garage	Site Employees	☐
Parking lot	Contractors	☐
Cooling towers	Contractors	☐
Sewage treatment	Site Employees	☐
345 switchyard	Substation Employees	☐
138 switchyard	Substation Employees	☐
Security checkpoint	Security	☐
Various	Security	☐
Various	Delivery People	☐
Various	Environs Monitoring Contractor	☐
County Line Road Fishing Area	General Public	☐