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10CFR50, Appendix E

5928-03-20230

November 17, 2003

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

Subject: Peach Bottom Atomic Power Station, Units 2 & 3
Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

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

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

EP-AA-1007, Revision 9, "Exelon Nuclear Radiological Plan Annex For Peach Bottom Atomic Power Station"
EP-AA-1008, Revision 5, "Exelon Nuclear Radiological Plan Annex For Limerick Generating Station"
EP-AA-110-301, Revision 2, "Core Damage Assessment"
EP-MA-113-100, Revision 2, "Assembly and Site Evacuation"

Enclosed are revised Emergency Plan Procedures for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3; Limerick Generating Station (LGS), Units 1 and 2; and Three Mile Island, (TMI) Unit 1. These procedures are required to be submitted within thirty (30) days of their revision in accordance with 10CFR50, Appendix E, and 10CFR50.4.

A045

Also, enclosed are copies of a computer generated report index identifying the latest revisions of the LGS, PBAPS, and TMI procedures.

If you have any questions or require additional information, please do not hesitate to contact us.

Very truly yours,

Michael P. Gallagher

M. P. Gallagher
Director - Licensing & Regulatory Affairs
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Enclosures

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ENCLOSURE 1

LIMERICK GENERATING STATION, UNITS 1 & 2
PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 & 3
THREE MILE ISLAND, UNIT 1

Docket Nos. 50-352
50-353
50-277
50-278
50-289

License Nos. NPF-39
NPF-85
DPR-44
DPR-56
DPR-50

EMERGENCY RESPONSE PROCEDURES

EP-AA-1007, "Exelon Nuclear Radiological Plan Annex for Peach Bottom Atomic Power Station," Revision 9
EP-AA-1008, "Exelon Nuclear Radiological Plan Annex for Limerick Generating Station," Revision 5
EP-AA-110-301, "Core Damage Assessment," Revision 2
EP-MA-113-100, "Assembly and Site Evacuation," Revision 2

Exelon Nuclear

EP-AA-1007
Revision 9

EXELON NUCLEAR

RADIOLOGICAL EMERGENCY PLAN ANNEX FOR PEACH BOTTOM ATOMIC POWER STATION

Submitted: ______________________ Date: 9/9/03
Mid-Atlantic Emergency Preparedness Manager

Authorized: ______________________ Date: 9/9/03
Corporate Functional Area Manager

Authorized: ______________________ Date: 9/26/03
Vice President, Licensing and Regulatory Affairs

Table of Contents

Section **Page**

Section 1: Introduction

1.1 Facility Description .... PBAPS 1-1
1.2 Emergency Planning Zones .... PBAPS 1-1
1.3 Participating Governmental Agencies .... PBAPS 1-2
Table PBAPS 1-1: Offsite Radiological Emergency Response Organizations and Response Plans .... PBAPS 1-5
Figure PBAPS 1-1 10-Mile Plume Exposure Pathway .... PBAPS 1-7
Figure PBAPS 1-2 50-Mile Ingestion Pathway .... PBAPS 1-8

Section 2: Organizational Control of Emergencies

2.1 Shift Organization Staffing .... PBAPS 2-1
2.2 Emergency Response Organization (ERO) Staffing .... PBAPS 2-2
2.3 Emergency Response Organization (ERO) Training .... PBAPS 2-3
2.4 Non-Exelon Nuclear Support Groups .... PBAPS 2-3
2.5 Nuclear Steam Systems Supplier (NSSS) .... PBAPS 2-3
2.6 Architect/Engineer .... PBAPS 2-3
Table PBAPS 2-1 Minimum Staffing Requirements .... PBAPS 2-4
Figure PBAPS 2-1 Exelon Overall ERO Command Structure .... PBAPS 2-7
Figure PBAPS 2-2 Emergency Onsite Organization .... PBAPS 2-8
Figure PBAPS 2-3 Emergency Offsite Organization .... PBAPS 2-9
Figure PBAPS 2-4 Emergency Public Information Organization .... PBAPS 2-10

Section 3: Classification of Emergencies

3.1 Emergency Action Levels (EALs) .... PBAPS 3-1
3.2 EAL Technical Basis .... PBAPS 3-3
3.3 General EAL Implementation Philosophy .... PBAPS 3-4
Table PBAPS 3-1 EAL Matrix .... PBAPS 3-5
Table PBAPS 3-2 EAL Technical Basis .... PBAPS 3-5

Section 4: Emergency Measures

4.1 Notification of the Emergency Organization .... PBAPS 4-1
4.2 Assessment Actions .... PBAPS 4-2
4.3 Protective Actions for the Offsite Public .... PBAPS 4-2
4.4 Protective Actions for Onsite Personnel .... PBAPS 4-6
4.5 Severe Accident Management (SAM) .... PBAPS 4-7
Figure PBAPS 4-1 Plant-Based PAR Determination Flowchart .... PBAPS 4-9
Figure PBAPS 4-2 Off-Site Assembly Locations .... PBAPS 4-10

Table of Contents

Section | Page

Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities....................PBAPS 5-1
5.2 Assessment Resources....................PBAPS 5-2
5.3 Protective Facilities and Equipment....................PBAPS 5-6
5.4 First Aid and Medical Facilities....................PBAPS 5-6
5.5 Communications....................PBAPS 5-7
5.6 Independent Spent Fuel Storage (ISFS)....................PBAPS 5-11
Table PBAPS 5-1 Emergency Supplies and Equipment....................PBAPS 5-12
Figure PBAPS 5-1 Emergency Radio Links....................PBAPS 5-14

APPENDICES

Appendix 1: NUREG-0654 Cross-Reference
Appendix 2: Site-Specific Letters of Agreement

Table of Contents

Section **Page**

REVISION HISTORY

REVISION	REVISION DATE	EFFECTIVE DATE
0	August 2002	August 30, 2002
1	September 2002	September 2002
2	November 2002	November 2002
3	January 2003	January 2003
4	February 2003	February 20, 2003
5	February 2003	March 2003
6	April 2003	May 9, 2003
7	May 2003	June 30, 2003
8	September 2003	September 18, 2003
9	October 2003	

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 Radiological Emergency Plan, Station Annexes, emergency plan implementing procedures, and associated program administrative documents. The Exelon Nuclear Standardized Radiological 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 Peach Bottom Atomic Power 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 Standardized Radiological Emergency Plan per EP-AA-120, "Emergency Plan Administration".

1.1 Facility Description

The Peach Bottom Atomic Power Station (PBAPS) is a fixed nuclear facility operated by Exelon Nuclear. The station consists of one High Temperature Gas Cooled Reactor designated as Unit 1, which is in the SAFSTOR status of decommissioning, two operating Boiling Water Reactors designated as Units 2 and 3, and an Independent Spent Fuel Storage Installation (ISFSI).

The PBAPS station is located partly in York County and partly in Lancaster County in southeastern Pennsylvania, on the west shore of Conowingo Pond, near the mouth of Rock Run Creek. The plant is about 38 miles NNE of Baltimore, MD; 65 miles WSW of Philadelphia, PA; 45 miles SE of Harrisburg, PA; and 20 miles SSE of Lancaster, PA. Conowingo Pond is a reservoir formed by the backwater of Conowingo Dam on the Susquehanna River; the dam is located about 9 miles downstream from PBAPS. The nearest communities are Delta, PA, and Cardiff, MD, located approximately 4 and 6 miles WSW of the site, respectively.

For more specific site location information, refer to the Updated Final Safety Analysis Report (UFSAR) for Peach Bottom Atomic Power Station.

1.2 Emergency Planning Zones

The Plume Exposure Emergency Planning Zone (EPZ) for Peach Bottom Atomic Power Station shall be an area surrounding the Station with a radius of about ten miles. The exact physical boundaries are determined by the Commonwealth of Pennsylvania, State of Maryland, and affected Counties). Refer to Figure PBAPS 1-1.

The Ingestion Pathway Emergency Planning Zone (EPZ) for Peach Bottom Atomic Power Station shall be an area surrounding the Station with a radius of about 50 miles. Refer to Figure PBAPS 1-2.

1.3 Participating Governmental Agencies

The overall responsibility for the management of the effects of accidental off-site releases of radioactivity resulting from either a nuclear power plant or a transportation accident rests with state and local governments.

The Commonwealth of Pennsylvania organizations having prime responsibility in matters of radiation hazards are the Pennsylvania Emergency Management Agency (PEMA) and the Bureau of Radiation Protection (BRP) of the Pennsylvania Department of Environmental Protection. State of Maryland organizations having primary responsibility in matters of radiation hazards are the Maryland Emergency Management Agency (MEMA) and the Technical Support Program of the Maryland Department of the Environment (MDE).

County and local governments are responsible for the protection of public health and safety within their jurisdiction. Similarly, organizations in the Commonwealth of Pennsylvania and States of Maryland, Delaware, and New Jersey are responsible for the protection of the public in their states. Cooperation with the States of Delaware and New Jersey is necessary because these states are within the Ingestion Pathway EPZ.

These civil agencies will respond to provide support in the event of an emergency in the areas indicated below.

1.3.1 Pennsylvania Emergency Management Agency (PEMA)

Responsibilities of PEMA are outlined in Annex E, "Radiological Emergency Response to Nuclear Power Plant Incidents" of the Commonwealth of Pennsylvania Emergency Operations Plan.

1.3.2 Department of Environmental Protection, Bureau of Radiation Protection (DEP/BRP)

Responsibilities of DEP/BRP are outlined in Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan.

1.3.3 Pennsylvania State Police

Responsibilities of the State Police are set forth in Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan.

1.3.4 Maryland Emergency Management Agency (MEMA)

MEMA responsibilities are outlined in Annex Q, "Fixed Nuclear Facility Radiological Emergency Plan".

1.3.5 Maryland Department of the Environment, Emergency Operations and Technical Support Program

Responsibilities of MDE Emergency Operations and Technical Support Program are outlined in Annex Q, "Fixed Nuclear Facility Radiological Emergency Plan".

1.3.6 Maryland State Police

Responsibilities of the State Police are set forth in Annex Q, "Fixed Nuclear Facility Radiological Emergency Plan".

1.3.7 State Of Delaware

The State of Delaware's border is located within the 50-mile Ingestion Pathway for PBAPS. The State would be notified if protective actions are required within that area. No direct support is provided to PBAPS.

1.3.8 State Of New Jersey

The State of New Jersey's border is located within the 50-mile Ingestion Pathway for PBAPS. The State would be notified if protective actions are required within that area. No direct support is provided to PBAPS.

1.3.9 County Governments

County government agencies have agreements regarding responsibilities for coping with emergencies. These agencies include three counties in Pennsylvania, York, Lancaster, and Chester; and two counties in Maryland, Cecil and Harford.

a. Pennsylvania Counties

Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan defines "risk counties" as those within a 10-mile radius of a fixed nuclear facility. For Peach Bottom, the risk counties are:

- York County
- Lancaster County
- Chester County

The responsibilities assigned to these counties are in Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan.

b. Maryland Counties

Harford and Cecil Counties in Maryland may potentially be affected by an incident at the Peach Bottom Atomic Power Station. Responsibilities assigned to these counties are outlined in Annex Q, " Fixed Nuclear Facility Radiological Emergency Plan".

Refer to Table PBAPS 1-1 for a list of offsite radiological emergency response organizations and response plans in support of the Peach Bottom Atomic Power Station's Emergency Preparedness Program.

Table PBAPS 1-1: Offsite Radiological Emergency Response Organizations and Response Plans

The following state, local and emergency plans are available and filed under separate cover.

- Annex E - "Radiological Emergency Response to Nuclear Power Plant Incidents" - to Commonwealth of Pennsylvania Emergency Operations Plan.

- Chester County Radiological Emergency Response Plan for Incidents at Peach Bottom Atomic Power Station:

  Municipality

  West Nottingham Township

  School District

  Oxford

- State of Maryland Disaster Assistance Plan, Annex Q, Radiological Emergency Plan.

- Lancaster County Emergency Operations Plan, Annex E, Part 2 - PBAPS

  Municipalities

Fulton Township	Drumore Township
Martic Township	Quarryville Borough
Little Britain Township	Providence Township
East Drumore Township	

  School District

Solanco	Penn Manor

Table PBAPS 1-1: Offsite Radiological Emergency Response Organizations and Response Plans (Cont'd)

- York County Emergency Operations Plan, Annex E, Part 2 - PBAPS

Municipalities

Lower Chanceford Township	Fawn Grove Township
Fawn Borough	Delta Borough
Peach Bottom Township	

School Districts

Red Lion

South Eastern

- Harford County Emergency Operations Plan - PBAPS

School District

Harford County

- Cecil County Emergency Operations Plan - PBAPS

School

Conowingo Elementary

- State of Delaware Emergency Plan

- State of New Jersey Emergency Plan

Figure PBAPS 1-1: 10-Mile Plume Exposure Pathway EPZ

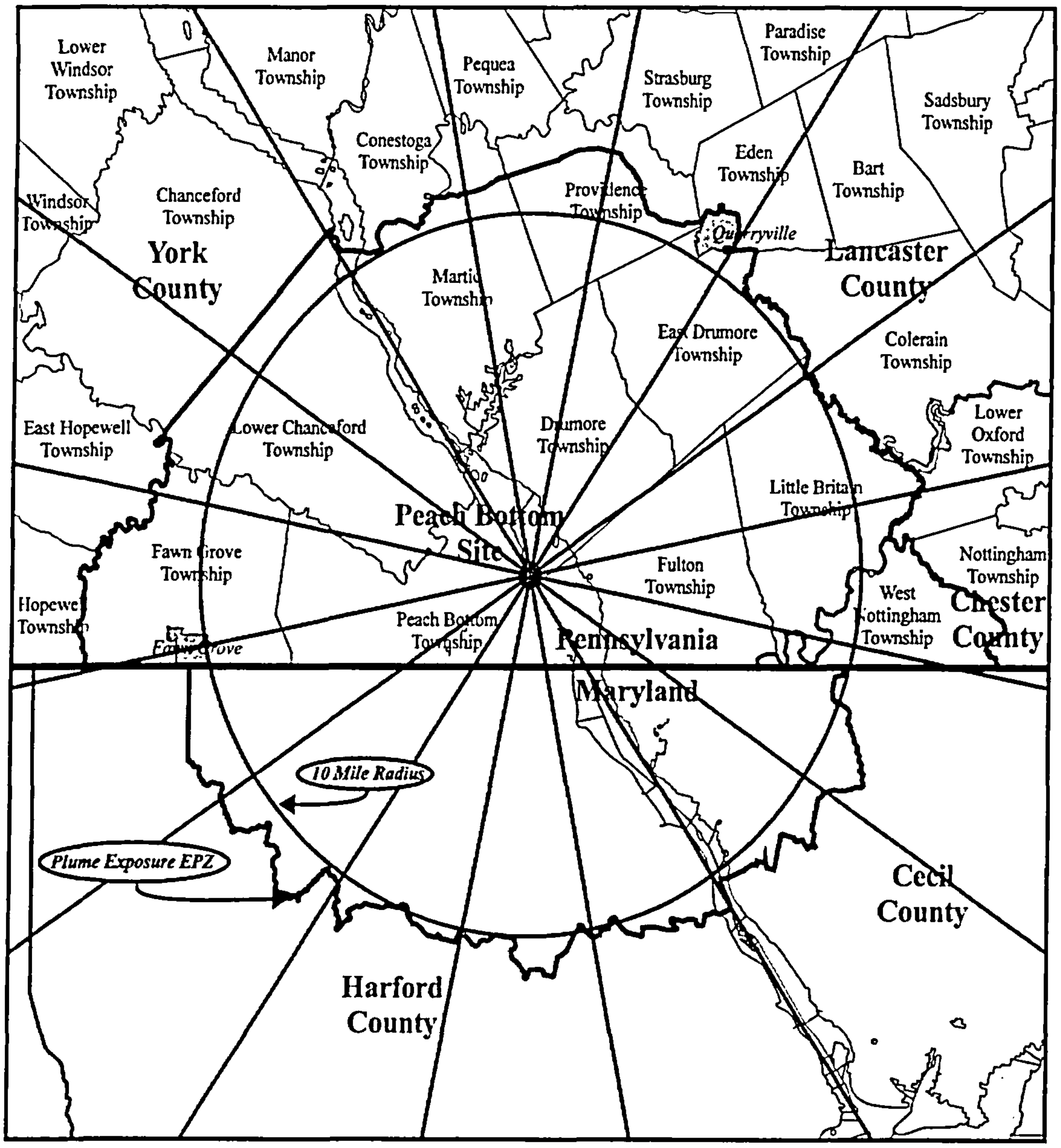
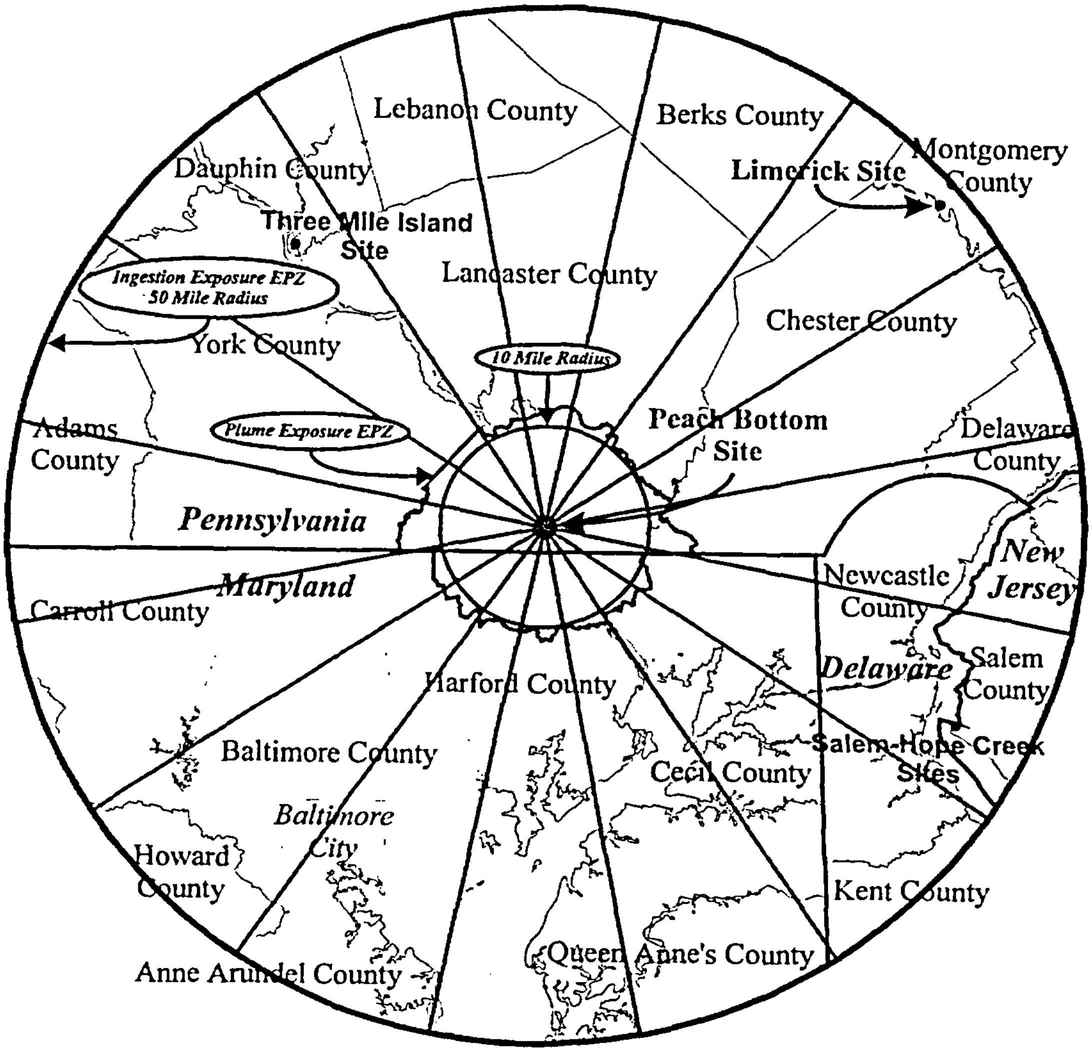


Figure PBAPS 1-2: 50-Mile Ingestion Pathway EPZ



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 PBAPS UFSAR Figure 13.2.2. Once an emergency is declared, the Emergency Response Organization (ERO) is activated as described in Section B.4 of the Exelon Nuclear Standardized Radiological Emergency Plan.

2.1 Shift Organization Staffing

Required on-shift staffing in support of emergency response activities is 16 people. This on-shift staffing level exceeds the Exelon Nuclear Standardized Radiological Emergency Plan commitment of ten (10) people, based on existing commitments supporting the elimination of 30-minute augmentation goal from Table B-1 of NUREG-0654/FEMA-REP-1.

A listing of minimum shift complement is provided in Table PBAPS 2-1 of the Annex for Peach Bottom Atomic Power Station (PBAPS). Based on existing on-shift staffing commitments, the "Minimum Shift Size" for the purposes of NUREG-0654, Table B-1 comparison is 16 persons versus the 10 persons specified in the Exelon Nuclear Standardized Radiological Emergency Plan. These six (6) additional on-shift positions include:

- 2nd Emergency Communicator
- Two Field Survey Team Members
- One Radwaste Operator (Equipment Operator)
- One Instrument and Controls Technician
- 3rd Radiation Protection Technician (two at affected station and 3rd at unaffected station performing dose assessor function).

2.1.1 Shift Dose Assessment

The on-shift dose assessment function will be performed by a shift Radiation Protection Technician (RPT) at Limerick Generating Station. However, Peach Bottom Atomic Power Station will maintain the capability to perform a shift dose assessment, if necessary.

2.1.2 Shift Emergency Communicators

The Shift Communicator performs notifications to the State and County organizations until relieved by the TSC, and assists in the initiation of the ERO Callout System as directed. The Communicator position is staffed by a designated on-shift individual capable of responding to the Control Room immediately in support of the initiation of offsite notifications within 15 minutes of event classification.

A 2nd on-shift individual will be designated to support communications with the NRC over the Emergency Notification System (ENS) until relieved by the TSC.

Activation of the automated ERO call out system will be performed by the Shift Manager, but may be delegated to a Control Room Emergency Communicator or available on-shift staff.

2.1.3 Shift Technical Advisor (STA) / Incident Assessor

Section B.1 of the Exelon Nuclear Standardized Radiological Emergency Plan outlines the On-Shift Emergency Response Organization Assignment of the STA. Peach Bottom Atomic Power Station has deemed the following as an acceptable method of implementing Section B.1 in reference to the STA.

The responsibilities of the STA are delineated on OP-AA-101-111, "Roles and Responsibilities of On-Shift Personnel." If the STA is the Shift Manager or Unit Supervisor, then another Senior Reactor Operator (SRO) shall assist as Incident Assessor during unexpected conditions and transients. Per Table B-1, the on-shift STA or Incident Assessor shall also provide core/thermal hydraulics support to Control Room staff.

2.2 Emergency Response Organization (ERO) Staffing

Refer to Table PBAPS 2-1 of the PBAPS Annex, "Minimum Staffing Requirements", for a comparison against the Exelon Nuclear Standardized Radiological Emergency Plan of 60-minute and full augmentation commitments.

2.2.1 Emergency Onsite Organization (Figure PBAPS 2-2)

No changes in augmentation positions or staffing levels for the Technical Support Center (TSC), Operations Support Center (OSC) and Control Room from that specified in the Exelon Nuclear Standardized Radiological Emergency Plan.

2.2.2 Emergency Offsite Organization (Figure PBAPS 2-3)

Based on existing interface and staffing agreements, representatives from the Commonwealth of Pennsylvania and State of Maryland will respond to the Emergency Operations Facility (EOF), allowing direct face-to-face communications. As such, the State Environs Communicator position, listed under the Exelon Nuclear Standardized Radiological Emergency Plan, is not staffed at the Coatesville EOF. Rather the EOF Environmental Coordinator will interface directly with State representatives present in the EOF.

An EOF Access Controller has been added to the Full Augmentation complement to support existing facility access control measures.

2.2.3 Emergency Public Information Organization (Figure PBAPS 2-4)

Based on the co-location of the EOF with the Joint Public Information Center (JPIC) the following Emergency News Center (ENC) functions, as described in Sections B.5.c and B.7 of the Exelon Nuclear Standardized Radiological Emergency Plan, have been eliminated or consolidated with corresponding JPIC positions. These differences in staffing are:

- Public Information Liaison was deleted.
- Radiation Protection Spokesperson was incorporated into the Radiological Advisor position
- Technical Spokesperson was incorporated into the Technical Advisor position

2.3 Emergency Response Organization (ERO) Training

Training is conducted in accordance with Section O.5 of the Exelon Nuclear Standardized Radiological Emergency Plan per TQ-AA-113, "ERO Training and Qualification." Retraining is performed on an annual basis, which is defined as every 12 months ± 3 months (25% grace period).

2.4 Non-Exelon Nuclear Support Groups

- Agreements exist on file with or are verified current annually by the MAROG Corporate Emergency Preparedness Group for the following support agencies listed in Appendix 2 of the Exelon Nuclear Radiological Emergency Plan Annex for PBAPS.

Additionally, Exelon Nuclear has contractual agreements common within Exelon Nuclear with several companies whose services would be available in the event of a radiological emergency. These agencies are listed in Appendix 3 of the Exelon Nuclear Standardized Radiological Emergency Plan. Emergency response coordination with governmental agencies and other support organizations is discussed in Section A of the Standard Plan.

2.5 Nuclear Steam Systems Supplier (NSSS)

General Electric Company maintains an Emergency Response Organization, which can provide technical assistance from their home office or at the site.

2.6 Architect/Engineer

Bechtel or other contractors may be involved in the technical analysis or construction activities associated with the emergency response or recovery operation. Each such organization will designate a lead representative who will have the same responsibilities, within their scope of work, as described for the NSSS Contractor.

Table PBAPS 2-1: Minimum Staffing Requirements

Functional Area	Major Tasks	Emergency Positions		[b]Minimum Shift Size	[a]60 Minute Augmentation	Full Augmentation
1. Plant Operations and Assessment of Operational Aspects	Control Room Staff	Shift Manager		1		
		Control Room Supervisor		1		
		Reactor Operator		2		
		Equipment Operator		2		
2. Emergency Direction and Control[c]	Command and Control / Emergency Operations	Shift Emergency Director	(CR)	1[f]		
		Station Emergency Director	(TSC)		1	
		Corporate Emergency Director	(EOF)		1	
3. Notification & Communication	Emergency Communications	Shift Personnel[d]		2		
		TSC Director	(TSC)		1	
		EOF Director	(EOF)		1	
		State/Local Communicator			1 (EOF)	1 (TSC)
		ENS Communicator			1 (TSC)	1 (EOF)
		HPN Communicator			1 (EOF)	1 (TSC)
	Plant Status	Operations Communicator	(CR/TSC)			2
	In-Plant Team Control	Damage Control Comm.	(CR/TSC/OSC)			3
	Technical Activities	Technical Communicator	(TSC)			1
	Governmental	EOC Communicator	(EOF)			1
		State EOC Liaison[h]	(PEMA/MEMA)			2
		Regulatory Liaison	(EOF)			1
4. Radiological Accident Assessment and Support of Operational Accident Assessment	Offsite Dose Assessment	Radiation Protection Personnel[e]		1		
		Dose Assessment Coordinator	(EOF)		1	
		Dose Assessor	(EOF)			1
		Radiation Controls Coordinator	(TSC)			1
	Offsite Surveys	Environmental Coordinator	(EOF)		1	
		Field Team Communicator	(EOF)			1
		Off-Site Field Team Personnel [m]		2	2	(g)
	Onsite Surveys	RP Personnel			2	(g)
	In-plant Surveys	RP Technicians		1	2	(g)
	Chemistry	Chemistry Personnel[n]		1	1	(g)
	RP Supervisory	Radiation Protection Manager	(TSC/EOF)		2	

Table PBAPS 2-1: Minimum Staffing Requirements (Cont'd)

Functional Area	Major Tasks	Emergency Positions		[b]Minimum Shift Size	[a]60 Minute Augmentation	Full Augmentation
5. Plant System Engineering, Repair and Corrective Actions	Technical Support	STA / Incident Assessor[p]	(CR)	1		
		Technical Manager	(TSC)		1	
		Core/Thermal Hydraulics Engineer	(TSC)	1[f]	1	
		Mechanical Engineer	(TSC)		1	
		Electrical Engineer	(TSC)		1	
		SAMG Decision Maker	(TSC)		1[f]	
		SAMG Evaluator	(TSC)		2[f]	
		Operations Manager	(TSC)		1	
		Radiation Controls Engineer	(TSC)			1
	Repair and Corrective Actions	Mechanical Maintenance/[n]	(OSC)	1[f]	2	(g)
		Rad Waste Operator		1		(g)
		Electrical Maintenance/[n]	(OSC)	1[f]	2	(g)
		Instrument & Control (I&C)	(OSC)	1		
		Maintenance Manager	(TSC)		1	
		OSC Director	(OSC)		1	
		Assistant OSC Director	(OSC)			1
		OPs Lead & Support Personnel	(OSC)			(g)
	Accident Analysis	Technical Support Manager	(EOF)			1
		Operations Advisor	(EOF)			1
		Technical Advisor	(EOF)			1
6. In-Plant Protective Actions	Radiation Protection	RP Personnel[e]		3[f]	4	(g)
7. Fire Fighting	--	Fire Brigade		(i)		
8. First Aid and Rescue Operations	--	Plant Personnel		2[f]		(g)
9. Site Access Control and Personnel Accountability	Security & Accountability	Security Team Personnel		(k)	(k)	
		Security Coordinator[q]	(TSC/Cantera EOF)			2
	EOF Security	Access Controller	(EOF)			1
10. Resource Allocation and Administration	Logistics / Administration	Logistics Manager	(EOF)		1	
		Logistics Coordinator	(TSC)			1
		Administrative Coordinator	(EOF)			1
		Clerical Staff	(TSC/OSC/EOF)			(g)
		Events Recorder	(EOF)			1
		Computer Specialist	(EOF)			1
		SUB-TOTAL:		**16**	**34**	**28+**

Table PBAPS 2-1: Minimum Staffing Requirements (Cont'd)

Functional Area	Major Tasks	Emergency Positions	[b]Minimum Shift Size	[a]60 Minute Augmentation	Full Augmentation
11. Public Information	Media Interface	Corporate Spokesperson (JPIC)		1	
		Rad Protection Spokesperson/Advisor (JPIC)			1
		Technical Spokesperson/Advisor (JPIC)			1
	Information Development	Public Information Director (JPIC)		1	
		News Writer (JPIC)			1
	Media Monitoring and Rumor Control	Communications Department (JPIC)			(g)
	Facility Operation and Control	JPIC Director (JPIC)		1	
		JPIC Coordinator (JPIC)			1
		Administrative Coordinator (JPIC)			1
		Events Recorder (EOF)			1
		Clerical Staff (JPIC)			(g)
		Access Controls (JPIC)			1
		SUB-TOTAL:	0	3[l]	7+

	[b]Minimum Shift Size	Total On-Call[l] Minimum Staff	Total Full Augmentation
TOTAL:	16	37	35+

(a) Response time is based on optimum travel conditions.

(b) For each unaffected nuclear unit in operation, maintain at least one Control Room Supervisor, one Reactor Operator, and one Equipment Operator, except that units sharing a Control Room may share a Control Room Supervisor if all functions are covered.

(c) Overall direction of facility response to be assumed by the Corporate Emergency Director (EOF) when all centers are fully manned. Direction of minute-to-minute facility operations and "non-delegable" responsibilities for event classification and emergency exposure controls remain with the Station Emergency Director (TSC). The Shift Manager, as Shift Emergency Director, shall function as acting Station Emergency Director prior to TSC activation.

(d) Refer to Section 2.1.2 for a description of shift emergency communication staffing.

(e) Refer to Section 2.1.1 for description of on-shift dose assessment staffing.

(f) May be provided by personnel assigned other functions. Personnel can fulfill multiple functions.

(g) Personnel numbers depend on the type and extent of the emergency.

(h) Staffing of the County EOC Liaison position is not required based on agreements with offsite agencies; however, every effort will be made to dispatch an Exelon Nuclear representative upon request from County EOC Director.

(i) Fire Brigade per UFSAR / TRM, as applicable.

(k) Per Security Plan.

(l) The following Emergency Public Information Organization personnel will be designated "minimum staffing" (on-call) positions but are not subject to the 60-minute response time requirement: Corporate Spokesperson, Public Information Director and JPIC Director.

(m) Each Field Survey Team consists of a Lead and Driver. Primarily the Lead will be an RP Technician (RPT); however, additional personnel qualified as RadWorker and trained in radiological exposure, ALARA principles and contamination control measures, may also serve as Lead to provide for long-term staff relief.

(n) OSC Group Leads can be used initially to fill 60-minute augmentation technical/craft positions in Maintenance, RP and Chemistry.

(p) Refer to Section 2.1.3 for description of on-shift STA/Incident Assessor staffing requirements.

(q) TSC Security Coordinator position will be staffed by PBAPS Security personnel. The EOF Security Coordinator position will be staffed by Corporate Security personnel at the Mid-West ROG Cantera Offices and will be contacted as part of the TSC activation process.

Figure PBAPS 2-1: Exelon Overall ERO Command Structure

Bolded Boxes indicate minimum staffing positions.

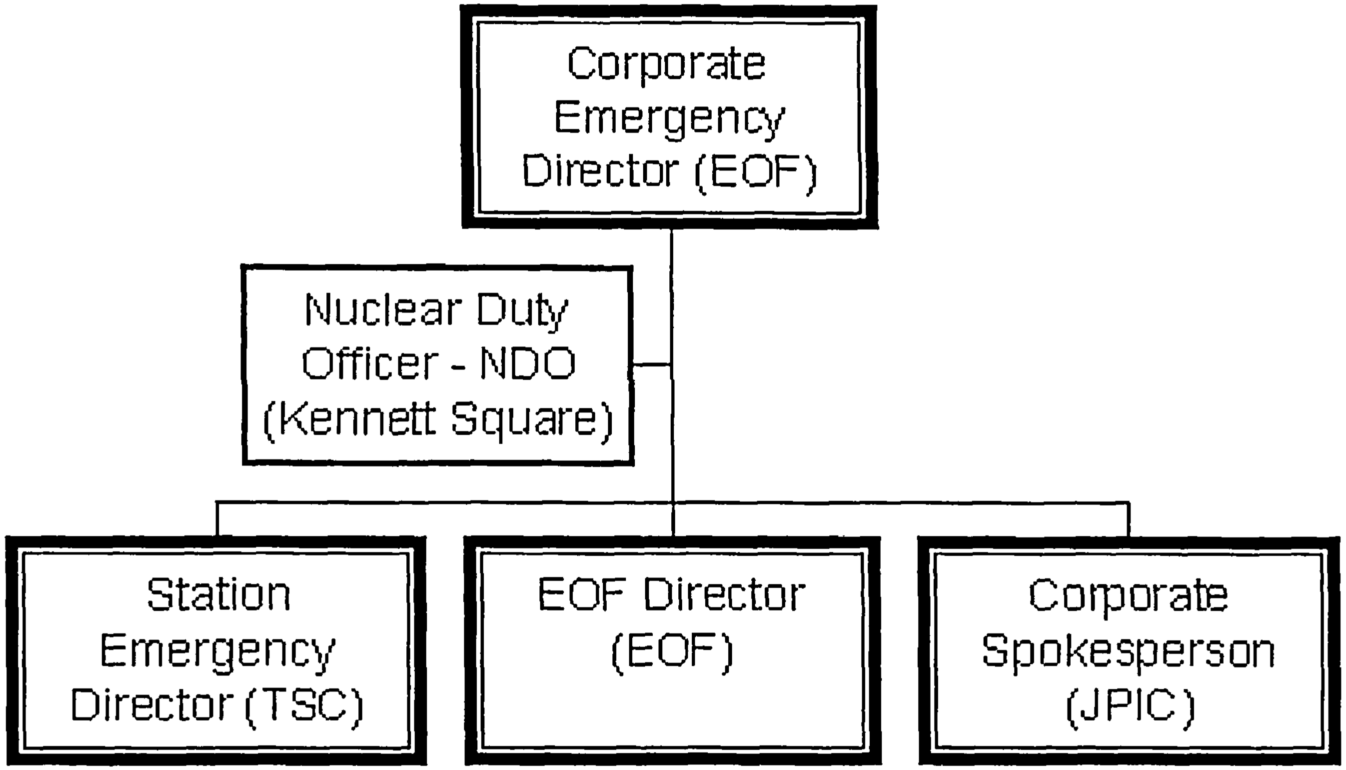
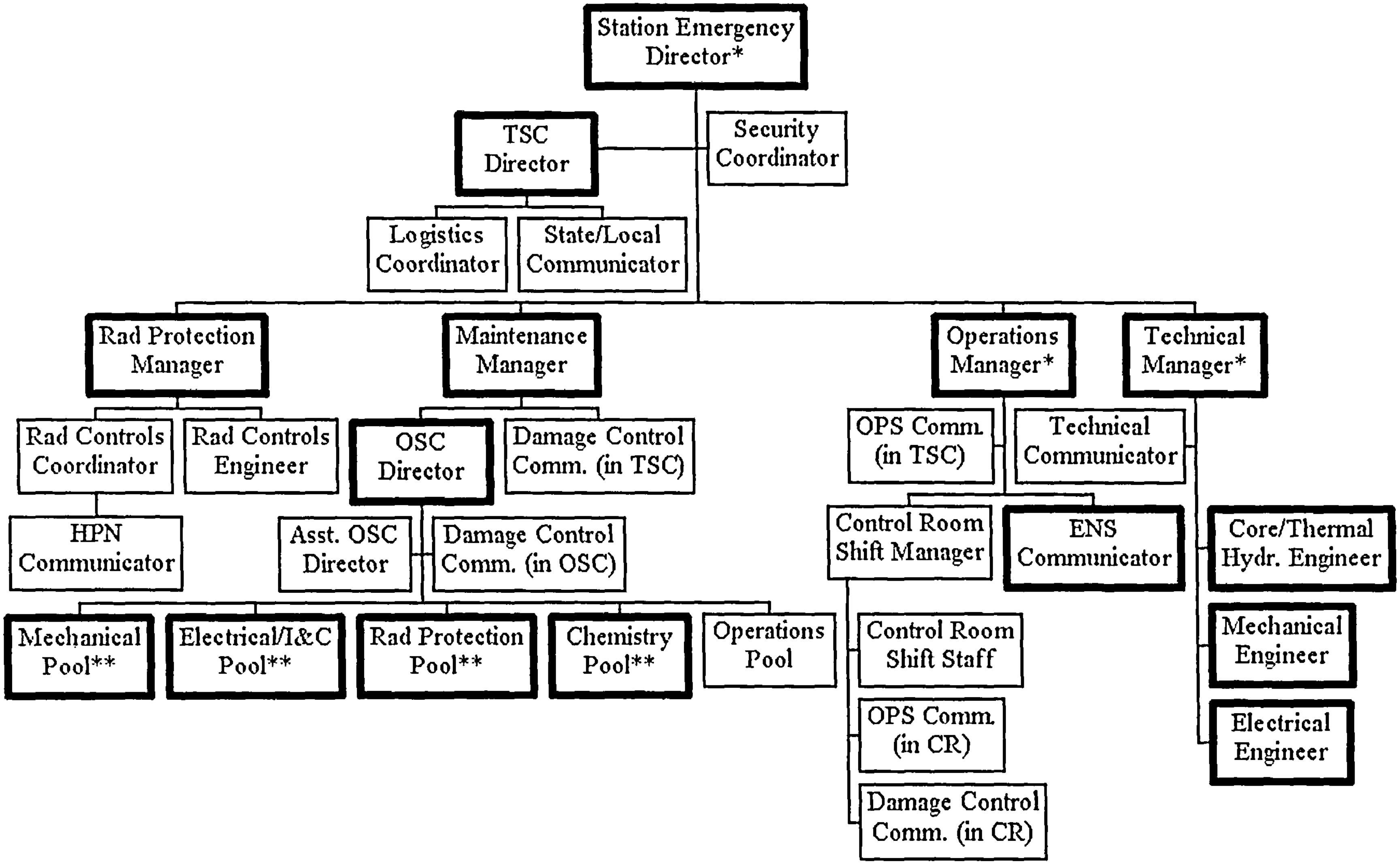


Figure PBAPS 2-2: Emergency Onsite Organization

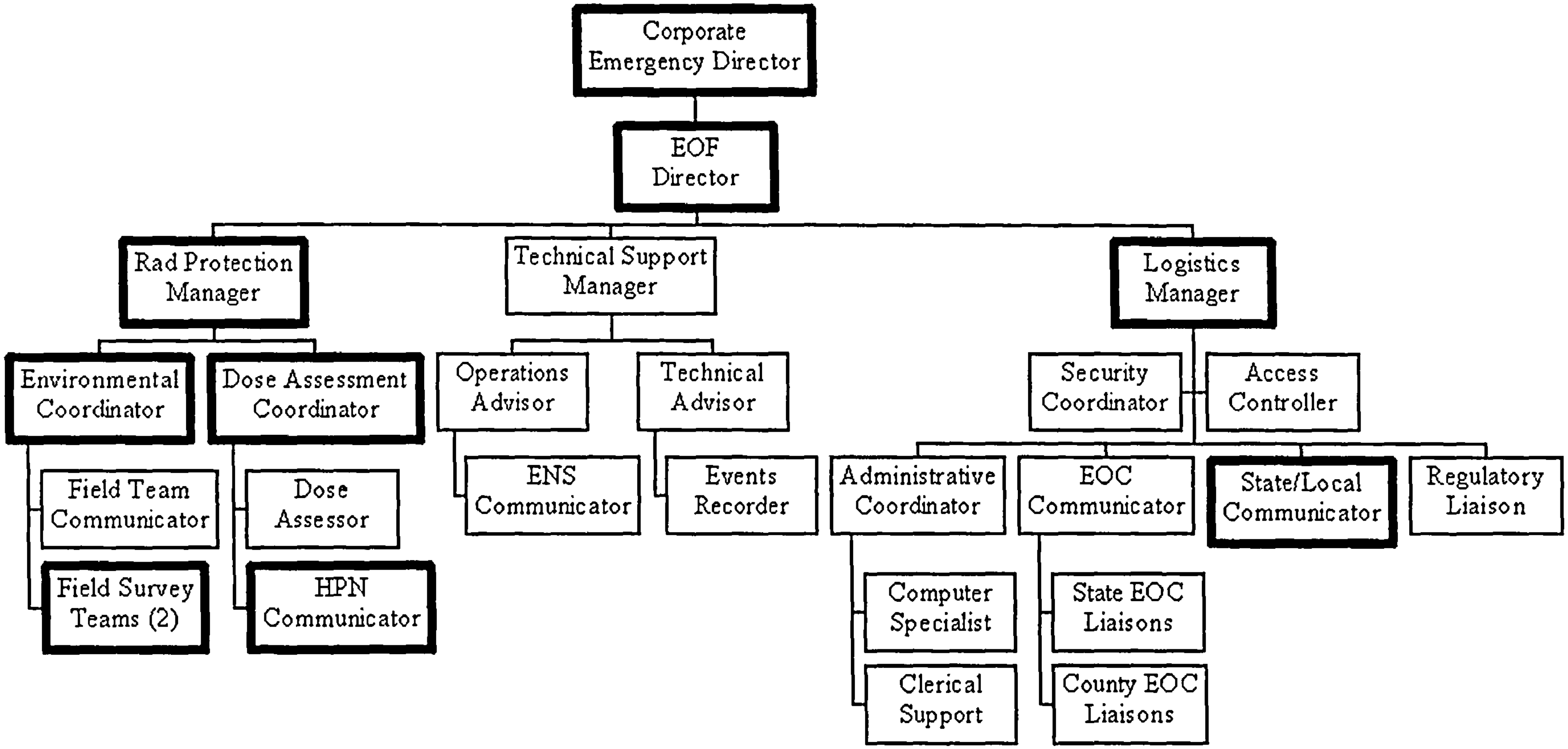


Bolded Boxes indicate minimum staffing positions.

* SAMG functions may be assigned to other qualified personnel. Minimum staffing requires 1 Decision Maker and 2 Evaluators.

** Refer to Table B-1 for required staffing levels

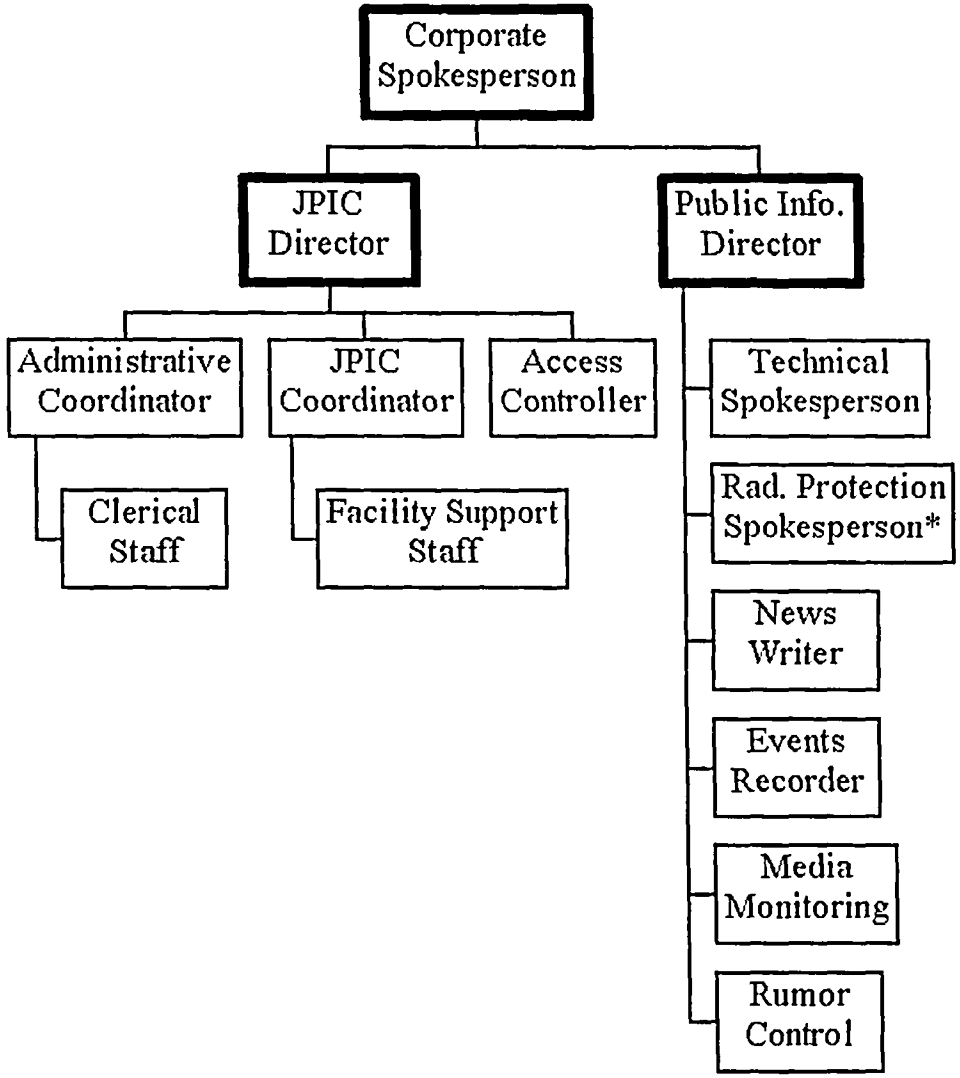
Figure PBAPS 2-3: Emergency Offsite Organization



Bolded Boxes indicate minimum staffing positions.

* EOF Security Coordinator position staffed by Corporate Security from MWROG Cantera Offices

Figure PBAPS 2-4: Emergency Public Information Organization



Bolded Boxes indicate minimum staffing positions.

* Radiation Protection Spokesperson / Advisor may be staffed by a qualified consultant.

Section 3: Classification of Emergencies

Section D of the Exelon Nuclear Standardized Radiological Emergency Plan describes five (5) Emergency Classes. The first four are the Unusual Event, Alert, Site Area Emergency and General Emergency, and are listed from least severe to most severe according to relative threat to the health and safety of the public and emergency workers. The fifth level is Recovery and is considered as a phase of the emergency. Recovery is not considered as part of the event classification logic contained in Section 3.0 of the Annex, but rather is entered by meeting criteria provided in Section M of the Exelon Nuclear Standardized Radiological Emergency Plan.

Site specific definitions are provided for terms to be used for that particular Initiating Conditions /Threshold Values and may not be applicable to other uses of that term in any other EAL, at other sites, in the Exelon Nuclear Standardized Radiological Emergency Plan or procedures. Also included are the technical bases, which were used to develop the EAL.

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 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.

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.

3.1 Emergency Action Levels (EALs)

Emergency Action Levels are the measurable, observable detailed conditions that must be met in order to classify the event. Classification shall not be made without referencing, comparing and satisfying the threshold values specified in the Emergency Action Levels. Mode Applicability provides the unit conditions when the Emergency Action Levels represent a threat. The Basis provides definitions of terms, explanations and justification for including the Initiating Condition and Emergency Action Level. Definitions are provided for terms having specific meaning as they relate to this procedure.

Unusual Event, Alert, Site Area Emergency, and General Emergency classifications are entered by meeting designated Emergency Action Levels (EALs) Threshold Values. These values are based on the criteria established under Revision 2 to NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels" (dated January 1992), and are labeled based on the four Recognition Categories outlined in NUMARC/NESP-007:

- Abnormal Radiological Levels / Effluents
- Fission Product Barrier Degradation
- System Malfunctions
- Hazards and Other Conditions

EAL Threshold Values are sorted under common Initiating Conditions (ICs). These ICs can be Symptom- or Event-based, and applicable to all or only designated Operational Conditions / Modes OPCONs. The Initiating Conditions (IC) and associated EAL Threshold Values are summarized in the EAL Matrix (Table PBAPS 3-1) according to Recognition Categories.

To aid user in identifying applicable ICs, they are further sorted under the following Event Sub-Categories, and appropriate Mode designator provided:

- *Abnormal Radiological Levels / Effluents ("R")*
  - Radiological Effluents
  - Abnormal Radiation Levels
  - Coolant Activity
- *Fission Product Barrier Degradation ("F")*
  - Fuel Clad
  - Reactor Coolant System, referred to as "RCS"
  - Primary Containment, referred to as "Containment"
- *System Malfunctions ("M")*
  - Loss of AC Power
  - Loss of DC Power
  - Failure of Reactor Protection System
  - Decay Heat Removal
  - Loss of Annunciators
  - RCS Leakage / RPV Draindown
  - MSL Break (with Isolation)
  - Loss of Communications
  - Technical Specifications
  - Irradiated Fuel Accidents
- *Hazards and Other Conditions ("H")*
  - Security Events
  - Control Room Evacuation
  - Natural or Man-Made Events
  - Fire / Explosion
  - Toxic or Flammable Gases
  - Discretionary

An emergency is classified by assessing plant conditions and comparing abnormal conditions to ICs and Threshold Values for each EAL, based on the designated Operational Condition (MODE). Modes 1 through 5 are defined in the Technical Specifications for Units 2 and 3 based on Reactor Mode Switch Position specific plant conditions. "Defueled" Mode was established for classification purposes under NUMARC/NESP-007 to reflect conditions where all fuel has been removed from the Reactor Pressure Vessel.

MODE	TITLE
1	Power Operation
2	Start-up
3	Hot Shutdown
4	Cold Shutdown
5	Refueling
D	Defueled

The EAL 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.

An appropriate EAL numbering system is provided as a user aid. ICs are coded with a two letter and one number code. For example: **HA1**

The first letter is the Recognition Category designator. In this case, **H** stands for "Hazards and Other Conditions". The second letter is the Classification Level: "U" for Unusual Event, "A" for Alert, "S" for Site Area Emergency, and "G" for General Emergency. The number is a sequential number for that Recognition Category series. All Initiating Conditions, which are describing the severity of a common condition (series), will have the same number (e.g. HA1, HA2, etc.).

A Fission Product Barrier (FPB) Table is provided as a subset to the Recognition Category "F" (FPB Degradation) of the EAL Matrix. This table is used to determine the integrity of the Fuel Clad, RCS and Containment Barriers based on EAL Threshold values established in accordance with NUMARC/NESP-007 (e.g., Intact, LOSS, or POTENTIAL LOSS).

3.2 EAL Technical Basis

Table PBAPS 3-2 serves as the Technical Basis for the EAL Matrix. The table consists of the following sections for each Initiating Condition (IC), sorted by Recognition Category:

- Initiating Condition
- Threshold Value
- Mode Applicability
- Basis (includes deviations from NUMARC/NESP-007 as appropriate)

Table PBAPS 3-2 provides the EAL user with the background and justification behind the EAL Threshold Values identified using the guidance set forth in NUMARC/NESP-007.

For a radiological liquid release, the emergency action level is based on calculated off-site dose from a chemistry sample. Shift Supervision utilizes emergency response procedures to notify risk counties and to obtain river water samples.

3.3 General EAL Implementation Philosophy

A broad spectrum of discretion in classifying events is provided in the "Discretionary" category under Hazards and Other Conditions and the Fission Product Barrier Matrix in Table PBAPS 3-1. In using the "Discretionary" category and in classifying emergencies under circumstances which are not a straight-forward use of the EAL's, ERO members should be mindful than an approach is needed which is conservative with respect to public, plant, and personnel safety and with respect to ensuring the adequacy of personnel and technical support. Conservative decisions must be made if the ED has any doubt regarding the health and safety of the public.

Declaring an Unusual Event provides the Company and off-site agencies the opportunity for early information regarding the event and for early activation of resources and may be considered a "no consequence decision." Conversely, not declaring an Unusual Event when there are credible (but, not clear) bases for doing so, would appear to be less than open or candid and could have serious adverse consequences. Although the consequences of declaring an Unusual Event are limited, inappropriate classifications do not accurately indicate the significance of the event to the public and emergency responders and should be avoided.

At the Alert, Site Area and General Emergency levels, clearly the threat to the plant and to the public is at a heightened level. Rapid application of resources and preparation for providing for the public health and safety are appropriate. Because of the magnitude of resource mobilization and the potential disruption of normal public activities, an overly conservative or an inappropriately early declaration of these levels is not advisable.

Events that meet the Emergency Action Level criteria for event declaration, but which are terminated before they are identified and declared, should still be classified and reported, but not declared to implement the Emergency Plan.

All EALs may not consider trends, rates of change, or status changes in equipment availability. In the event of rapidly changing parameters trending toward an increased emergency classification, it may be appropriate to decide that the higher level EAL will be exceeded and escalate the classification early. In the event of trends toward a decreased emergency classification, parameter values must be below the EALs to de-escalate.

In the event of a "spike" which rapidly exceeds and then exits an EAL condition, entry into the Emergency Plan or escalation to the higher classification "in retrospect" is not appropriate unless the "spike" is indicative of continuing degrading conditions which will lead to an escalated emergency classification level. This statement does not apply if the EAL includes a "spike". Spurious alarms or parameters, which are known to be invalid indicators of actual plant conditions or of the emergency classification, should not be used to declare emergency classifications.

TABLE PBAPS 3-1: Emergency Action Level (EAL) Matrix

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
ABNORMAL RAD LEVELS / EFFLUENTS				
Radiological Effluents	**RG1** Actual or Projected Site Boundary Dose Using Actual Meteorology > 1000 mRem TEDE **OR** > 5000 mRem CDE Thyroid **MODES: ALL** EAL Threshold Value 1 Radiological release in excess of Table R1 "General Emergency" threshold **AND** Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "General Emergency" thresholds **OR** 2 Radiological releases exceed **ANY** Table R2 column "General Emergency" threshold	**RS1** Actual or Projected Site Boundary Using Dose Actual Meteorology > 100 mRem TEDE **OR** > 500 mRem CDE Thyroid **MODES: ALL** EAL Threshold Value 1 Radiological release in excess of Table R1 "Site Area Emergency" threshold **AND** Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "Site Area Emergency" thresholds **OR** 2 Radiological releases exceed **ANY** Table R2 column "Site Area Emergency" threshold	**RA1** Release > 200 X ODCM Limit for ≥ 15 minutes **MODES: ALL** EAL Threshold Value 1 Unplanned radiological release lasting ≥ 15 minutes in excess of Table R1 "Alert" threshold **AND** Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "Alert" thresholds **OR** 2 Unplanned radiological releases lasting ≥ 15 minutes in excess of **ANY** Table R2 column "Alert" threshold	**RU1** Release > 2 X ODCM Limit for ≥ 60 minutes **MODES: ALL** EAL Threshold Value 1 Unplanned radiological release lasting ≥ 60 minutes in excess of Table R1 "Unusual Event" threshold **AND** Releases **CANNOT** be determined in < 60 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "Unusual Event" thresholds **OR** 2 Unplanned radiological releases lasting ≥ 60 minutes in excess of **ANY** Table R2 column "Unusual Event" threshold
Abnormal Rad Levels	Not Applicable	Not Applicable	**RA2** In-Plant Radiation Levels Impede Plant Operations **MODES: ALL** EAL Threshold Value: 1 Radiation readings > 15 mR/hr in **EITHER** of the following: • Main Control Room **OR** • Central Alarm Station **AND** Increase is **NOT** due to an anticipated temporary increase from a planned event **OR** 2 In-plant radiation reading > 5 R/hr **AND** Increase is **NOT** due to an anticipated temporary increase from a planned event **AND** Access is required to affected area(s) per SE-1 or SE-10	**RU2** Rise In Plant Radiation Levels by a Factor of 1000 **MODES: ALL** EAL Threshold Value 1. Radiation readings indicate an unplanned rise by a factor of 1000 over normal levels **RU3** High MSL or Off-gas Radiation Levels **MODES: 1,2,3** EAL Threshold Value 1. SJAE Discharge Radiation > 2.5E+3 mR/hr **OR** 2 Main Steam Line Hi-Hi radiation Alarm (10xNFPB)
Coolant Activity	Not Applicable	Not Applicable	Not Applicable	**RU4** High coolant activity **MODES: ALL** EAL Threshold Value 1 Reactor coolant activity > 4 μCi/gm I-131 dose equivalent

Table R1 -- Effluent Monitor Thresholds

	General Emergency	Site Area Emergency	Alert	Unusual Event
Main Stack	> 3.23E+10 μCi/sec	> 3.23E+9 μCi/sec	> 200X Hi-Hi alarm	> 2X Hi-Hi alarm
	(RI-0-17-050A/B)			
Vent Stack	> 2.02E+8 μCi/sec	> 2.02E+7 μCi/sec	> 200X Hi-Hi alarm	> 2X Hi-Hi alarm
	(RI-2979A/B Unit 2 or RI-3979A/B Unit 3)			
Torus Vent	≥ 1.0E+7 cpm (OSH)	Not Applicable	Not Applicable	Not Applicable
	(RIS-80291 Unit 2 or RIS-90291 Unit 3)			
Service Water ESW HPSW Radwaste Discharge	Not Applicable	Not Applicable	> 200X Hi-Hi alarm	> 2X Hi-Hi alarm

Table R2 Dose Assessment Thresholds

* Refers to dose/dose rates at or beyond the Site Boundary, based on a 1 hour release duration

Method	General Emergency	Site Area Emergency	Alert	Unusual Event
Sample Analysis	Not Applicable	Not Applicable	> 200 X ODCM limits	> 2 X ODCM Limits
Field Team Monitoring*	> 1000 mRem/hr Whole Body OR > 5000 mRem CDE Thyroid	> 100 mRem TEDE OR > 500 mRem CDE Thyroid	> 3 mRem/hr Whole Body OR > 9 mRem CDE Thyroid	Not Applicable
Dose Assessment*	> 1000 mRem TEDE OR > 5000 mRem CDE Thyroid	> 100 mRem TEDE OR > 500 mRem CDE Thyroid	> 2.8 mRem/hr TEDE OR > 8.5 mRem CDE Thyroid	>0.114 mRem/hr TEDE OR > 0.342 mRem CDE Thyroid

TABLE PBAPS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

FISSION PRODUCT BARRIER MATRIX (Applicability: Modes 1, 2 & 3 ONLY) MODES: 1,2,3

FISSION PRODUCT BARRIER STATUS	FG1: GENERAL EMERGENCY				FS1: SITE AREA EMERGENCY								FA1: ALERT				FU1: UNUSUAL EVENT	
Fuel Clad – LOSS	X	X		X	X		X		X				X					
Fuel Clad – POTENTIAL LOSS			X			X		X		X				X				
Reactor Coolant System – LOSS	X	X	X		X			X			X				X			
Reactor Coolant System – POTENTIAL LOSS				X		X	X					X				X		
Primary Containment – LOSS	X		X	X					X	X	X	X					X	
Primary Containment – POTENTIAL LOSS		X																X

	1. FUEL CLAD BARRIER		2. REACTOR COOLANT SYSTEM BARRIER		3. PRIMARY CONTAINMENT BARRIER	
	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
a. Reactor Pressure Vessel (RPV) Water Level	1 RPV water level < -195 inches	2 RPV water level < -172inches	1 RPV water Level < -172 inches	2 RPV water level CANNOT be determined	Not Applicable	1 ANY of the following direct entry into SAMP-1 and SAMP-2. • T-111 • T-116 • T-117
b. Drywell (DW) High Range Rad Monitor	1 DW high range rad monitor reading > 7.8E+4 R/hr	Not Applicable	1. DW high range rad monitor reading > 15R/hr	Not Applicable	Not Applicable	1 DW high range rad monitor reading > 6.0E+5 R/hr
c. Drywell (DW) Pressure	Not Applicable	Not Applicable	1. Drywell pressure > 2.0 psig **AND** Indication of RCS leak inside Drywell	Not Applicable	1 Rapid, unexplained drop in DW pressure following an initial rise **OR** 2 DW pressure response not consistent with LOCA conditions indicating a Containment breach	3 DW pressure > 49 PSIG
d. Breached / Bypassed	1 Coolant activity > 300 uCi/gm I-131 dose equivalent **OR** 2 Core damage calculations indicate > 2.6% fuel clad damage	Not Applicable	1 Unisolable MSL Break indicated by the failure of BOTH MSIVs in ANY one line to close **AND** EITHER of the following • High MSL Flow and High Steam Tunnel Temperature **OR** • Direct report of steam release [NOTE Refer to MA8 for ISOLABLE MSL Break] **OR** 2 SRV is stuck open or cycling **AND** Indication of a LOSS of the Fuel Clad Barrier per the Fission Product Barrier Matrix	3 RCS Leakage > 50 gpm [NOTE Refer to MU7 for RCS leakage < 50 gpm] **OR** 4 Unisolable primary system leakage outside of CNTMT that results in: a Exceeding EITHER of the following T-103 Action Levels. • Table SC/T-3 (Temperature) **OR** • Table SC/R-1 (Radiation) **OR** b. SCRAM initiated per T-103 due to temperature or radiation levels	1 Failure of ALL automatic isolation valves in ANY one line penetrating Primary Containment to close resulting from an isolation actuation signal **AND** Downstream pathway exists to environment **OR** 2. Intentional venting per T-200 is required **OR** 3. Unisolable primary system leakage outside of CNTMT that results in. a Exceeding EITHER of the following T-103 Action Levels • Table SC/T-3 (Temperature) **OR** • Table SC/R-1 (Radiation) **OR** b SCRAM initiated per T-103 due to temperature or radiation levels	Not Applicable
e. Drywell Hydrogen Concentration	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable	1. Drywell H_2 > 6 % **AND** Drywell O_2 > 5 %
f. Discretionary	1. ANY condition that indicates a LOSS of the Fuel Clad Barrier	2. ANY condition that indicates a POTENTIAL LOSS of the Fuel Clad Barrier	1. ANY condition that indicates a LOSS of the Reactor Coolant System Barrier	2 ANY condition that indicates a POTENTIAL LOSS of the Reactor Coolant System Barrier	1 ANY condition that indicates a LOSS or of the Primary Containment Barrier	2. ANY condition that indicates a LOSS or of the POTENTIAL Primary Containment Barrier

TABLE PBAPS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
SYSTEM MALFUNCTIONS					
Loss of AC Power	**MG1** Prolonged Loss of ALL Offsite AC Power AND Prolonged Loss of ALL Onsite AC Power **MODES: 1,2,3** EAL Threshold Value 1 Loss of offsite power to ALL 4 KV Safeguard Busses **AND** ALL four of the 4 KV Safeguards Busses are de-energized for > 15 minutes **AND** ANY of the following • Restoration of at least one 4KV emergency bus in ≤ 2 hours is NOT likely **OR** • Reactor water level CANNOT be maintained > -172 inches **OR** • Torus temperature CANNOT be maintained on the "SAFE" side of the HCTL Curve (T-102, T/T-1)	**MS1** Loss of ALL Offsite AC Power AND Loss of ALL Onsite AC Power to Essential Busses **MODES: 1,2,3** EAL Threshold Value 1 Loss of offsite power to ALL 4 KV Safeguard Busses **AND** ALL four of the 4 KV Safeguards Busses are de-energized for > 15 minutes	**MA1** AC Power to Essential Buses Reduced to a Single Source for > 15 minutes **MODES: 1,2,3** EAL Threshold Value 1 Loss of offsite power to ALL 4 KV Safeguard Busses **AND** Three of four of the 4 KV Safeguards Busses are de-energized for > 15 minutes **MA2** Loss of ALL Offsite AC Power AND Loss of ALL Onsite AC Power to Essential Busses **MODES: 4,5,D** EAL Threshold Value 1. Loss of offsite power to ALL 4 KV Safeguard Busses **AND** ALL four of the 4 KV Safeguards Busses are de-energized for > 15 minutes	**MU1** Loss of ALL Offsite AC Power for > 15 minutes to Essential Busses **MODES: ALL** EAL Threshold Value 1 Loss of offsite power to ALL 4 KV Safeguard Busses for >15 minutes	Loss of AC Power
Loss of DC Power	Not Applicable	**MS3** Loss of ALL Required T S Safety-Related 125 VDC Power Sources **MODES: 1,2,3** EAL Threshold Value 1 Loss of ALL required T S safety related 125 VDC power sources for > 15 minutes as indicated by < 107 5 VDC on Panels 2(3)0D21, 22, 23, 24	Not Applicable	**MU3** Loss of ALL Required T S Safety-Related 125 VDC Power Sources **MODES: 4,5** EAL Threshold Value 1 Loss of ALL required T S. safety related 125 VDC power sources for > 15 minutes as indicated by < 107 5 VDC on Panels 2(3)0D21, 22, 23, 24	Loss of DC Power
Failure of Reactor Protection System	**MG4** Auto and Manual SCRAM NOT Successful, AND Loss of Core Cooling or Heat Sink **MODES: 1,2** EAL Threshold Value: 1. Failure of automatic RPS, ARI AND Manual SCRAM /ARI to shutdown the reactor as defined by EITHER of the following criteria • Reactor Power > 4% **OR** • Torus temperature greater than 110°F AND boron injection is required **AND** EITHER of the following criteria are met • Torus temperature CANNOT be maintained on the "SAFE" side of the HCTL Curve (T-102, T/T-1) **OR** • Reactor water level < -195 inches	**MS4** Auto and Manual SCRAM NOT Successful **MODES: 1,2** EAL Threshold Value: 1 Failure of automatic RPS, ARI AND Manual SCRAM /ARI to shutdown the reactor as defined by EITHER of the following criteria • Reactor power > 4% **OR** • Torus temperature greater than 110°F AND boron injection is required	**MA4** Auto SCRAM NOT Successful **MODES: 1,2** EAL Threshold Value: 1. RPS set point has been exceeded for an automatic SCRAM **AND** Failure of automatic RPS to achieve a state in which the reactor is shutdown under all conditions without boron injection	Not Applicable	Failure of Reactor Protection System
Decay Heat Removal	Not Applicable	**MS5** Complete Loss of Functions Needed to Achieve AND Maintain Hot Shutdown **MODES: 1,2,3** EAL Threshold Value 1. Loss of functions required for hot shutdown as evidenced by T-102 T/T leg directing a T-112 Emergency Blowdown	**MA5** Inability to Maintain Plant in Cold Shutdown **MODES: 4,5** EAL Threshold Value 1 Unplanned loss of ALL T.S. required decay heat removal systems **AND** EITHER of the following. • RCS temperature exceeding 212 °F for ≥ 15 minutes with a heat removal function restored **OR** • Uncontrolled RCS temperature rise approaching 212 °F with NO heat removal function restored	Not Applicable	Decay Heat Removal

TABLE PBAPS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
SYSTEM MALFUNCTIONS (cont.)					
Loss of Annunciators	Not Applicable	**MS6** Inability to Monitor a Significant Transient In Progress **MODES: 1,2,3** EAL Threshold Value: 1. A significant transient is in progress (Table M-1) AND ALL of the following are lost • Safety system annunciators (Table M-2) • Safety function indicators (Table M-3) • Plant Monitoring System	**MA6** Loss of Annunciators or Indicators Requiring Increased Surveillance **MODES: 1,2,3** EAL Threshold Value: 1. Unplanned loss for > 15 minutes of MOST (NOTE 1) or ALL of EITHER • Safety system annunciators (Table M-2) OR • Safety function indicators (Table M-3) AND Increased surveillance is required to safely operate the unit(s) AND EITHER of the following • A significant plant transient is in progress (Table M-1) OR • Plant Monitoring System is unavailable	**MU6** Unplanned Loss of Annunciators OR Indicators for > 15 minutes **MODES: 1,2,3** EAL Threshold Value: 1. Unplanned loss for > 15 minutes of MOST (NOTE 1) or ALL of EITHER • Safety system annunciators (Table M-2) OR • Safety function indicators (Table M-3) AND Increased surveillance is required to safely operate the unit(s)	Loss of Annunciators
RCS Leakage / RPV Draindown	Not Applicable	**MS7** Loss of Water Level in the Reactor Vessel That Has or Will Uncover Fuel in the Reactor Vessel **MODES: 4,5** EAL Threshold Value: 1. RPV water level < -172 inches	Not Applicable	**MU7** Reactor Coolant System Leakage **MODES: 1,2,3** EAL Threshold Value: 1. Unidentified primary system leakage > 10 gpm into the Drywell OR 2. Identified primary system leakage > 25 gpm into the Drywell [NOTE: Refer to Fission Product Barrier Matrix (2.d.3) for RCS leakage > 50 gpm.]	RCS Leakage / RPV Draindown
MSL Break (with Isolation)	Not Applicable	Not Applicable	**MA8** Main Steam Line Break **MODES: 1,2,3** EAL Threshold Value: 1. MSL Break indicated by EITHER of the following • High MSL Flow and High Steam Tunnel Temperature annunciators OR • Direct report of MSL steam release AND MSL break is successfully isolated [NOTE: REFER to Fission Product Barrier Matrix (2.d.1) for possible event escalation if break is unisolable.]	Not Applicable	MSL Break (with Isolation)
Loss of Communications	Not Applicable	Not Applicable	Not Applicable	**MU9** Unplanned Loss of ALL Onsite OR Offsite Communications Capabilities **MODES: ALL** EAL Threshold Value: 1. ALL onsite communications equipment lost (Table M-4) OR 2. ALL offsite communications equipment lost (Table M-5)	Loss of Communications
Technical Specs	Not Applicable	Not Applicable	Not Applicable	**MU10** Inability to Reach Required Operating Mode Within Technical Specification Time Limits **MODES: 1,2,3** EAL Threshold Value: 1. Inability to reach required operating mode within Tech Spec LCO action completion time	Technical Specs

Table M-1: Significant Plant Transients	Table M-2: Safety System Annunciators	Table M-3: Safety Function Indicators	Table M-4: Onsite Communications Equipment	Table M-5: Offsite Communications Equipment	NOTE 1
• SCRAM • Recirc Runback (> 25% thermal power change) • Sustained Power Oscillations (25% peak to peak) • Stuck Open Relief Valves • ECCS Injection	• ECCS • Containment Isolation • Reactor Trip • Process Radiation Monitoring	• Reactor Power • Decay Heat Removal • Containment Safety Functions	• Site Phones (GTE System) • OMNI System • Plant Public Address (PA) • Station Radio	• Site Phones (GTE System) • OMNI System • NRC (ENS) • PA State Radio • Load Dispatcher Radio	"MOST" refers to a loss of ~75% or a significant risk that a degraded plant condition could go undetected. Use is not intended to require a detailed count of annunciators/indicators

TABLE PBAPS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
SYSTEM MALFUNCTIONS (cont.)					
Irradiated Fuel Accidents	Not Applicable	Not Applicable	**MA11** Major Damage OR Uncovering of Spent Fuel **MODES: ALL** EAL Threshold Value 1 Unplanned general area radiation > 500 mR/hr on the Refuel Floor (Table M-6) OR 2 Report or visual observation that irradiated fuel is uncovered OR 3 Water level < 232 ft 3 inches plant elevation for the Spent Fuel Pool that will result in irradiated fuel uncovering	**MU11** Potential Damage OR Uncovering of Spent Fuel **MODES: ALL** EAL Threshold Value 1 Uncontrolled water level drop in Spent Fuel Pool that cannot be quickly terminated with ALL irradiated fuel assemblies remaining covered by water	Irradiated Fuel Accidents
Irradiated Fuel Accidents	Not Applicable	Not Applicable	**MA12** Loss of Water Level That Has OR Will Uncover Irradiated Fuel **MODE: 5*** EAL Threshold Value (* With the Rx Refueling Cavity Flooded) 1 Water level < 458 inches above RPV instrument zero for the Reactor Refueling Cavity AND Loss of water level will result in irradiated fuel uncovering	**MU12** Uncontrolled Water Level Decrease in Reactor Refueling Cavity **MODES: ALL** EAL Threshold Value 1 Unexpected Skimmer Surge Tank low level alarm AND Visual observation of an uncontrolled drop in water level below the fuel pool skimmer surge tank inlet that cannot be quickly terminated	Irradiated Fuel Accidents
Irradiated Fuel Accidents	Not Applicable	Not Applicable	Not Applicable	**MU13** Independent Spent Fuel Storage Installation (ISFSI) **MODES: ALL** EAL Threshold Value 1. EITHER of the following criteria is met for dry storage of spent fuel • > 600 mR/hr, 1 ft. away OR • > 1200 mR/hr at the external surface	Irradiated Fuel Accidents
HAZARDS AND OTHER CONDITIONS					
Security Events	**HG1** Security Event Resulting in Loss of Ability to Reach AND Maintain Cold Shutdown **MODES: ALL** EAL Threshold Value 1 Loss of physical control of the Control Room due to a security event OR 2 Loss of physical control of the remote shutdown capability due to a security event	**HS1** Confirmed Security Event in a Vital Area **MODES: ALL** EAL Threshold Value 1. Intrusion into plant Vital Area by a hostile force OR 2. Confirmed bomb, sabotage or sabotage device discovered in a Vital Area	**HA1** Confirmed Security Event in a Plant Protected Area **MODES: ALL** EAL Threshold Value 1. Intrusion into a Protected Area or ISFSI by a hostile force OR 2 Confirmed bomb, sabotage or sabotage device discovered in a Protected Area or ISFSI	**HU1** Confirmed Security Event That Indicates a Potential Degradation in Level of Plant Safety **MODES: ALL** EAL Threshold Value 1 A credible threat to the station reported by the NRC OR 2 BOTH of the following criteria met for a credible threat reported by any other outside agency as determined per SY-AA-101-132, "Threat Assessment". • Is specifically directed towards the station • Is imminent (≤ 2 hours) OR 3 Attempted intrusion and attack on a Protected Area or ISFSI OR 4 Attempted sabotage discovered within a Protected Area or ISFSI OR 5. Hostage/Extortion situation that threatens normal plant operations	Security Events

Table M-6: Refuel Floor ARMs

• 3-7 (7-9), Steam Separator Pool	• 3-9 (7-11), Fuel Pool
• 3-8 (7-10), Refuel Slot	• 3-10 (7-12), Refueling Bridge

TABLE PBAPS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
HAZARDS AND OTHER CONDITIONS (cont.)					
Control Room Evacuation	Not Applicable	**HS2** Control Room Evacuation Initiated **MODES: ALL** AND Plant Control CANNOT be re-established in ≤ 15 minutes EAL Threshold Value 1 Control Room evacuation initiated **AND** Control of the plant **CANNOT** be re-established in ≤ 15 minutes per SE-1 or SE-10	**HA2** Control Room Evacuation Initiated EAL Threshold Value 1 Entry into SE-1 or SE-10 for Control Room evacuation	Not Applicable	**Control Room Evacuation**
Natural or Man-Made Events	Not Applicable	Not Applicable	**HA3** Natural OR Destructive Phenomena Affecting a Vital Area **MODES: ALL** EAL Threshold Value 1 Earthquake > 0.05 g (Operating Basis Earthquake, OBE) as determined by procedure SO 67.7A **OR** 2 Tornado or wind speeds > 75 mph causing damage to Plant Vital Structures (Table H-1) **OR** 3 Report of visible structural damage to ANY Plant Vital Structure (Table H-1) **OR** 4 Vehicle crash affecting a plant vital function contained in a Plant Vital Structure (Table H-1) **OR** 5 Turbine failure generated missiles result in visible structural damage or penetration to ANY Plant Vital Structures (Table H-1) **OR** 6 Abnormal river level, as indicated by **EITHER** • > 116 ft (high level) **OR** • < 92.5 ft (low level) 7 Flooding in 2 or more areas designated in T-103, Table SC/L-2 requiring a plant shutdown	**HU3** Natural OR Destructive Phenomena Affecting the Protected Area **MODES: ALL** EAL Threshold Value 1 Earthquake > 0.01 g as determined by procedure SO 67.7A **OR** 2 Report by plant personnel of a tornado strike within Protected Area **OR** 3 Wind speeds > 75 mph as indicated on Site Meteorological instrumentation for > 15 minutes **OR** 4 Vehicle crash within the Protected Area Boundary that may potentially damage plant functions required for safe shutdown of the plant **OR** 5 Report of turbine failure resulting in casing penetration or damage to generator seals **OR** 6 Assessment by Control Room that a natural or destructive phenomena has occurred affecting the Protected Area **OR** 7 Abnormal river level, as indicated by **EITHER**. • > 112 ft. (high level) **OR** • < 98.5 ft (low level)	**Natural or Man-Made Events**

Table H-1: Plant Vital Structures
• Power Block
• Diesel Generator Building
• Emergency Pump Structure
• Inner Screen Structure
• Emergency Cooling Tower

TABLE PBAPS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS (cont.)				
Fire / Explosion	Not Applicable	Not Applicable	**HA4** Fire OR Explosion Affecting Operability of Safety Systems Required for Safe Shutdown **MODES: ALL** EAL Threshold Value 1. ANY of the following are made potentially inoperable due to a fire or explosion • <u>2 or more</u> Safe Shutdown Systems (Table H-2) • <u>2 or more</u> subsystems of a Safe Shutdown System (Table H-2) as defined by Tech Specs • <u>1 or more</u> Plant Vital Structures containing Safe Shutdown Equipment (Table H-1) AND Safe Shutdown System or Plant Vital Structure is required for the present Operational Condition	**HU4** Fire Within the Protected Area Boundary NOT Extinguished in ≤ 15 minutes of Detection **MODES: ALL** EAL Threshold Value 1 Fire within or impacting a Plant Vital Structure (Table H-1) AND Fire is NOT extinguished in ≤ 15 minutes of EITHER • Control Room notification • Verification of alarm OR 2 Report by plant personnel of an explosion within the Protected Area Boundary resulting in visible damage to a permanent structure or equipment
Toxic or Flammable Gases	Not Applicable	Not Applicable	**HA5** Release of Toxic OR Flammable Gases Within a Facility Structure Which Jeopardizes Operation of Systems Required to Maintain Safe Operation OR to Establish or Maintain Cold Shutdown **MODES: ALL** EAL Threshold Value 1 Report or detection of toxic gases within Plant Vital Structures (Table H-1) in concentrations that will be life threatening to plant personnel OR 2. Report or detection of flammable gases within Plant Vital Structures (Table H-1) in concentrations affecting the safe operation of the plant	**HU5** Release of Toxic OR Flammable Gases Deemed Detrimental to Safe Operation of the Plant **MODES: ALL** EAL Threshold Value 1 Report or detection of toxic or flammable gases that could enter within the site area in amounts that can affect normal operation of the plant OR 2 Report by Local, County or State officials for potential evacuation of site personnel based on an offsite event.
Discretionary	**HG6** Conditions Indicate Imminent Core Damage OR Release Affecting the Public **MODES: ALL** EAL Threshold Value 1 Actual or imminent core degradation and potential loss of containment OR 2 Potential uncontrolled radionuclide release, which can reasonably be expected to exceed 1 Rem TEDE or 5 Rem CDE Thyroid plume exposure levels at the Site Boundary	**HS6** Conditions Indicate Actual OR Likely Failure of Plant Functions Needed for Public Protection **MODES: ALL** EAL Threshold Value 1 Other conditions exist which in the judgment of the Emergency Director indicate actual or likely major failures of plant functions needed for protection of the public	**HA6** Conditions Indicate Actual OR Potential Substantial Degradation of the Level of Plant Safety **MODES: ALL** EAL Threshold Value 1. Other conditions exist which in the judgment of the Emergency Director indicate that plant safety systems may be degraded and that increased monitoring of plant functions is warranted	**HU6** Conditions Indicate a Potential Degradation in the Level of Plant Safety **MODES: ALL** EAL Threshold Value 1 ANY of the following occur, which in the judgment of the Emergency Director indicate a potential degradation in the level of safety of the plant: • Aircraft crash on-site • Train derailment on-site • Near-site explosion, which may adversely affect normal site activities OR 2 Other conditions exist which in the judgment of the Emergency Director indicate a potential degradation in the level of safety of the plant

Table H-1: Plant Vital Structures
• Power Block
• Diesel Generator Building
• Emergency Pump Structure
• Inner Screen Structure
• Emergency Cooling Tower

Table H-2: Safe Shutdown Systems	
• Diesel Generators	• HPSW
• HPCI	• ECW
• Core Spray	• Control Room Emergency Ventilation
• SBGT	• ADS
• PCIS (Primary CNTMT Isolation System)	• RHR (all modes)
• 4 KV Safeguards Busses	• ESW
• RCIC	• CAC / CAD

TABLE OF CONTENTS

TAB "R" – ABNORMAL RADIATION LEVELS / EFFLUENTS (Category "R")

1. **Radiological Effluents**
   - RG1 (Modes: ALL) .... PBAPS 3-15
   - RS1 (Modes: ALL) .... PBAPS 3-18
   - RA1 (Modes: ALL) .... PBAPS 3-20
   - RU1 (Modes: ALL) .... PBAPS 3-23
2. **Abnormal Radiation Levels**
   - RA2 (Modes: ALL) .... PBAPS 3-26
   - RU2 (Modes: ALL) .... PBAPS 3-28
   - RU3 (Modes: 1, 2 & 3) .... PBAPS 3-29
3. **Coolant Activity**
   - RU4 (Modes: ALL) .... PBAPS 3-30

TAB "F" – FISSION PRODUCT BARRIER DEGRADATION

- FG1 (Modes: 1, 2 & 3) .... LGS 3-31
- FS1 (Modes: 1, 2 & 3) .... LGS 3-33
- FA1 (Modes: 1, 2 & 3) .... LGS 3-34
- FU1 (Modes: 1, 2 & 3) .... LGS 3-35

1. **Fuel Clad Barrier**
   - Reactor Pressure Vessel Water Level (1.a) .... PBAPS 3-36
   - Drywell Radiation (1.b) .... PBAPS 3-37
   - Drywell Pressure (1.c) .... PBAPS 3-40
   - Breached / Bypassed (1.d) .... PBAPS 3-41
   - Drywell Hydrogen Concentration (1.e) .... PBAPS 3-43
   - Discretionary (1.f) .... PBAPS 3-44
2. **Reactor Coolant System Barrier**
   - Reactor Pressure Vessel Water Level (2.a) .... PBAPS 3-45
   - Drywell Radiation (2.b) .... PBAPS 3-46
   - Drywell Pressure (2.c) .... PBAPS 3-47
   - Breached / Bypassed (2.d) .... PBAPS 3-48
   - Drywell Hydrogen Concentration (2.e) .... PBAPS 3-51
   - Discretionary (2.f) .... PBAPS 3-52
3. **Primary Containment Barrier**
   - Reactor Pressure Vessel Water Level (3.a) .... PBAPS 3-53
   - Drywell Radiation (3.b) .... PBAPS 3-54
   - Drywell Pressure (3.c) .... PBAPS 3-55
   - Breached / Bypassed (3.d) .... PBAPS 3-56
   - Drywell Hydrogen Concentration (3.e) .... PBAPS 3-58
   - Discretionary (3.f) .... PBAPS 3-59

TABLE OF CONTENTS

TAB "M" – SYSTEM MALFUNCTIONS

1. Loss of AC Power (Operating)
   - MG1 (Modes: 1, 2 & 3) .... PBAPS 3-60
   - MS1 (Modes: 1, 2 & 3) .... PBAPS 3-62
   - MA1 (Modes: 1, 2 & 3) .... PBAPS 3-63
   - MU1 (Modes: ALL) .... PBAPS 3-64
2. Loss of AC Power (Shutdown)
   - MA2 (Modes: 4, 5 & Defuel) .... PBAPS 3-65
3. Loss of DC Power
   - MS3 (Modes: 1, 2 & 3) .... PBAPS 3-66
   - MU3 (Modes: 4 & 5) .... PBAPS 3-67
4. Failure of Reactor Protection System
   - MG4 (Modes: 1 & 2) .... PBAPS 3-68
   - MS4 (Modes: 1 & 2) .... PBAPS 3-70
   - MA4 (Modes: 1 & 2) .... PBAPS 3-72
5. Decay Heat Removal
   - MS5 (Modes; 1, 2 & 3) .... PBAPS 3-74
   - MA5 (Modes: 4 & 5) .... PBAPS 3-75
6. Loss of Annunciators
   - MS6 (Modes: 1, 2 & 3) .... PBAPS 3-77
   - MA6 (Modes: 1, 2 & 3) .... PBAPS 3-78
   - MU6 (Modes: 1, 2 & 3) .... PBAPS 3-80
7. RCS Leakage / RPV Draindown
   - MS7 (Modes: 4 & 5) .... PBAPS 3-81
   - MU7 (Modes: 1, 2, 3 & 4) .... PBAPS 3-83
8. Main Steam Line Break (with isolation)
   - MA8 (Modes: 1, 2 & 3) .... PBAPS 3-85
9. Loss of Communications
   - MU9 (Modes: ALL) .... PBAPS 3-86
10. Technical Specifications
    - MU10 (Modes: 1, 2 & 3) .... PBAPS 3-87
11. Irradiated Fuel Accidents (Spent Fuel Pool)
    - MA11 (Modes: ALL) .... PBAPS 3-88
    - MU11 (Modes: ALL) .... PBAPS 3-90
12. Irradiated Fuel Accidents (Reactor Refuel Cavity)
    - MA12 (Modes: 5 – With the Rx Refueling Cavity Flooded) ... PBAPS 3-91
    - MU12 (Modes: ALL) .... PBAPS 3-93
13. Irradiated Fuel Accidents (Dry Spent Fuel Storage Facility)
    - MU13 (Modes: ALL) .... PBAPS 3-94

TABLE OF CONTENTS

TAB "H" – HAZARDS AND OTHER CONDITIONS

1. Security Events
   - HG1 (Modes: ALL) .... PBAPS 3-96
   - HS1 (Modes: ALL) .... PBAPS 3-97
   - HA1 (Modes: ALL) .... PBAPS 3-98
   - HU1 (Modes: ALL) .... PBAPS 3-99
2. Control Room Evacuation
   - HS2 (Modes: ALL) .... PBAPS 3-100
   - HA2 (Modes: ALL) .... PBAPS 3-101
3. Natural or Man-Made Events
   - HA3 (Modes: ALL) .... PBAPS 3-102
   - HU3 (Modes: ALL) .... PBAPS 3-105
4. Fire / Explosion
   - HA4 (Modes: ALL) .... PBAPS 3-108
   - HU4 (Modes: ALL) .... PBAPS 3-110
5. Toxic or Flammable Gases
   - HA5 (Modes: ALL) .... PBAPS 3-112
   - HU5 (Modes: ALL) .... PBAPS 3-114
6. Discretionary
   - HG6 (Modes: ALL) .... PBAPS 3-115
   - HS6 (Modes: ALL) .... PBAPS 3-116
   - HA6 (Modes: ALL) .... PBAPS 3-117
   - HU6 (Modes: ALL) .... PBAPS 3-118

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RG1

INITIATING CONDITION

Actual or Projected Site Boundary Dose Using Actual Meteorology:

> 1000 mRem TEDE

OR

> 5000 mRem CDE Thyroid

EAL THRESHOLD VALUES

1. Radiological release in excess of Table R1 "General Emergency" threshold
   AND
   Release **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "General Emergency" thresholds
   OR
2. Radiological releases exceed ANY Table R2 column "General Emergency" threshold.

Table R1: Effluent Monitor Thresholds	
	General Emergency
Main Stack	> 3.23E+10 µCi/sec (RI-0-17-050A/B)
Vent Stacks	> 2.02E+8 µCi/sec (RI-2979A/B Unit 2 or RI-3979A/B Unit 3)
Torus Vents	**≥ 1.0E+7 cpm (Off-Scale High)** (RIS-80291 Unit 2 or RIS-90291 Unit 3)

Table R2: Dose Assessment Thresholds	
Method	**General Emergency**
Sample	N/A
Field Team Monitoring*	> 1000 mRem/hr Whole Body **OR** > 5000 mRem CDE Thyroid
Dose Projection*	> 1000 mRem TEDE **OR** > 5000 mRem CDE Thyroid
* At or beyond Site Boundary based on a 1 hour release duration	

MODE APPLICABILITY:

ALL

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RG1 – Cont'd

BASIS: (References)

Site Boundary - For classification and dose projection purposes, the Site Boundary is the Exclusion Area Boundary, a 2700 foot radius around the plant. The actual boundary is specified in the ODCM.

Total Effective Dose Equivalent (TEDE) The sum of the deep dose equivalent (for external exposure) and the committed effective dose equivalent (for internal exposure) and 4 days of deposition exposure.

Committed Dose Equivalent (CDE) The Dose Equivalent to organs or tissues of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake. Thyroid values are taken from EPA-400, Table 5-4 to be consistent with the NRC RASCAL dose assessment program used by the Pennsylvania Emergency Management Agency (PEMA) / Bureau of Radiation Protection (BRP). Actual meteorology is used, since it gives the most accurate dose projection.

Table R1:

Effluent Monitors - Classification is based on the instantaneous release rate value if NO dose projections can be performed or verified within 15 minutes of meeting or exceeding the specified Release Rate value.

For a gaseous release from the Torus Vents, **if** a valid effluent monitor reading (Unit 2: RIS-80291 or Unit 3: RIS-90291) indicates "off-scale high" (≥ 1.0E+7 cpm) AND field monitoring teams are not available to immediately confirm actual offsite dose below Table R-2 thresholds, **then DECLARE** a General Emergency classification. Since effluent monitor is "off-scale high", Table R-2 projected offsite dose thresholds cannot be confirmed using DAPAR for the Torus Vents.

NOTE: (For calculation basis purposes) The threshold value for the Torus Vent was calculated at 6.95E+9 cpm, which is "Off-Scale High" (≥ 1.0E+7 cpm).

Monitor indications are calculated using the computerized dose model with UFSAR (gap release) source terms applicable to each monitored pathway in conjunction with annual average meteorology per NUMARC/NESP-007 (Revision 2). Calculations assume:

	Main Stack	Vent Stack	Torus Vent
Time After Shutdown (TAS)	0 hours	0 hours	0 hours
Release Duration	1 hour	1 hour	1 hour
Core Damage	10%	10%	10%
Flow Rate	33,000 cfm	474,000 cfm (U2) 450,000 cfm (U3)	9570 cfm (U2) 9570 cfm (U3)
Highest Annual Average X/Q (per ODCM)	9.97E-8 sec/m^3	5.33E-7 sec/m^3	5.33E-7 sec/m^3
Process Reduction Factors (per NUREG-1228)			
• CNMT Hold Up / Spray OFF	< 2 hours	< 2 hours	< 2 hours
• Reactor Bldg. Hold Up	< 2 hours	< 2 hours	N/A
• Standby Gas Treatment Filters	YES	N/A	N/A

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RG1 – Cont'd

BASIS: (References)

Table R2:

Field Team Monitoring – 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. The assumed release duration is 1 hour. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption media followed by field analysis are used for determining the iodine value.

Dose Projection – Any calculated dose projection of 1 Rem TEDE or 5 Rem CDE Thyroid is classified based on U.S. Environmental Protective Action (EPA) guidelines established under EPA-400-R-92-001 (May 1992). Source term, release elevation and release duration inputs are options and should reflect actual release conditions. Actual meteorology should also be used to reflect actual release conditions.

Since effluent monitor is "off-scale high", Table R-2 projected offsite dose thresholds cannot be confirmed using DAPAR for the Torus Vents.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RS1

INITIATING CONDITION

Actual or Projected Site Boundary Dose Using Actual Meteorology:

>100 mRem TEDE
OR
>500 mRem CDE Thyroid

EAL THRESHOLD VALUES

1. Radiological release in excess of Table R1 "Site Area Emergency" threshold
   AND
   Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "Site Area Emergency" thresholds

   OR

2. Radiological releases exceed ANY Table R2 column "Site Area Emergency" threshold.

Table R1: Effluent Monitor Thresholds	
	Site Area Emergency
Main Stack	> 3.23E+9 μCi/sec (RI-0-17-050A/B)
Vent Stacks	> 2.02E+7 μCi/sec (RI-2979A/B Unit 2 or RI-3979A/B Unit 3)
Torus Vents	Not Applicable

Table R2: Dose Assessment Thresholds	
Method	**Site Area Emergency**
Sample	N/A
Field Team Monitoring*	> 100 mRem/hr Whole Body **OR** > 500 mRem CDE Thyroid
Dose Projection*	> 100 mRem TEDE **OR** > 500 mRem CDE Thyroid
* At or beyond Site Boundary based on a 1 hour release duration	

MODE APPLICABILITY:

ALL

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RS1 – Cont'd

BASIS: (References)

Table R1:

Effluent Monitors - Classification is based on the instantaneous release rate value if no dose projections can be performed or verified within 15 minutes of meeting or exceeding the specified Release Rate value.

For a gaseous release from the Torus Vents, **if** a valid effluent monitor reading (Unit 2: RIS-80291 or Unit 3: RIS-90291) indicates "off-scale high" (≥ 1.0E+7 cpm) AND field monitoring teams are not available to immediately confirm actual offsite dose below Table R-2 thresholds, **then REFER** to RG1 for evaluation of a General Emergency classification. Since effluent monitor is "off-scale high", Table R-2 projected offsite dose thresholds cannot be confirmed using DAPAR for the Torus Vents.

NOTE: (For calculation basis purposes) The threshold value for the Torus Vent was calculated at 6.95E+8 cpm, which is "Off-Scale High" (≥ 1.0E+7 cpm). As such, user shall default to RG1 (General Emergency) based on effluent monitor reading for the Torus Vent.

Monitor indications are calculated using the computerized dose model with UFSAR (gap release) source terms applicable to each monitored pathway in conjunction with annual average meteorology per NUMARC/NESP-007 (Revision 2). Calculations assume:

	Main Stack	Vent Stack	Torus Vent
Time After Shutdown (TAS)	0 hours	0 hours	0 hours
Release Duration	1 hour	1 hour	1 hour
Core Damage	10%	10%	10%
Flow Rate	33,000 cfm	474,000 cfm (U2) 450,000 cfm (U3)	9570 cfm (U2) 9570 cfm (U3)
Highest Annual Average X/Q (per ODCM)	9.97E-8 sec/m^3	5.33E-7 sec/m^3	5.33E-7 sec/m^3
Process Reduction Factors (per NUREG-1228)			
• CNMT Hold Up / Spray OFF	< 2 hours	< 2 hours	< 2 hours
• Reactor Bldg. Hold Up	< 2 hours	< 2 hours	N/A
• Standby Gas Treatment Filters	YES	N/A	N/A

Table R2:

Field Team Monitoring – 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. The assumed release duration is 1 hour. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption media followed by field analysis are used for determining the iodine value.

Dose Projection – Any calculated dose projection of 100 mRem TEDE or 500 mRem CDE Thyroid is classified based on 10% of the guidelines established under EPA-400-R-92-001 (May 1992). Source term, release elevation and release duration inputs are options and should reflect actual release conditions. Actual meteorology should also be used to reflect actual release conditions.

Since effluent monitor is "off-scale high", Table R-2 projected offsite dose thresholds cannot be confirmed using DAPAR for the Torus Vents.

This event will be escalated to a General Emergency when actual or projected doses exceed EPA-400-R-92-001 Protective Action Guidelines per IC RG1.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA1

INITIATING CONDITION

Release > 200 X ODCM Limit for ≥ 15 minutes

EAL THRESHOLD VALUES

1. Unplanned radiological release lasting ≥ 15 minutes in excess of Table R1 "Alert" thresholds
   AND
   Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "Alert" thresholds
   OR
2. Unplanned radiological releases lasting ≥ 15 minutes in excess of ANY Table R2 column "Alert" threshold

Table R1: Effluent Monitor Thresholds	
	Alert
• **Main Stack** (RI-0-17-050A/B) • **Vent Stack** (RI-2779A/B Unit 2 or RI-3979A/B Unit 3) • **Radwaste Discharge** • **Service Water** • **Emergency Service Water** • **High Pressure Service Water**	> 200 X Hi Hi alarm set point

Table R2: Dose Assessment Thresholds	
Method	**Alert**
Sample	> 200 X ODCM limits
Field Team Monitoring*	> 3 mRem/hr Whole Body **OR** > 9 mRem CDE Thyroid
Dose Projection*	> 2.8 mRem/hr TEDE **OR** > 8.5 mRem CDE Thyroid
* **At or beyond Site Boundary based on a 1 hour release duration**	

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA1 – Cont'd

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned - Any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm set points, etc.) on the applicable permit

It is not intended that the release be averaged over 15 minutes. A release of this greater magnitude that cannot be terminated in 15 minutes represents an uncontrolled situation that is an actual or potential substantial degradation of the level of safety of the plant. The degradation in plant control implied by the fact that the release cannot be terminated in 15 minutes is the primary concern.

Further, 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, unless release rate confirmation (sample, field survey or dose projection) is likely within this 15 minute period.

Table R1:

Effluent Monitors - EAL thresholds are based on 200 times the ODCM limits. For event classification purposes, the HI-HI Radiation alarms for the Main Stack and Vent Stacks are used to represent the ODCM limit. These alarm set points are using the calculation methodology as outlined in Sections 2.2 of the ODCM, which uses the highest annual average X/Q value for the designated sectors in accordance with NUMARC/NESP-007 (AA1). The HI Radiation alarm set points are also set conservatively to indicate when a release may approach ODCM limits assuming multiple release points.

Table R2:

It is intended that the event be declared as soon as it is determined that the release will exceed two hundred times ODCM for greater than 15 minutes.

Samples - Grab samples are used to determine release concentrations or rates to confirm effluent monitor readings or when the effluent monitors are out of services.

Field Team Monitoring - 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. The assumed release duration is 1 hour. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption media followed by field analysis are used for determining the iodine value.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA1 – Cont'd

BASIS (References) – Cont'd

Readings are based on equivalent dose model triggers representing 200X ODCM limit. Values are rounded up to the next highest unit based on conservatism of dose projections and to allow reading on survey meter.

Dose Projection – This EAL includes a 15-minute average for the dose projection with the release point radiation monitor above two hundred times the Hi Hi alarm set point value for the entire 15 minutes. It is not intended that the release be averaged over 15 minutes, but exceed threshold for 15 minutes.

TEDE and CDE Thyroid thresholds used represent offsite dose triggers built into the dose model to conservatively reflect 200X ODCM limit.

Releases in excess of 200 times the ODCM limits that continue for > 15 minutes represent an uncontrolled situation and hence a potential degradation in the level of safety. The primary concern is the final integrated dose [100 times greater than the Unusual Event] and the degradation in plant control implied by the fact that the release was not isolated within 15 minutes.

This event will be escalated to a Site Area Emergency when actual or projected doses are determined to exceed 10CFR20 annual average population exposure limits per IC RS1.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU1

INITIATING CONDITION

Release > 2 X ODCM Limit for ≥ 60 minutes

EAL THRESHOLD VALUE

1. Unplanned radiological release lasting ≥ 60 minutes in excess of Table R1 "Unusual Event" threshold
   AND
   Releases **CANNOT** be determined in < 60 minutes from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "Unusual Event" thresholds

   OR

2. Unplanned radiological releases lasting ≥ 60 minutes in excess of ANY Table R2 column "Unusual Event" threshold

Table R1: Effluent Monitor Thresholds	
	Unusual Event
• **Main Stack** (RI-0-17-050A/B) • **Vent Stack** (RI-2779A/B Unit 2 or RI-3979A/B Unit 3) • **Radwaste Discharge** • **Service Water** • **Emergency Service Water** • **High Pressure Service Water**	> 2 X Hi Hi alarm set point

Table R2: Dose Assessment Thresholds	
Method	**Unusual Event**
Sample	> 2 X ODCM limits
Field Team Monitoring*	N/A
Dose Projection*	> 0.114 mRem/hr TEDE **OR** > 0.342 mRem CDE Thyroid
* At or beyond Site Boundary based on a 1 hour release duration	

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU1 – Cont'd

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned - Any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm set points, etc.) on the applicable permit

Unplanned releases in excess of two times the site ODCM 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 limits for 30 minutes does not exceed this EAL. Further, 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, if it is unlikely that an ODCM calculation can be performed within 60 minutes of exceeding the EAL threshold.

Table R1:

Effluent Monitors - EAL thresholds are based on two times the ODCM limits. For event classification purposes, the HI-HI Radiation alarms for the Main Stack and Vent Stacks are used to represent the ODCM limit. These alarm set points are using the calculation methodology as outlined in Sections 2.2 of the which uses the highest annual average X/Q value for the designated sectors in accordance with NUMARC/NESP-007 (AU1). The HI Radiation alarm set points are also set conservatively to indicate when a release may approach ODCM limits assuming multiple release points.

Table R2:

It is intended that the event be declared as soon as it is determined that the release will exceed two times ODCM for greater than 60 minutes.

Samples - Grab samples are used to determine release concentrations or rates to confirm effluent monitor readings or when the effluent monitors are out of services.

Dose Projection – This EAL includes a 60-minute average for the dose projection with the release point radiation monitor above two times the Hi Hi alarm set point value for the entire 60 minutes. It is not intended that the release be averaged over 60 minutes, but exceed threshold for 60 minutes.

TEDE and CDE Thyroid thresholds used represent offsite dose triggers built into the dose model to conservatively reflect 2X ODCM limit.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU1 – Cont'd

BASIS (References)

Releases in excess of 2 times the ODCM limits that continue for > 60 minutes represent an uncontrolled situation and hence a potential degradation in the level of safety. The final integrated dose is very low and is not the primary concern. Rather it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

This event will be escalated to an Alert when release is determined to be >200 x ODCM Limit for greater than or equal to 15 minutes per IC RA1.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA2

INITIATING CONDITION

In-Plant Radiation Levels Impede Plant Operations

EAL THRESHOLD VALUES

1. Radiation readings > 15 mR/hr in **EITHER** of the following:
   - Main Control Room

     OR
   - Central Alarm Station

   AND

   Increase is **NOT** due to an anticipated temporary increase from a planned event.

   OR

2. In-plant radiation readings > 5 R/hr

   AND

   Increase is **NOT** due to an anticipated temporary increase from a planned event.

   AND

   Access is required to affected area(s) per SE-1 or SE-10

MODE APPLICABILITY

ALL

BASIS (References)

This EAL addresses elevated radiation levels that impede necessary access to operator stations, or other areas containing equipment that must be operated manually in order to maintain safe operation or to perform a safe shutdown. The concern of the EAL is a loss of control of radioactive material causing high radiation levels. As such, this EAL is not intended to apply to anticipated temporary increases in radiation levels due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, controlled movement of radiological sources, or expected increases in area radiation levels due to normal operation of plant systems / components, etc.).

The impaired ability to operate the plant is to be considered as the actual or potential substantial degradation of the level of safety of the plant. The cause of the rise in radiation levels is not the major concern of this EAL. For example, a dose rate of 15 mR/hr in the control room or hi radiation monitor readings may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, the fission product barrier table may indicate a SAE or GE. This EAL could result in declaration of an Alert at one unit due to a radioactivity release or radiation shine resulting from a major accident at the other unit.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA2 – Cont'd

BASIS (References) – Cont'd

Threshold Value 1 - The value of 15 mRem/hr is derived from the general design criteria (GDC) value of 5 REM in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737 "Clarification of TMI Action Plan Requirements" provides that the 15 mRem/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an ALERT.

Plant normal and emergency procedures may be implemented without requiring access to any areas except the Control Room and Central Alarm Station to be continuously occupied. The Radwaste Control Room is not required to be continuously occupied in order to maintain plant safety functions since inputs to radwaste will be isolated with a secondary containment isolation and releases can only be performed manually.

Threshold Value 2 - Areas requiring infrequent access and dose rate values are based on those specified in procedures SE-1 or SE-10. EAL is applicable only when procedures SE-1 or SE-10 direct access, in other words, when you are in those procedures. Therefore, if you were in procedures SE-1 or SE-10 and you needed direct access to a particular area and at the time radiation levels were > 5 R/hr, classification under this EAL is appropriate. Just having radiation levels > 5 R/hr in those areas defined in SE-1 or SE-10, when access is not directed per procedure, does not warrant classification under this EAL.

The single value of 5 R/hr was selected because it is based on radiation levels which result in exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits (i.e., 10 CFR 20), and in doing so, will impede necessary access. Stay times for levels up to that value are, generally several minutes, enough time to enter an area and manually operate the equipment. Dose rates > 5 R/hr will impede necessary access.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU2

INITIATING CONDITION

Rise in Plant Radiation Levels By a Factor of 1000

EAL THRESHOLD VALUE

1. Radiation readings indicate an unplanned rise by a factor of 1000 over normal levels

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned - Not the result of an intended evolution and requiring corrective or mitigative actions.

Normal Levels – Normal radiation levels can be considered as the highest reading in the past 24-hour period, excluding the current peak value, as determined by recorder charts, surveys, logs, etc.

Classification of an UNUSUAL EVENT is warranted as a precursor to more serious events. The concern of this EAL is the loss of control of radioactive material representing a potential degradation of the level of safety of the plant. The Threshold Value tends to have a long lead-time relative to a radiological release and thus the threat to public health and safety is very low. In light of the elevated dose rates the Emergency Director should evaluate how these conditions will affect the other unit.

This event will be escalated to an Alert when in-plant radiation levels impede plant operations per IC RA2.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU3

INITIATING CONDITION

High Main Steam Line (MSL) OR Off-gas Radiation Levels

EAL THRESHOLD VALUES

1. SJAE Discharge Radiation > 2.5E+3 mRem/hr

   OR

2. Main Steam Line Hi-Hi Radiation Alarm (10xNFPB)

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

Threshold Value 1: The steam jet air ejector discharge (Off-Gas) radiation monitor in the Control Room would be one of the first indicators of a degrading core. This instrument takes a sample before the recombiner. This indicator of elevated activity is equivalent to Technical Specification limit and is considered to be a precursor of more serious problems.

Threshold Value 2: Main Steam Line (MSL) Hi-Hi Radiation alarm > 10 times normal full power background (NFPB), may be indicative of minor fuel cladding degradation. The MSL Hi-Hi radiation condition requires a manual Main Steam Isolation Valve (MSIV) closure and a reactor scram. This transient may result in the introduction of fission product gases (previously contained in the gap area) to be suddenly released into the coolant due to the rapid down power transient and subsequent collapse of voids in the coolant.

This level of steam line activity is indicative of the release of gap activity to the coolant, rather than a major failure of the fuel clad. However, the mechanics that caused MSL radiation to rise to this level indicate there are a degradation of Fuel Clad integrity.

This EAL is NOT intended to apply to cases caused by resin intrusion or other known factors that are not directly indicative of fuel cladding degradation, but rather coolant chemistry issues.

This event will escalate to a Site Area Emergency based on a MSL break per **EITHER** the Fission Product Barrier Matrix (2.d.1 & 3.d.1) for an unisolable break, **OR** an Alert based on MA8 for an isolable MSL break scenario.

DEVIATION: The MODE applicability [1,2,3] is a deviation from NUMARC [all] in that, the SJAE Radiation Monitor and Main Steam Line Radiation Monitors will only be a valid indication of Fuel Clad Degradation in those Modes. There are no other monitors, which can be used as an indicator of Fuel Clad Degradation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU4

INITIATING CONDITION

High Coolant Activity

EAL THRESHOLD VALUES

Reactor coolant activity > 4 μCi/gm I-131 dose equivalent

MODE APPLICABILITY:

ALL

BASIS: (References)

Coolant activity in excess of Technical Specifications (> 4 μCi/gm) is considered to be a precursor of more serious problems. The Technical Specification limit reflects a degrading or degraded core condition. This level is chosen to be above any possible short duration spikes under normal conditions. 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.

This event will be escalated to an Alert when Reactor Coolant activity exceeds 300 μCi/gm Dose Equivalent Iodine 131 per Fission Product Barrier Matrix 1.d.1.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FG1

INITIATING CONDITION

LOSS of 2 Fission Product Barriers and POTENTIAL LOSS of the Third Barrier

EAL THRESHOLD VALUE

Comparison of conditions / values with those listed in Fission Product Barrier Matrix indicates:

LOSS of ANY Two Barriers

AND

POTENTIAL LOSS of Third Barrier

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Conditions / events required to cause the loss of 2 Fission Product Barriers with the potential loss of the third could reasonably be expected to cause a release beyond the immediate site area exceeding EPA Protective Action Guidelines.

Guidance for development of this EAL was taken from Recognition Category F in the NUMARC/NEI Methodology for Development of Emergency Action Levels.

A barrier LOSS shall also constitute a POTENTIAL LOSS for classification purposes.

NOTES:

1. Although the logic used for these initiating conditions appears overly complex, it is necessary to reflect the following considerations:

   - The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Primary Containment barrier. Unusual Event Initiating Conditions (ICs) associated with RCS and Fuel Clad barriers are addressed under the other plant condition EALs.

   - At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from General Emergency. For example, if the Fuel Clad barrier and RCS barrier "Loss" EALs existed, this would indicate to the Emergency Director that, in addition to offsite dose assessments, the ED must focus on continual assessments of radioactive inventory and containment integrity. If, on the other hand, both Fuel Clad barrier and RCS barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FG1 – Cont'd

BASIS (References) – Cont'd

- The ability to escalate to higher emergency classes as an event gets worse must be maintained. For example, RCS leakage steadily rising would represent an increasing risk to public health and safety.

2. Fission Product Barrier ICs must be capable of addressing event dynamics. An IMMINENT (i.e., within 1 to 2 hours) Loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

3. The Fuel Clad barrier is the cladding tubes that contain the fuel pellets.

4. The RCS Barrier is the reactor coolant system pressure boundary and includes the reactor vessel and all reactor coolant system piping up to the isolation valves.

5. The Primary Containment Barrier includes the drywell, the torus, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves.

6. If a "Loss" condition is satisfied, the "Potential Loss" category can be considered satisfied. This is also applicable to conditions where there is a "Loss" indication with no corresponding "Potential Loss" condition.

7. For all conditions listed in Fission Product Barrier Table, the barrier failure column is only satisfied if it fails when called upon to mitigate an accident. For example, failure of both 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 operations, it will be an active Technical Specification Action Statement. However, during accident conditions, this will represent a breach of Primary Containment.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FS1

INITIATING CONDITION / EAL THRESHOLD VALUE

LOSS of BOTH Fuel Clad and RCS Barriers

OR

POTENTIAL LOSS of BOTH Fuel Clad and RCS Barriers

OR

POTENTIAL LOSS of EITHER the Fuel Clad or RCS Barrier, AND a LOSS of ANY Additional Barrier

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Loss of 2 Fission Product Barriers would be a major failure of plant systems needed for protection of the public.

Guidance for development of this EAL was taken from Recognition Category F in the NUMARC/NEI Methodology for Development of Emergency Action Levels.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FA1

INITIATING CONDITION

ANY LOSS or ANY POTENTIAL LOSS of **EITHER** the Fuel Cladding or Reactor Coolant System

EAL THRESHOLD VALUE

Comparison of conditions / values with those listed in Fission Product Barrier Matrix indicates:

LOSS <u>or</u> POTENTIAL LOSS of the Fuel Cladding Barrier

OR

LOSS <u>or</u> POTENTIAL LOSS of the Reactor Coolant System Barrier

MODE APPLICABILITY

1, 2, 3

BASIS (References)

The Fuel Cladding and the Reactor Coolant System are weighted more heavily than the Containment Barrier.

A LOSS or POTENTIAL LOSS of either the Fuel Cladding or the Reactor Coolant System would be a substantial degradation in the level of plant safety.

Guidance for development of this EAL was taken from Recognition Category F in the NUMARC/NEI Methodology for Development of Emergency Action Levels.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FU1

INITIATING CONDITION

ANY LOSS or ANY POTENTIAL LOSS of Containment

EAL THRESHOLD VALUE

Comparison of conditions / values with those listed in Fission Product Barrier Matrix indicates:

LOSS of the Containment Barrier

OR

POTENTIAL LOSS of the Containment Barrier

MODE APPLICABILITY

1, 2, 3

BASIS (References)

The Fuel Cladding and the Reactor Coolant System are weighted more heavily than the Containment Barrier.

Loss of the Containment would be a potential degradation in the level of plant safety.

Guidance for development of this EAL was taken from Recognition Category F in the NUMARC/NEI Methodology for Development of Emergency Action Levels.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.a

INITIATING CONDITION

Reactor Pressure Vessel (RPV) Water Level

THRESHOLD VALUE

LOSS: .... 1. RPV water level < -195 inches

POTENTIAL LOSS:........... 2. RPV water Level < -172 inches

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - [Threshold Value #1] Value of -195 inches corresponds to the level, which is used in the TRIP guidance to indicate challenge of core cooling. This is the Minimum Steam Cooling RPV Water Level, and as such, is the lowest RPV level at which the submerged portion of the core will generate sufficient steam to prevent any clad in the uncovered portion of the core from exceeding 1500°F. This RPV level is utilized to preclude fuel damage when RPV level is below the Top of Active Fuel (TAF).

Core submergence is the preferred method of core cooling and as such, the failure to re-establish RPV water level above the top of active fuel for an extended period of time could lead to significant fuel damage. RPV level < -195" could be indicative of a large break Loss Of Coolant Accident (LOCA) where ECCS Systems are designed to maintain level at 2/3 core height, or a small LOCA with the inability of emergency core cooling systems to reflood the RPV.

POTENTIAL LOSS – [Threshold Value #2] Core submergence is the preferred method of core cooling, and as such, the failure to re-establish RPV water level above TAF for an extended period of time could lead to significant fuel damage.

A level of < -172 inches also corresponds to the EAL for a RCS Barrier LOSS (IC RCS 2.a.1). Thus, this EAL indicates a LOSS of RCS barrier and a POTENTIAL LOSS of the Fuel Clad Barrier.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.b

INITIATING CONDITION

Drywell (DW) High Range Rad Monitor

THRESHOLD VALUE

LOSS: ....1. DW high range rad monitor reading > 7.8E+4 R/hr

POTENTIAL LOSS: ....NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - The intent is not to verify criteria used in calculation (e.g., release of reactor coolant into drywell), but rather to classify once EAL threshold value is reached or exceeded. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for RCS barrier under RCS 2.b.1. Thus, this EAL indicates a loss of both Fuel Clad barrier and RCS barrier.

[Calculation Basis] The 7.8E+4 R/hr reading on a drywell high range gamma radiation monitor RI-8(9)103A,B,C,D is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell. The reading was calculated assuming an instantaneous release and dispersal of the Reactor Coolant noble gas and iodine inventory into the Primary Containment (direct reading not shine) at a coolant concentration of 300 μCi/gm Dose Equivalent Iodine 131.

This calculation is as follows:

Using Curve 3 [1%] of Figure 4-1 of attached "Containment Radiation Monitor Dose Rate Curves":

Time after Shutdown = 1 hour (more conservative due to lower value for EAL)

1% fuel clad damage: dose rate = 30,000 R/hr

Extrapolating to 2.6%: (30,000 R/hr/1%)(2.6) = 78,000 R/hr

2.6% clad damage is based upon NUREG-1228 core damage analysis, and by virtue of its release into containment, the loss of the Reactor Coolant barrier (detailed calculations are contained in the Basis for Fission Product Barrier IC FC 1.d.1).

POTENTIAL LOSS – NONE

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

Containment Radiation Monitor Dose Rate Curves

Percent of Fuel Inventory Airborne in the Containment vs. Approximate Source and Damage Estimate

Curve No.	% Fuel* Inventory Released	Approximate Source and Damage Estimate
1	100.	100% TID-14844, 100% fuel damage, potential core melt.
	50.	50% TID noble gases, TMI source.
2	10.	10% TID, 100% NRC gap activity, total clad failure, partial core uncovered.
	3.	3% TID, 100% WASH-1400 gap activity, major clad failure.
3	1.	1% TID, 10% NRC gap, Max. 10% clad failure.
4	0.1	0.1% TID, 10% NRC gap, 1% clad failure, local heating of 5-10 fuel assemblies.
5	0.01	0.01% TID, 0.1% NRC gap, clad failure of 3/4 of a fuel assembly (36 rods) .
6	10^{-3}	0.01% NRC gap, clad failure of a few rods
	10^{-4}	100% coolant release with spiking.
7	$5X10^{-6}$	100% coolant inventory release.
	10^{-6}	Upper range of normal airborne noble gas activity in containment.

* 100% Fuel Inventory = 100% Noble Gases +25% Iodine + 1% particulates.

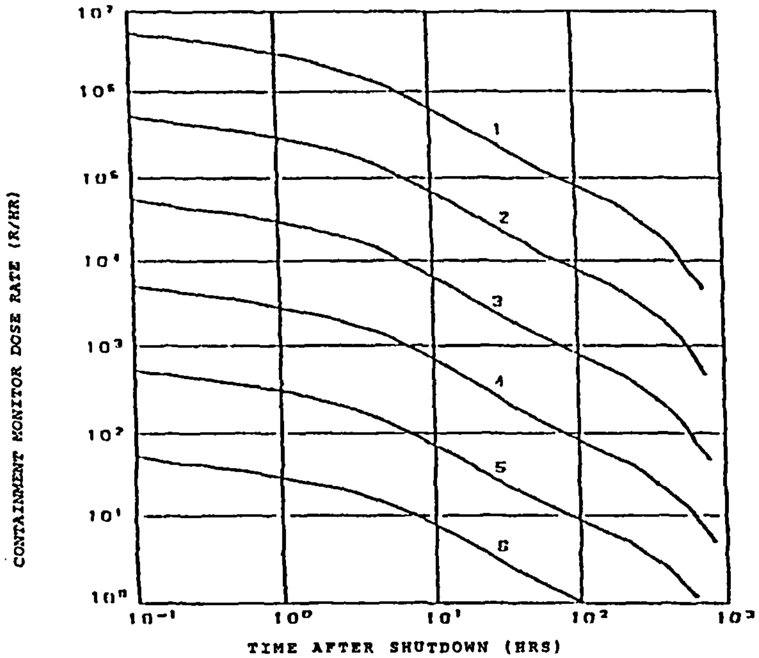
NOTES (1) These curves account for the finite containment volume and shield wall seen by the detector but do not account for any monitor physical or shielding characteristics or calibration uncertainties.

(2) The curves assume that both airborne noble gases and iodines are significant. Sprays (if used) would make the iodine and particulate contribution (presently about 50%) insignificant. However, particulate plate out on the monitor casing and direct shine doses from components may make the readings unreliable.

(3) Curve uncertainties are on the order of a factor of 5 to 10.

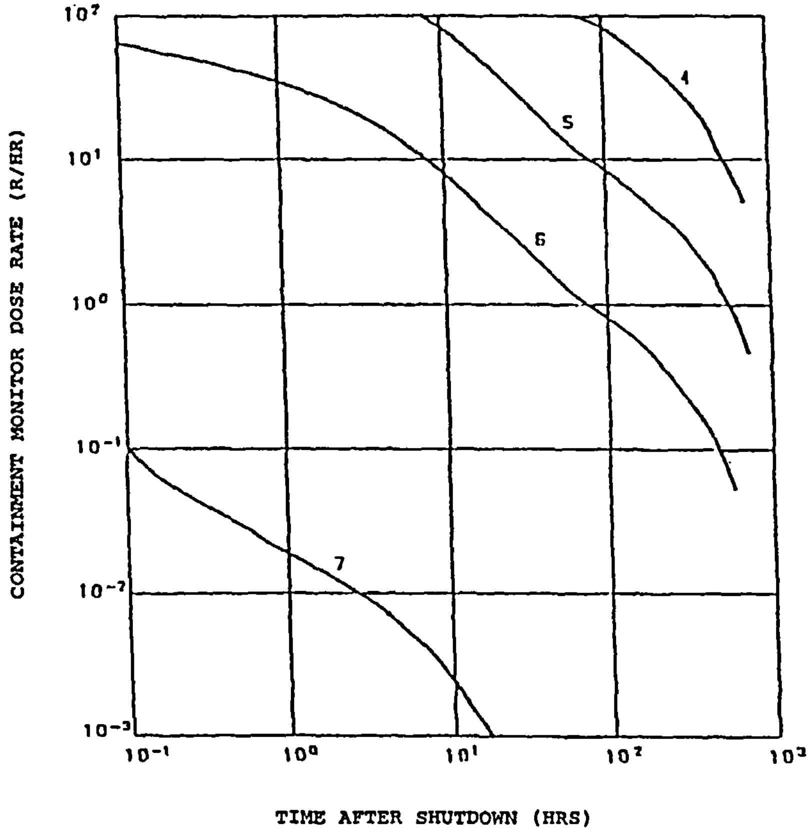
RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

PBAPS CONTAINMENT RADIATION MONITOR CURVES



PEACH BOTTOM MONITOR RESPONSE CURVES
FOR
PRIMARY CONTAINMENT LOW RANGE MONITORS

CURVE INDEX
1. 100% FUEL INVENTORY (100% TID-14844)
2. 10% FUEL INVENTORY (100% GAP ACTIVITY/R.G. 1.25)
3. 1% FUEL INVENTORY (10% NRC GAP-CLAD FAILURE)
4. 0.1% FUEL INVENTORY
5. 0.01% FUEL INVENTORY
6. 0.001% FUEL INVENTORY



PEACH BOTTOM MONITOR RESPONSE CURVES FOR PRIMARY CONTAINMENT LOW RANGE MONITORS

CURVE INDEX
4 - 0.1% FUEL INVENTORY
5 - .01% FUEL INVENTORY
6 - .001% FUEL INVENTORY
7 - 100% COOLANT ACTIVITY

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.c

INITIATING CONDITION

Drywell (DW) Pressure

THRESHOLD VALUE

Not Applicable

MODE APPLICABILITY

Not Applicable

BASIS: (References)

Not applicable

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.d

INITIATING CONDITION

Breached / Bypassed

THRESHOLD VALUE

LOSS: ....1. Coolant Activity > 300 µCi/gm I-131 dose equivalent [T04513]

OR

2. Core damage calculations indicate > 2.6% fuel clad damage

POTENTIAL LOSS: ....NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - A reactor coolant sample activity of greater than > 300 µCi/gm was determined to indicate significant clad heating and is indicative of the loss of the fuel clad barrier. This concentration is well above that expected for Iodine spikes and corresponds to 2.6% clad damage. 2.6% fuel clad damage is based upon NUREG-1228 core damage analysis.

Calculation of 300 µCi/cc equivalence to percent fuel clad damage is as follows: (For purposes of this calculation, cc and gm are considered equivalent.)

Iodine Isotope	Dose Factors (Reg Guide 1.109)	Ci/MWe Values (Time After Shutdown = 0) (NUREG-1228)
I-131	4.39E-3	85000
I-132	5.23E-5	120000
I-133	1.04E-3	170000
I-134	1.37E-5	190000
I-135	2.14E-4	150000

Time After Shutdown (T = 0) Ratios

R_{132} = 120000/85000(I-131) = 1.41(I-131)
R_{133} = 170000/85000(I-131) = 2.00(I-131)
R_{134} = 190000/85000(I-131) = 2.24(I-131)
R_{135} = 150000/85000(I-131) = 1.76(I-131)

Equation for Dose Equivalent Iodine (DEI_{131})

$$DEI_{131} = \frac{A_{131}DF_{131} + (R_{132})A_{131}DF_{132} + (R_{133})A_{131}DF_{133} + (R_{134})A_{131}DF_{134} + (R_{135})A_{131}DF_{135}}{DF_{131}}$$

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.d (Cont'd)

BASIS: (References) – Cont'd

$$300 = \frac{A_{131}4.39E-3 + 1.41\,A_{131}5.23E-5 + 2.00\,A_{131}1.04E-3 + 2.24\,A_{131}1.37E-5 + 1.76\,A_{131}2.14E-4}{4.39E-3}$$

$$300 = \frac{6.95E-3\,A_{131}}{4.39E-3}$$

Solve for A_{131} assuming DEI_{131} = 300 µCi/cc

Therefore: A_{131} = 189 µCi/cc I-131

Clad damage fraction (NUREG-1228, Table 4.1) = .02

Full Power = 1150 MWe

Clad Activity I-131 = (Ci/MWe) (MWe) (Clad Damage Fraction)
= (85000Ci/MWe) (1150MWe) (.02)
= 1.96E6 Ci

Reactor Water Volume = 2.67E8 cc (ERP-C-1410)

Total Coolant Activity I-131 = (A_{131}) (Rx Water Volume) (Ci/µCi)
= (189 µCi/cc) (2.67E8cc) (1.0E-6Ci/ µCi)
= 5.05E4Ci

Percent Clad Damage = Total Coolant Activity/Clad Activity I-131
= (5.05E4) / (1.96E6)
= 2.6%

<u>POTENTIAL LOSS</u> - NONE

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.e

INITIATING CONDITION

Drywell Hydrogen Concentration

THRESHOLD VALUE

Not Applicable

MODE APPLICABILITY

Not Applicable

BASIS: (References)

Not applicable

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.f

INITIATING CONDITION

Discretionary

THRESHOLD VALUE

LOSS: ....1. Any condition in the judgment of the Emergency Director that indicates a LOSS of the Fuel Clad barrier

POTENTIAL LOSS: ....2. Any condition in the judgment of the Emergency Director that indicates a POTENTIAL LOSS of the Fuel Clad barrier

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL, as a factor in Emergency Director judgment, that the barrier may be considered lost or potentially lost. (See also IC, MG1, "Prolonged Loss of ALL Offsite AC Power AND Prolonged Loss of ALL Onsite AC Power", for additional information.)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.a

INITIATING CONDITION

Reactor Pressure Vessel (RPV) Water Level

THRESHOLD VALUE

LOSS:....................................1. Reactor water level < -172 inches

POTENTIAL LOSS:............2. Reactor water level **CANNOT** be determined

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS – Core submergence is the preferred method of core cooling, and as such, the failure to re-establish RPV water level above TAF for an extended period of time could lead to significant fuel damage.

A level of < -172 inches also corresponds to the EAL for a Fuel Clad Barrier POTENTIAL LOSS (IC FC 1.a.2). Thus, this EAL indicates a LOSS of RCS barrier and a POTENTIAL LOSS of the Fuel Clad Barrier.

POTENTIAL LOSS - Inability to determine Reactor Pressure Vessel (RPV) level prevents assurance of adequate core cooling by methods that rely on being able to determine RPV water level (i.e., submergence or Minimum Steam Cooling RPV Water Level). TRIP procedures will provide criteria and strategies when RPV level CANNOT be determined.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.b

INITIATING CONDITION

Drywell (DW) High Range Rad Monitor

THRESHOLD VALUE

LOSS: .... 1. DW high range rad monitor reading > 15 R/hr

POTENTIAL LOSS: .... NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - The intent is not to verify criteria used in calculation (e.g., RCS breach), but rather to classify once EAL threshold value is reached or exceeded. This reading is less than that specified for a Fuel Clad Barrier LOSS (under IC Fuel Clad 1.b.1). Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading rises to that value specified under IC Fuel Clad 1.b.1, Fuel Clad damage would also be indicated.

[Calculation Basis] The 15 R/hr reading is a value, which indicates the release of reactor coolant to the drywell. The value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with concentrations corresponding to 0.001% Total Isotopic Distribution (TID) into the drywell atmosphere.

Using Curve 6 [0.001%] of the "Containment Radiation Monitor Dose Rate Curves" under Fuel Clad 1.b:

Time after Shutdown = 0.1 hour

0.001% TID = 17 R/hr

This is rounded to 15 R/hr for human factors considerations

POTENTIAL LOSS - NONE

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.c

INITIATING CONDITION

Drywell (DW) Pressure

THRESHOLD VALUE

LOSS:....................................1. Drywell pressure > 2.0 psig

AND

Indication of RCS leak inside Drywell

POTENTIAL LOSS: ....NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - The 2.0 psig drywell pressure is based on the drywell high pressure alarm set point and indicates a LOCA. If drywell pressure exceeds 2.0 psig, there is indication that a leak of sufficient magnitude exists that prevents drywell pressure stabilization.

Cycling of safety relief valves to reduce primary system overpressure when no fuel damage is indicated, is NOT considered reactor coolant leakage.

Primary containment pressure rises due solely to loss of containment heat removal capability are also NOT considered to exceed this threshold.

POTENTIAL LOSS- NONE

DEVIATION: The EAL, as stated in NUMARC/NESP-007, contains only the drywell pressure. A qualifier was added as a human factor reminder to the Emergency Director that use of this EAL is for accident scenarios only:

"**AND**

Indication of a RCS leak inside drywell"

Thus, a Drywell Pressure rise due to the loss of Drywell Cooling will not require an emergency classification.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.d

INITIATING CONDITION

Breached / Bypassed

THRESHOLD VALUE

LOSS: .... 1. Unisolable Main Steam Line (MSL) break as indicated by the failure of **BOTH** MSIVs in **ANY** one line to close

AND

EITHER of the following:

- High MSL Flow and High Steam Tunnel Temperature annunciators

  OR

- Direct report of steam release

OR

2. SRV is stuck open or cycling

   AND

   Indication of a LOSS of the Fuel Clad Barrier per the Fission Product Barrier Matrix

POTENTIAL LOSS:........... 3. RCS leakage > 50 gpm

OR

4. Unisolable primary system leakage outside Containment that results in:

   a. Exceeding **EITHER** of the following T-103 Action Levels:
      - Table SC/T-3 (Temperature)

        OR
      - Table SC/R-1 (Radiation)

   OR

   b. SCRAM initiated per T-103 due to temperature or radiation levels.

MODE APPLICABILITY

1, 2, 3

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.d (Cont'd)

BASIS: (References)

LOSS – [Threshold Value #1] Hi Steam Flow and Hi Steam Tunnel Temperature Annunciators are both indicators of a Main Steam Line Break. Both parameters will cause an isolation of the MSIV's. Should both valves in any one line fail to isolate, this event would be also considered a LOSS of Primary Containment (per IC 3.d.1) and appropriately classified as a Site Area Emergency.

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 visual 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.

Refer to MA8 for classification of an Alert due to an isolable Main Steam Line Break.

Design basis accident analyses of a Main Steam Line Break outside of secondary containment shows that even if MSIV closure occurs within design limits, dose consequences offsite from a "puff" release would be in excess of 10 mRem.

LOSS – [Threshold Value #2] Loss of the RCS Barrier based on an open safety relief valve (SRV) is dependent on other events. If an SRV is stuck open or cycling and no other emergency condition exists, an emergency declaration is not appropriate. However, if the fuel is damaged and the relief valve is allowing the fission products to escape into the Drywell (Containment), a LOSS of the RCS Barrier has occurred.

POTENTIAL LOSS – [Threshold Value #3] The potential loss of RCS based on leakage >50 gpm 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, a break propagation leading to a significantly larger loss of inventory is possible. RCS leakage is measured by the normal primary system leakage monitoring system and is leakage into the drywell. Under certain conditions, this system may be isolated due to elevated drywell pressure caused by the leak. In that case, a LOSS of RCS will be indicated and this "potential loss" of RCS would not impact the classification.

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 Value #4] Potential loss of RCS based on primary system leakage outside Containment is determined from site-specific area temperatures or radiation levels per T-103, which indicate a direct path from the RCS to areas outside primary containment. Initiation of a SCRAM per T-103 prior to reaching the Action Levels for area temperature and radiation meets the intent of EAL.

T-103 Action Levels based on area water level (Table SC/L-2) are evaluated separately for an Alert classification under IC HA3 (Natural or Destructive Phenomena Affecting a Vital Area).

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.d (Cont'd)

BASIS:(References) – Cont'd

Terms:

- Unisolable - Refers to a leak that cannot be isolated from the Control Room. When evaluating this EAL for unisolable primary system leakage, it is appropriate to attempt isolation from the Control Room prior to classification.
- Primary System - The pipes, valves and other equipment which connect directly to the RPV such that a reduction in RPV pressure will cause a drop in the flowrate of steam or water being discharged through a break in the system.
- Primary System Leakage - 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 (e.g. 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 BARRIER DEGRADATION

RCS 2.e

INITIATING CONDITION

Drywell Hydrogen Concentration

THRESHOLD VALUE

Not Applicable

MODE APPLICABILITY

Not Applicable

BASIS: (References)

Not applicable

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.f

INITIATING CONDITION

Discretionary

THRESHOLD VALUE

LOSS: 1. Any condition in the judgment of the Emergency Director that indicates a LOSS of the Reactor Coolant System barrier

POTENTIAL LOSS: 2. Any condition in the judgment of the Emergency Director that indicates a POTENTIAL LOSS of the Reactor Coolant System barrier

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. (See also IC, MG1, "Prolonged Loss of ALL Offsite AC Power AND Prolonged Loss of ALL Onsite AC Power" for additional information.)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.a

INITIATING CONDITION

Reactor Pressure Vessel (RPV) Water Level

THRESHOLD VALUE

LOSS: ....NONE

POTENTIAL LOSS: ....1. ANY of the following direct entry into SAMP-1 and SAMP-2:

- T-111
- T-116
- T-117

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS- NONE

POTENTIAL LOSS - The entry into SAMP-1 (RPV and Primary Containment Flooding) and SAMP-2 (Containment and Radioactivity Release Control) indicates that the reactor core cannot be adequately cooled and the Primary Containment is required to be flooded to submerge the core and preserve Primary Containment integrity. Concurrent entry and execution of SAMP-2 with SAMP-1 properly coordinates Primary Containment control functions with RPV and Primary Containment injection.
Entry into Severe Accident Management Procedures (SAMP) is directed by the TRIP procedures when adequate core cooling requirements cannot be satisfied and core damage has or may occur.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.b

INITIATING CONDITION

Drywell (DW) High Range Rad Monitor

THRESHOLD VALUE

LOSS: .... NONE

POTENTIAL LOSS:........... 1. Drywell high range rad monitor reading > 6.0E+5 R/hr

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - NONE

POTENTIAL LOSS – The intent is not to verify criteria used in calculation, but rather to classify once EAL threshold value is reached or exceeded.

[Calculation Basis] A drywell high range gamma radiation monitor RI-8(9)103A,B,C,D reading > 6.0E+5 R/hr indicates significant fuel damage, well in excess of that required for the loss of the RCS and Fuel Clad. As stated in Section 3.8 of NUMARC/NESP-007, a major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

The reading was calculated assuming an instantaneous release of the Reactor Coolant volume into the Primary Containment (direct reading not shine) where the value corresponds to a release of approximately 20% of the gap region. This calculation is as follows:

Using Curve 3 [1%] of the "Containment Radiation Monitor Dose Rate Curves" under Fuel Clad 1.b:

Time after Shutdown = 1 hour (more conservative due to lower value for EAL)

1% fuel clad damage: dose rate = 30,000 R/hr

Extrapolating to 20%: (30,000 R/hr/1%)(20) = 600,000 R/hr

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.c

INITIATING CONDITION

Drywell (DW) Pressure

THRESHOLD VALUE

LOSS:....................................1. Rapid, unexplained drop in DW pressure following an initial rise

OR

2. DW pressure response not consistent with LOCA conditions indicating a Containment breach

POTENTIAL LOSS:............3. DW pressure > 49 psig

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS – [Threshold Value #1] A rapid unexplained loss of Drywell pressure not due to use of containment sprays following an initial pressure rise indicates a loss of containment integrity.

LOSS – [Threshold Value #2] Drywell pressure should rise as a result of mass and energy release into the containment from a Loss of Coolant Accident (LOCA). Thus, Drywell pressure NOT rising under these conditions indicates a breach of containment integrity.

POTENTIAL LOSS - [Threshold Value #3] A Drywell pressure 49 psig is equal to the peak pressure expected from a Design Basis Accident (DBA) LOCA and 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.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.d

INITIATING CONDITION

Breached / Bypassed

THRESHOLD VALUE

LOSS:.................................. 1. Failure of ALL automatic isolation valves in ANY one line penetrating Primary Containment to close resulting from an isolation actuation signal

AND

Downstream pathway exists to the environment

OR

2. Intentional venting per T-200 is required

OR

3. Unisolable primary system leakage outside of Containment that results in:

   a. Exceeding **EITHER** of the following T-103 Action Levels:
      - Table SC/T-3 (Temperature)
        OR
      - Table SC/R-1 (Radiation)

   OR

   b. SCRAM initiated per T-103 due to temperature or radiation levels.

POTENTIAL LOSS:........... NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS – [Threshold #1] A failure of all Primary Containment isolation valves in any one line indicates a breach of the primary containment integrity as described in the primary containment Limiting Conditions for Operation. 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 operations, it will be addressed by a Technical Specification Action Statement. However, during accident conditions, this will represent a breach of Primary Containment. The criteria "from an automatic isolation actuation signal" is used to define that accident conditions are present and that isolation is required.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.d (Cont'd)

BASIS: (REFERENCES) – Cont'd

The breach is NOT isolable from the Control Room OR an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation from the Control Room should be made prior to the accident classification. If Operator actions from the Control Room are successful, then this IC is not applicable and REFER to IC MA8. 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.

LOSS – [Threshold #2] Intentional venting of the primary containment per T-200 procedures to the secondary containment and/or the environment is considered to be a breach of the primary containment for the purposes of accident classification.

LOSS – [Threshold #3] Loss of Primary Containment Barrier based on primary system leakage outside Containment is determined from site-specific area temperatures or radiation levels per T-103, which indicate a direct path from the RCS to areas outside primary containment. Initiation of a SCRAM per T-103 prior to reaching the Action Levels for area temperature and radiation meets the intent of the EAL.

T-103 Action Levels based on area water level (Table SC/L-2) are evaluated separately for an Alert classification under IC HA3 (Natural or Destructive Phenomena Affecting a Vital Area).

POTENTIAL LOSS - NONE

Terms:

- Unisolable - A leak that cannot be isolated from the Control Room. When evaluating this EAL for unisolable primary system leakage, it is appropriate to attempt isolation from the Control Room prior to classification.
- Primary System - The pipes, valves and other equipment which connect directly to the RPV such that a reduction in RPV pressure will cause a drop in the flowrate of steam or water being discharged through a break in the system.
- Containment Bypassed – The unintentional opening of, or leakage through, penetration isolations (e.g., equipment / personnel access hatches / airlocks, dampers / valves, etc.), such that a path to the environment exists.
- Primary System Leakage - 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 (e.g. 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 BARRIER DEGRADATION

CONTAINMENT 3.e

INITIATING CONDITION

Drywell Hydrogen Concentration

THRESHOLD VALUE

LOSS: .... NONE

POTENTIAL LOSS: .... 1. Drywell Hydrogen (H_2) > 6%
AND
Drywell Oxygen (O_2) > 5%

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - NONE

POTENTIAL LOSS - The specified value of 6% hydrogen and 5% oxygen concentration is the minimum, which can support a deflagration. Combustion of hydrogen in the deflagration concentration range creates a traveling flame causing a rapid rise in primary containment pressure. A deflagration may result in a peak primary containment pressure high enough to rupture the primary containment or damage the Drywell-to-Torus boundary.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.f

INITIATING CONDITION

Discretionary

THRESHOLD VALUE

LOSS:

1. Any condition in the judgment of the Emergency Director that indicates a LOSS of the Primary Containment barrier

POTENTIAL LOSS:

2. Any condition in the judgment of the Emergency Director that indicates a POTENTIAL LOSS of the Primary Containment barrier

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Primary Containment Barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in the Emergency Director's judgment that the barrier may be considered lost or potentially lost. See also IC, MG1, "Prolonged Loss of ALL Offsite AC Power AND Prolonged Loss of ALL Onsite AC Power" for additional information.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG1

INITIATING CONDITION

Prolonged Loss of **ALL** Offsite AC Power **AND** Prolonged Loss of **ALL** Onsite AC Power

EAL THRESHOLD VALUES

1. Loss of offsite power to **ALL** 4 KV Safeguard Busses

   AND

   ALL four of the 4 KV Safeguard Busses are de-energized for > 15 minutes

   AND

   ANY of the following:

   - Restoration of at least one 4 KV emergency bus in ≤ 2 hours is **NOT** likely

     OR

   - Reactor water level **CANNOT** be maintained > -172 inches

     OR

   - Torus temperature **CANNOT** be maintained on the **"SAFE"** side of the Heat Capacity Temperature Limit (HCTL) Curve (T-102, T/T-1)

MODE APPLICABILITY

1, 2, 3

BASIS (References)

When evaluating this EAL for Torus level outside of the Heat Capacity Temperature Limit Curve, High or Low, it is appropriate to consider the operation to be on the "UNSAFE" side.

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment. The two hours to restore AC power is based on the site blackout coping analysis as described below. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

10 CFR 50.2 defines Station Blackout (SBO) as complete loss of AC power to essential and non-essential buses. SBO does not include loss of AC Power to busses fed by station batteries through inverters, nor does it assume a concurrent single failure or design basis accident.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG1 – Cont'd

BASIS (References) – Cont'd

Successful extended SBO coping depends on ability to keep HPCI/RCIC available for injection, and ability to maintain RPV depressurized for low pressure injection should HPCI and RCIC become unavailable. 125V DC provides control power for HPCI, RCIC and SRVs.

The criteria for "Restoration of at least one 4 KV emergency bus in ≤ 2 hours" is based on Peach Bottom Calculation PE-017. From the Peach Bottom DBD, "The 125/250 VDC system battery capacity requirements are based on supplying DC power with the batteries as the sole source for two (2) hours following a LOOP / LOCA." PE-017 also identifies that the LOOP / LOCA battery loads are worst case and bound the SBO battery loads for these first 2 hours.

The significance of a station blackout relative to the loss of fission product release barriers is that all three barriers will eventually be lost due to the inability to remove heat from the fuel and the containment. Although the RCS will be intact the longest, eventually SRVs will operate in the relief mode due to RPV over-pressurization and if the containment has already failed then there is a direct bypass of the RCS boundary.

Implementation of this EAL is based on the number of powered 4 KV buses per unit.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS1

INITIATING CONDITION

Loss of **ALL** Offsite AC Power **AND** Loss of **ALL** Onsite AC Power to Essential Busses

EAL THRESHOLD VALUES

1. Loss of offsite power to ALL 4 KV Safeguard Busses

   AND

   ALL four of the 4 KV Safeguard Busses are de-energized for > 15 minutes

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Control Room annunciators would indicate that all offsite and onsite AC power feeds have been lost. Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal, High Pressure Service Water, and Emergency Service Water. Although instrumentation (supplied through instrument inverters) and DC power systems would be available, their operability would be limited to the amount of stored energy contained in their respective batteries. Instrumentation, communication equipment, and in-plant lighting and ventilation will be significantly hampered by the loss of all AC power.

Fifteen (15) minutes has been selected to allow adequate time to cross tie or address diesel generator failures and to exclude transient or momentary power losses. It is not necessary to wait for 15 minutes if it can be determined in less than 15 minutes that the power loss is not transient or momentary.

Implementation of this EAL is based on the number of powered 4 KV buses per unit.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA1

INITIATING CONDITION

AC Power to Essential Busses Reduced to a Single Source for > 15 minutes

EAL THRESHOLD VALUE

1. Loss of offsite power to ALL 4 KV Safeguard Busses

   AND

   Three of four of the 4 KV Safeguard Busses are de-energized for > 15 minutes

MODE APPLICABILITY

1, 2, 3

BASIS (References)

The reduction of available reliable power sources to a condition where ANY additional single failure will result in a station blackout is a substantial degradation in the level of safety of the plant. That is, the Unit is down to its last source of AC power. Loss of the single power supply would escalate to a SITE AREA EMERGENCY via IC MS1.

This EAL is intended to provide an escalation from "Loss of offsite Power for greater than 15 minutes." This condition is a degradation of the offsite and onsite power systems such that any additional failure would result in a station blackout. Fifteen (15) minutes has been selected to allow adequate time to cross tie or address diesel generator failures and to exclude transient or momentary power losses. However, an Alert should be declared in less than 15 minutes if it can be determined in less than 15 minutes that the power loss is not transient or momentary.

Depending on the 4 KV AC bus that remains energized there is a disparity in the systems that may be available. The ability to remove heat from the containment via Torus cooling may be lost due to the need to operate the remaining available RHR pump in other than Torus cooling (e.g., LPCI). As such there is a decrease in the systems available to remove heat transferred to the containment and there is an ongoing release of energy from the reactor to the containment (via SRVs, HPCI and/or RCIC operation). The ability to cool the nuclear fuel, remove decay heat, and control containment parameters is severely limited. Should equipment be unavailable prior to the loss of power, functions necessary to maintain the plant in a cold shutdown condition may be threatened.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU1

INITIATING CONDITION

Loss of ALL Offsite AC Power for > 15 minutes to Essential Busses

EAL THRESHOLD VALUE

1. Loss of offsite power to ALL 4 KV Safeguard Busses for >15 minutes

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned – Not the result of an intended evolution and requiring corrective or mitigative actions

This EAL addresses the loss of offsite AC power supplying the station. Offsite power is fed through 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer. Loss of offsite power will cause a reactor scram and a containment isolation. All four (4) emergency Diesel Generators will be available to carry the essential loads for each unit (the four Diesel Generators are shared between each unit). Balance of Plant systems that would assist in plant operations (i.e., condensate pumps, etc.) may be unavailable due the loss of power.

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (Station Blackout). Fifteen minutes was selected as a threshold to allow adequate time to cross tie or address diesel generator failures and to exclude transient or momentary power losses.

The Emergency Director must also consider the impact to the unaffected unit due to the loss of power to balance of plant equipment on common or shared systems.

Escalation of this event to an Alert would be based on having a loss of all offsite AC power coincident with onsite AC power being reduced to a single power source in Modes 1, 2, and 3 or having a loss of all offsite and onsite AC power in Modes 4 or 5.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA2

INITIATING CONDITION

Loss of **ALL** Offsite AC Power **AND** Loss of **ALL** Onsite AC Power to Essential Busses

EAL THRESHOLD VALUES

1. Loss of offsite power to **ALL** 4 KV Safeguard Busses

   AND

   ALL four of the 4 KV Safeguard Busses are de-energized for > 15 minutes

MODE APPLICABILITY

4, 5, Defueled

BASIS (References)

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode, the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, raising the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL.

Fifteen (15) minutes has been selected to allow adequate time to cross tie or address diesel generator failures and to exclude transient or momentary power losses. However, an Alert should be declared in less than 15 minutes if it can be determined in less than 15 minutes that the power loss is not transient or momentary.

Implementation of this EAL is based on the number of powered 4 KV buses per unit.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS3

INITIATING CONDITION

Loss of ALL Required T.S. Safety-Related 125 VDC Power Sources

EAL THRESHOLD VALUE

1. Loss of ALL required T.S. safety related 125 VDC power sources for > 15 minutes as indicated by < 107.5 VDC on Panels 2(3)0D21, 22, 23, 24.

MODE APPLICABILITY

1, 2, 3

BASIS (References)

A loss of all DC power compromises the ability to monitor and control plant functions. 125 Volt DC system provides control power to engineered safety features valve actuation, diesel generator auxiliaries, plant alarm and indication circuits as well as the control power for the associated load group. If 125 Volt DC power is lost for an extended period of time (greater than 15 minutes) critical plant functions required to maintain safe plant conditions may not operate and core uncovery with subsequent reactor coolant system and primary containment failure might occur. Refer to SE-13, "Loss of a 125 or 250 VDC Safety-Related Bus".

107.45 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment. This EAL uses 107.5 VDC for human factors concerns. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate those loads.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU3

INITIATING CONDITION

Loss of ALL Required T.S. Safety-Related 125 VDC Power Sources

EAL THRESHOLD VALUE

1. Loss of ALL required T.S. safety-related 125 VDC power sources for > 15 minutes as indicated by < 107.5 VDC on DC Panels 2(3)0D21, 22, 23, 24.

MODE APPLICABILITY

4, 5

BASIS (References)

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

107.45 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment. The value of 107.5 VDC will be used for human factors concerns. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate those loads.

Unplanned is included in this IC and EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, plants will perform maintenance on a Train related basis during shutdown periods. It is intended that the loss of a required train be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will occur.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG4

INITIATING CONDITION

Auto and Manual SCRAM **NOT** Successful, **AND** Loss of Core Cooling or Heat Sink

EAL THRESHOLD VALUES

1. Failure of automatic RPS, ARI and Manual SCRAM/ARI to shutdown the reactor as defined by **EITHER** of the following criteria:
   - Reactor Power > 4%

     OR
   - Torus temperature is greater than 110°F **AND** boron injection is required

   AND

   EITHER of the following criteria are met:
   - Torus Temperature **CANNOT** be maintained on the "**SAFE**" side of the Heat Capacity Temperature Limit (HCTL) curve (T-102, T/T-1)

     OR
   - Reactor water level < -195 inches

MODE APPLICABILITY:

1, 2

BASIS: (References)

Manual SCRAM - Any set of actions by the reactor operators at the reactor control console which causes control rods to be insert sufficiently to reduce reactor power to a condition where it will remain shutdown under all conditions without the use of boron injection (i.e., mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

Automatic actuation of the ARI system is a backup to the MANUAL SCRAM and, as a result, does not constitute a successful MANUAL SCRAM.

Boron Injection Initiation Temperature (BIIT) is defined as 110°F Torus temperature per the T-101 Basis for "RPV Control".

This EAL is not applicable if a manual scram is initiated and no RPS set points are exceeded. Taking the mode switch to shutdown is considered a manual scram action. Note that although placing the Mode Switch in "shutdown" is a manual scram action, when the Mode Switch passes through the "startup / hot standby" position the Nuclear Instrumentation Scram Setpoint is lowered. If reactor power is greater than the setpoint, an automatic scram will be initiated. If the RPS then fails to initiate a scram, then this should be evaluated as an automatic RPS set point being exceeded and a failure of the automatic scram.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG4 – Cont'd

BASIS: (References) – Cont'd

A valid automatic and/or manual scram signal is present as indicted by control room indications and/or alarms. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, reactor pressure, Torus temperature trend) can be used to determined if reactor power is greater than 4% power. The Reactor Protection System (RPS) is designed to function to shut down the reactor (either manually or automatically). The RPS system is "fail safe," that is, it de-energizes to function. An Anticipated Transient Without Scram (ATWS) event can be caused by either a failure of RPS (electrical ATWS) or the Control Rod Drive system to insert the control rods (hydraulic ATWS).

The TRIP procedures establish 4% power (APRM downscales) as a power level sufficient to challenge Primary Containment heat removal capabilities should this energy be directed to the Primary Containment. If APRM downscale setpoint is achieved, but Torus temperature is greater than Boron Injection Temperature, a precursor exists for a threat to Primary Containment.

In addition, control room instrumentation indicates that operation is on the "UNSAFE" side of the HCTL Curve (T-102, T/T-1) or RPV level is < -195 inches. When Torus level is outside of the Heat Capacity Temperature Limit (HCTL) Curve (High or Low), it is appropriate to consider operation to be on the "UNSAFE" side. Failure of all automatic and manual trip functions coincident with a high Torus temperature will place the plant in a condition where reactivity control capability is jeopardized and heat removal capability is severely limited. RPV level less than -195 inches corresponds to the level, which is used in the TRIP procedures to indicate a challenge to adequate core cooling.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS4

INITIATING CONDITION

Auto and Manual SCRAM **NOT** Successful

EAL THRESHOLD VALUE

1. Failure of RPS, ARI and Manual SCRAM/ARI to shutdown the reactor as defined by **EITHER** of the following criteria:
   - Reactor Power > 4%

   OR
   - Torus temperature is greater than 110°F **AND** boron injection is required

MODE APPLICABILITY:

1, 2

BASIS: (References)

Manual SCRAM - Any set of actions by the reactor operator(s) at the reactor control console which causes control rods to insert sufficiently to reduce reactor power to a condition where it will remain shutdown under all conditions without the use of boron injection (i.e., mode switch to shutdown, manual scram push buttons, or manual ARI initiation).

Automatic actuation of the ARI system is a backup to the MANUAL SCRAM and, as a result, does not constitute a successful MANUAL SCRAM.

Boron Injection Initiation Temperature (BIIT) is defined as 110°F Torus temperature per the T-101 Basis for "RPV Control".

This EAL is not applicable if a manual scram is initiated and no RPS set points are exceeded. Taking the mode switch to shutdown is considered a manual scram action. Note that although placing the Mode Switch in "shutdown" is a manual scram action, when the Mode Switch passes through the "startup / hot standby" position the Nuclear Instrumentation Scram Setpoint is lowered. If reactor power is greater than the setpoint, an automatic scram will be initiated. If the RPS then fails to initiate a scram, then this should be evaluated as an automatic RPS set point being exceeded and a failure of the automatic scram.

A valid automatic and/or manual scram signal is present as indicated by control room indications and/or alarms. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, reactor pressure, Torus temperature trend) can be used to determined if reactor power is greater than 4% power. The Reactor Protection System (RPS) is designed to function to shut down the reactor (either manually or automatically). The RPS system is "fail safe," that is, it de-energizes to function. An Anticipated Transient Without Scram (ATWS) event can be caused by either a failure of RPS (electrical ATWS) or the Control Rod Drive system to insert the control rods (hydraulic ATWS).

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS4 – Cont'd

BASIS: (References) – Cont'd

The TRIP procedures establish 4% power (APRM downscales) as a power level sufficient to challenge Primary Containment heat removal capabilities should this energy be directed to the Primary Containment. If APRM downscale setpoint is achieved, but Torus temperature is greater than Boron Injection Temperature, a precursor exists for a threat to Primary Containment and thus a Site Area Emergency is warranted.

This event escalation is based on rising Torus temperature or lowering RPV water level that would result in the loss of containment integrity and the inability to remove the heat generated from the fuel per MG4.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA4

INITIATING CONDITION

Auto SCRAM NOT Successful

EAL THRESHOLD VALUE

1. RPS set point has been exceeded for an automatic SCRAM

   AND

   Failure of automatic RPS to achieve a state in which the reactor is shutdown under all conditions without boron injection

MODE APPLICABILITY

1, 2

BASIS (References)

This EAL is not applicable if a manual scram is initiated <u>and no RPS set points are exceeded</u>. Taking the mode switch to shutdown is considered a manual scram action. Note that although placing the Mode Switch in "shutdown" is a manual scram action, when the Mode Switch passes through the "startup / hot standby" position the Nuclear Instrumentation Scram Setpoint is lowered. If reactor power is greater than the setpoint, an automatic scram will be initiated. If the RPS then fails to initiate a scram, then this should be evaluated as an automatic RPS set point being exceeded and a failure of the automatic scram.

Entry into this EAL is based on a reactor parameter actually exceeding a RPS set point and the reactor is not brought to a state in which the reactor is shutdown under all conditions without boron injection and maintained at that state with automatic RPS functions. The parameter must exceed the RPS set point by a significant margin eliminating minor set point drifts, which are accounted for in the Technical Specification Margin of Safety. Subsequent manual scram actions were successful in bringing the reactor to a state in which the reactor is shutdown under all conditions without boron injection. Confirmation indications include control room annunciators, APRM/WRNM power level, Reactor Period, and Control rod position indication.

When partial control rod insertion occurs following a scram signal (either manual or automatic) judgment should be applied as to whether the Reactor will remain "Shutdown Under All Conditions Without Boron" and if classification should occur. Multiple control rods failing to insert beyond the Maximum Subcritical Banked Withdraw Position (MSBWP) may require actions to fully insert the control rods. TRIP guidance will govern the insertion of these control rods.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA4 – Cont'd

BASIS (References) – Cont'd

This condition is more than a potential degradation of a safety system in that a front line automatic protection system does 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 the Fuel Clad or RCS Barriers.

A scram is considered unsuccessful if it does not result in achieving a state in which the reactor will remain shutdown under all conditions without boron injection.

This EAL would be escalated to a Site Area Emergency with a failure of both manual and automatic scram signals and either Reactor power > 4% (APRM downscales) or Torus temperature greater than Boron Injection Temperature.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS5

INITIATING CONDITION

Complete Loss of Functions Needed to Achieve AND Maintain Hot Shutdown

EAL THRESHOLD VALUES

1. Loss of functions required for Hot Shutdown as evidenced by T-102 T/T leg, directing a T-112 Emergency Blowdown

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other EALs. The loss of heat removal function is indicated by T-102 T/T leg requiring an Emergency Blowdown, which is directed when the Heat Capacity Temperature Limit (HCTL) curve is exceeded.

The EAL is concerned with Torus temperature. It is not appropriate to make a Site Area Emergency classification for the condition where the T-102 Torus Level (T/L) leg alone directs a T-112 Emergency Blowdown since the Emergency Blowdown is performed PRIOR to those Torus levels which may cause a loss of containment capability due to uncovering downcomers or excessive SRV tailpipe stresses.

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. Escalation to General Emergency would be via Effluent Release/In-Plant Radiation, Emergency Director Judgment, or Fission Product Barrier Degradation ICs.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5

INITIATING CONDITION

Inability to Maintain Plant in Cold Shutdown

EAL THRESHOLD VALUE

1. Unplanned loss of ALL T.S. required decay heat removal systems
   AND
   EITHER of the following:
   - RCS temperature exceeding 212 °F for $\geq$ 15 minutes with a heat removal function restored
     OR
   - Uncontrolled RCS temperature rise approaching 212 °F with NO heat removal function restored

MODE APPLICABILITY

4, 5

BASIS (References)

Uncontrolled - A temperature rise that is not the result of a planned evolution

This EAL addresses complete loss of functions required for core cooling during refueling and cold shutdown modes. A loss of Technical Specifications components is paired with exceeding temperature limits to acknowledge additional plant capabilities to maintain plant cooling. Escalation to Site Area Emergency or General Emergency would be via Effluent Release/In-Plant Radiation or Emergency Director Judgment ICs.

The statement "Temporary Loss of ALL Tech Spec Required Decay Heat Removal Systems" is intended to represent a complete loss of functions available, or an inadequate ability, to provide core cooling during the Cold Shutdown and Refueling Modes, including alternate decay heat removal methods. This EAL allows for actions taken in ON-125, "Loss of Shutdown Cooling - Procedure," to reestablish RHR in the Shutdown Cooling Mode or provide for alternate methods of decay heat removal, with the intent of maintaining RCS temperature below 212° F.

For loss of an in-service Decay Heat Removal system with other decay heat removal methods available, actions taken to provide for restoration of a decay heat removal function may require time to implement. If the event results in RCS temperature "momentarily" (for less than 15 minutes) rising above 212°F with heat removal capability restored, Emergency Director/Shift Management judgment will be required to determine whether heat removal systems are adequate to prevent boiling in the core and restoration of RCS temperature control.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5 – Cont'd

BASIS (References) – Cont'd

Momentary (not to exceed 15 minutes) unplanned excursions above 212° F, when alternate decay heat removal capabilities exist, should not be classified under this EAL.

The EAL guidance related to uncontrolled temperature rise is necessary to preserve the anticipatory philosophy of NUREG-0654 for events starting from temperatures much lower than the cold shutdown temperature limit.

This EAL is concerned with the ability to keep the reactor core temperature less than 212 °F. The criteria of uncontrolled Reactor Coolant temperature rise > 212 °F is met as soon as it becomes known that sufficient cooling cannot be restored in time to maintain the temperature < 212 °F, regardless of the current temperature. The inability to establish alternate methods of decay heat removal indicates that either alternate methods are unavailable to cool the core in the RPV or when the steam is transferred to the Torus, Torus cooling is unavailable. Loss of Torus cooling will result in a continuing, uncontrolled rise in reactor coolant temperature.

Special Test Exception 3.10.8 allows for temperature to rise above 212 °F during hydrostatic testing. The limit of 212 °F in this EAL does not apply under those conditions as that is not an "Uncontrolled Temperature rise."

Escalation to the Site Area Emergency is by EAL MS7, "Loss of Water Level in the Reactor Vessel that has or will uncover Fuel in the Reactor Vessel," or by Effluent Release/In-Plant Radiation RS1.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6

INITIATING CONDITION

Inability to Monitor a Significant Transient in Progress

EAL THRESHOLD VALUE

1. A significant plant transient is in progress (Table M-1)

   AND

   ALL of the following are lost:

   - Safety system annunciators (Table M-2)
   - Safety function indicators (Table M-3)
   - Plant Monitoring System

Table M-1 Significant Plant Transients	Table M-2 Safety System Annunciators	Table M-3 Safety Function Indicators
• SCRAM • Recirc Runbacks (> 25% thermal power change) • Sustained Power Oscillations (25% peak to peak) • Stuck open relief valves • ECCS Injection	• ECCS • Containment Isolation • Reactor Trip • Process Radiation Monitoring	• Reactor Power • Decay Heat Removal • Containment Safety Functions

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

This EAL recognizes the difficulty associated in monitoring conditions without normal annunciators. In the opinion of the Shift Supervisor, this loss of annunciators requires increased surveillance to safely operate the plant. This EAL represents an increase in severity above MA6 in that the Plant Monitoring System can not provide compensatory indication, and that a significant transient is in progress.

Planned maintenance or testing activities are included in this EAL due to the significance of this event. Control Room panels with annunciators and the restoration is included in ON-123, Loss of Control Room Annunciators.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6

INITIATING CONDITION

Loss of Annunciators OR Indicators Requiring Increased Surveillance

EAL THRESHOLD VALUES

1. Unplanned loss for > 15 minutes of **MOST** or **ALL** of **EITHER**:
   - Safety system annunciators (Table M-2)
     OR
   - Safety function indicators (Table M-3) for > 15 minutes

   AND
   Increased surveillance is required to safely operate the unit(s)

   AND
   EITHER of the following:
   - A significant plant transient is in progress (Table M-1)
     OR
   - Plant Monitoring System is unavailable

Table M-1 Significant Plant Transients	Table M-2 Safety System Annunciators	Table M-3 Safety Function Indicators
• SCRAM • Recirc Runbacks (> 25% thermal power change) • Sustained Power Oscillations (25% peak to peak) • Stuck open relief valves • ECCS Injection	• ECCS • Containment Isolation • Reactor Trip • Process Radiation Monitoring	• Reactor Power • Decay Heat Removal • Containment Safety Functions

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

MOST - 75% of safety system annunciators or indicators are lost or a significant risk that a degraded plant condition could go undetected exists. The use and definition of MOST is not intended to require a detailed count of lost annunciators or indicators but should be used as a guide to assess the ability to monitor the operation of the plant.

UNPLANNED - Loss of annunciators or indicators is not the result of scheduled maintenance or testing.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6 – Cont'd

BASIS: (References) – Cont'd

This EAL recognizes the difficulty associated in monitoring conditions without normal annunciators. It is not intended that a detailed count of instrumentation be performed, but by the use of the judgment of the Shift Supervisor as the threshold for determining the severity of the plant conditions. This judgment is supported by the specific opinion of the Shift Supervisor that additional operating personnel will be required to provide increased monitoring of systems needed to safely operate the plant.

This EAL represents an increase in severity above MU6 in that the Plant Monitoring System (PMS) cannot provide compensatory indication, or that a significant transient is in progress.

Fifteen minutes is used as a threshold to exclude transient or momentary power loses. Control Room panels with annunciators and direction for restoration is included in ON-123, Loss of Control Room Annunciators.

This EAL is not applicable in cold shutdown or refueling modes due to the limited number of safety systems required for operation.

This event will be escalated to a Site Area Emergency if a transient is in progress, the Plant Monitoring System is unavailable, and a loss of annunciators occurs.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU6

INITIATING CONDITION

Unplanned Loss of Annunciators OR Indicators for > 15 minutes

EAL THRESHOLD VALUE

1. Unplanned loss for > 15 minutes of **MOST** or **ALL** of **EITHER:**
   - Safety system annunciators (Table M-2)
   OR
   - Safety function indicators (Table M-3)

   AND

   Increased surveillance is required to safely operate the unit(s)

Table M-2 Safety System Annunciators	Table M-3 Safety Function Indicators
• ECCS • Containment Isolation • Reactor Trip • Process Radiation Monitoring	• Reactor Power • Decay Heat Removal • Containment Safety Functions

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

MOST - 75% of safety system annunciators or indicators are lost **OR** a significant risk that a degraded plant condition could go undetected exists. The use and definition of MOST is not intended to require a detailed count of lost annunciators or indicators but should be used as a guide to assess the ability to monitor the operation of the plant.

UNPLANNED - Loss of annunciators or indicators is NOT the result of scheduled maintenance or testing.

This EAL recognizes the difficulty associated in monitoring conditions without normal annunciators. It is not intended that a detailed count of instrumentation be performed, but by the use of the judgment of the Shift Supervisor as the threshold for determining the severity of the plant conditions. This judgment is supported by the specific opinion of the Shift Supervisor that additional operating personnel will be required to provide increased monitoring of systems needed to safely operate the plant. The Plant Monitoring System (PMS) is available to provide compensatory indication. Fifteen minutes is used as a threshold to exclude transient or momentary power losses. Unplanned loss of annunciators excludes scheduled maintenance and testing activities. Control Room panels with annunciators and direction for response are included in ON-122, Loss of Main Control Room Annunciators.

This EAL is not applicable in cold shutdown or refueling modes due to the limited number of safety systems required for operation.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS7

INITIATING CONDITION

Loss of Water Level in the Reactor Vessel That Has **OR** Will Uncover Fuel in the Reactor Vessel

EAL THRESHOLD VALUES

1. Reactor water level < -172 inches

MODE APPLICABILITY

4, 5

BASIS (References)

The indicator for "core is or will be uncovered" is Reactor Pressure Vessel Water level below the Top of Active Fuel (TAF), -172 inches as indicated on RPV Fuel Zone Level Instruments LI-2(3)-02-3-091 or LI-2(3)-02-3-113. Core submergence ensures adequate core cooling. When RPV level drops below the top of active fuel, the ability to remove the decay heat generated from the nuclear fuel becomes suspect and the Fuel Clad Fission Product barrier can no longer be considered intact. Sustained partial or total core uncovery can result in the release of a significant amount of fission products to the reactor coolant.

Under the conditions specified by this IC, severe core damage can occur and reactor coolant system pressure boundary integrity may not be assured. It is intended to address concerns raised by NRC Office for Analysis and Evaluation of Operational Data (AEOD) report AEOD/EG09, "BWR Operating Experience Involving Inadvertent Draining of the Reactor Vessel," dated August 8, 1986. This report states:

> In broadest terms, the dominant causes of inadvertent reactor vessel draining are related to the operational and design problems associated with the residual heat removal system when it is entering into or exiting from the shutdown cooling mode. During this transitional period, water is drawn from the reactor vessel, cooled by the residual heat removal system heat exchangers (from the cooling provided by the service water system), and returned to the reactor vessel. First, there are piping and valves in the residual heat removal system, which are common to both the shutdown cooling mode and other modes of operation such as low pressure coolant injection and Torus cooling. These valves, when improperly positioned, provide a drain path for reactor coolant to flow from the reactor vessel to the Torus or the radwaste system. Second, there is no comprehensive valve interlock arrangement for all shutdown cooling. Collectively, these factors have contributed to the inadvertent draining of the reactor vessel.

Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via effluent release EAL.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS7 – Cont'd

BASIS (References) – Cont'd

DEVIATION: During EAL review and approval process, it was determined that the condition stated in NUMARC NESP-007, SS5, 1.a "Loss of all decay heat removal cooling as determined by (site-specific) procedure" is not necessary to conclude that the plant condition warrants a Site Area Emergency. Therefore, that sample NUMARC EAL was not included in this EAL.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU7

INITIATING CONDITION

Reactor Coolant System leakage

EAL THRESHOLD VALUES

1. Unidentified primary system leakage > 10 gpm into the Drywell

   OR

2. Identified primary system leakage > 25 gpm into the Drywell

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Utilizing the leak before break methodology, it is anticipated that there will be indication(s) of minor reactor coolant system boundary integrity loss prior to this fault escalating to a major leak or rupture. Detection of low levels of leakage while pressurized is utilized to monitor for the potential of catastrophic failures. Leakage not associated with catastrophic failure potential such as SRV leakage, should not be considered in this EAL.

Identified and unidentified Primary System Leakage is measured by the normal primary system leakage monitoring system and is leakage into the drywell.

This EAL is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, it is considered to be a potential degradation of the level of safety of the plant. The value of 10 gpm unidentified leakage is significantly higher than the expected pressurized leak rate from the reactor coolant system. The 10 gpm value for the unidentified pressure boundary leakage was selected as it is twice the Technical Specification value, indicating an increase beyond that assumed in Safety Analysis. It also is observable with normal control room indications. The EAL for identified leakage is set at a higher value (25 gpm) due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.

Technical Specification LCO required actions would necessitate a plant shutdown and subsequent depressurization, unless the source of the leak can be isolated, identified, and/or stopped. Actions initiated by plant staff would include close monitoring of the calculated break size such that any sudden or gradual rise in leak rate would be identified. A slow power reduction and gradual depressurization would be necessitated due to the possibility that a sudden power and/or pressure surge could potentially worsen the break or cause a catastrophic failure.

The leak rate of 10 gpm may cause a high drywell pressure indication. Other indications of a leak of this magnitude would include a rise in drywell temperature or radiation.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU7 – Cont'd

BASIS (References) – Cont'd

This event will escalate to an Alert based upon high Drywell pressure per Fission Product Barrier Matrix 2.c.1.

DEVIATION: NUMARC/NESP-007 Example EAL SU5.1.a identifies pressure boundary leakage. There is no Peach Bottom EAL listed for pressure boundary leakage specifically since it is a subset of unidentified leakage. Peach Bottom Tech. Specs. require a shutdown if any pressure boundary leakage is found.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA8

INITIATING CONDITION

Main Steam Line Break

EAL THRESHOLD VALUES

1. MSL Break indicated by EITHER of the following:
   - High MSL Flow and High Steam Tunnel Temperature annunciators
     OR
   - Direct report of steam release

   AND

   MSL break is successfully isolated.

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Design basis accident analyses of a Main Steam Line Break outside of secondary containment shows that even if MSIV closure occurs within design limits, dose consequences offsite from a "puff" release would be in excess of 10 mRem.

Hi Steam Flow Annunciator and Hi Steam Tunnel Temperature Annunciator are both indicators of a Main Steam Line Break. Both parameters will cause an isolation of the MSIV's. Should both valves in any one line fail to isolate, this event would be also considered a LOSS of Primary Containment (per IC 3.d.1) and as loss of RCS (per IC 2.d.1). This would then appropriately be classified as a Site Area Emergency.

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 visual 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.

Loss of the RCS Barrier due to an unisolable MSL break is covered under Fission Product Barrier Matrix (IC 2.d.1).

DEVIATION: NUMARC/NESP-007, Table 3 (RC Example EAL #1) EAL placed as a separate Alert threshold under previous NRC submittal to cover an isolable MSL break outside secondary containment. If the Main Steam Line (MSL) isolates as designed, this condition does not constitute a loss of RCS barrier. However, this condition was included as an event-based EAL due to the potential dose consequences associated with this event. This is consistent with the recommendations provided in the Industry-developed Questions and Answers on NUMARC/NESP-007 guidance, which was endorsed by the NRC in a letter dated June 10, 1993.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU9

INITIATING CONDITION

Unplanned loss of **ALL** onsite **OR** offsite communications capabilities

EAL THRESHOLD VALUES

1. **ALL** onsite communications equipment lost (Table M-4)

   OR

2. **ALL** offsite communications capability lost (Table M-5)

Table M-4 Onsite Communications Equipment	Table M-5 Offsite Communications Equipment
• Station Phones • OMNI System • Plant Public Address (PA) • Station Radio	• Station Phones • OMNI System • NRC (ENS) • PA State Radio • Load Dispatcher Radio

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned - The loss of communication is not a result of planned maintenance or surveillance activities

This EAL recognizes a loss of communication ability that significantly degrades the plant operations staff's ability to perform tasks necessary for plant operations or the ability to communicate with offsite authorities. This EAL is separated into two groups of communications, Onsite and Offsite. A complete loss of either group is so severe, that the Unusual Event declaration is warranted.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU10

INITIATING CONDITION

Inability to reach required operating mode within Technical Specification time limits

EAL THRESHOLD VALUE

1. Inability to reach required operating mode within Tech. Spec. LCO action completion time

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required operating mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate Notification of an Unusual Event is required when it is determined that the there is an inability to bring the plant to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of an Unusual Event is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other various ICs address other required Technical Specification shutdowns that involve precursors to more serious events.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA11

INITIATING CONDITION

Major Damage **OR** Uncovering of Spent Fuel

EAL THRESHOLD VALUES

1. Unplanned general area radiation > 500 mR/hr on the Refuel Floor (Table M-6)

   OR

2. Report or visual observation that irradiated fuel is uncovered

   OR

3. Water level < 232 ft. 3 inches Plant elevation for the Spent Fuel Pool that will result in irradiated fuel uncovering

Table M-6 Refuel Floor ARMs
• 3-7 (7-9), Steam Separator Pool • 3-8 (7-10), Refuel Slot • 3-9 (7-11), Fuel Pool • 3-10 (7-12), Refueling Bridge

MODE APPLICABILITY

ALL

BASIS: (References)

Offsite doses during these accidents would be well below the EPA Protective Action Guidelines and the classification as an Alert is therefore appropriate. This radiation level could also be caused by an inadvertent criticality and is included even though the probability of this event occurring is low. Radiation levels rise above 500 mR/hr, which were expected during a planned evolution, should not cause an Alert to be declared. Additionally, surveys, which identify "hot spots" greater than 500 mR/hr should not cause an Alert to be declared.

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in NUMARC/NESP-007 IC AU2, "Unexpected Rise in Plant Radiation or Airborne Concentration."

NUREG-0818, "Emergency Action Levels for Light Water Reactors," forms the basis for this EAL. The areas where irradiated fuel is located forms the basis for Table M-6. Unexpected radiation levels, which are at least 100 times higher than the normal background will generally indicate a fuel handling accident or loss of water covering the irradiated fuel. Readings may be from refuel floor Area Radiation Monitors or taken during a qualified radiological survey.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA11 – Cont'd

BASIS: (References) – Cont'd

The value 232 feet and 3 inches plant elevation is the Tech. Spec. Limit and an uncontrolled level drop that would uncover irradiated fuel is an indicator of a lowering in the level of safety of the plant.

There is time available to take corrective actions, and there is little potential for substantial fuel damage. In addition, 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 that risk of injury is low. In addition, NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel" presents the following in its discussion:

> In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of one mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85 in the event of an accident with decayed spent fuel.
>
> Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU11

INITIATING CONDITION

Potential Damage OR Uncovering of Spent Fuel

EAL THRESHOLD VALUE

1. Uncontrolled water level drop in the Spent Fuel Pool that cannot be quickly terminated with ALL irradiated fuel assemblies remaining covered by water

MODE APPLICABILITY

ALL

BASIS: (References)

Uncontrolled - An unexplained level drop that cannot be quickly terminated and is not the result of a planned evolution. The event should not be considered terminated if continuous make up is required and should not preclude classification of the Unusual Event.

Threshold 1: This event tends to have a long lead time relative to potential for radiological release outside the site boundary, thus impact to public health and safety is very low.

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR all occurring since 1984, explicit coverage of these types of events via this EAL is appropriate given their potential for elevated doses to plant staff. Classification as an Unusual Event is warranted as a precursor to a more serious event.

This event will be escalated to an Alert as a result of uncovery of a fuel assembly and/or indication of high radiation levels on the refueling floor.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA12

INITIATING CONDITION

Loss of Water Level That Has **OR** Will Uncover Irradiated Fuel

EAL THRESHOLD VALUES

1. Water level < 458 inches above RPV instrument zero for the Reactor Refueling Cavity

   AND

   Loss of water level will result in irradiated fuel uncovering

MODE APPLICABILITY

5 (with Reactor Refueling Cavity flooded)

BASIS: (References)

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in NUMARC/NESP-007 IC AU2, "Unexpected Rise in Plant Radiation or Airborne Concentration."

NUREG-0818, "Emergency Action Levels for Light Water Reactors," forms the basis for this EAL.

There is time available to take corrective actions, and there is little potential for substantial fuel damage. In addition, 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 that risk of injury is low. In addition, NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel" presents the following in its discussion:

> In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of one mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85 in the event of an accident with decayed spent fuel.
>
> Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate.

The value 458 inches above RPV instrument zero is the Tech. Spec. Limit and an uncontrolled level drop that would uncover irradiated fuel is an indicator of a lowering in the level of safety of the plant. Escalation would occur via Effluent Release, In-plant radiation, or Emergency Director Judgment.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA12 – Cont'd

BASIS: (References) – Cont'd

The MODE applicability [5 With Reactor Refueling Cavity Flooded] is a deviation from NUMARC [all] in that the EAL is only applicable in that plant condition. This adds clarity to the EAL to ensure that it will not be applied under plant conditions where a classification is not warranted.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU12

INITIATING CONDITION

Uncontrolled Water Level Decrease in Reactor Refueling Cavity

EAL THRESHOLD VALUE

1. Unexpected Skimmer Surge Tank low level alarm

   AND

   Visual observation of an uncontrolled drop in water level below the fuel pool skimmer surge tank inlet that cannot be quickly terminated

MODE APPLICABILITY

ALL

BASIS (References)

Unexpected - An alarm that is not a result of a planned evolution

Uncontrolled - An unexplained level drop that cannot be quickly terminated and is not the result of a planned evolution. The event should not be considered terminated if continuous make up is required and should not preclude classification of the Unusual Event.

A drop in the Spent Fuel Pool level or the RPV [when in refueling and flooded up with the gates removed] will result in a control room annunciator Fuel Pool and Cleanup System Trouble Alarm. This Control Room alarm directs an operator to be dispatched to a local alarm panel, which will identify the Skimmer Surge Tank low level alarm. This alarm is validated with visual observation of a lowering Spent Fuel Pool level. If the spent fuel pool level drops below the inlet to the skimmer surge tank, without a planned event such as removing a large piece of equipment, there must be a leak in the spent fuel pool or the RPV. This event has a long lead time relative to potential for radiological release outside the site boundary, thus the impact to public health and safety is very low. Classification as an Unusual Event is warranted as a precursor to a more serious event.

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR all occurring since 1984, explicit coverage of these types of events via this EAL is appropriate given their potential for elevated doses to plant staff. Classification as an Unusual Event is warranted as a precursor to a more serious event.

This event will be escalated to an Alert as a result of uncovery of a fuel assembly and/or indication of high radiation levels on the refueling floor.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU13

INITIATING CONDITION

Independent Spent Fuel Storage Installation (ISFSI)

EAL THRESHOLD VALUE

1. **EITHER** of the following criteria is met for the dry storage of spent fuel:
   - > 600 mR/hr at 1 ft. away

     OR
   - > 1200 mR/hr at external surface

MODE APPLICABILITY

ALL

BASIS: (References)

Threshold 1: This EAL applies to potential emergency conditions, which might develop during use of the Independent Spent Fuel Storage Installation and dry cask storage system. This EAL provides for an Unusual Event classification, which may be entered in the event that conditions occur which have the potential for damaging or degrading the fuel, but no releases of radioactive material requiring offsite response or monitoring are expected. Consistent with the NUMARC guidance, escalations above the Unusual Event are not warranted.

Accidents associated with the dry cask storage system include natural and man-made events that are postulated to affect the storage system. The limiting impacts to the system include loss of shielding capability and loss of confinement. The loss of shielding results in higher direct radiation to the environment from the cask while the loss of confinement results in a release of materials from within the cask to the environment at a postulated leak rate.

Loss of confinement for the dry storage system is evaluated in TN-68, Safety Analysis Report, Section 7. Two scenarios are considered, one for off-normal conditions and one for hypothetical accident conditions. Dose calculations are included in section 7.3.2.1. In the extremely unlikely event that one of these scenarios did occur, the event would be addressed by the EALs under Category R, "Abnormal Radiological Levels / Effluents".

Loss of shielding for the dry storage system is evaluated in TN-68, Safety Analysis Report, Section 5. Dose calculations are included in Table 5.1-2 for both normal and accident conditions. The value of 600 mR/hr one foot away OR 1200 mR/hr at the external surface is determined for several reasons. According to the TN-68, Safety Analysis Report, Table 5.1-2, Summary of Average Dose Rates, the maximum expected surface dose rates will be 529.5 mR/hr (see note 2). Consequently, the value of 1200 mR/hr is sufficiently above normal conditions as to preclude inappropriate classifications.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU13 – Cont'd

BASIS: (References) – Cont'd

Also, the value of 1200 mR/hr is sufficiently below the 1467 mR/hr found in Table 5.1-2 for the cask surface radiological reading for accident conditions. Therefore, 1200 mR/hr from a loss of shielding accident would trigger an Unusual Event classification.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HG1

INITIATING CONDITION

Security Event Resulting in Loss of Ability to Reach **AND** Maintain Cold Shutdown

EAL THRESHOLD VALUES

1. Loss of physical control of the Control Room due to a security event.

   OR

2. Loss of physical control of the remote shutdown capability due to a security event.

MODE APPLICABILITY

ALL

BASIS (References)

This class of security event represents conditions under which a hostile force has taken physical control of areas required to reach and maintain cold shutdown. Loss of Remote Shutdown Capability would occur if the control function of the Remote Shutdown Panels were lost.

Security events, which meet the threshold for declaration of a General Emergency, are physical loss of the Control Room or the Remote and Alternate Shutdown Panels.

This situation leaves the plant in a very unstable condition with a high potential of multiple barrier failures.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS1

INITIATING CONDITION

Confirmed Security Event in a Vital Area

EAL THRESHOLD VALUES

1. Intrusion into plant Vital Area by a hostile force.

   OR

2. Confirmed bomb, sabotage or sabotage device discovered in a Vital Area

MODE APPLICABILITY

ALL

BASIS (References)

This class of security event represents an escalated threat to plant safety above that contained in an Alert in that a hostile intrusion or attack has progressed from the Protected Area to a Vital Area. The Vital Areas are within the Protected Area and are generally controlled by key card readers. These areas contain vital equipment, which includes any equipment, system, device or material, the failure, destruction or release of could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems, which would be required to function to protect health and safety following such failure, destruction or release, are also considered vital.

Identification of Vital Areas can be accomplished through discussions with security.

This event will be escalated to a General Emergency based upon the loss of physical control of the Control Room or Remote Shutdown Capability

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA1

INITIATING CONDITION

Confirmed Security Event in a Plant Protected Area

EAL THRESHOLD VALUES

1. Intrusion into a Protected Area or ISFSI by a hostile force.

   OR

2. Confirmed bomb, sabotage or sabotage device discovered in a Protected Area or ISFSI

MODE APPLICABILITY

ALL

BASIS (References)

EALs #1 and #2 are applicable to ANY Protected Area as defined under the Station Nuclear Security Plan, including the on-site Independent Spent Fuel Storage Installation (ISFSI).

This class of security event represents an escalated threat to the level of safety of the plant. This event is satisfied if physical evidence supporting the hostile intrusion or attack exists. The Emergency Director will declare an Alert subsequent after consulting with the on-shift Security representative to determine the validity of the entry conditions.

This event will be escalated to a Site Area Emergency based upon a hostile intrusion or act in-plant Vital Areas.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU1

INITIATING CONDITION

Confirmed Security Event That Indicates a Potential Degradation in the Level of Plant Safety

EAL THRESHOLD VALUES

1. A credible threat to the station reported by the NRC.

   OR

2. **BOTH** of the following criteria are met for a credible threat reported by any other outside agency as determined per SY-AA-101-132, "Threat Assessment":
   - Is specifically directed towards the station.
   - Is imminent (≤ 2 hours).

   OR

3. Attempted intrusion and attack on a Protected Area or ISFSI

   OR

4. Attempted sabotage discovered within a Protected Area or ISFSI

   OR

5. Hostage/Extortion situation that threatens normal plant operations

MODE APPLICABILITY

ALL

BASIS (References)

EALs #3 and #4 are applicable to ANY Protected Area as defined under the Station Nuclear Security Plan, including the on-site Independent Spent Fuel Storage Installation (ISFSI).

A security threat that is identified as being directed towards the station and represents a potential degradation in the level of safety of the plant. A security threat is satisfied if physical evidence supporting the threat exists, if information independent from the actual threat exists, or if a specific group claims responsibility for the threat. The Shift Management will declare an Unusual Event subsequent to consulting with the on shift Security representative to determine the credibility of the security event per SY-AA-101-132 and the Physical Security Plan.

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 10 CFR 50.72 and will not cause an Unusual Event to be declared.

This event will be escalated to an Alert based upon a hostile intrusion or act within the Protected Area.

DEVIATION: A bomb device discovered within Plant Protected Area and outside the Plant Vital Areas is an Alert declaration as determined per the site Safeguards Contingency Plan and therefore is not included as an Unusual Event in the EAL scheme.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS2

INITIATING CONDITION

Control Room Evacuation Initiated **AND** Plant Control **CANNOT** be re-established in ≤ 15 minutes

EAL THRESHOLD VALUES

1. Control Room evacuation initiated

   AND

   Control of the plant **CANNOT** be re-established in ≤ 15 minutes per SE-1 or SE-10

MODE APPLICABILITY:

All

BASIS: (References)

The 15-minute time period starts when physical control of the plant is lost requiring Control Room evacuation OR when the required Control Room personnel have evacuated the Control Room.

Control - Placing all local control switches in local control necessary for operation from remote panels and the Shift Manager has determined that the systems for controlling reactivity, core cooling and heat sink functions are established.

Transfer of safety system control has not been performed in an expeditious manner but it is unknown if any damage has occurred to the fission product barriers. The 15 minute time limit for transfer of control is based on a reasonable time period for personnel to leave the control room, arrive at the remote shutdown area, and re-establish plant control to preclude core uncovery and/or core damage. During this transitional period the function of monitoring and/or controlling parameters necessary for plant safety may not be occurring and as a result there may be a threat to plant safety.

This event will be escalated based upon system malfunctions or damage consequences.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA2

INITIATING CONDITION

Control Room Evacuation Initiated

EAL THRESHOLD VALUE

1. Entry into SE-1 or SE-10 for Control Room evacuation

MODE APPLICABILITY

All

BASIS (References)

Control Room evacuation requires establishment of plant control from outside the control room (e.g., local control and remote shutdown panel) and support from the Technical Support Center and/or other emergency facilities as necessary. Control Room evacuation represents a serious plant situation since the level of control is not as complete as it would be without evacuation. The establishment of system control outside of the Control Room will bypass many protective trips and interlocks. In addition, many of the instruments and assessment tools available in the Control Room will not be available.

This event will be escalated to a Site Area Emergency if control cannot be re-established within fifteen minutes.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA3

INITIATING CONDITION

Natural **OR** Destructive Phenomena Affecting a Vital Area

EAL THRESHOLD VALUE

1. Earthquake > 0.05g (Operating Basis Earthquake, OBE) as determined by procedure SO 67.7.A
   OR
2. Tornado or wind speeds > 75 mph causing damage to Plant Vital Structures (Table H-1)
   OR
3. Report of visible structural damage to **ANY** Plant Vital Structure (Table H-1)
   OR
4. Vehicle crash affecting a plant vital function contained in a Plant Vital Structure (Table H-1)
   OR
5. Turbine failure generated missiles result in visible structural damage to or penetration of **ANY** Plant Vital Structures (Table H-1)
   OR
6. Abnormal River level, as indicated by **EITHER**:
   - >116 ft. (high level)
     OR
   - < 92.5 ft. (low level)

   OR
7. Flooding in 2 or more areas designated in T-103, Table SC/L-2 requiring a plant shutdown.

Table H-1 **Plant Vital Structures**
• Power Block • Diesel Generator Building • Emergency Pump Structure • Inner Screen Structure • Emergency Cooling Tower

MODE APPLICABILITY

ALL

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA3 – Cont'd

BASIS (References)

Each of these EALs is intended to address events that may have resulted in a plant vital 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 EAL to assess the actual magnitude of the damage.

Threshold Value 1 – This EAL addresses an earthquake that exceeds the Operating Basis Earthquake level of .05 g and is beyond design basis limits . An earthquake of this magnitude may be sufficient to cause damage to safety related systems and functions. The Max Credible Earthquake for PBAPS is 0.12 g per UFSAR Section 1.6; therefore, this EAL is conservative and warrants an Alert classification.

> Confirmed – As used in this EAL, a call to the National Earthquake Center is the primary confirmation source. Other confirmation includes reports from television or radio stations, or reports from university monitoring stations.

Threshold Value 2 – This EAL is based on FSAR design basis. Wind loads of this magnitude can cause damage to safety functions. This EAL addresses events where a Plant Vital Structure has been struck with high winds, and thus damage may have occurred to safe shutdown systems.

Threshold Value 3 – Structural damage should be of sufficient force, that in the Emergency Director's judgment, the potential exists to affect the operation of systems and functions required for safe shutdown of the plant. This EAL specifies a Plant Vital Structure, which contain systems and functions required for safe shutdown of the plant.

Threshold Value 4 – The intent is to address such items as aircraft, train, barge or large motor vehicles (e.g. cranes, etc.). Automobiles, trucks and forklifts are also vehicles within the context of this EAL; however, the key is whether or not the vehicle can potentially affect a plant vital function, located within a designated Plant Vital Structure.

Threshold Value 5 – Missile impacts including rotating equipment or turbine failure causing casing penetration.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA3 – Cont'd

BASIS (References) – Cont'd

Threshold Value 6 – High River level > 116 feet is indication of the river being in flood. This level is capable of causing flooding that can affect Plant Vital Structures. General grade at the site of Units 2 & 3 has been established at a nominal 115 feet elevation in the area surrounding the Turbine Hall and other structures on the river side of the plant. Top of ground floor of the structures in this area is at 116 feet elevation. No attempt should be made to determine the magnitude of flooding. This is a long lead time event but this level is ground elevation of the reactor building and intake pump structure so classification as an Alert Event is appropriate. The evidence of flooding is sufficient for declaration.

Low River level < 92.5 feet is indication of a potential loss of Conowingo Pond and subsequent loss of the main condenser circulating water pumps if water level continues to drop.

Threshold Value 7 – Flooding in vital areas that affect operability of safety-related systems or components. The source of the flooding need not be known.

This event will be escalated to a higher emergency classification based upon damage consequences covered under other various EAL Sections.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU3

INITIATING CONDITION

Natural OR Destructive Phenomena Affecting the Protected Area

EAL THRESHOLD VALUE

1. Earthquake > 0.01g as determined by procedure SO 67.7.A
   OR
2. Report by plant personnel of a tornado strike within the Protected Area
   OR
3. Wind speeds > 75 mph as indicated on Site Meteorological instrumentation for > 15 minutes
   OR
4. Vehicle crash within the Protected Area Boundary that may potentially damage plant structures containing functions required for safe shutdown of the plant.
   OR
5. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.
   OR
6. Control Room assessment indicates that a phenomena has occurred affecting the Protected Areas
   OR
7. Abnormal River level, as indicated by **EITHER:**
   - >112 ft. (high level)
     OR
   - < 98.5 ft. (low level)

MODE APPLICABILITY:

ALL

BASIS: (References)

This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure or by the radiological releases and would be classified under the Fission Product Barrier Matrix or Event Category R "Abnormal Radiological Conditions/Effluents".

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU3 – Cont'd

BASIS: (References) – Cont'd

The Emergency Director should consider how these Threshold Values may affect both units due to the affects of common or shared plant systems.

Threshold Value 1 - This EAL addresses a sensed earthquake. The magnitude of .01g is the lowest detectable earthquake measured on PBAPS seismic instrumentation per procedure SO 67.7.A. An earthquake of this magnitude may be sufficient to cause minor damage to plant structures or equipment within the Protected Area. Damage is considered to be minor, as it would not affect physical or structural integrity. This event is not expected to affect the capabilities of plant safety functions. This event will be escalated to an Alert if the earthquake reaches an Operating Basis Earthquake (OBE).

Threshold Values 2 & 3 - A tornado touching down within the Protected Area or wind speeds > 75 mph within the Owner Controlled Area are of sufficient velocity to have the potential to cause damage to Plant Vital Structures. The value of 75 mph was selected to maintain consistency with plant value and to coincide with the Beaufort Scale for Hurricane wind speed winds of 73-136 mph. These criteria are indicative of unstable weather conditions and represent a potential degradation in the level of safety of the plant. Verification of a tornado will be by direct observation and reporting by station personnel. Verification of wind speeds > 75 mph will be via meteorological data in the control room. This event will be escalated to an Alert if the tornado or high wind speeds strike a Plant Vital Structure.

Threshold Value 4 - This criterion is intended to address such items as plane, helicopter, train or other "vehicle" crashes that may potentially damage plant structures containing functions required for safe shutdown of the plant. Automobiles, trucks and forklifts are also vehicles within the context of this EAL; however, the key is whether or not the vehicle can potentially cause significant damage to plant structures. If the crash is confirmed to affect a plant vital function contained in a designated Plant Vital Structure, the event may be escalated to an Alert classification.

Threshold Value 5 - This criterion 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 leakage of combustible fluids (e.g., lubricating oils) and gases (e.g., hydrogen) to the plant environs. Actual fires and flammable gas build up are appropriately classified via other EALs. Turbine failure of sufficient magnitude to cause observable damage to the turbine casing or seals of the turbine generator raises the potential for leakage of combustible fluids and gases (Hydrogen cooling) to the Turbine Building. The damage should be readily observable and should not require equipment disassembly to locate.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU3 – Cont'd

BASIS: (References) – Cont'd

Threshold Value 6 – This criterion allows for the control room to determine that an event has occurred and take appropriate action based on personal assessment as opposed to verification (e.g., an earthquake is felt but does not register on any plant-specific instrumentation, etc.)

Threshold Value 7 – Cooling water is pumped from the normal heat sink (Conowingo Pond) via the pump structure. An alternate suction supply and discharge path (from the emergency heat sink – which consists of an induced draft cooling tower with an integral storage reservoir) is available in the unlikely event of a Conowingo Dam failure.

High River level of greater than 112 feet: At this level open grating in the operating floor of the Circulating Water Pump Structure will allow water from the circulating water bays to rise into the structure during postulated external flooding conditions. Per the UFSAR, "The configuration of the circulating water system would likely trip at a flood elevation of about 113 feet. Therefore, a river elevation of 11 feet was chosen as the elevation at which a flood-related shutdown is initiated." The use of a threshold of 112 feet for the Unusual Event would represent a condition above T.S., but prior to the postulated loss of circulating water.

Low River level of less than 98.5 feet: This is the plant low water design level and consistent with T.S. 3.7.2 (Minimum Water Level in Pump Bay). Per the UFSAR, with the river level is 104 feet when an uncontrolled release of about 350,000 cfs is passed through the Conowingo Dam and there is no in flow into pond, it will require about 1-1/2 hours to drop level to 98.5 feet.

This event will be escalated to an Alert classification based continuation of the river situation.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA4

INITIATING CONDITION

Fire **OR** Explosion Affecting Operability of Safety Systems Required for Safe Shutdown

EAL THRESHOLD VALUES

1. ANY of the following are made potentially inoperable by a fire or explosion:
   - 2 or more Safe Shutdown Systems (Table H-2)
   - 2 or more subsystems, as defined by Tech. Specs., of a Safe Shutdown System (Table H-2).
   - 1 or more Plant Vital Structures containing Safe Shutdown Equipment (Table H-1)

   AND

   Safe Shutdown System or Plant Vital Structure is required for the present Operational Condition

Table H-1 Plant Vital Structures
• Power Block • Diesel Generator Building • Emergency Pump Structure • Inner Screen Structure • Emergency Cooling Tower

Table H-2: Safe Shutdown Systems		
• Diesel Generators • HPCI • Core Spray • SBGT • PCIS (Primary CNTMT Isolation System)	• 4 KV Safeguard Busses • RCIC • HPSW • ECW • Control Room Emergency Ventilation	• ADS • RHR (all modes) • ESW • CAC / CAD

MODE APPLICABILITY:

ALL

BASIS: (References)

Explosion - A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to nearby structures or equipment.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA4 – Cont'd

BASIS: (References) – Cont'd

Fire - combustion characterized by the generation of heat and smoke. Sources of smoke such as overheated electrical equipment and slipping drive belts, for example, do not constitute fires. Observation of a flame is preferred, but is NOT required if large quantities of smoke and heat are observed.

The primary concern of this EAL is the magnitude of the fire and the effects on Safe Shutdown Systems required for the present Operational Condition. A Safe Shutdown System is defined as any system required to maintain safe operation or to establish or maintain Cold Shutdown. A system being "inoperable" means that it is incapable of performing the design function. For example, the LPCI System is intended to maintain adequate core cooling by covering the core to at least 2/3 core height following a DBA LOCA. In order for the system to be unable to maintain its intended function, multiple loops would need to be disabled by the fire. In addition to indication of degraded system performance, potential inoperability may be determined by visual observation and other control room indications such as loss of indicating lights.

Safe Shutdown Analysis is consulted to determine systems required for the applicable mode.

Two examples of applying this methodology are as follows:

- Diesel Generators and 4 KV Safeguard Buses: The fire disables multiple Diesel Generators or 4 KV Safeguard Buses so that the number of emergency power systems available would be lowered to below what would be required to mitigate an accident under the current operating conditions. For 100% power, this could be conservatively interpreted as at least two Diesel Generators or 4 KV Buses disabled.
- RHR - LPCI Mode: The fire disables multiple loops of LPCI so that adequate core submergence could not be assured following a DBA LOCA. For 100% power, this could also be conservatively interpreted as at least two loops disabled.

The EAL includes the condition that the fire must make "TWO OR MORE" subsystems (as defined by Tec. Specs.) or "TWO OR MORE" systems inoperable. In those cases where it is believed that the fire may have caused damage to *Safety Systems*, then an Alert declaration is warranted, since the full extent of the damage may not be known. For Plant Vital Structure damage, classification is required under this EAL if the structure houses or otherwise supports *Safety Systems* required for the present Operational Condition.

Degraded system performance or observation of damage that could degrade system performance is used as the indicator that the safe shutdown system was actually affected or made inoperable. A report of damage should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of damage. The occurrence of the fire or explosion with reports of damage (e.g., deformation, scorching) is sufficient for declaration.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU4

INITIATING CONDITION

Fire Within Protected Area Boundary NOT Extinguished in ≤ 15 minutes of Detection

EAL THRESHOLD VALUE

1. Fire within or impacting a plant Vital Structure (Table H-1)

   AND

   Fire is **NOT** extinguished in ≤ 15 minutes of **EITHER**:
   - Control Room notification

     OR
   - Verification of alarm

   OR

2. Report by plant personnel of an explosion within the Protected Area Boundary resulting in visible damage to a permanent structure or equipment

Table H-1 Plant Vital Structures
• Power Block • Diesel Generator Building • Emergency Pump Structure • Inner Screen Structure • Emergency Cooling Tower

MODE APPLICABILITY

ALL

BASIS (References)

Verification - Determination is made that the fire alarm is not spurious.

Explosion - A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to nearby structures or equipment.

The purpose of this IC is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. This excludes such items as fires within administration buildings, wastebasket fires, and other small fires of no safety consequence. This IC applies to buildings and areas contiguous to plant vital areas or other significant buildings or areas. The intent of this IC is not to include buildings (e.g., warehouses) or areas that are not contiguous or immediately adjacent to plant vital areas.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU4 – Cont'd

BASIS (References) – Cont'd

Verification of the alarm in this context means those actions taken in the control room to determine that the control room alarm is not spurious.

This EAL addresses fires in Plant Vital Structures that house safety systems. These fires may be precursors to damage to safety systems contained in these structures. There are no areas/buildings contiguous to Plant Vital Structures, which could affect a safety system in one of the listed Plant Vital Structures except for those already on the list. Therefore, no additional areas/buildings are considered for this EAL.

Verification that a fire exists is by operator actions to confirm that fire alarms received in the Control Room are not spurious or by any verbal notification by plant personnel. Fifteen minutes has been established to allow plant staff to respond and control small fires or to verify that no fire exists.

This event will be escalated to an Alert if the fire damages redundant trains of plant safety systems required for the current operating condition.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA5

INITIATING CONDITION

Release of Toxic **OR** Flammable Gases Within a Facility Structure Which Jeopardizes Operation of Systems Required to Maintain Safe Operation **OR** to Establish or Maintain Cold Shutdown

EAL THRESHOLD VALUE

1. Report or detection of toxic gases within Plant Vital Structures (Table H-1) in concentrations that will be life threatening to plant personnel.

   OR

2. Report or detection of flammable gases within Plant Vital Structures (Table H-1) in concentrations affecting the safe operation of the plant

Table H-1 Plant Vital Structures
• Power Block • Diesel Generator Building • Emergency Pump Structure • Inner Screen Structure • Emergency Cooling Tower

MODE APPLICABILITY

ALL

BASIS (References)

Gases within the site boundary that are above life-threatening or flammable concentrations, and have exceeded those concentrations within a Plant Vital Structure (as defined under Table H-1), should be declared as an Alert.

This IC is based on gases that have entered a plant structure affecting the safe operation of the plant. This IC applies to buildings and areas contiguous to plant Vital Areas or other significant buildings or areas. The intent of this IC is not to include buildings (e.g., warehouses) or other areas that are not contiguous or immediately adjacent to Plant Vital Areas. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred.

Concentrations above life-threatening or flammable concentrations that result from planned maintenance or repair activities on-site, where planned contingency measures are identified to monitor and control gas(es), do not require classification.

Threshold #1: Toxic gas concentration results in an atmosphere that is immediately harmful to unprotected personnel, and would preclude access to any such affected area. However, access into the affected area does not have to be required for classification purposes.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA5 – Cont'd

BASIS (References) – Cont'd

Threshold #2: Flammable gas, such as hydrogen and acetylene, are routinely used to maintain plant systems or to repair equipment / components. This EAL addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage / personnel injury.

This event will be escalated to higher classifications based on damage consequences covered under other various EAL Sections. 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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU5

INITIATING CONDITION

Release of Toxic OR Flammable Gases Deemed Detrimental to Safe Operation of the Plant

EAL THRESHOLD VALUES

1. Report or detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant.

   OR

2. Report by Local, County or State officials for potential evacuation of site personnel based on an offsite event.

MODE APPLICABILITY

ALL

BASIS: (References)

Gases within the site boundary that are above life-threatening or flammable concentrations, and have not exceeded those concentrations within a Plant Vital Structure (as defined under Table H-1 in IC HA1), should be declared as an Unusual Event.

A toxic/flammable gas is considered to be any substance that is dangerous to life or limb by reason of inhalation or skin contact. It should not be construed to include confined spaces that must be ventilated prior to entry.

This IC is based on releases in concentrations within the site boundary that will affect the health of plant personnel or the safe operation of the plant with the plant being within the evacuation area of an offsite event (e.g., tanker truck accident releasing toxic gases, etc.). The evacuation area is determined from the DOT Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials.

Concentrations above life-threatening or flammable concentrations that result from planned maintenance or repair activities on-site, where planned contingency measures are identified to monitor and control gas(es), do not require classification.

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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HG6

INITIATING CONDITION

Conditions Indicate Imminent Core Damage **OR** Release Affecting the Public

EAL THRESHOLD VALUES

1. Actual or imminent core degradation with potential loss of containment.

   OR

2. Potential uncontrolled radionuclide release, which can reasonably be expected to exceed 1 Rem TEDE, or 5 Rem CDE Thyroid plume exposure levels at the Site Boundary

MODE APPLICABILITY

ALL

BASIS (References)

General Emergency - Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Imminent - Mitigation actions have been ineffective and trended information indicates that the event or condition will occur within 2 hours.

Potential - Mitigation actions are not effective and trended information indicates that the parameters are outside desirable bands and not stable or improving.

This EAL allows the Emergency Director to declare a General Emergency upon the determination of an actual or imminent substantial core degradation or melting with the potential for loss of containment integrity, but is not explicitly addressed by other EALs.

Releases may exceed the EPA Protective Action Guidelines for more than the immediate site area and will be classified under Event Category R, "Abnormal Radiological Levels/Effluents".

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS6

INITIATING CONDITION

Conditions Indicate Actual OR Likely Failure of Plant Functions Needed for Public Protection

EAL THRESHOLD VALUE

1. Other conditions exist which in the judgment of the Emergency Director indicate actual or likely major failures of plant functions needed for protection of the public.

MODE APPLICABILITY

ALL

BASIS (References)

Site Area Emergency – Events are in process or have occurred which involve actual or likely failure of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels, which exceed EPA Protective Action Guideline exposure levels except near the site boundary.

This EAL allows the Emergency Director to declare a Site Area Emergency upon the determination of an actual or likely major failure of plant functions needed for protection of the public, but is not explicitly addressed by other EALs.

Releases are not expected to result in exposure levels, which exceed the EPA Protective Action Guidelines except within the site boundary and will be classified under Event Category R, "Abnormal Radiological Levels/Effluents".

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA6

INITIATING CONDITION

Conditions Indicate Actual OR Potential Substantial Degradation of the Level of Plant Safety

EAL THRESHOLD VALUE

1. Other conditions, exist which in the judgment of the Emergency Director indicate that plant safety systems may be degraded and that increased monitoring of plant functions is warranted.

MODE APPLICABILITY

ALL

BASIS (References)

Alert - Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

This EAL allows the Emergency Director to declare an Alert upon the determination that the level of safety of the plant has substantially degraded but is not explicitly addressed by other EALs. This includes a determination by Shift Management that the TSC and OSC should be activated and command and control functions should be transferred for the event to be effectively mitigated. Transfer of command and control functions is used as an initiator since an event significant to warrant transfer is a substantial reduction in the level of safety of the plant. Other examples are:

Internal flooding affects the operability of plant safety systems required to establish or maintain cold shutdown.

Releases that are expected will be limited to a small fraction of the EPA Protective Action Guidelines and will be classified under Event Category R, “Abnormal Radiological Levels/Effluents”.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU6

INITIATING CONDITION

Conditions Indicate a Potential Degradation in the Level of Plant Safety

EAL THRESHOLD VALUE

1. ANY of the following occur, which in the judgment of the Emergency Director indicate a potential degradation in the level of safety of the plant:
   - Aircraft crash on-site
   - Train derailment on-site
   - Near-site explosion, which may adversely affect normal site activities

   OR
2. Other conditions exist which in the judgment of the Emergency Director indicate a potential degradation in the level of safety of the plant

MODE APPLICABILITY

ALL

BASIS (References)

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event emergency class.

> Unusual Event – Events are in process of have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

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 response 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.

It is also intended that the Emergency Director's judgment not be limited by any list of events as defined here or as augmented by the site. This list is provided solely as examples for consideration and it is recognized that actual events may not always follow a pre-conceived description

Section 4: Emergency Measures

4.1 Notification of the Emergency Organization

Notifications for the Peach Bottom Atomic Power Station are made to the following additional State and local agencies in accordance with Section E.3 of the Exelon Nuclear Standardized Radiological Emergency Plan:

- Maryland Emergency Management Agency (MEMA)
- Pennsylvania Emergency Management Agency (PEMA)
- Cecil County Emergency Management & Civil Defense Agency
- Chester County Department of Emergency Services
- Harford County Division of Emergency Operations
- Lancaster County Emergency Management Agency
- York County Emergency Services

Notification of PEMA and the risk counties will be directed by the Emergency Director within 15 minutes of initial event classification, reclassification, or a change in a protective action recommendation (PAR) due to plant conditions or meteorological changes per Section E.3 of the Exelon Nuclear Standardized Radiological Emergency Plan. In addition, once the EOF is activated, the Corporate Emergency Director will contact the Senior Pennsylvania State Official as designated by PEMA following the decision to recommend a protective action for the general public.

Upon notification of an emergency at Peach Bottom Atomic Power Station, the Pennsylvania Bureau of Radiation Protection (BRP) and Maryland Department of the Environment (MDE) will contact the appropriate station to verify that an emergency exists and to obtain technical information, and then makes recommendations to PEMA and MEMA respectively, regarding protective actions for the public. The BRP/ MDE Support Plan For Fixed Nuclear Facility Incidents utilizes the Protective Action Guidelines in the U.S. Environmental Protection Agency (EPA) 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents".

Exelon Nuclear will provide follow-up information to the BRP/MDE or other off-site authorities. The follow-up information will keep these authorities apprised of existing or potential radiological releases, meteorological conditions, projected doses and contamination levels, licensee actions, recommend protective actions and other information pertinent to the authorities responsibilities. The information may be provided over open communication paths or in person to BRP/MDE personnel.

4.2 Assessment Actions

The effluent radiation monitoring system provides indications of gross releases of gaseous and liquid radioactivity. By applying calibration factors, meteorological data, or river flow, the gross indications are used to calculate approximate release rates in µCi/sec and dose rates at specific distances along the release pathways. Particulate and iodine analysis depends on collecting installed filter papers and charcoal cartridges for analysis in the counting room. Similar calculation procedures are applied to approximate release rates and dose rates due to iodine.

Detectors are strategically located throughout the plant. These detectors indicate and alarm locally and in the Control Room. They serve the purpose of indicating current dose rates in those areas and are used for local evacuation action levels and re-entry operations.

Certain plant operating systems contain radiation monitors. These systems are described in the PBAPS UFSAR.

Portable monitoring instruments and sampling equipment consist of such items that are utilized and maintained on-site by the Chemistry and Health Physics sections for normal day-to-day plant operations and are thus available for emergency operations.

4.2.1 Core Damage Assessment Methodology

Core damage information is used to refine dose assessments and confirm or extend initial protective action recommendations. Limerick Generating Station utilizes NEDC-33045P-A, "Methods of Estimating Core Damage in BWRs" (Revision 0, July 2001) , as the basis for the methodology for post-accident core damage assessment. This methodology utilizes real-time plant indications in addition to samples of plant fluids and atmospheres. Core damage is qualitatively evaluated per NRC Core Condition Categories (1-10) as shown in the table below:

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

4.3 Protective Actions for the Offsite Public

For incidents at PBAPS, PEMA coordinates with MEMA and contacts York, Lancaster and Chester County Emergency Management Agencies to assure that local plans have been implemented. MEMA likewise contacts Cecil and Harford Counties in the event of emergency at PBAPS to assure that all plans have been implemented. County and local governments have primary responsibilities for implementing protective measures for the public following a nuclear incident.

The BRP and MDE serve as lead State agency, in Pennsylvania and Maryland respectively, for technical assistance to other state agencies, county, and local governments regarding radiological health and accident assessment. In the absence of communications with the state, recommendations for protective actions shall be made directly to county emergency operations centers from the station.

4.3.1 Alert and Notification System (ANS) Sirens

Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan and Annex Q of the Maryland Radiological Emergency Plan address notification to the general public and others regarding protective actions. An Alert Notification System, which is intended for use by the counties, in conjunction with the Emergency Alert System (EAS) to provide notification to the general public, has been installed.

Alerting of the EPZ population is provided by a siren system that was installed and is maintained by Exelon Nuclear. The system consists of high-powered rotating electro-mechanical sirens mounted on Class 1 utility poles throughout the Plume Exposure Pathway (10-Mile EPZ). Personnel at the risk county communication centers operate the sirens. The Pennsylvania Emergency Management Agency (PEMA), in conjunction with Maryland and the risk counties, coordinates the activation of the siren system for Peach Bottom Atomic Power Station.

The siren system meets or exceeds the acoustic coverage requirements outlined in NUREG-0654/FEMA-REP-1 and FEMA-REP-10. The location of each siren site was determined by a computer-based sound propagation model.

The sirens are controlled by digitally encoded radio signals transmitted by a transceiver at the station. Each risk county has control of the sirens that are physically located in that county. The sirens can be activated on an individual, municipal, county, or EPZ-wide basis. A controller located at the station serves as a backup to the county controllers. After the system is activated, each siren reports the result of its activation back to the respective county controller and the controller at the station. The siren system is tested regularly to ensure its operability.

Annex E (to the PA Emergency Operations Plan) and Annex Q (to the Maryland REP) delineate risk counties as responsible to:

- Develop a system for rapid notification (in priority order) of county and local government heads, key staff, emergency forces, volunteer organizations, schools, hospitals, nursing homes, business, and industry;
- Ensure that the alert and notification system is operable on an around-the-clock basis;
- Prepare and disseminate public information material on protective actions to provide clear instructions to the population at risk;
- Prepare and maintain material current for dissemination through the EAS; and

- Include provisions in the alert plan for notification of transients.

PEMA/MEMA will notify other states within the Ingestion Pathway EPZ should such action be necessary.

Annex E (to the PA Emergency Operations Plan) and Annex Q (to the Maryland REP) also call for each risk county to promptly activate their alert notification system, when appropriate. EAS radio stations will be activated and instructed as to which prepared message to use. Detailed messages with specific instructions to the public will be provided to the EAS stations by state and county public information officers on a timely basis. Various state agencies will assist the counties in assuring notifications of transients.

4.3.2 Evacuation Time Estimates

The evacuation time estimates (ETE) were developed in coordination with the Commonwealth of Pennsylvania and State of Maryland to assess the relative feasibility of an evacuation of the 10-Mile EPZ for the Peach Bottom Atomic Power Station. The evacuation times are based on a detailed consideration of the EPZ roadway network and population distribution. The ETE Study, maintained separately by Emergency Preparedness, presents representative evacuation times for daytime and nighttime scenarios under various weather conditions for the evacuation of various areas around the Peach Bottom Atomic Power Station, once a decision has been made to evacuate.

4.3.3 Potassium Iodide (KI)

The Department of Health, Commonwealth of Pennsylvania, is responsible for providing advice to PEMA on the planning for the use, stockpiling and distribution of Potassium Iodide (KI) or other thyroid blocking agents and such other radiological health materials as may be required for the protection of the general public. Their decision shall also be based on U.S. FDA guidance.

The use of KI in the State of Maryland will be in accordance with state health laws and under the direction of State and County Medical Officials.

Based on agreement with the Commonwealth of Pennsylvania and State of Maryland, PBAPS will recommend to government officials that the general public be notified to take KI at a General Emergency classification in those sectors were an evacuation has been recommended. This notification will be approved by the Emergency Director in Command and Control of PAR decision-making and off-site notifications, and performed as part of the State / local notifications described under Sections II.B.4 and II.E.3 of the Exelon Nuclear Standardized Radiological Emergency Plan.

4.3.4 Public Information

a. Publications

Public information on protective actions is prepared and disseminated annually to provide clear instructions to the population-at-risk. Exelon Nuclear assists PEMA/MEMA and risk counties in the preparation and distribution of their respective public information..

Pamphlets outlining public education response actions are readily available for transients in the 10-Mile EPZ. In addition, emergency information is provided to the operators of other recreational areas in the 10-Mile EPZ, as defined by the Commonwealth of Pennsylvania, State of Maryland and risk counties.

These public information publications (including telephone book emergency information, etc.) instruct the public to go indoors and turn on their radios when they hear the ANS sirens operating. These publications identify the local radio stations to which the public should tune in for information related to the emergency. Additional materials (e.g., such as rumor control numbers, evacuation routes, information on inadvertent siren soundings, etc.) may also be included in these publications based on agreements with responsible State and risk county agencies.

b. News Media Education

Information kits are available to news media personnel. These kits include information on a variety of nuclear power plant related subjects.

4.3.5 Protective Action Recommendations (PARs) for the General Public

Figure PBAPS 4-1, "Plant-Based PAR Determination Flowchart", illustrates affected downwind sectors based on wind direction, using the generic plant-based event logic as outlined in Figure J-1 of the Exelon Nuclear Standardized Radiological Emergency Plan.

Further evaluation of PAR based on dose assessments shall be performed in accordance with Section II.J.10.m.2 of the Exelon Nuclear Standardized Radiological Emergency Plan..

4.4 Protective Actions for Onsite Personnel

4.4.1 Plant Evacuation

Exelon Nuclear personnel and contractors filling emergency response organization positions are considered essential personnel. As such, they will report to their emergency response locations. They will not evacuate unless specifically directed by the Emergency Director. All other personnel are considered non-essential.

In-plant evacuation is initiated primarily by area radiation monitor alarms and continuous air monitor alarms, but is also applicable for fire alarms, explosions, toxic material conditions, as well as radiation, contamination, and airborne radioactivity surveys which indicate conditions above applicable limits. Notification for personnel to proceed with in-plant evacuation will be via a local alarm or an announcement on the plant PA system. The affected area and evacuation assembly areas (if appropriate) will be announced. The immediate response by individuals in the vicinity of such an alarm or announcement is evacuation to an unaffected area or designed assembly area. In the absence of readily available radiological survey information or other logical assessment of conditions, evacuation will be, at least, to a point where other area radiation monitors, continuous air monitors, or observation of local conditions show that the area is not affected.

Assigned plant personnel report to the scene to evaluate conditions, to provide information to the Control Room, and to perform other emergency functions such as personnel accountability, decontamination, medical assistance, and control of the hazard.

Notification of a Site Evacuation is accomplished by activating the Evacuation Alarm System followed by an announcement over the plant PA system. The evacuation assembly area(s) are announced. Evacuation assembly areas are illustrated in Figure PBAPS 4-2. Non-essential personnel will exit via the security exit points and will proceed to the parking lot for transportation. Evacuees are expected to use their personal vehicles in evacuating to the designated evacuation assembly area(s). Designated evacuation assembly areas are located out side the protected area. Plant access roads are maintained clear during the winter months, travel on these roads is expected to be possible at all times.

Plant visitors who have not completed the required training program are escorted at all times. This ensures proper response under emergency conditions. Visitors at the station shall follow the lead of their escorts to the assembly areas.

4.4.2 Personnel Accountability

The Security personnel shall follow security procedures for personnel accountability. For evacuations, information from evacuees is an important means of accounting for plant personnel. For Site Evacuations, non-essential personnel and those ERO members whose facility is located outside the Protected Area are accounted for at the security exit point. Emergency response personnel responding to the OSC within the Protected Area are accounted for by badging into designated card readers.

4.4.3 Monitoring of Evacuees

Evacuees from the Peach Bottom Site are checked for contamination. Necessary personnel and vehicle decontamination efforts are initiated at the evacuation assembly area using in-plant equipment or emergency kit supplies. Priority for decontamination shall be given to personnel found to have the highest levels of contamination. Any personnel suspected, or known, to have ingested or inhaled radioactive material shall be given a whole body count, as soon as conditions permit, to assess their internal exposure.

The registering and monitoring of the general public evacuating from the Plume Exposure Pathway EPZ, as described in Section II.J.12 of the Exelon Nuclear Standardized Radiological Emergency Plan, will occur at designated facilities per the respective State and County Radiological Emergency Response Plans.

4.5 Severe Accident Management

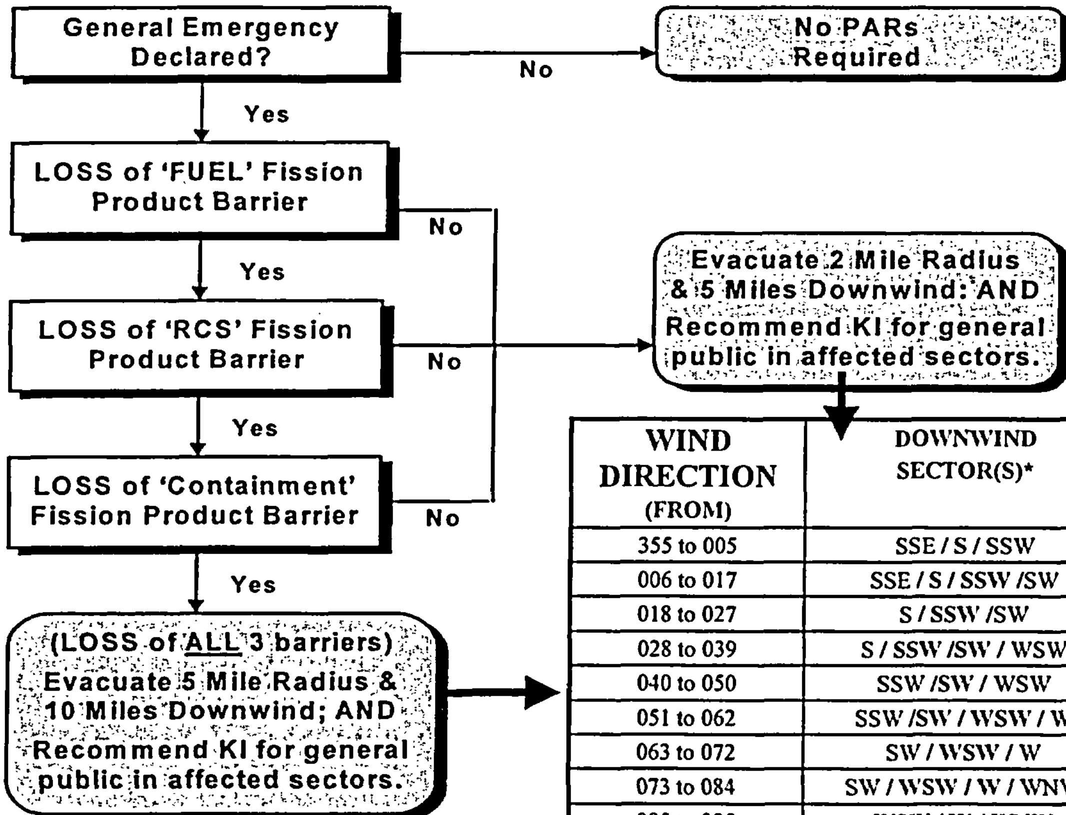
Accident management consists of those actions taken during the course of an accident, by the Emergency Response Organization (ERO), specifically: plant operations, technical support, and plant management staff in order to:

- Prevent the accident from progressing to core damage;
- Terminate core damage once it begins;
- Maintain the capability of the containment as long as possible; and
- Minimize on-site and off-site releases and their effects.

The later three actions constitute a subset of accident management, referred to as Severe Accident Management (SAM) or severe accident mitigation. The Severe Accident Management Plan (SAMP) procedures provide sound technical strategies for maximizing the effectiveness of equipment and personnel in preventing, mitigating and terminating severe accidents.

Implementation of SAMP procedures is a collaborative effort between the Shift Manager and the Station Emergency Director in the TSC (once activated). The Station Emergency Director maintains ultimate responsibility for direction of mitigating strategies. Designated TSC Technical and Operations Support personnel are also trained to assist with decision-making by evaluating plant conditions using the SAM Technical Support Guidelines (TSG).

Figure PBAPS 4-1: Plant-Based PAR Determination Flowchart



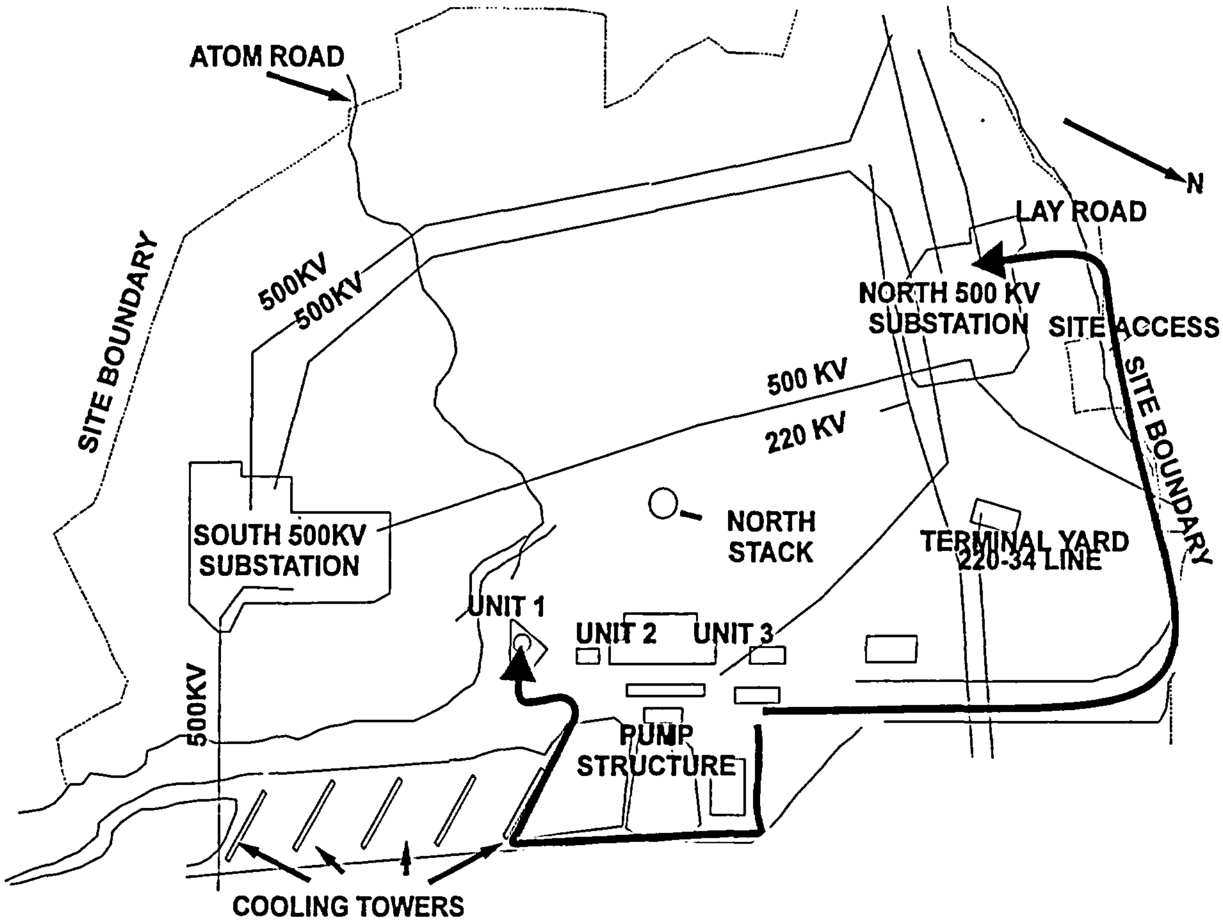
WIND DIRECTION (FROM)	DOWNWIND SECTOR(S)*
355 to 005	SSE / S / SSW
006 to 017	SSE / S / SSW /SW
018 to 027	S / SSW /SW
028 to 039	S / SSW /SW / WSW
040 to 050	SSW /SW / WSW
051 to 062	SSW /SW / WSW / W
063 to 072	SW / WSW / W
073 to 084	SW / WSW / W / WNW
085 to 095	WSW / W / WNW
096 to 107	WSW / W / WNW / NW
108 to 117	W / WNW / NW
118 to 129	W / WNW / NW / NNW
130 to 140	WNW / NW / NNW
141 to 152	WNW / NW / NNW / N
153 to 162	NW / NNW / N
163 to 174	NW / NNW / N / NNE
175 to 185	NNW / N / NNE
186 to 197	NNW / N / NNE / NE
198 to 207	N / NNE / NE
208 to 219	N / NNE / NE / ENE
220 to 230	NNE / NE / ENE
231 to 242	NNE / NE / ENE / E
243 to 252	NE / ENE / E
253 to 264	NE / ENE / E / ESE
265 to 275	ENE / E / ESE
276 to 287	ENE / E / ESE / SE
288 to 297	E / ESE / SE
298 to 309	E / ESE / SE / SSE
310 to 320	ESE / SE / SSE
321 to 332	ESE / SE / SSE / S
333 to 342	SE / SSE / S
343 to 354	SE / SSE / S / SSW

*BOLD refers to affected sector(s)

NOTE: Ensure dose-based PARs are evaluated when a release is in progress.*

* Dose projections are NOT required to support the decisions in the plant-based PAR flowcharts. However, it is expected that a dose projection be performed as soon as possible by the facility in Command and Control at a General Emergency with a release in progress.

Figure PBAPS 4-2: Off – Site Assembly Location



TYPE OF EVACUATION	EVACUATION ASSEMBLY AREAS
LOCAL EVACUATION	Announced on PA System
SITE EVACUATION	Peach Bottom Atomic Power Station Unit 1, North Sub-Station

Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities

5.1.1 Station Control Room

The Peach Bottom Atomic Power Station Control Room shall be the initial onsite center of emergency control. The Control Room is located on the 165' elevation of the Turbine Building (Control Structure). The ventilation system, shielding, and structural integrity are designed and built to permit continuous occupancy during the postulated design basis accident.

5.1.2 Technical Support Center (TSC)

Peach Bottom Atomic Power Station has established a Technical Support Center (TSC) located on the 3rd floor of the Training Center. The TSC fully meets the requirements of Section H.1.b of the Exelon Nuclear Standardized Radiological Emergency Plan and conforms to Section 8.2.1 of Supp. 1, NUREG-0737.

5.1.3 Operational Support Center (OSC)

Peach Bottom Atomic Power Station has designated an Operational Support Center (OSC). The OSC is located in a 2nd floor conference room at the Site Administrative Building. The OSC conforms to the requirements of Section H.1.c of the Exelon Nuclear Standardized Radiological 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.

In the event the OSC is not habitable, personnel report to backup facilities that can be designated based upon specific event conditions.

5.1.4 Emergency Operations Facility (EOF)

The dedicated Emergency Operations Facility (EOF) is located on Exelon property at 175 North Caln Road, Coatesville, PA. The EOF supports both Peach Bottom and Limerick, and is located approximately 31 miles from Peach Bottom Atomic Power Station. Separate offices are provided for Exelon Nuclear, NRC, Maryland and Pennsylvania representatives and other emergency personnel.

Plant Monitoring System data is available through the Emergency Preparedness Data System (EPDS) at the EOF. The EOF equipment includes:

a. Supplies and equipment for EOF personnel, and

b. Sanitary and food preparation facilities.

5.1.5 Joint Public Information Center (JPIC)

The Joint Public Information Center (JPIC) is the facility in which media personnel gather to receive information related to the emergency event. The JPIC is co-located with the EOF at 175 North Caln Road, Coatesville, Pennsylvania.

5.2 Assessment Resources

5.2.1 Geophysical Monitors

a. Onsite Meteorological Monitoring Program

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. Meteorological data is provided to the station Control Room from Meteorological Towers. Data include wind speed, wind direction, and temperature. Meteorological monitoring is described in the PBAPS UFSAR.

b. Seismic Monitoring

Seismic instrumentation includes time-history strong motion pressure triaxial seismic monitor accelerographs located in secondary containment. Peak recording accelerographs, and seismic switches are discussed in the PBAPS UFSAR.

5.2.2 Radiation Monitoring Equipment

For radiological assessments, instrumentation includes area radiation monitors (ARMs), ventilation effluent radiation monitors, liquid effluent radiation monitors, stack effluent monitors, primary containment radiation monitors and miscellaneous process radiation monitors (Refer to PBAPS UFSAR Section 7 for additional information). Data from these sources would be augmented by plant and field surveys for radiation and airborne levels.

a. Radiological Effluent Gaseous Monitoring

PBAPS has five points of release of radioactive material to the atmosphere. These are the Main Off-Gas Stack, Units 2 and 3 Roof Vents and Torus Hardened Vents. Sample systems are installed for three pathways, Main Stack and two Roof Vents. The sample systems consist of isokinetic sample lines containing particulate/iodine filters, and separate sample lines to shielded gas chambers. Detector outputs associated with the gas chambers are recorded in the Control Room. Roof Vent and Main Stack flow rates are also recorded in the Control Room.

The roof vent radiation monitoring system continuously monitors the noble gas being discharged from the Peach Bottom Unit 2 and Unit 3 roof vents. Each unit has two independent monitoring stations. The monitoring stations use scintillation detectors, which read out digitally in the Control Room.

A representative sample of the Torus Hardened Vent (THV) effluent can be obtained by utilizing the Post Accident Sampling System (PASS). The PASS is capable of sampling containment atmosphere prior to and during the use of the Torus Hardened Vent. The THV radiation monitoring system consists of GM type radiation detectors. One monitor is externally mounted to the vent. Both monitors readout in cpm, and are displayed on a digital monitor in the Control Room.

The refuel floor exhaust is combined with other building exhaust streams and is monitored by the Ventilation Stack Radiation Monitoring system for each unit. All alarm functions and readouts are in the Main Control Room. There are also several Area Radiation Monitors on the refuel floors that provide both local and Main Control Room alarm and readout.

Peach Bottoms' gas chamber detector recorder readouts are converted to uCi/sec of noble gas using calibration data and effluent flow rates for each point of release. The uCi/sec Iodine and particulates are determined from the filter and charcoal cartridge samples. The dose projection system then relates meteorological and radiological data to project dose rates along the plume pathway for selected distances. Appropriate atmospheric distribution coefficients are selected for distances of interest from the point of release. Dose rates at these distances are calculated using this data.

b. Radiological Effluent Liquid Monitoring

Liquid releases are made on a batch basis from waste sample tanks. The contents of these tanks are circulated prior to sampling and analysis and release in the discharge canal. Release forms are prepared to authorize releases to the discharge canal. Potentially, plant system leaks could cause discharge to the canal directly. Radiation monitors are located on certain process water systems that indicate abnormal radioactivity levels. A point of release sampling system is located at the end of the discharge canal.

c. Laboratory Facilities

Chemical laboratories are in the Plant Entrance and Radiochemistry Laboratory (PEARL) at PBAPS. A radiochemistry section is provided. The laboratories are adjacent to the counting room for convenience in transporting prepared samples for counting.

5.2.3 Data Acquisition Methods

a. Plant Monitoring System (PMS)

The PBAPS Main Control Room (MCR) and Technical Support Center (TSC) use an emergency facility data system to aid in assessing plant response and status during emergencies. PMS is a computer-based real-time data acquisition and display system, which gathers and records, selected plant parameters for display.

The system displays are designed to aid the Control Room operator in the performance of emergency response procedures. These displays provide information pertinent to reactor core cooling, reactor coolant system integrity, reactivity control, containment integrity, and power system status. These displays are also available to personnel in the TSC.

PMS also provides concise displays of parameters selected for post-accident monitoring. These displays are designed to aid TSC personnel in assessing plant conditions and in assisting Main Control Room personnel in recovering from abnormal or accident conditions and in mitigating their consequences. The displays include parameter versus time and parameter versus parameter trending.

PMS utilizes high-speed data recording, long-term data storage and a transient analysis program package to aid the Technical Support Center staff in reconstructing the accident sequence as well as tracking the plant steady state and dynamic behavior prior to and through the course of an event. PMS displays are available in the Main Control Room and TSC, and EOF through EPDS interactive color graphic display consoles. Hardcopy output devices are available at each location. Provisions have been made to share data with State Liaisons located in the EOF.

b. Emergency Preparedness Data System (EPDS)

The Emergency Preparedness Data System (EPDS) is an emergency facility data system to aid in assessing plant response and status during emergencies. EPDS is a computer based real-time data acquisition and display system, which acquires, stores and re-packages data from PMS for display in the Technical Support Center and Emergency Operations Facility.

5.2.4 Onsite Fire Detection Instrumentation

PBAPS is afforded fire protection from various systems, selected for their applicability in coping with the several possible types of fires. These systems include an extensive fire water system, carbon dioxide system, air foam system, dry chemical system, heat and smoke detectors as well as portable fire extinguishers located throughout the plant. These systems have alarm outputs located in the Control Room. Fire protection systems are described in the PBAPS UFSAR.

5.2.5 Facilities and Equipment for Offsite Monitoring

Off-site Radiological Environmental Monitoring Program is described in the Offsite Dose Calculations Manual (ODCM). Installed radiological monitoring equipment and facilities, including process, area, and effluent, are described in the PBAPS UFSAR. Sets of instruments are available for emergency use by field survey teams. The field survey teams perform field surveys to locate and track the plume and to determine depositing of activity on the ground.

Emergency kits contain radiation survey equipment, which enables the Field Survey Teams to obtain dose rates, surface contamination, and airborne contamination including radioiodine measurements to supplement calculations based on effluent data. These emergency kits are located at facilities outside the plant for ready accessibility. The equipment in these kits is dedicated for emergency use only.

Concurrent field sampling and analysis for radioiodine provides the capability to detect 10^{-7} µCi/cc I-131, per NUREG-0654, FEMA-REP-1.

The services of Normandeau Associates Inc. (NAI) are contracted to provide for the collection of environmental media samples (e.g., water, grass vegetation, etc.) under emergency conditions and there transport to an offsite laboratory for analysis.

5.2.6 Site Hydrological Characteristics

A list of downstream users is maintained to ensure that they are notified. Should contamination of site drinking water sources be suspected, water samples shall be analyzed.

There are river water level indicators in the PBAPS Control Room. These level indicators continuously indicate river levels, which are also input to the process computer for periodic logging, and high and low level alarms. In addition to the river water indicators in the PBAPS Control Room, river levels at Conowingo Dam (downstream) and Muddy Run Pump Storage Station (upstream) are recorded in the Conowingo Control Room. Conowingo Station engineers receive upstream river stages and weather information, which are used to predict river levels and flow rates up to four days in advance. This information is available to the PBAPS Control Room personnel.

5.3 Protective Facilities and Equipment

a. Emergency Supplies

Refer to Table PBAPS 5-1 for a listing of Emergency Supplies and Equipment.

b. Maintenance Equipment

Maintenance equipment consists of normal and special purpose tools and devices utilized in the course of maintenance functions throughout the station. Maintenance and Radiation Protection personnel responding to the OSC are cognizant of the locations of equipment, which may normally be required in an emergency condition. The Maintenance supervision has access to keys for tool storage, shops, and other locations where maintenance equipment may be stored.

5.4 First Aid and Medical Facilities

EMT kits are located in designated areas and are checked and replenished as necessary. Stretchers are also provided at designated locations.

5.4.1 Decontamination and Medical Response

On-site personnel decontamination facilities for emergency conditions include showers and sinks, which drain to the liquid radioactive, waste processing system, at the primary health physics decontamination area in the plant. Special decontamination materials and personnel decontamination procedures are available in the area for use under the direction of health physics supervision. Provisions are made for medical decontamination when personnel are transported to hospitals.

5.4.2 Emergency Medical Assistance Program (EMAP)

An Emergency Medical Assistance Program plan has been established to provide for consultation and definitive care for radiation accident victims. The EMAP distinguishes three levels of medical care:

1. First aid, decontamination, and preliminary patient evaluation at the site

2. Emergency care and patient stabilization in a supporting hospital

3. When necessary, definitive evaluation and treatment.

The EMAP provides for a Radiation Emergency Medical Team (REM Team) to respond to accident 24 hours per day. The team consists of experienced physicians, board certified health physicists, and technicians. It has portable medical and health physics equipment to render emergency treatment at accident sites and conduct the initial evaluation of the radiation status of patients as well as the environment.

For on-site medical assistance, REM Team capabilities include:

1. Consultation and actual assistance to site medical response personnel and the attending Physician

2. Assistance in personnel decontamination

The Emergency Medical Assistance Program (EMAP) consultant has access to extensive laboratory facilities, which provide the capability for radiochemical analysis of plant and environmental samples.

Bioassay including whole body counting, gamma spectroscopy and personnel dosimetry processing are among the capabilities of EMAP.

5.4.3 Medical Transportation

Transportation of injured personnel, who may or may not be radioactively contaminated, to medical treatment facilities is provided by local ambulance services. (Refer to Section 2.4 of the Peach Bottom Annex.)

5.5 Communications

Refer to Section F.1 of the Exelon Nuclear Standardized Radiological Emergency Plan for a description of dedicated communications lines to support both offsite and inter-facility communications.

5.5.1 Intra-Plant Public Address (PA) System

Peach Bottom utilizes a 3-channel system permitting simultaneous use of one page line and two party lines. Loudspeakers powered by individual amplifiers are located throughout the plant and in remote structures. The in-plant system and several remote buildings are powered from two separate emergency busses through automatic transfer switches. Other remote buildings are provided with local power.

The Peach Bottom Public Address system has also been equipped with an advanced page line control system for the enhancement of page announcements throughout the site. This control system provides improved sound quality for emergency announcements made to and from the main control room. It is also capable of screening out page announcements that do not originate from designated page announcement control points such as the control room, TSC, OSC, security locations, etc.

Local area PA announcements can still be conducted by the use of the emergency page button, and the entire system can be reverted back to allow announcements from all locations as required during emergency conditions. The primary purpose of the screening function is to reduce the number of locations where site wide page announcements can originate.

The Peach Bottom PA stations in the plant can only make pages (loudspeaker announcements) to key/central locations (Main Control Room, security station and TSC). General PA announcements over all the plant speakers can only be made from the Main Control Room, Security CAS & SAS stations, OSC, and TSC areas. This system of controlling the PA page announcements dedicates the PA system to reporting emergencies and communications to the Main Control Room.

Capability exists to warn individuals in the vicinity of the river through the river warning system utilizing the plant PA system.

Peach Bottom's Main Control Room has priority page abilities that allow the MCR announcements to override normal plant page announcements.

5.5.2 Private Branch Exchange (PBX) Telephone System

The PBAPS main commercial telephone system (PBX) provides telephone communications capabilities throughout the plant, remote structures, and with off-site parties. Extensions are located in the Main Control Room, the TSC, and the OSC. The power supply for this system consists of one on-site source with an 8-hour battery backup.

The PECO Energy Main Office and Exelon Nuclear headquarters are also served by separate commercial telephone systems (PBX's). All PECO Energy and Exelon Nuclear's PBX's are networked together to create a fully-integrated voice network, providing call management and network redundancy.

5.5.3 Dedicated Emergency PBX Telephone System

The PBAPS dedicated emergency PBX telephone system provides rapid and reliable communications in the event of an emergency. It is independent of the main PBX switch. The dedicated emergency PBX allows rapid dialing and conferencing of emergency response personnel. Extensions are located in the Control Room, the TSC, the OSC, the EOF, and the JPIC. Tie line access capability is provided both through the Peach Bottom main PBX switch and the Limerick dedicated emergency PBX switch. Exhibit 5-1, "Emergency Communications Links", provides a simplified diagram of conference capabilities. The system is powered by the Conowingo underwater line and has a battery backup.

Dedicated lines are provided between the PBAPS Control Room, PBAPS substations, and Exelon Nuclear System Operations located at the Corporate Headquarters.

5.5.4 Intra-Plant Maintenance Telephone System

The intra-plant maintenance telephone system is a part of the PBX system and consists of telephone jacks into which telephone sets may be plugged. The telephone jacks are in various plant locations (predominantly in areas of high maintenance activity) and have the effect of expanding the PBX capability.

5.5.5 EOF/JPIC Private Branch Exchange (PBX)

A dedicated PBX is installed at the Coatesville EOF/JPIC. This switch will control telephone communications in and between the facility, other Exelon locations, and non-Exelon locations. In the event of a PBX failure, outside dial capability is available through trunk lines from the Coatesville Service Building. The EOF/JPIC PBX switch is powered by a source that is backed by a 4-hour uninterruptible power supply and an emergency diesel generator. The UPS is designed to allow sufficient time to bridge any power interruption caused by switching to diesel-supplied power.

5.5.6 Data and Facsimile Transmission Lines

Various data lines are provided to interface computer systems and facsimile machines located at Peach Bottom, Limerick, EOF/JPIC.

5.5.7 Trunk Lines

Incoming and outgoing central office trunk lines are provided from the local telephone company. These lines are used to access the Public Switched Telephone Network.

5.5.8 Tie Lines

Two-way tie lines are provided between LGSS, PBAPS, Corporate Main Office, Exelon Nuclear, and the EOF. Communication lines are maintained between PBAPS and Conowingo Dam. These can be used if conditions warrant securing of the plant in the event of a flood or failure at Conowingo Dam.

The tie lines are available to emergency personnel to allow communications between the sites and Exelon Nuclear locations supporting the emergency.

Company tie lines are utilized to route NRC communications (e.g., ENS, HPN and counterpart circuits) from between Exelon Nuclear emergency response facilities for Peach Bottom Atomic Power Station.

5.5.9 Emergency PBX T-1 Circuit Lines

Two dedicated T-1 circuits between the Limerick Generating Station and Peach Bottom Atomic Power Station emergency PBX telephone systems are provided for calls within and outside the Exelon voice network. This linkage also allows the continuation of 2-way commercial telephone service in the event that one of the two main commercial telephone system PBX's becomes inoperable or unavailable.

5.5.10 Microwave Tie Lines

Dedicated microwave tie lines exist between LGS, PBAPS, Main Office, Exelon Nuclear, and the EOF/JPIC. The microwave system is backed up by eight hours of battery. In addition, communication lines exist between LGS, PBAPS, Main Office, the Nuclear Group Headquarters, and the EOF/JPIC.

5.5.11 Radio Equipment

A fixed base radio system with multiple channels provides primary/backup outside communication capability as shown in Figure PBAPS 5-1, "Emergency Radio Links.".

A separate group of fixed radio channels provides primary/backup communications between in-plant user groups. These channels function through a distributed antenna system located on-site to ensure proper coverage of the area.

The fixed base radio repeaters, antenna system, and radio consoles are powered from a variety of emergency AC buses (diesel backup) and dedicated alternate battery supplies.

A supplementary radio communication system at PBAPS operating on the "ACS/Fire" channel is installed at the six alternate shutdown control stations in the plant. This system is battery backed up for a minimum of 16 hours. The radio channels for this system are designed to survive an automatic isolation on any line faults produced by a Control Room fire.

5.5.12 Evacuation Alarm System

The Evacuation Alarm System consists of a siren tone generator, PA system speakers, a roof siren, and evacuation alarm beacons. The siren tone generator injects an audible evacuation alarm in the PA system, which is broadcast over the PA system speakers. The evacuation alarm beacons provide an audible and visual alarm through two mechanical sirens and flashing red beacon on each beacon unit. The evacuation alarm beacons are installed in all high noise areas of the plant and in areas not covered by the PA system. A selector switch in the Control Room manually initiates the evacuation alarm.

5.6 Independent Spent Fuel Storage (ISFS)

Accidents associated with dry cask storage system include natural and man-made events that are postulated to affect the storage system. The limiting impacts to the system include: (1) loss of shielding capability, and (2) loss of confinement to the system. The loss of shielding results in higher direct radiation from the cask to the environment, while the loss of confinement results in a release of materials from within the cask to the environment at a postulated leak rate.

Monitoring of the fuel storage system would provide the means to detect the accident condition and initiate corrective actions. Continued assessment would be provided to the Emergency Director by in-field radiological monitoring. Emergency response procedures include guidance for performing dose projections and may be supplemented by data obtained from ERO dose assessment and environmental monitoring personnel.

Table PBAPS 5-1: Emergency Supplies and Equipment

The following is a listing of typical equipment available for use during emergencies. While specific equipment designations and items may be subject to change, equivalent emergency activity capabilities will be maintained. Procedures define the specific locations, types, and amounts of equipment for emergency use and define requirements for applicable surveillance, testing, maintenance, and inventory activities to ensure that the equipment is in a state of readiness.

1.0	PROTECTIVE	LOCATIONS STORED OR AVAIL
	Anti-C Clothing	2, 7, 8, 10
	Dosimetry	2, 4, 9, 10
	Respirator/Filters	2, 4, 10, 13
	Self Contained Breathing Apparatus	1, 2, 10, 13
	Radiation signs, rope and tape	2, 7, 8, 13
	Potassium Iodide	2, 7, 8, 10
2.0	RADIATION MONITORING	LOCATIONS STORED OR AVAIL
	Air Sampler	2, 4, 7, 8, 10, 13
	Geiger Counter	1, 2, 4, 7, 8, 10, 13
	Ion Chamber	1, 2, 4, 7, 8, 10, 13
	Frisker	3
	Radiation Survey Forms	2, 7, 8, 10, 13
	Smears	2, 7, 8, 10, 13
	Swipes	2, 7, 8, 10, 13
3.0	SEARCH AND RESCUE	LOCATIONS STORED OR AVAIL
	Flashlight	3
	Blanket	3
	Stretcher	3
	Rope	3
4.0	DECISION AIDS	LOCATIONS STORED OR AVAIL
	Nuclear Emergency Plan	1, 2, 4, 5, 13, 15
	PBAPS EP Procedures	1, 2, 4, 5, 6, 7, 8, 13, 14, 15
	Maps	2, 4, 5, 7
	Prints (Aperture Cards)	1, 4
	Drawings (Aperture Cards)	1, 4

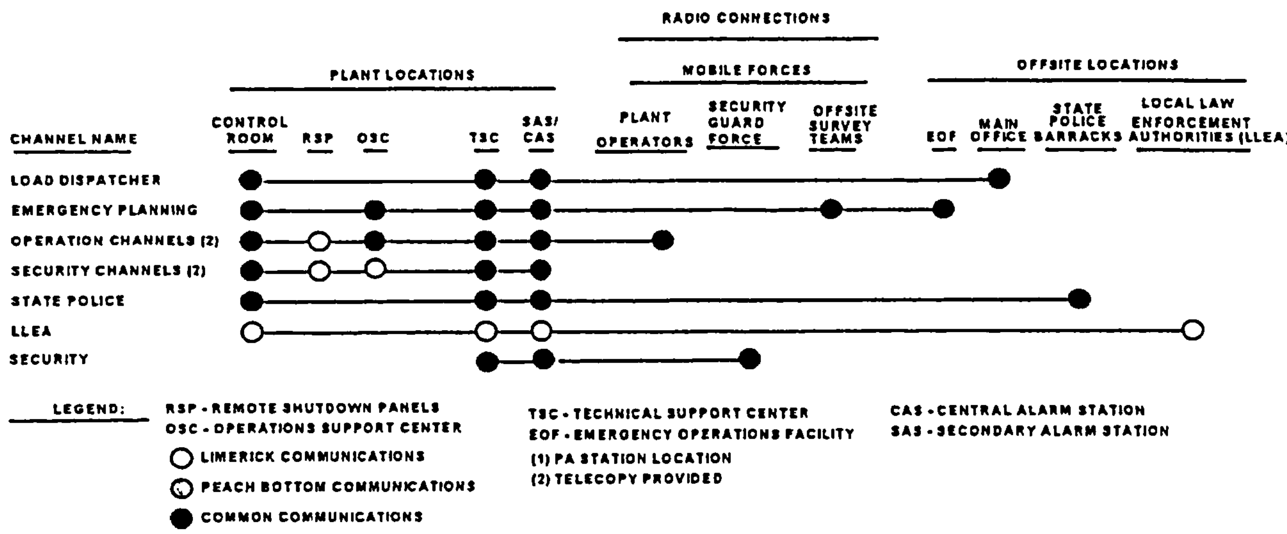
Table PBAPS 5-1: Emergency Supplies and Equipment (Cont'd)

5.0	COMMUNICATIONS	LOCATIONS STORED OR AVAIL
	Base Stations	1, 2, 4, 5, 14
	Mobile Radios	1, 2, 5, 7, 14
6.0	DECONTAMINATION	LOCATIONS STORED OR AVAIL
	Soap	8, 16
	Detergent	8, 16
	Hose	8
	Brushes	8, 16
	Sponges	8, 16
	Buckets	8

LOCATION KEY

1	Control Room Area
2	Operations Support Center
3	Strategically located throughout Station
4	Technical Support Center
5	Emergency Operations Facility
6	Alternate Chemistry Laboratory
7	Field Monitoring Kits
8	Evacuation Assembly Area Kits
9	Personnel Dosimetry Office
10	Peach Bottom Unit 1
13	Health Physics
14	Security Building
15	Joint Public Information Center
16	Decontamination Room

Figure PBAPS 5-1: Emergency Radio Links



APPENDIX 1: NUREG-0654 CROSS-REFERENCE

Annex Section	NUREG-0654
1.0	Part I, Section A
1.1	Part I, Section B
1.2	Part I, Section D
1.3	Part I, Section F
Table PBAPS 1-1	Part I, Section F
Figure PBAPS 1-1	Part II, Section J.10
Figure PBAPS 1-2	Part II, Section J.11
2.0	Part II, Section B.1
2.1	Part II, Section B.5
2.2	Part II, Section A.3
2.3	Part II, Section C.3
2.4	Part II, Section C.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.f
4.3.1	Part II, Section E.6
4.3.2	Part II, Section J.8
4.3.3	Part II, Section J.6.c
4.3.4a	Part II, Section G.1 & 2
4.3.4b	Part II, Section G.5
4.3.5	Part II, Section J.7
4.4.1	Part II, Sections I.2 & 3.a
4.4.2	Part II, Section J.5
4.4.3	Part II, Section J.3
Figure PBAPS 4-1	Part II, Section J.7
Figure PBAPS 4-2	Part II, Section J.4
5.1	Part II, Section H.1-2, & G.3.a
5.2.1	Part II, Section H.5.a & 8
5.2.2	Part II, Section H.5.b, H.6.c &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, I.9-10
5.2.6	Part II, Section H.5.a & 6.a
5.3	Part II, Section H.9-10
5.4	Part II, Section L.1 & 2
5.5	Part II, Section F.1
Table PBAPS 5-1	Part II, Section H.11
Figure PBAPS 5-1	Part II, Section F.1.d
Appendix 1	Part II, Section P.8
Appendix 2	Part II, Section J.8

APPENDIX 2: SITE-SPECIFIC LETTERS OF AGREEMENT

The following is a listing of letters of agreement and contracts specific to emergency response activities in support of Peach Bottom Atomic Power Station. Letters of agreement and contracts common to the multiple Exelon Nuclear stations are listed under Appendix 3 to the Exelon Nuclear Standardized Radiological Emergency Plan.

- Chester County Department of Emergency Services (Letter on File)
- Lancaster County Emergency Management Agency (Letter on File)
- York County Emergency Management Agency (Letter on File)
- Pennsylvania Emergency Management Agency (Letter on File)
- Pennsylvania Department of Environmental Resources / Bureau of Radiation Protection (Letter on File)
- Pennsylvania State Police#
- Memo of Understanding (Letter on File) with Maryland Emergency Management Agency (MEMA), which includes the following support agencies:
  - Maryland Department of the Environment / Radiological Health Program,
  - Harford County Division of Emergency Operations, and
  - Cecil County Emergency Management Agency
- Porter Consultants, Inc. (P.O.)
- Delta-Cardiff Volunteer Fire / Ambulance Company (Letter on File)
- Harford Memorial Hospital (Letter on File)
- York Hospital (Letter on File)

Agreements with State and local law enforcement agencies maintained by Station Security under the Nuclear Station Security Plan.

Exelon Nuclear

EP-AA-1008
Revision 5

EXELON NUCLEAR

RADIOLOGICAL EMERGENCY PLAN ANNEX FOR LIMERICK GENERATING STATION

Submitted: John Kupacka Date: 9/9/03
Mid-Atlantic Emergency Preparedness Manager

Approved: C. Amone Date: 9/9/03
Corporate Functional Area Manager

Authorized: Date: 9/26/03
Vice President, Licensing and Regulatory Affairs

Section 1: Introduction

1.1 Facility Description ....LGS 1-1
1.2 Emergency Planning Zones ....LGS 1-1
1.3 Participating Governmental Agencies ....LGS 1-2
Table LGS 1-1: Offsite Radiological Emergency Response Organizations and Response Plans ....LGS 1-4
Figure LGS 1-1 10-Mile Plume Exposure Pathway ....LGS 1-5
Figure LGS 1-2 50-Mile Ingestion Pathway ....LGS 1-6

Section 2: Organizational Control of Emergencies

2.1 Shift Organization Staffing ....LGS 2-1
2.2 Emergency Response Organization (ERO) Staffing ....LGS 2-2
2.3 Emergency Response Organization (ERO) Training ....LGS 2-3
2.4 Non-Exelon Nuclear Support Groups ....LGS 2-3
2.5 Nuclear Steam Systems Supplier (NSSS) ....LGS 2-3
2.6 Architect/Engineer ....LGS 2-3
Table LGS 2-1 Minimum Staffing Requirements ....LGS 2-4
Figure LGS 2-1 Exelon Overall ERO Command Structure ....LGS 2-7
Figure LGS 2-2 Emergency Onsite Organization ....LGS 2-8
Figure LGS 2-3 Emergency Offsite Organization ....LGS 2-9
Figure LGS 2-4 Emergency Public Information Organization ....LGS 2-10

Section 3: Classification of Emergencies

3.1 Emergency Action Levels (EALs) ....LGS 3-1
3.2 EAL Technical Basis ....LGS 3-3
3.3 General EAL Implementation Philosophy ....LGS 3-4
Table LGS 3-1 Emergency Action Levels ....LGS 3-5
Table LGS 3-2 LGS EAL Technical Basis ....LGS 3-12

Section 4: Emergency Measures

4.1 Notification of the Emergency Organization ....LGS 4-1
4.2 Assessment Actions ....LGS 4-1
4.3 Protective Actions for the Offsite Public ....LGS 4-2
4.4 Protective Actions for Onsite Personnel ....LGS 4-5
4.5 Severe Accident Management ....LGS 4-7
Figure LGS 4-1 10-Mile EPZ Permanent Resident Population Distribution ....LGS 4-8
Figure LGS 4-2 Plant-Based PAR Determination Flowchart ....LGS 4-9
Figure LGS 4-3 Off-Site Assembly Locations ....LGS 4-10

Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities .... LGS 5-1
5.2 Assessment Resources .... LGS 5-2
5.3 Protective Facilities and Equipment .... LGS 5-5
5.4 First Aid and Medical Facilities .... LGS 5-5
5.5 Communications .... LGS 5-6
Table LGS 5-1 Emergency Supplies and Equipment .... LGS 5-10
Figure LGS 5-1 Emergency Radio Links .... LGS 5-12

APPENDICES

Appendix 1: NUREG-0654 Cross-Reference

Appendix 2: Site-Specific Letters of Agreement

REVISION HISTORY

REVISION	REVISION DATE	EFFECTIVE DATE
0	August 2002	August 30, 2002
1	October 2002	November 1, 2002
2	November 2002	November 2002
3	February 2003	February 29, 2003
4	September 2003	September 18, 2003
5	October 2003	

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 Radiological Emergency Plan, Station Annexes, emergency plan implementing procedures, and associated program administrative documents. The Exelon Nuclear Standardized Radiological 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 Limerick Generating 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 Standardized Radiological Emergency Plan per EP-AA-120, "Emergency Plan Administration".

1.1 Facility Description

The Limerick Generating Station (LGS) is a fixed nuclear electrical generating facility operated by Exelon Nuclear and licensed by the USNRC. The station includes two boiling water reactor (BWR) electrical generating units.

The Limerick station is located partly in Montgomery County and partly in Chester County Pennsylvania on the Schuylkill River about 1.7 miles southeast of the limits of the Borough of Pottstown. The Schuylkill River passes through the site and separates the western portion, which is located in East Coventry Township, Chester County, from the eastern portion, which is partly in Limerick Township and partly in Lower Pottsgrove Township, both in Montgomery County. Major plant structures are in Limerick Township.

For more specific site location information, refer to the Updated Final Safety Analysis Report (UFSAR) for Limerick Generating Station.

1.2 Emergency Planning Zones

The Plume Exposure Emergency Planning Zone (EPZ) for Limerick Generating Station shall be an area surrounding the Station with a radius of about ten miles. The exact physical boundaries are determined by the Commonwealth of Pennsylvania and affected Counties. Refer to Figure LGS 1-1.

The Ingestion Pathway Emergency Planning Zone (EPZ) for Limerick Generating Station shall be an area surrounding the Station with a radius of about 50 miles. Refer to Figure LGS 1-2.

1.3 Participating Governmental Agencies

The overall responsibility for the management of the effects of accidental off-site releases of radioactivity resulting from either a nuclear power plant or a transportation accident rests with state and local governments.

The Commonwealth organizations having prime responsibility in matters of radiation hazards are the Pennsylvania Emergency Management Agency and the Bureau of Radiation Protection (BRP) of the Pennsylvania Department of Environmental Protection. County and local governments are responsible for the protection of public health and safety within their jurisdiction. Similarly, organizations in the Commonwealth of Pennsylvania and States of Maryland, Delaware, and New Jersey are responsible for the protection of the public in their states. Cooperation with the States of Maryland, Delaware and New Jersey is necessary because these states are within the Ingestion Pathway EPZ.

These civil agencies will respond to provide support in the event of an emergency in the areas indicated below.

1.3.1 Pennsylvania Emergency Management Agency (PEMA)

Responsibilities of PEMA are outlined in Annex E, "Radiological Emergency Response to Nuclear Power Plant Incidents" of the Commonwealth of Pennsylvania Emergency Operations Plan.

1.3.2 Department of Environmental Protection, Bureau of Radiation Protection (DEP/BRP)

Responsibilities of DEP/BRP are outlined in Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan.

1.3.3 Pennsylvania State Police

Responsibilities of the State Police are set forth in Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan.

1.3.4 County Governments

Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan defines "risk counties" as those within a 10-mile radius of a fixed nuclear facility. For LGS, the risk counties are:

a. Montgomery County

b. Chester County

c. Berks County

The responsibilities assigned to these Counties are outlined in Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan.

1.3.5 State Of Maryland

The State of Maryland's border is located within the 50-mile Ingestion Pathway for LGS. The State would be notified if protective actions were required within that area. No direct support is provided to LGS.

1.3.6 State Of New Jersey

The State of New Jersey's border is located within the 50-mile Ingestion Pathway for LGS. The State would be notified if protective actions were required within that area. No direct support is provided to LGS.

1.3.7 State Of Delaware

The State of Delaware's border is located within the 50-mile Ingestion Pathway for LGS. The State would be notified if protective actions were required within that area. No direct support is provided to LGS.

Refer to Table LGS 1-1 for a list of offsite radiological emergency response organizations and response plans in support of the Limerick Generating Station's Emergency Preparedness Program.

Table LGS 1-1: Offsite Radiological Emergency Response Organizations and Response Plans

The following state, local and emergency plans are available and filed under separate cover.

- Annex E - "Radiological Emergency Response to Nuclear Power Plant Incidents" - to Commonwealth of Pennsylvania Emergency Operations Plan.
- Montgomery County Radiological Emergency Response Plan for Incidents at LGS.

Municipalities

Collegeville Borough	Douglass Township	Green Lane Borough
Limerick Township	Marlborough Township	Lower Pottsgrove Township
Lower Frederick Township	Lower Salford Township	Lower Providence Township
Perkiomen Township	New Hanover Township	Royersford Borough
Pottstown Borough	Skippack Township	Schwenksville Borough
Upper Frederick Township	Trappe Borough	Upper Providence Township
Upper Pottsgrove Township	West Pottsgrove Township	Upper Salford Township

School Districts

Methacton	Perkiomen Valley	Pottsgrove
Souderton	Pottstown	Spring-Ford
Upper Perkiomen		

- Chester County Radiological Emergency Response Plan for Incidents at LGS.

Municipalities

Charlestown Township	East Pikeland Township	East Coventry TownshipEast
Nantmeal Township	East Vincent Township	North Coventry Township
Phoenixville Borough	Schuylkill Township	South Coventry Township
Spring City Borough	Upper Uwchlan Township	Uwchlan Township
Warwick Township	West Pikeland Township	West Vincent Township

School Districts

Downingtown	Great Valley
Phoenixville Area	Owen J. Roberts

- Berks County Radiological Emergency Response Plan for Incidents at LGS.

Municipalities

Amity Township	Boyertown Borough	Colebrookdale Township
Douglass Township	Earl Township	Union Township
Washington Township		

School Districts

Boyertown	Daniel Boone

- State of Delaware Emergency Plan
- State of New Jersey Emergency Plan
- State of Maryland Emergency Plan

Figure LGS 1-1: 10-Mile Plume Exposure Pathway EPZ

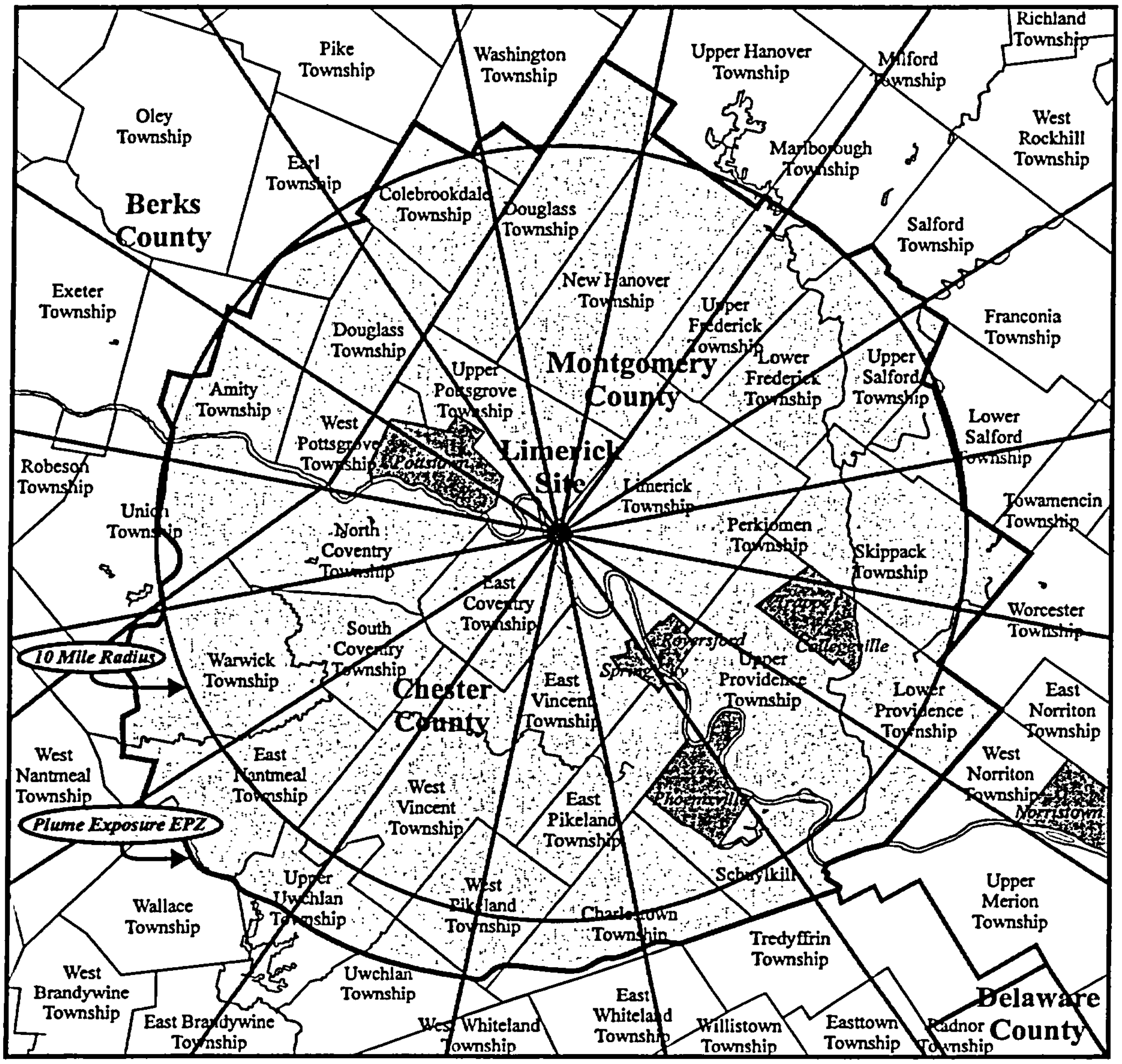
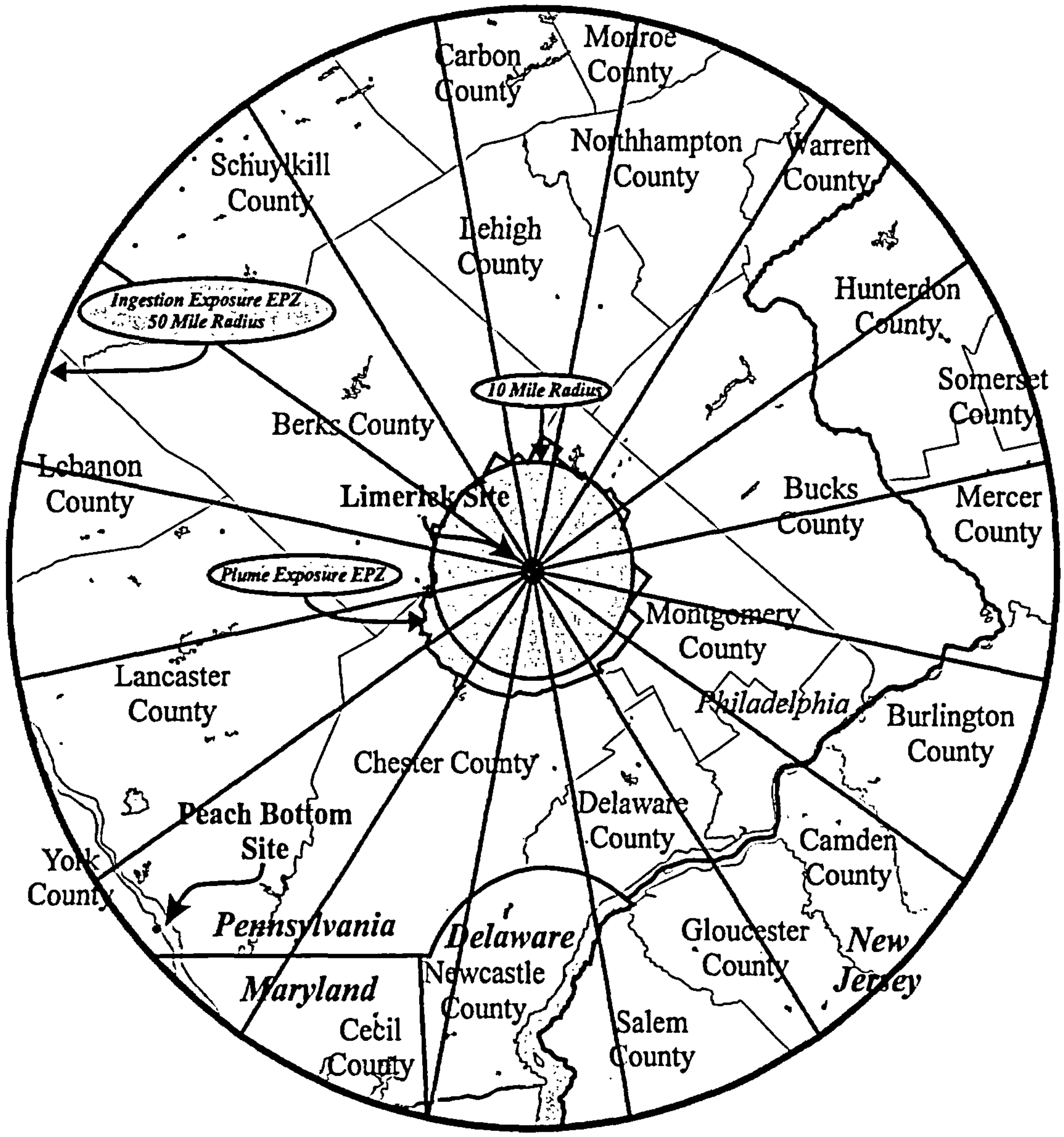


Figure LGS 1-2: 50-Mile Ingestion Pathway EPZ



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 LGS UFSAR Figure 13.1.2. Once an emergency is declared, the Emergency Response Organization (ERO) is activated as described in Section B.4 of the Exelon Nuclear Standardized Radiological Emergency Plan.

2.1 Shift Organization Staffing

Required on-shift staffing in support of emergency response activities is 16 people. This on-shift staffing level exceeds the Exelon Nuclear Standardized Radiological Emergency Plan commitment of ten (10) people, based on existing commitments supporting the elimination of 30-minute augmentation goal from Table B-1 to NUREG-0654/FEMA-REP-1.

A listing of minimum shift complement is provided in Table LGS 2-1 of the Annex for Limerick Generating Station. Based on existing on-shift staffing commitments, the "Minimum Shift Size" for the purposes of NUREG-0654, Table B-1 comparison is 16 persons versus the 10 persons specified in the Exelon Nuclear Standardized Radiological Emergency Plan. These six (6) additional on-shift positions include:

- 2nd Emergency Communicator
- Two Field Survey Team Members
- One Radwaste Operator (Equipment Operator)
- One Instrument and Controls Technician
- 3rd Radiation Protection Technician (two at affected station and 3rd at unaffected station performing dose assessor function).

2.1.1 Shift Dose Assessment

The on-shift dose assessment function will be performed by a shift Radiation Protection Technician (RPT) at Peach Bottom Atomic Power Station. However, Limerick Generating Station will maintain the capability to perform a shift dose assessment, if necessary.

2.1.2 Shift Communicator

The Shift Communicator performs notifications to the State and County organizations until relieved by the TSC, and assists in the initiation of the ERO Callout System as directed. The Communicator position is staffed by a designated on-shift individual capable of responding to the Control Room immediately in support of the initiation of offsite notifications within 15 minutes of event classification.

A 2nd on-shift individual will be designated to support communications with the NRC over the Emergency Notification System (ENS) until relieved by the TSC.

Activation of the automated ERO call out system will be performed by the Shift Manager, but may be delegated to a Control Room Emergency Communicator or available on-shift staff.

2.1.3 Shift Technical Advisor (STA) / Incident Assessor

Section B.1 of the Exelon Nuclear Standardized Radiological Emergency Plan outlines the On-Shift Emergency Response Organization Assignment of the STA. Limerick Generating Station has deemed the following as an acceptable method of implementing Section B.1 in reference to the STA.

The responsibilities of the STA are delineated on OP-AA-101-111, "Roles and Responsibilities of On-Shift Personnel." If the STA is the Shift Manager or Unit Supervisor, then another Senior Reactor Operator (SRO) shall assist as Incident Assessor during unexpected conditions and transients. Per Table B-1, the on-shift STA or Incident Assessor shall also provide core/thermal hydraulics support to Control Room staff.

2.2 Emergency Response Organization (ERO) Staffing

Refer to Table LGS 2-1 of the LGS Annex, "Minimum Staffing Requirements", for a comparison against the Exelon Nuclear Standardized Radiological Emergency Plan of 60-minute and full augmentation commitments.

2.2.1 Emergency Onsite Organization (Figure LGS 2-2)

No changes in augmentation positions or staffing levels for the Technical Support Center (TSC), Operations Support Center (OSC) and Control Room from that specified in the Exelon Nuclear Standardized Radiological Emergency Plan.

2.2.2 Emergency Offsite Organization (Figure LGS 2-3)

Based on existing interface and staffing agreements, representatives from the Commonwealth of Pennsylvania will respond to the Emergency Operations Facility (EOF), allowing direct face-to-face communications. As such, the State Environs Communicator position, listed under the Exelon Nuclear Standardized Radiological Emergency Plan, is not staffed at the Coatesville EOF. Rather the EOF Environmental Coordinator will interface directly with State representatives present in the EOF.

An EOF Access Controller has been added to the Full Augmentation complement to support existing facility access control measures.

2.2.3 Emergency Public Information Organization (Figure LGS 2-4)

Based on the co-location of the EOF with the Joint Public Information Center (JPIC) the following Emergency News Center (ENC) functions, as described in Sections B.5.c and B.7 of the Exelon Nuclear Standardized Radiological Emergency Plan, have been eliminated or consolidated with corresponding JPIC positions.

These differences in staffing are:

- Public Information Liaison was deleted.
- Radiation Protection Spokesperson position was incorporated into the Radiological Advisor
- Technical Spokesperson was incorporated into the Technical Advisor position

2.3 Emergency Response Organization (ERO) Training

Training is conducted in accordance with Section O.5 of the Exelon Nuclear Standardized Radiological Emergency Plan per TQ-AA-113, "ERO Training and Qualification." Retraining is performed on an annual basis, which is defined as every 12 months ± 3 months (25% grace period).

2.4 Non-Exelon Nuclear Support Groups

Agreements exist on file with or are verified current annually by the MAROG Corporate Emergency Preparedness Group for the support agencies listed in Appendix 2 of the Exelon Nuclear Radiological Emergency Plan Annex for LGS.

Additionally, Exelon Nuclear has contractual agreements common within Exelon Nuclear with several companies whose services would be available in the event of a radiological emergency. These agencies are also listed in Appendix 3 of the Exelon Nuclear Standardized Radiological Emergency Plan.

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

2.5 Nuclear Steam Systems Supplier (NSSS)

General Electric Company maintains an Emergency Response Organization, which can provide technical assistance from their home office or at the site.

2.6 Architect/Engineer

Bechtel or other contractors may be involved in the technical analysis or construction activities associated with the emergency response or recovery operation. Each such organization will designate a lead representative who will have the same responsibilities, within their scope of work, as described for the NSSS Contractor.

Table LGS 2-1: Minimum Staffing Requirements

Functional Area	Major Tasks	Emergency Positions	[b]Minimum Shift Size	[a]60 Minute Augmentation	Full Augmentation
1. Plant Operations and Assessment of Operational Aspects	Control Room Staff	Shift Manager	1		
		Control Room Supervisor	1		
		Reactor Operator	2		
		Equipment Operator	2		
2. Emergency Direction and Control[c]	Command and Control / Emergency Operations	Shift Emergency Director (CR)	1[f]		
		Station Emergency Director (TSC)		1	
		Corporate Emergency Director (EOF)		1	
3. Notification & Communication	Emergency Communications	Shift Personnel[d]	2		
		TSC Director (TSC)		1	
		EOF Director (EOF)		1	
		State/Local Communicator		1 (EOF)	1 (TSC)
		ENS Communicator		1 (TSC)	1 (EOF)
		HPN Communicator		1 (EOF)	1 (TSC)
	Plant Status	Operations Communicator (CR/TSC)			2
	In-Plant Team Control	Damage Control Comm. (CR/TSC/OSC)			3
	Technical Activities	Technical Communicator (TSC)			1
	Governmental	EOC Communicator (EOF)			1
		State EOC Liaison[h] (PEMA)			1
		Regulatory Liaison (EOF)			1
4. Radiological Accident Assessment and Support of Operational Accident Assessment	Offsite Dose Assessment	Radiation Protection Personnel[e]	1		
		Dose Assessment Coordinator (EOF)		1	
		Dose Assessor (EOF)			1
		Radiation Controls Coordinator (TSC)			1
	Offsite Surveys	Environmental Coordinator (EOF)		1	
		Field Team Communicator (EOF)			1
		Off-Site Field Team Personnel [m]	2	2	(g)
	Onsite Surveys	RP Personnel		2	(g)
	In-plant Surveys	RP Technicians	1	2	(g)
	Chemistry	Chemistry Personnel[n]	1	1	(g)
	RP Supervisory	Radiation Protection Manager (TSC/EOF)		2	

Table LGS 2-1: Minimum Staffing Requirements (Cont'd)

Functional Area	Major Tasks	Emergency Positions	[b]Minimum Shift Size	[a]60 Minute Augmentation	Full Augmentation
5. Plant System Engineering, Repair and Corrective Actions	Technical Support	STA / Incident Assessor[p] (CR)	1		
		Technical Manager (TSC)		1	
		Core/Thermal Hydraulics Engineer (TSC)	1[f]	1	
		Mechanical Engineer (TSC)		1	
		Electrical Engineer (TSC)		1	
		SAMG Decision Maker (TSC)		1[f]	
		SAMG Evaluator (TSC)		2[f]	
		Operations Manager (TSC)		1	
		Radiation Controls Engineer (TSC)			1
	Repair and Corrective Actions	Mechanical Maintenance/[n] (OSC)	1[f]	2	(g)
		Rad Waste Operator	1		(g)
		Electrical Maintenance/[n] (OSC)	1[f]	2	(g)
		Instrument & Control (I&C) (OSC)	1		
		Maintenance Manager (TSC)		1	
		OSC Director (OSC)		1	
		Assistant OSC Director (OSC)			1
		OPs Lead & Support Personnel (OSC)			(g)
	Accident Analysis	Technical Support Manager (EOF)			1
		Operations Advisor (EOF)			1
		Technical Advisor (EOF)			1
6. In-Plant Protective Actions	Radiation Protection	RP Personnel[e]	3[f]	4	(g)
7. Fire Fighting	--	Fire Brigade	(i)		
8. First Aid and Rescue Operations	--	Plant Personnel	2[f]		(g)
9. Site Access Control and Personnel Accountability	Security & Accountability	Security Team Personnel	(k)	(k)	
		Security Coordinator[q] (TSC/Cantera EOF)			2
	EOF Security	Access Control (EOF)			1
10. Resource Allocation and Administration	Logistics / Administration	Logistics Manager (EOF)		1	
		Logistics Coordinator (TSC)			1
		Administrative Coordinator (EOF)			1
		Clerical Staff (TSC/OSC/EOF)			(g)
		Events Recorder (EOF)			1
		Computer Specialist (EOF)			1
		SUB-TOTAL:	16	34	27+

Table LGS 2-1: Minimum Staffing Requirements (Cont'd)

Functional Area	Major Tasks	Emergency Positions	[b]Minimum Shift Size	[a]60 Minute Augmentation	Full Augmentation
11. Public Information	Media Interface	Corporate Spokesperson (JPIC)		1	
		Rad Protection Spokesperson/Advisor(JPIC)			1
		Technical Spokesperson/Advisor (JPIC)			1
	Information Development	Public Information Director (JPIC)		1	
		News Writer (JPIC)			1
	Media Monitoring and Rumor Control	Communications Department (JPIC)			(g)
	Facility Operation and Control	JPIC Director (JPIC)		1	
		JPIC Coordinator (JPIC)			1
		Administrative Coordinator (JPIC)			1
		Events Recorder (JPIC)			1
		Clerical Support (JPIC)			(g)
		Access Control (JPIC)			1
		SUB-TOTAL:	0	3[l]	7+

	[b]Minimum Shift Size	Total Minimum Staff	Total Full Augmentation
TOTAL:	16	37	34+

[a] Response time is based on optimum travel conditions.

[b] For each unaffected nuclear unit in operation, maintain at least one Control Room Supervisor, one Reactor Operator, and one Equipment Operator, except that units sharing a Control Room may share a Control Room Supervisor if all functions are covered.

[c] Overall direction of facility response to be assumed by the Corporate Emergency Director (EOF) when all centers are fully manned. Direction of minute-to-minute facility operations and "non-delegable" responsibilities for event classification and emergency exposure controls remain with the Station Emergency Director (TSC). The Shift Manager, as Shift Emergency Director, shall function as acting Station Emergency Director prior to TSC activation.

[d] Refer to Section 2.1.2 for a description of shift communicator staffing.

[e] Refer to Section 2.1.1 for description of on-shift dose assessment staffing.

[f] May be provided by personnel assigned other functions. Personnel can fulfill multiple functions.

[g] Personnel numbers depend on the type and extent of the emergency.

[h] Staffing of the County EOC Liaison position is not required based on agreements with offsite agencies; however, every effort will be made to dispatch an Exelon Nuclear representative upon request from County EOC Director.

[i] Fire Brigade per FSAR / TRM, as applicable.

[k] Per Security Plan.

[l] The following Emergency Public Information Organization personnel will be designated "minimum staffing" (on-call) positions, but are not subject to the 60-minute response time requirement: Corporate Spokesperson, Public Information Director and JPIC Director.

[m] Each Field Survey Team consists of a Lead and Driver. Primarily the Lead will be an RP Technician (RPT); however, additional personnel qualified as RadWorker and trained in radiological exposure, ALARA principles and contamination control measures, may also serve as Lead to provide for long-term staff relief.

[n] OSC Group Leads can be used initially to fill 60-minute augmentation technical/craft positions in Maintenance, RP and Chemistry.

[p] Refer to Section 2.1.3 for description of on-shift STA/Incident Assessor staffing requirements.

[q] TSC Security Coordinator position will be staffed by LGS Security personnel. The EOF Security Coordinator position will be staffed by Corporate Security personnel at the Mid-West ROG Cantera Offices and will be contacted as part of the TSC activation process.

Figure LGS 2-1: Exelon Overall ERO Command Structure

Bolded Boxes indicate minimum staffing positions.

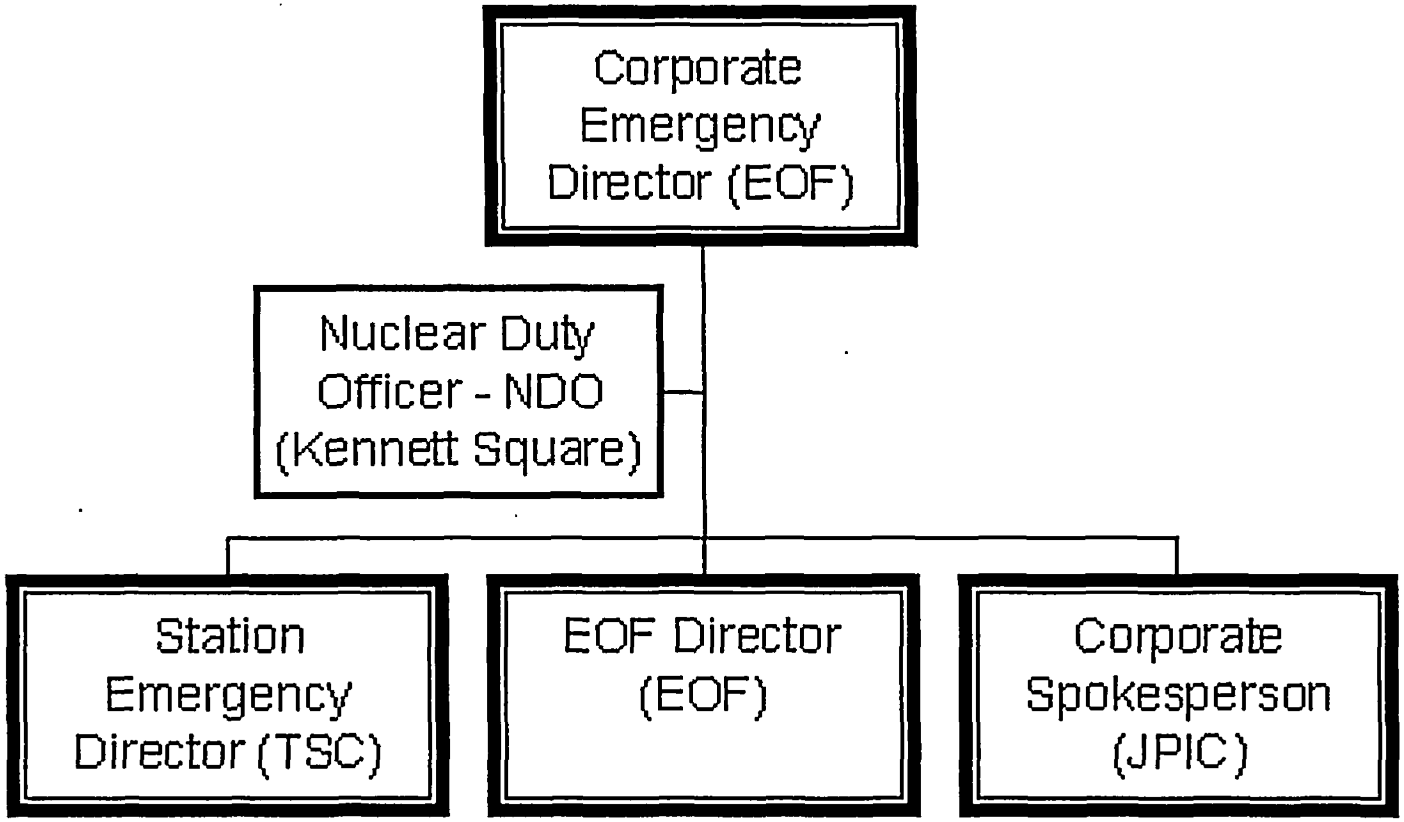
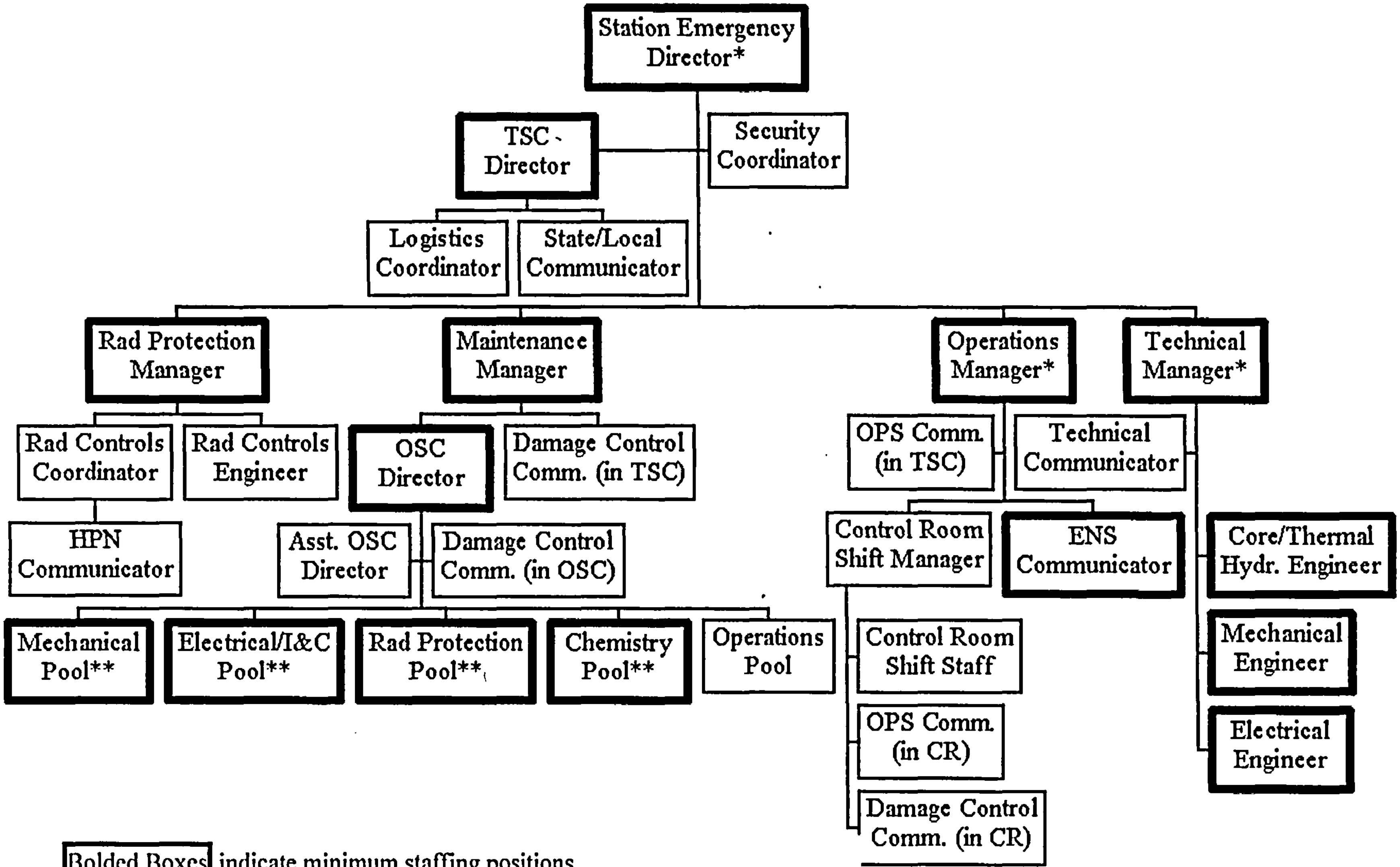


Figure LGS 2-2: Emergency Onsite Organization

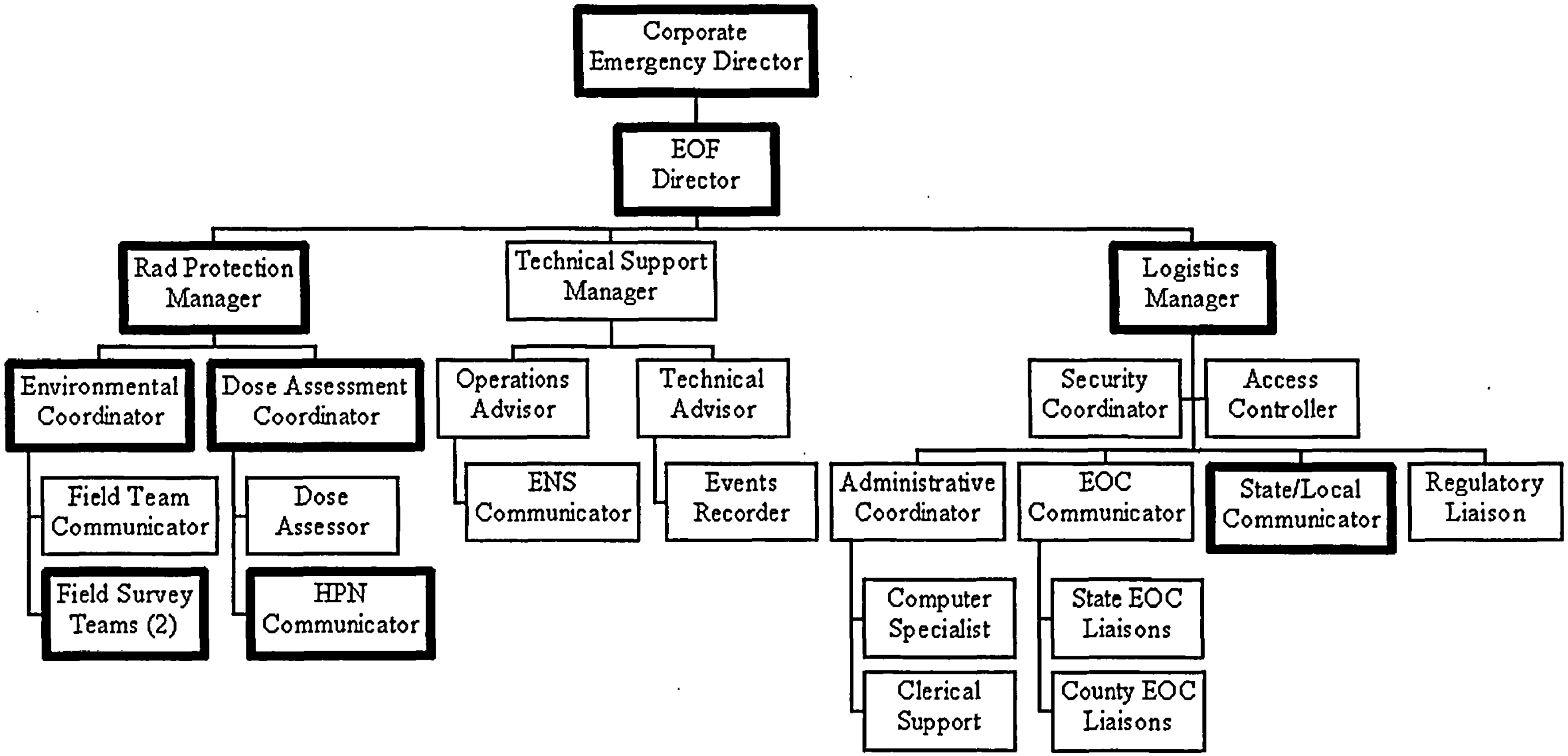


Bolded Boxes indicate minimum staffing positions.

* SAMG functions may be assigned to other qualified personnel. Minimum staffing requires 1 Decision Maker and 2 Evaluators.

** Refer to Table B-1 for required staffing levels

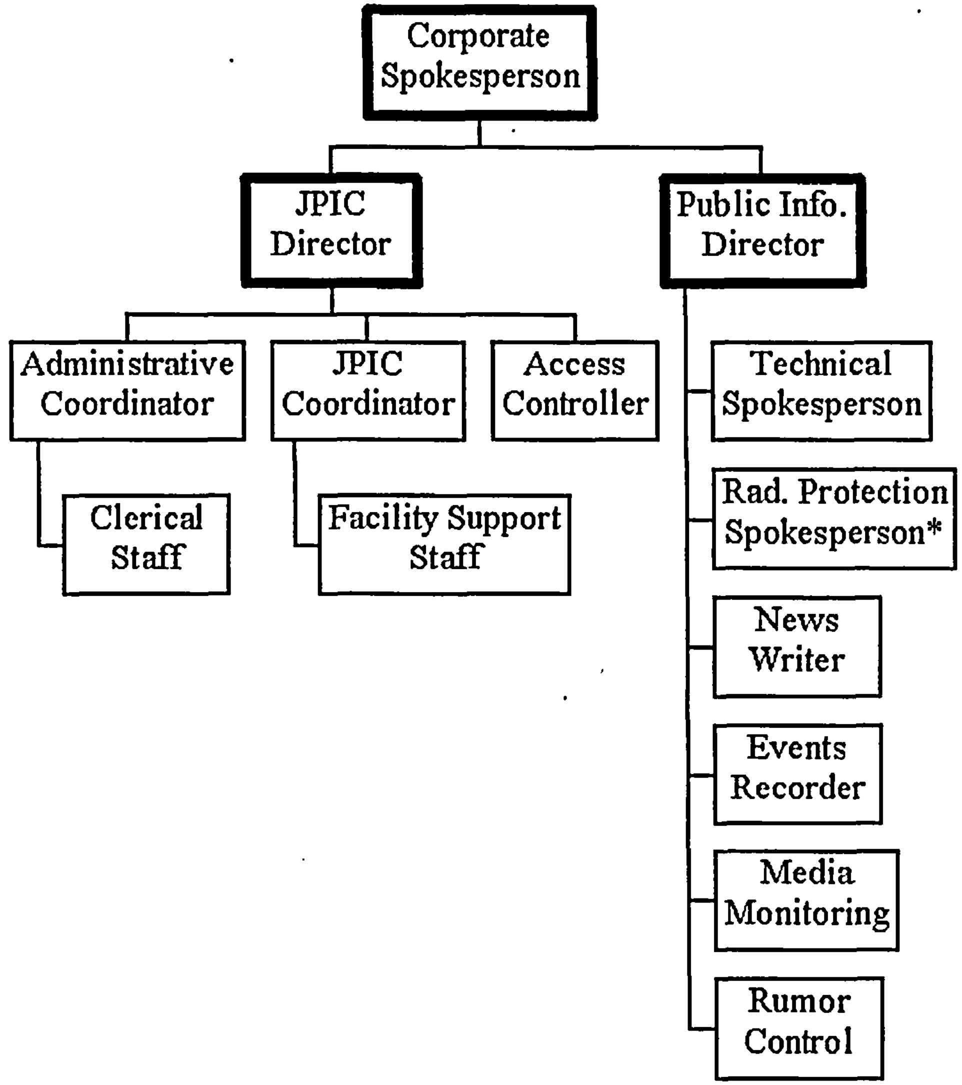
Figure LGS 2-3: Emergency Offsite Organization



Bolded Boxes indicate minimum staffing positions.

* EOF Security Coordinator position staffed by Corporate Security from MWROG Cantera Offices

Figure LGS 2-4: Emergency Public Information Organization



Bolded Boxes indicate minimum staffing positions.

* Radiation Protection Spokesperson / Advisor may be staffed by a qualified consultant.

Section 3: Classification of Emergencies

Section D of the Exelon Nuclear Standardized Radiological Emergency Plan describes five (5) Emergency Classes. The first four are the Unusual Event, Alert, Site Area Emergency and General Emergency, and are listed from least severe to most severe according to relative threat to the health and safety of the public and emergency workers. The fifth level is Recovery and is considered as a phase of the emergency. Recovery is not considered as part of the event classification logic contained in Section 3.0 of the Annex, but rather is entered by meeting criteria provided in Section M of the Exelon Nuclear Standardized Radiological Emergency Plan.

Site specific definitions are provided for terms to be used for that particular Initiating Conditions /Threshold Values and may not be applicable to other uses of that term in any other EAL, at other sites, in the Exelon Nuclear Standardized Radiological Emergency Plan or procedures. Also included are the technical bases, which were used to develop the EAL.

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 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.

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.

3.1 Emergency Action Levels (EALs)

Emergency Action Levels are the measurable, observable detailed conditions that must be met in order to classify the event. Classification shall not be made without referencing, comparing and satisfying the threshold values specified in the Emergency Action Levels. Mode Applicability provides the unit conditions when the Emergency Action Levels represent a threat. The Basis provides definitions of terms, explanations and justification for including the Initiating Condition and Emergency Action Level. Definitions are provided for terms having specific meaning as they relate to this procedure.

Unusual Event, Alert, Site Area Emergency, or General Emergency classifications are entered by meeting designated Emergency Action Levels (EALs) Threshold Values. These values are based on the criteria established under Revision 2 to NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels" (dated January 1992), and are labeled based on the four Recognition Categories outlined in NUMARC/NESP-007:

- Abnormal Radiological Levels / Effluents
- Fission Product Barrier Degradation
- System Malfunctions
- Hazards and Other Conditions

EAL Threshold Values are sorted under common Initiating Conditions (ICs). These ICs can be Symptom- or Event-based, and applicable to all or only designated Operational Conditions / Modes (OPCONs). The Initiating Conditions (IC) and associated EAL Threshold Values are summarized in the EAL Matrix (Table LGS 3-1) according to Recognition Categories.

To aid user in identifying applicable ICs, they are further sorted under the following Event Sub-Categories, and appropriate Mode designator provided:

- *Abnormal Radiological Levels / Effluents ("R")*
  - Radiological Effluents
  - Abnormal Radiation Levels
  - Coolant Activity
- *Fission Product Barrier Degradation ("F")*
  - Fuel Clad
  - Reactor Coolant System, referred to as "RCS"
  - Primary Containment, referred to as "Containment"
- *System Malfunctions ("M")*
  - Loss of AC Power
  - Loss of DC Power
  - Failure of Reactor Protection System
  - Decay Heat Removal
  - Loss of Annunciators
  - RCS Leakage / RPV Draindown
  - MSL Break
  - Loss of Communications
  - Technical Specifications
  - Irradiated Fuel Accidents
- *Hazards and Other Conditions ("H")*
  - Security Events
  - Control Room Evacuation
  - Natural or Man-Made Events
  - Fire / Explosion
  - Toxic or Flammable Gases
  - Discretionary

An emergency is classified by assessing plant conditions and comparing abnormal conditions to ICs and Threshold Values for each EAL, based on the designated Operational Condition (MODE). Modes 1 through 5 are defined in the Technical Specifications (T.S.), for Units 1 and 2, based on Reactor Mode Switch Position and specific plant conditions. "Defueled" Mode was established for classification purposes under NUMARC/NESP-007 to reflect conditions where all fuel has been removed from the Reactor Pressure Vessel.

MODE	TITLE
1	Power Operation
2	Start-up
3	Hot Shutdown
4	Cold Shutdown
5	Refueling
D	Defueled

The EAL 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.

An appropriate EAL numbering system is provided as a user aid. ICs are coded with a two letter and one number code. For example: HA1

The first letter is the Recognition Category designator. In this case, H stands for "Hazards and Other Conditions". The second letter is the Classification Level: "U" for Unusual Event, "A" for Alert, "S" for Site Area Emergency, and "G" for General Emergency. The number is a sequential number for that Recognition Category series. All Initiating Conditions, which are describing the severity of a common condition (series), will have the same number (e.g. HA1, HA2, etc.).

A Fission Product Barrier (FPB) Table is provided as a subset to the Recognition Category "F" (FPB Degradation) of the EAL Matrix. This table is used to determine the integrity of the Fuel Clad, RCS and Containment Barriers based on EAL Threshold values established in accordance with NUMARC/NESP-007 (e.g., Intact, LOSS, or POTENTIAL LOSS).

3.2 EAL Technical Basis

Table LGS 3-2 serves as the Technical Basis for the EAL Matrix. The table consists of the following sections for each Initiating Condition (IC), sorted by Recognition Category:

- Initiating Condition
- Threshold Value
- Mode Applicability
- Basis (includes deviations from NUMARC/NESP-007 as appropriate)

Table LGS 3-2 provides the EAL user with the background and justification behind the EAL Threshold Values identified using the guidance set forth in NUMARC/NESP-007.

For a radiological liquid release, the emergency action level is based on calculated off-site dose from a chemistry sample. Shift Supervision utilizes emergency response procedures to notify risk counties and to obtain river water samples.

3.3 General EAL Implementation Philosophy

A broad spectrum of discretion in classifying events is provided in the "Discretionary" category under Hazards and Other Conditions and the Fission Product Barrier Matrix in Table LGS 3-1. In using the "Discretionary" category and in classifying emergencies under circumstances which are not a straight-forward use of the EAL's, ERO members should be mindful than an approach is needed which is conservative with respect to public, plant, and personnel safety and with respect to ensuring the adequacy of personnel and technical support. Conservative decisions must be made if the ED has any doubt regarding the health and safety of the public.

Declaring an Unusual Event provides the Company and off-site agencies the opportunity for early information regarding the event and for early activation of resources and may be considered a "no consequence decision." Conversely, not declaring an Unusual Event when there are credible (but, not clear) bases for doing so, would appear to be less than open or candid and could have serious adverse consequences. Although the consequences of declaring an Unusual Event are limited, inappropriate classifications do not accurately indicate the significance of the event to the public and emergency responders and should be avoided.

At the Alert, Site Area and General Emergency levels, clearly the threat to the plant and to the public is at a heightened level. Rapid application of resources and preparation for providing for the public health and safety are appropriate. Because of the magnitude of resource mobilization and the potential disruption of normal public activities, an overly conservative or an inappropriately early declaration of these levels is not advisable.

Events that meet the Emergency Action Level criteria for event declaration, but which are terminated before they are identified and declared, should still be classified and reported, but not declared to implement the Emergency Plan.

All EALs may not consider trends, rates of change, or status changes in equipment availability. In the event of rapidly changing parameters trending toward an increased emergency classification, it may be appropriate to decide that the higher level EAL will be exceeded and escalate the classification early. In the event of trends toward a decreased emergency classification, parameter values must be below the EALs to de-escalate.

In the event of a "spike" which rapidly exceeds and then exits an EAL condition, entry into the Emergency Plan or escalation to the higher classification "in retrospect" is not appropriate unless the "spike" is indicative of continuing degrading conditions which will lead to an escalated emergency classification level. This statement does not apply if the EAL includes a "spike". Spurious alarms or parameters, which are known to be invalid indicators of actual plant conditions or of the emergency classification, should not be used to declare emergency classifications.

TABLE LGS 3-1: Emergency Action Level (EAL) Matrix

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
ABNORMAL RAD LEVELS / EFFLUENTS					
Radiological Effluents	**RG1** Actual or Projected Site Boundary Dose Using Actual Meteorology: **MODES: ALL** > 1000 mRem TEDE **OR** > 5000 mRem CDE Thyroid EAL Threshold Value: 1. Radiological release in excess of Table R1 "General Emergency" threshold **AND** Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 Dose Assessment "General Emergency" thresholds **OR** 2. Radiological releases exceed ANY Table R2 column "General Emergency" threshold.	**RS1** Actual or Projected Site Boundary Dose Using Actual Meteorology: **MODES: ALL** > 100 mRem TEDE **OR** > 500 mRem CDE Thyroid EAL Threshold Value: 1. Radiological release in excess of Table R1 "Site Area Emergency" threshold **AND** Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 Dose Assessment "Site Area Emergency" thresholds **OR** 2. Radiological releases exceed ANY Table R2 column "Site Area Emergency" threshold	**RA1** Release > 200 X ODCM Limit for ≥ 15 minutes **MODES: ALL** EAL Threshold Value: 1. Unplanned radiological release lasting ≥ 15 minutes in excess of Table R1 "Alert" threshold **AND** Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 Dose Assessment "Alert" thresholds **OR** 2. Unplanned radiological releases lasting ≥ 15 minutes in excess of ANY Table R2 column "Alert" threshold	**RU1** Release > 2 X ODCM Limit for ≥ 60 minutes **MODES: ALL** EAL Threshold Value: 1. Unplanned radiological release lasting ≥ 60 minutes in excess of Table R1 "Unusual Event" threshold **AND** Releases **CANNOT** be determined in < 60 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 Dose Assessment "Unusual Event" thresholds **OR** 2. Unplanned radiological releases lasting ≥ 60 minutes in excess of ANY Table R2 column "Unusual Event" threshold	Radiological Effluents
Abnormal Rad Levels	Not Applicable	Not Applicable	**RA2** In-Plant Radiation Levels Impede Plant Operations **MODES: ALL** EAL Threshold Value: 1. Radiation readings > 15 mR/hr in EITHER of the following: • Main Control Room **OR** • Central Alarm Station **AND** Increase is NOT due to an anticipated increase from a planned event. **OR** 2. In-plant radiation reading > 5 R/hr **AND** Access is required to affected area(s) per SE-1, SE-6 or FSSG **AND** Increase is NOT due to an anticipated increase from a planned event.	**RU2** Rise In Plant Radiation Levels by a Factor of 1000 **MODES: ALL** EAL Threshold Value: 1. Radiation monitor readings indicate an unplanned rise by a factor of 1000 over normal levels. **RU3** High MSL or Off-gas Radiation Levels **MODES: 1,2,3** EAL Threshold Value: 1. SJAE Radiation (Offgas Monitor) reading > 2.1E+4 mR/hr **OR** 2. Main Steam Line Hi-Hi radiation Alarm (3xNFPB)	Abnormal Rad Levels
Coolant Activity	Not Applicable	Not Applicable	Not Applicable	**RU4** High coolant activity **MODES: ALL** EAL Threshold Value: 1. Reactor coolant activity > 4 μCi/gm I-131 dose equivalent	Coolant Activity

Table R1 -- Effluent Monitor Thresholds

	General Emergency	Site Area Emergency	Alert	Unusual Event
North Stack	> 3.00E+8 μCi/sec	> 3.00E+7 μCi/sec	> 200X Hi-Hi alarm	> 2X Hi-Hi alarm
	(WR Monitor: RIX-26-076-4)		(Normal Range Monitors: RY26-075A-3 / RY26-075B-3)	
South Stack	> 8.84E-2 μCi/cc	> 8.84E-3 μCi/cc	> 200X Hi-Hi alarm	> 2X Hi-Hi alarm
	(Unit 1: RY26-185A-3 / RY26-185-B-3 OR Unit 2: RY26-285A-3 / RY26-285-B-3)			
Radwaste Discharge	Not Applicable	Not Applicable	> 200X Hi-Hi alarm	> 2 X Hi-Hi alarm
Service Water	Not Applicable	Not Applicable	> 200X Hi-Hi alarm	> 2 X Hi-Hi alarm
RHRSW	Not Applicable	Not Applicable	> 200X Hi-Hi alarm	> 2 X Hi-Hi alarm

Table R2 Dose Assessment Thresholds

* Refers to dose/dose rates at or beyond the Site Boundary, based on a 1 hour release duration

Method	General Emergency	Site Area Emergency	Alert	Unusual Event
Sample Analysis	None	None	> 200X ODCM limits	> 2X ODCM limits
Field Team Monitoring*	> 1000 mRem/hr Whole Body **OR** > 5000 mRem CDE Thyroid	>100 mRem/hr Whole Body **OR** > 500 mRem CDE Thyroid	> 3 mRem/hr Whole Body **OR** > 9 mRem CDE Thyroid	Not Applicable
Dose Assessment*	> 1000 mRem TEDE **OR** > 5000 mRem CDE Thyroid	> 100 mRem TEDE **OR** > 500 mRem CDE Thyroid	> 2.8 mRem TEDE **OR** > 8.5 mRem CDE Thyroid	>0.114 mRem TEDE **OR** > 0.342 mRem CDE Thyroid

TABLE LGS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

FISSION PRODUCT BARRIER MATRIX (Applicability: Modes 1, 2 & 3 ONLY) MODES: 1,2,3

FISSION PRODUCT BARRIER STATUS	FG1: GENERAL EMERGENCY				FS1: SITE AREA EMERGENCY								FA1: ALERT				FU1: UNUSUAL EVENT	
Fuel Clad – LOSS	X	X		X	X		X		X				X					
Fuel Clad – POTENTIAL LOSS			X			X		X		X				X				
Reactor Coolant System – LOSS	X	X	X		X			X			X				X			
Reactor Coolant System – POTENTIAL LOSS				X		X	X					X				X		
Primary Containment – LOSS	X		X	X					X	X	X	X					X	
Primary Containment – POTENTIAL LOSS		X																X

	1. FUEL CLAD BARRIER		2. REACTOR COOLANT SYSTEM BARRIER		3. PRIMARY CONTAINMENT BARRIER	
	LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
a. Reactor Pressure Vessel (RPV) Water Level	1. RPV water level < -186 inches	2. RPV water level < -161 inches	1. RPV water Level < -161 inches	2. RPV water level CANNOT be determined	Not Applicable	1. ANY of the following direct entry into SAMP-1 and SAMP-2: • T-111 • T-116 • T-117
b. Drywell (DW) Radiation Monitor	1. DW radiation monitor reading > 4 2E+4 R/hr	Not Applicable	1. DW radiation monitor reading > 15R/hr	Not Applicable	Not Applicable	1. DW radiation monitor reading > 3.0E+5 R/hr
c. Drywell (DW) Pressure	Not Applicable	Not Applicable	1. Drywell pressure > 1.68 psig **AND** Indication of RCS leak inside Drywell	Not Applicable	1. Rapid, unexplained drop in DW pressure following an initial rise **OR** 2. DW pressure response not consistent with LOCA conditions indicating a Containment breach	3. DW pressure > 44 PSIG
d. Breached / Bypassed	1. Coolant activity > 300 uCi/gm I-131 dose equivalent **OR** 2. Core damage calculations indicate > 2.8% fuel clad damage	Not Applicable	1. <u>Unisolable</u> MSL Break indicated by the failure of BOTH MSIVs in ANY one line to close **AND** **EITHER** of the following: • High MSL Flow and High Steam Tunnel Temperature annunciators **OR** • Direct report of steam release [<u>NOTE</u>: Refer to MA8 for ISOLABLE MSL Break] **OR** 2. SRV is stuck open or cycling **AND** Indication of a LOSS of the Fuel Clad Barrier per Fission Product Barrier Matrix	3. RCS Leakage > 50 gpm [<u>NOTE</u>: Refer to MU7 for RCS leakage < 50 gpm.] **OR** 4. <u>Unisolable</u> primary system breach per T-103, as indicated by meeting conditions for **EITHER**: a. T-103, Step SCC/T-8 **OR** b. T-103, Step SCC/RAD-10	1. Failure of BOTH MSIVs in ANY one line to close *on* an isolation signal **AND** Downstream pathway exists to the turbine, condenser or directly into Turbine Enclosure. **OR** 2. Intentional venting per T-200 or T-228 is required **OR** 3. <u>Unisolable</u> primary system breach per T-103, as indicated by meeting conditions for **EITHER**: a. T-103, Step SCC/T-8 **OR** b. T-103, Step SCC/RAD-10	Not Applicable
e. Drywell Hydrogen Concentration	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable	1. Drywell H_2 > 6 % **AND** Drywell O_2 > 5 %
f. Discretionary	1. ANY condition that indicates a LOSS of the Fuel Clad Barrier	2. ANY condition that indicates a POTENTIAL LOSS of the Fuel Clad Barrier	1. ANY condition that indicates a LOSS of the Reactor Coolant System Barrier	2. ANY condition that indicates a POTENTIAL LOSS of the Reactor Coolant System Barrier	1. ANY condition that indicates a LOSS or of the Primary Containment Barrier	2. ANY condition that indicates a LOSS or of the POTENTIAL Primary Containment Barrier

TABLE LGS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
SYSTEM MALFUNCTIONS					
Loss of AC Power	**MG1** Prolonged Loss of ALL Offsite AC Power AND Prolonged Loss of ALL Onsite AC Power **MODES: 1,2,3** EAL Threshold Value: 1. Loss of offsite power to ALL 4 KV Safeguard Busses AND ALL four of the 4 KV Safeguard Busses are de-energized for > 15 minutes AND ANY of the following: • Restoration of at least one 4KV emergency bus in ≤ 2 hours is NOT likely OR • Reactor water level CANNOT be maintained > -161 inches OR • Suppression Pool temperature CANNOT be maintained on the "SAFE" side of the HCTL Curve (T-102, SP/T-1)	**MS1** Loss of ALL Offsite AC Power AND Loss of ALL Onsite AC Power to Essential Busses **MODES: 1,2,3** EAL Threshold Value: 1. Loss of offsite power to ALL 4 KV Safeguard Busses AND ALL four of the 4 KV Safeguard Busses are de-energized for > 15 minutes	**MA1** AC Power to Essential Busses Reduced to a Single Source for > 15 minutes **MODES: 1,2,3** EAL Threshold Value: 1. Loss of offsite power to ALL 4 KV Safeguard Busses AND Three of four of the 4 KV Safeguard Busses are de-energized for > 15 minutes **MA2** Loss of ALL Offsite AC Power AND Loss of ALL Onsite AC Power to Essential Busses **MODES: 4,5,D** EAL Threshold Value: 1. Loss of offsite power to ALL 4 KV Safeguard Busses AND ALL four of the 4 KV Safeguard Busses are de-energized for > 15 minutes	**MU1** Loss of ALL Offsite AC Power for > 15 minutes to Essential Busses **MODES: ALL** EAL Threshold Value: 1. Loss of offsite power to ALL 4 KV Safeguard Busses for > 15 minutes	Loss of AC Power
Loss of DC Power	Not Applicable	**MS3** Loss of ALL Required T.S. Safety-Related 125 VDC Power **MODES: 1,2,3** EAL Threshold Value: 1. Loss of ALL required T.S. safety related 125 VDC power for > 15 minutes as indicated by < 105 VDC on Panels 1(2)FA, B, C, D	Not Applicable	**MU3** Loss of BOTH Required T.S. Safety- Related 125 VDC Power Sources **MODES: 4,5** EAL Threshold Value: 1. Loss of BOTH required T.S. safety related 125 VDC power sources for > 15 minutes as indicated by < 105 VDC on Panels 1(2)FA, B, C, D	Loss of DC Power
Failure of Reactor Protection System	**MG4** Auto and Manual SCRAM NOT Successful, AND Loss of Core Cooling or Heat Sink **MODES: 1,2** EAL Threshold Value: 1. RPS set point has been exceeded for an automatic SCRAM AND Failure of automatic RPS, ARI AND Manual SCRAM/ARI to shutdown the reactor as defined by EITHER: • Reactor Power > 4% OR • Suppression Pool temperature greater than 110°F AND EITHER of the following criteria are met: • Suppression Pool temperature CANNOT be maintained on the "SAFE" side of the HCTL Curve (T-102, SP/T-1) OR • Reactor water level < -186 inches	**MS4** Auto and Manual SCRAM NOT Successful **MODES: 1,2** EAL Threshold Value: 1. RPS set point has been exceeded for an automatic SCRAM AND Failure of automatic RPS, ARI AND Manual SCRAM / ARI to shutdown the reactor as defined by EITHER: • Reactor Power > 4% OR • Suppression Pool temperature greater than 110°F	**MA4** Auto SCRAM NOT Successful **MODES: 1,2** EAL Threshold Value: 1. RPS set point has been exceeded for an automatic SCRAM AND Failure of automatic RPS to achieve a state in which the reactor is shutdown under all conditions without boron injection	Not Applicable	Failure of Reactor Protection System
Decay Heat Removal	Not Applicable	**MS5** Complete Loss of Functions Needed to Achieve AND Maintain Hot Shutdown **MODES: 1,2,3** EAL Threshold Value: 1. Loss of functions required for Hot Shutdown as evidenced by T-102 SP/T legs directing a T-112 Emergency Blowdown	**MA5** Inability to Maintain Plant in Cold Shutdown **MODES: 4,5** EAL Threshold Value: 1. Unplanned loss of ALL T.S. required decay heat removal systems AND EITHER of the following: • RCS temperature exceeding 200 °F for ≥ 15 minutes with a heat removal function restored OR • Uncontrolled RCS temperature rise approaching 200 °F with NO heat removal function restored	Not Applicable	Decay Heat Removal

TABLE LGS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
SYSTEM MALFUNCTIONS (cont.)					
Loss of Annunciators	None	**MS6** Inability to Monitor a Significant Transient In Progress **MODES: 1,2,3** EAL Threshold Value: 1. A significant transient is in progress (Table M-1) AND ALL of the following are lost: • Safety system annunciators (Table M-2) • Safety function indicators (Table M-3) • Plant Monitoring System	**MA6** Loss of Annunciators OR Indicators Requiring Increased Surveillance **MODES: 1,2,3** EAL Threshold Value: 1. Unplanned loss for > 15 minutes of MOST (NOTE 1) or ALL of EITHER: • Safety system annunciators (Table M-2) OR • Safety function indicators (Table M-3) AND Increased surveillance is required to safely operate the unit(s) AND EITHER of the following: • A significant plant transient is in progress (Table M-1) OR • Plant Monitoring System is unavailable	**MU6** Unplanned loss of Annunciators or Indicators for > 15 minutes **MODES: 1,2,3** EAL Threshold Value: 1. Unplanned loss for > 15 minutes of MOST (NOTE 1) or ALL of EITHER: • Safety system annunciators (Table M-2) OR • Safety function indicators (Table M-3) AND Increased surveillance is required to safely operate the unit(s)	Loss of Annunciators
RCS Leakage / RPV Draindown	None	**MS7** Loss of Water Level in the Reactor Vessel That Has OR Will Uncover Fuel in the Reactor Vessel **MODES: 4,5** EAL Threshold Value: 1. RPV water level < -161 inches	None	**MU7** Reactor Coolant System Leakage **MODES: 1,2,3,4** EAL Threshold Value: 1. Unidentified primary system leakage > 10 gpm into the Drywell OR 2. Identified primary system leakage > 25 gpm into the Drywell [NOTE: Refer to Fission Product Barrier Matrix (2 d 3) for RCS leakage > 50 gpm]	RCS Leakage / RPV Draindown
MSL Break (with Isolation)	None	None	**MA8** Main Steam Line Break **MODES: 1,2,3** EAL Threshold Value: 1. MSL Break indicated by EITHER of the following: • High MSL Flow and High Steam Tunnel Temperature annunciators OR • Direct report of MSL steam release AND MSL break is successfully isolated [NOTE: REFER to Fission Product Barrier Matrix (2 d.1) for possible event escalation if break is unisolable]	None	MSL Break (with Isolation)
Loss of Communications	None	None	None	**MU9** Unplanned Loss of ALL Onsite OR Offsite Communications Capabilities **MODES: ALL** EAL Threshold Value: 1. ALL onsite communications equipment lost (Table M-4) OR 2 ALL offsite communications equipment lost (Table M-5)	Loss of Communications
Technical Specs	None	None	None	**MU10** Inability to Reach Required Operating Mode Within Technical Specification Time Limits **MODES: 1,2,3** EAL Threshold Value: 1. Inability to reach required operating mode within Tech. Spec. LCO action completion time	Technical Specs

Table M-1: Significant Plant Transients	Table M-2: Safety System Annunciators	Table M-3: Safety Function Indicators	Table M-4: Onsite Communications Equipment	Table M-5: Offsite Communications Equipment	NOTE 1
• SCRAM • Recirc Runback (> 25% thermal power change) • Sustained Power Oscillations (25% peak to peak) • Stuck Open Relief Valves • ECCS Injection	• ECCS • Containment Isolation • Reactor Trip • Process Radiation Monitoring	• Reactor Power • Decay Heat Removal • Containment Safety Functions	• Station Phones • PRELUDE System • Plant Public Address (PA) • Station Radio	• Station Phones • PRELUDE System • NRC (ENS) • County Police Radio • Load Dispatcher Radio • PA State Police Radio	"MOST" refers to a loss of ~75% or a significant risk that a degraded plant condition could go undetected. Use is not intended to require a detailed count of annunciators/indicators.

TABLE LGS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
SYSTEM MALFUNCTIONS (cont.)					
Irradiated Fuel Accidents	None	None	**MA11** Major Damage OR Uncovering of Spent Fuel **MODES: ALL** EAL Threshold Value: 1. Unplanned general area radiation > 500 mR/hr on the Refuel Floor (Table M-6) **OR** 2. Report or visual observation that irradiated fuel is uncovered **OR** 3. Water level < 22 ft. above seated irradiated fuel for the Spent Fuel Pool that will result in uncovery	**MU11** Potential Damage OR Uncovering of Spent Fuel **MODES: ALL** EAL Threshold Value: 1. Uncontrolled water level drop in Spent Fuel Pool that cannot be quickly terminated with ALL irradiated fuel assemblies remaining covered by water	Irradiated Fuel Accidents
Irradiated Fuel Accidents	None	None	**MA12** Loss of Water Level That Has OR Will Uncover Irradiated Fuel **MODE: 5** EAL Threshold Value: 1. Water level < 22 ft. above RPV flange for the Reactor Refueling Cavity (484 inches RPV water level) **AND** Loss of water level has or will result in irradiated fuel uncovering	**MU12** Uncontrolled Water Level Decrease in Reactor Refueling Cavity **MODES: ALL** EAL Threshold Value: 1. Unexpected Fuel Pool Storage low level alarm **AND** Visual observation of an uncontrolled drop in water level below the fuel pool skimmer surge tank inlet that cannot be quickly terminated	Irradiated Fuel Accidents
HAZARDS AND OTHER CONDITIONS					
Security Events	**HG1** Security Event Resulting in Loss of Ability to Reach AND Maintain Cold Shutdown **MODES: ALL** EAL Threshold Value: 1. Loss of physical control of the Control Room due to a security event **OR** 2. Loss of physical control of the remote shutdown capability due to a security event	**HS1** Confirmed Security Event in a Vital Area **MODES: ALL** EAL Threshold Value: 1. Intrusion into plant Vital Area by a hostile force **OR** 2. Confirmed bomb, sabotage or sabotage device discovered in a Vital Area	**HA1** Confirmed Security Event in a Plant Protected Area **MODES: ALL** EAL Threshold Value: 1. Intrusion into the Protected Area by a hostile force **OR** 2. Confirmed bomb, sabotage or sabotage device discovered in the Protected Area	**HU1** Confirmed Security Event That Indicates a Potential Degradation in Level of Plant Safety **MODES: ALL** EAL Threshold Value: 1. A credible threat to the station reported by the NRC. **OR** 2. BOTH of the following criteria met for a credible threat reported by any other outside agency as determined per SY-AA-101-132, "Threat Assessment": • Is specifically directed towards the station. • Is imminent (≤ 2 hours) **OR** 3. Attempted intrusion and attack of the Protected Area **OR** 4. Attempted sabotage discovered within the Protected Area **OR** 5. Hostage/Extortion situation that threatens normal plant operations	Security Events
Control Room Evacuation	None	**HS2** Control Room Evacuation Initiated AND Plant Control CANNOT be re-established in ≤ 15 minutes **MODES: ALL** EAL Threshold Value: 1. Control Room evacuation initiated **AND** Control of the plant CANNOT be re-established in ≤ 15 minutes per SE-1 or SE-6	**HA2** Control Room Evacuation Initiated **MODES: ALL** EAL Threshold Value: 1. Entry into SE-1 or SE-6 for Control Room evacuation	None	Control Room Evacuation

Table M-6: Refuel Floor ARMs

• RIS29-M1-1(2)K600, Drywell Head Laydown • RIS30-M1-1(2)K600, Dryer / Separator Area • RIS31-M1-1(2)K600, Spent Fuel Pool	• RIS32-M1-1(2)K600, New Fuel Storage Vault • RIS33-M1-1(2)K600, Pool Plug Laydown

TABLE LGS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
HAZARDS AND OTHER CONDITIONS (cont.)					
Natural or Man-Made Events	None	None	**HA3** Natural OR Destructive Phenomena Affecting a Vital Area **MODES: ALL** EAL Threshold Value: 1. Earthquake > 0.075 g (Operating Basis Earthquake, OBE) as determined by procedure SE-5 **OR** 2. Tornado or wind speeds > 75 mph causing damage to Plant Vital Structures (Table H-1) **OR** 3. Report of visible structural damage to ANY Plant Vital Structure (Table H-1) **OR** 4. Vehicle crash affecting a plant vital function contained in a Plant Vital Structure (Table H-1) **OR** 5. Turbine failure generated missiles result in visible structural damage or penetration to ANY Plant Vital Structures (Table H-1) **OR** 6. Flooding in 2 or more areas designated in T-103, Table SCC-1 requiring a plant shutdown.	**HU3** Natural OR Destructive Phenomena Affecting the Protected Area **MODES: ALL** EAL Threshold Value: 1. Earthquake > 0 005 g as determined by procedure SE-5 **OR** 2. Report by plant personnel of a tornado strike within Protected Area **OR** 3. Wind speeds > 75 mph as indicated on Site Meteorological instrumentation for > 15 minutes **OR** 4. Vehicle crash within the Protected Area Boundary that may potentially damage plant structures containing function and systems required for safe shutdown of the plant **OR** 5. Report of turbine failure resulting in casing penetration or damage to generator seals **OR** 6. Assessment by Control Room that a natural or destructive phenomena has occurred affecting the Protected Area	Natural or Man-Made Events
Fire / Explosion	None	None	**HA4** Fire OR Explosion Affecting Operability of Safety Systems Required for Safe Shutdown **MODES: ALL** EAL Threshold Value: 1. ANY of the following are made potentially inoperable due to a fire or explosion: • 2 or more Safe Shutdown Systems (Table H-2) • 2 or more subsystems, as defined by Tech. Specs., of a Safe Shutdown System (Table H-2) • 1 or more Plant Vital Structures containing Safe Shutdown Equipment (Table H-1) **AND** Safe Shutdown System or Plant Vital Structure is required for the present Operational Condition	**HU4** Fire Within the Protected Area Boundary NOT Extinguished in ≤ 15 minutes of Detection **MODES: ALL** EAL Threshold Value: 1. Fire within or impacting a Plant Vital Structure (Table H-1) **AND** Fire is NOT extinguished in ≤ 15 minutes of EITHER: • Control Room notification • Verification of alarm **OR** 2. Report by plant personnel of an explosion within the Protected Area Boundary resulting in visible damage to a permanent structure or equipment	Fire / Explosion

Table H-1: Plant Vital Structures
• Reactor Enclosure
• Control Enclosure
• Turbine Enclosure
• Diesel Generator Enclosure
• Spray Pond Pump House / Spray Network

Table H-2: Safe Shutdown Systems	
• Diesel Generators	• RHR Service Water
• HPCI	• RERS
• Core Spray	• Control Room Ventilation
• SGTS	• RHR (all modes)
• PCIS (Primary CNTMT Isolation System)	• ESW
• 4 KV Safeguards Busses	• CAC
• RCIC	

TABLE LGS 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
HAZARDS AND OTHER CONDITIONS (cont.)					
Toxic or Flammable Gases	None	None	**HA5** Release of Toxic OR Flammable Gases Within a Facility Structure Which Jeopardizes Operation of Systems Required to Maintain Safe Operations OR to Establish or Maintain Cold Shutdown **MODES: ALL** EAL Threshold Value: 1. Report or detection of toxic gases within Plant Vital Structures (Table H-1) in concentrations that will be life threatening to plant personnel **OR** 2. Report or detection of flammable gases within Plant Vital Structures (Table H-1) in concentrations affecting the safe operation of the plant	**HU5** Release of Toxic OR Flammable Gases Deemed Detrimental to Safe Operation of the Plant **MODES: ALL** EAL Threshold Value: 1. Report or detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant **OR** 2. Report by Local, County, or State officials for a potential evacuation of site personnel based on an offsite event	**Toxic or Flammable Gases**
Discretionary	**HG6** Conditions Indicate Imminent Core Damage OR Release Affecting the Public **MODES: ALL** EAL Threshold Value: 1. Actual or imminent core degradation and potential loss of containment **OR** 2. Potential uncontrolled radio nuclide release, which can reasonably be expected to exceed 1 Rem TEDE or 5 Rem CDE Thyroid plume exposure levels at the Site Boundary	**HS6** Conditions Indicate Actual OR Likely Failure of Plant Functions Needed for Public Protection **MODES: ALL** EAL Threshold Value: 1. Other conditions exist which in the judgment of the Emergency Director indicate actual or likely major failures of plant functions needed for protection of the public	**HA6** Conditions Indicate Actual OR Potential Substantial Degradation of the Level of Plant Safety **MODES: ALL** EAL Threshold Value: 1. Other conditions exist which in the judgment of the Emergency Director indicate that plant safety systems may be degraded and that increased monitoring of plant functions is warranted.	**HU6** Conditions Indicate a Potential Degradation in the Level of Plant Safety **MODES: ALL** EAL Threshold Value: 1. ANY of the following occur, which in the judgment of the Emergency Director indicate a potential degradation in the level of safety of the plant: • Aircraft crash on-site • Train derailment on-site • Near-site explosion, which may adversely affect normal site activities **OR** 2. Other conditions exist which in the judgment of the Emergency Director indicate a potential degradation in the level of safety of the plant	**Discretionary**

Table H-1: Plant Vital Structures

- Reactor Enclosure
- Control Enclosure
- Turbine Enclosure
- Diesel Generator Enclosure
- Spray Pond Pump House / Spray Network

Table LGS 3-2: LGS EAL Technical Basis

TABLE OF CONTENTS

TAB "R" – ABNORMAL RADIATION LEVELS / EFFLUENTS (Category "R")

1. **Radiological Effluents**
   - RG1 (Modes: ALL) .... LGS 3-15
   - RS1 (Modes: ALL) .... LGS 3-18
   - RA1 (Modes: ALL) .... LGS 3-20
   - RU1 (Modes: ALL) .... LGS 3-23
2. **Abnormal Radiation Levels**
   - RA2 (Modes: ALL) .... LGS 3-26
   - RU2 (Modes: ALL) .... LGS 3-28
   - RU3 (Modes: 1, 2 & 3) .... LGS 3-29
3. **Coolant Activity**
   - RU4 (Modes: ALL) .... LGS 3-30

TAB "F" – FISSION PRODUCT BARRIER DEGRADATION

- FG1 (Modes: 1, 2 & 3) .... LGS 3-31
- FS1 (Modes: 1, 2 & 3) .... LGS 3-33
- FA1 (Modes: 1, 2 & 3) .... LGS 3-34
- FU1 (Modes: 1, 2 & 3) .... LGS 3-35

1. **Fuel Clad Barrier**
   - Reactor Pressure Vessel Water Level (1.a) .... LGS 3-36
   - Drywell Radiation (1.b) .... LGS 3-37
   - Drywell Pressure (1.c) .... LGS 3-40
   - Breached / Bypassed (1.d) .... LGS 3-41
   - Drywell Hydrogen Concentration (1.e) .... LGS 3-43
   - Discretionary (1.f) .... LGS 3-44
2. **Reactor Coolant System Barrier**
   - Reactor Pressure Vessel Water Level (2.a) .... LGS 3-45
   - Drywell Radiation (2.b) .... LGS 3-46
   - Drywell Pressure (2.c) .... LGS 3-47
   - Breached / Bypassed (2.d) .... LGS 3-48
   - Drywell Hydrogen Concentration (2.e) .... LGS 3-51
   - Discretionary (2.f) .... LGS 3-52
3. **Primary Containment Barrier**
   - Reactor Pressure Vessel Water Level (3.a) .... LGS 3-53
   - Drywell Radiation (3.b) .... LGS 3-54
   - Drywell Pressure (3.c) .... LGS 3-55
   - Breached / Bypassed (3.d) .... LGS 3-56
   - Drywell Hydrogen Concentration (3.e) .... LGS 3-58
   - Discretionary (3.f) .... LGS 3-59

TABLE OF CONTENTS

TAB "M" – SYSTEM MALFUNCTIONS

1. Loss of AC Power (Operating)
   - MG1 (Modes: 1, 2 & 3) .... LGS 3-60
   - MS1 (Modes: 1, 2 & 3) .... LGS 3-62
   - MA1 (Modes: 1, 2 & 3) .... LGS 3-63
   - MU1 (Modes: ALL) .... LGS 3-64
2. Loss of AC Power (Shutdown)
   - MA2 (Modes: 4, 5 & Defuel) .... LGS 3-65
3. Loss of DC Power
   - MS3 (Modes: 1, 2 & 3) .... LGS 3-66
   - MU3 (Modes: 4 & 5) .... LGS 3-68
4. Failure of Reactor Protection System
   - MG4 (Modes: 1 & 2) .... LGS 3-69
   - MS4 (Modes: 1 & 2) .... LGS 3-71
   - MA4 (Modes: 1 & 2) .... LGS 3-73
5. Decay Heat Removal
   - MS5 (Modes; 1, 2 & 3) .... LGS 3-75
   - MA5 (Modes: 4 & 5) .... LGS 3-76
6. Loss of Annunciators
   - MS6 (Modes: 1, 2 & 3) .... LGS 3-78
   - MA6 (Modes: 1, 2 & 3) .... LGS 3-79
   - MU6 (Modes: 1, 2 & 3) .... LGS 3-81
7. RCS Leakage / RPV Draindown
   - MS7 (Modes: 4 & 5) .... LGS 3-82
   - MU7 (Modes: 1, 2, 3 & 4) .... LGS 3-84
8. Main Steam Line Break (with isolation)
   - MA8 (Modes: 1, 2 & 3) .... LGS 3-86
9. Loss of Communications
   - MU9 (Modes: ALL) .... LGS 3-85
10. Technical Specifications
    - MU10 (Modes: 1, 2 & 3) .... LGS 3-86
11. Irradiated Fuel Accidents (Spent Fuel Pool)
    - MA11 (Modes: ALL) .... LGS 3-89
    - MU11 (Modes: ALL) .... LGS 3-91
12. Irradiated Fuel Accidents (Reactor Refuel Cavity)
    - MA12 (Modes: 5) .... LGS 3-90
    - MU12 (Modes: ALL) .... LGS 3-91

TABLE OF CONTENTS

TAB "H" – HAZARDS AND OTHER CONDITIONS

1. **Security Events**
   - HG1 (Modes: ALL) .... LGS 3-94
   - HS1 (Modes: ALL) .... LGS 3-95
   - HA1 (Modes: ALL) .... LGS 3-96
   - HU1 (Modes: ALL) .... LGS 3-97
2. **Control Room Evacuation**
   - HS2 (Modes: ALL) .... LGS 3-98
   - HA2 (Modes: ALL) .... LGS 3-99
3. **Natural or Man-Made Events**
   - HA3 (Modes: ALL) .... LGS 3-100
   - HU3 (Modes: ALL) .... LGS 3-102
4. **Fire / Explosion**
   - HA4 (Modes: ALL) .... LGS 3-104
   - HU4 (Modes: ALL) .... LGS 3-106
5. **Toxic or Flammable Gases**
   - HA5 (Modes: ALL) .... LGS 3-108
   - HU5 (Modes: ALL) .... LGS 3-110
6. **Discretionary**
   - HG6 (Modes: ALL) .... LGS 3-111
   - HS6 (Modes: ALL) .... LGS 3-112
   - HA6 (Modes: ALL) .... LGS 3-113
   - HU6 (Modes: ALL) .... LGS 3-114

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RG1

INITIATING CONDITION

Actual or Projected Site Boundary Dose Using Actual Meteorology:

> 1000 mRem TEDE

OR

> 5000 mRem CDE Thyroid

EAL THRESHOLD VALUES

1. Radiological release in excess of Table R1 "General Emergency" threshold
   AND
   Release **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "General Emergency" thresholds

   OR

2. Radiological releases exceed **ANY** Table R2 column "General Emergency" threshold.

Table R1: Effluent Monitor Thresholds	
	General Emergency
North Stack	> 3.00E+8 µCi/sec **(Wide Range Monitor: RIX-26-076-4)**
South Stack	> 8.84E-2 µCi/cc **(Unit 1: RY26-185A-3 / RY26-185B-3)** **(Unit 2: RY26-285A-3 / RY26-285B-3)**

Table R2: Dose Assessment Thresholds	
Method	**General Emergency**
Sample	N/A
Field Team Monitoring*	> 1000 mRem/hr for Whole Body **OR** > 5000 mRem CDE Thyroid
Dose Projection*	> 1000 mRem TEDE **OR** > 5000 mRem CDE Thyroid
*** At or beyond Site Boundary based on a 1 hour release duration**	

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RG1 – Cont'd

MODE APPLICABILITY:

ALL

BASIS: (References)

Site Boundary - For classification and dose projection purposes, the Site Boundary is the Exclusion Area Boundary, a 2500-foot radius around the plant. The actual boundary is specified in the ODCM.

Total Effective Dose Equivalent (TEDE) The sum of the deep dose equivalent (for external exposure) and the committed effective dose equivalent (for internal exposure) and 4 days of deposition exposure.

Committed Dose Equivalent (CDE) The Dose Equivalent to organs or tissues of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake. Thyroid values are taken from EPA-400, Table 5-4 to be consistent with the NRC RASCAL dose assessment program used by the Pennsylvania Emergency Management Agency (PEMA) / Bureau of Radiation Protection (BRP). Actual meteorology is used, since it gives the most accurate dose projection.

Table R1:

Effluent Monitors - Classification is based on the instantaneous release rate value if NO dose projections can be performed or verified within 15 minutes of meeting or exceeding the specified Release Rate value. Monitor indications are calculated using the computerized dose model with UFSAR (gap release) source terms applicable to each monitored pathway in conjunction with annual average meteorology per NUMARC/NESP-007 (Revision 2). Calculations assume:

	North Stack	South Stack
Time After Shutdown (TAS)	0 hours	0 hours
Release Duration	1 hour	1 hour
Core Damage	10%	10%
Flow Rate (per ODCM)	668,450 cfm	234,000 cfm
Annual Average X/Q (per ODCM)	1.1E-5 sec/m^3	1.1E-5 sec/m^3
Process Reduction Factors (per NUREG-1228)		
• CNMT Hold Up / Spray OFF	< 2 hours	< 2 hours
• Reactor Bldg. Hold Up	< 2 hours	< 2 hours
• Standby Gas Treatment Filters	YES	N/A
• RERS Filters	YES	N/A

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RG1 – Cont'd

BASIS: (References)

Table R2:

Field Team Monitoring – 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. The assumed release duration is 1 hour. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption media followed by field analysis are used for determining the iodine value.

Dose Projection – Any calculated dose projection of 1 Rem TEDE or 5 Rem CDE Thyroid is classified based on U.S. Environmental Protective Action (EPA) guidelines established under EPA-400-R-92-001 (May 1992). Source term, release elevation and release duration inputs are options and should reflect actual release conditions. Actual meteorology should also be used to reflect actual release conditions.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RS1

INITIATING CONDITION

Actual or Projected Site Boundary Dose Using Actual Meteorology:

>100 mRem TEDE
OR
>500 mRem CDE Thyroid

EAL THRESHOLD VALUES

1. Radiological release in excess of Table R1 "Site Area Emergency" threshold
   AND
   Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "Site Area Emergency" thresholds

   OR

2. Radiological releases exceed **ANY** Table R2 column "Site Area Emergency" threshold.

Table R1: Effluent Monitor Thresholds

	Site Area Emergency
North Stack	> 3.00E+7 µCi/sec (Wide Range Monitor: RIX-26-076-4)
South Stack	> 8.84E-3 µCi/cc (Unit 1: RY26-185A-3 / RY26-185B-3) (Unit 2: RY26-285A-3 / RY26-285B-3)

Table R2: Dose Assessment Thresholds

Method	Site Area Emergency
Sample	N/A
Field Team Monitoring*	> 100 mRem/hr Whole Body **OR** > 500 mRem CDE Thyroid
Dose Projection*	> 100 mRem TEDE **OR** > 500 mRem CDE Thyroid

* At or beyond Site Boundary based on a 1 hour release duration

MODE APPLICABILITY:

ALL

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RS1 – Cont'd

BASIS: (References)

Table R1:

Effluent Monitors - Classification is based on the instantaneous release rate value if NO dose projections can be performed or verified within 15 minutes of meeting or exceeding the specified Release Rate value. Monitor indications are calculated using the computerized dose model with UFSAR (gap release) source terms applicable to each monitored pathway in conjunction with annual average meteorology per NUMARC/NESP-007 (Revision 2). Calculations assume:

	North Stack	South Stack
Time After Shutdown (TAS)	0 hours	0 hours
Release Duration	1 hour	1 hour
Core Damage	10%	10%
Flow Rate (per ODCM)	668,450 cfm	234,000 cfm
Annual Average X/Q (per ODCM)	1.1E-5 sec/m^3	1.1E-5 sec/m^3
Process Reduction Factors (per NUREG-1228)		
• CNMT Hold Up / Spray OFF	< 2 hours	< 2 hours
• Reactor Bldg. Hold Up	< 2 hours	< 2 hours
• Standby Gas Treatment Filters	YES	N/A
• RERS Filters	YES	N/A

Table R2:

Field Team Monitoring – 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. The assumed release duration is 1 hour. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption media followed by field analysis is used for determining the iodine value.

Dose Projection – Any calculated dose projection of 100 mRem TEDE or 500 mRem CDE Thyroid is classified based on 10% of the guidelines established under EPA-400-R-92-001 (May 1992). Source term, release elevation and release duration inputs are options and should reflect actual release conditions. Actual meteorology should also be used to reflect actual release conditions.

This event will be escalated to a General Emergency when actual or projected doses exceed EPA-400-R-92-001 Protective Action Guidelines per IC RG1.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA1

INITIATING CONDITION

Release > 200 X ODCM Limit for ≥ 15 minutes

EAL THRESHOLD VALUES

1. Unplanned radiological release lasting ≥ 15 minutes in excess of Table R1 "Alert" thresholds
   AND
   Releases **CANNOT** be determined in < 15 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "Alert" thresholds
   OR
2. Unplanned radiological releases lasting ≥ 15 minutes in excess of ANY Table R2 column "Alert" threshold

Table R1: Effluent Monitor Thresholds	
	Alert
• **North Stack (RY26-075A-3 / RY26-075B-3)** • **South Stack (U1: RY26-185A-3 / RY26-185B-3) (U2: RY26-285A-3 / RY26-285B-3)** • **Radwaste Discharge** • **Service Water** • **RHRSW**	> 200 X Hi Hi alarm set point

Table R2: Dose Assessment Thresholds	
Method	**Alert**
Sample	> 200 X ODCM limit
Field Team Monitoring*	> 3 mRem Whole Body **OR** > 9 mRem CDE Thyroid
Dose Projection*	> 2.8 mRem TEDE **OR** > 8.5 mRem CDE Thyroid
*** At or beyond Site Boundary based on a 1 hour release duration**	

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA1 – Cont'd

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned – Any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm set points, etc.) on the applicable permit

It is not intended that the release be averaged over 15 minutes. A release of this greater magnitude that cannot be terminated in 15 minutes represents an uncontrolled situation that is an actual or potential substantial degradation of the level of safety of the plant. The degradation in plant control implied by the fact that the release cannot be terminated in 15 minutes is the primary concern.

Further, 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 15 minutes, unless release rate confirmation (sample, field survey or dose projection) is likely within this 15 minute period.

Table R1:

Effluent Monitors – EAL thresholds are based on 200 times the ODCM limits. For event classification purposes, the HI-HI Radiation alarms for the North Vent and South Vent are used to represent the ODCM limit. These alarm set points are using the calculation methodology as outlined in Sections 2.2 of the ODCM, which uses the highest annual average X/Q value for the designated sectors in accordance with NUMARC/NESP-007 (AA1). The HI Radiation alarm set points are also set conservatively to indicate when a release may approach ODCM limits assuming multiple release points.

Table R2:

It is intended that the event be declared as soon as it is determined that the release will exceed two hundred times ODCM for greater than 15 minutes.

Samples – Grab samples are used to determine release concentrations or rates to confirm effluent monitor readings or when the effluent monitors are out of services.

Field Team Monitoring – 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. The assumed release duration is 1 hour. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption media followed by field analysis is used for determining the iodine value.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA1 – Cont'd

BASIS (References) – Cont'd

Readings are based on equivalent dose model triggers representing 200X ODCM limit. Values are rounded up to the next highest unit based on conservatism of dose projections and to allow reading on survey meter.

Dose Projection – This EAL includes a 15-minute average for the dose projection with the release point radiation monitor above two hundred times the Hi Hi alarm set point value for the entire 15 minutes. It is not intended that the release be averaged over 15 minutes, but exceed threshold for 15 minutes.

TEDE and CDE Thyroid thresholds used represent offsite dose triggers built into the dose model to conservatively reflect 200X ODCM limit.

Releases in excess of 200 times the ODCM limits that continue for > 15 minutes represent an uncontrolled situation and hence a potential degradation in the level of safety. The primary concern is the final integrated dose [100 times greater than the Unusual Event] and the degradation in plant control implied by the fact that the release was not isolated within 15 minutes.

This event will be escalated to a Site Area Emergency when actual or projected doses are determined to exceed 10CFR20 annual average population exposure limits per IC RS1.

Table LGS 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU1

INITIATING CONDITION

Release > 2 X ODCM Limit for ≥ 60 minutes

EAL THRESHOLD VALUE

1. Unplanned radiological release lasting ≥ 60 minutes in excess of Table R1 "Unusual Event" threshold
   AND
   Releases **CANNOT** be determined in < 60 minutes (from the time Table R1 threshold was exceeded) to be below Table R2 **Dose Assessment** "Unusual Event" thresholds
   OR
2. Unplanned radiological releases lasting ≥ 60 minutes in excess of **ANY** Table R2 column "Unusual Event" threshold

Table R1: Effluent Monitor Thresholds

	Unusual Event
• North Stack (RY26-075A-3 / RY26-075B-3) • South Stack (U1: RY26-185A-3 / RY26-185B-3) (U2: RY26-285A-3 / RY26-285B-3) • Radwaste Discharge • Service Water • RHRSW	> 2 X Hi Hi alarm set point

Table R2: Dose Assessment Thresholds

Method	Unusual Event
Sample	> 2 X ODCM limit
Field Team Monitoring	N/A
Dose Projection*	> 0.114 mRem TEDE **OR** > 0.342 mRem CDE Thyroid

* At or beyond Site Boundary based on a 1 hour release duration

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU1 – Cont'd

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned - Any release for which a radioactive discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm set points, etc.) on the applicable permit

Unplanned releases in excess of two times the site ODCM 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 limits for 30 minutes does not exceed this EAL. Further, 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, if it is unlikely that an ODCM calculation can be performed within 60 minutes of exceeding the EAL threshold.

Table R1:

Effluent Monitors – EAL thresholds are based on two times the ODCM limits. For event classification purposes, the HI-HI Radiation alarms for the North Vent and South Vent are used to represent the ODCM limit. These alarm set points are using the calculation methodology as outlined in Sections 2.2 of the which uses the highest annual average X/Q value for the designated sectors in accordance with NUMARC/NESP-007 (AU1). The HI Radiation alarm set points are also set conservatively to indicate when a release may approach ODCM limits assuming multiple release points.

Table R2:

It is intended that the event be declared as soon as it is determined that the release will exceed two times ODCM for greater than 60 minutes.

Samples – Grab samples are used to determine release concentrations or rates to confirm effluent monitor readings or when the effluent monitors are out of services.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU1 – Cont'd

BASIS (References)

Dose Projection – This EAL includes a 60 minute average for the dose projection with the release point radiation monitor above two times the Hi Hi alarm set point value for the entire 60 minutes. It is not intended that the release be averaged over 60 minutes, but exceed threshold for 60 minutes.

TEDE and CDE Thyroid thresholds used represent offsite dose triggers built into the dose model to conservatively reflect 200X ODCM limit.

Releases in excess of 2 times the ODCM limits, that continue for > 60 minutes represent an uncontrolled situation and hence a potential degradation in the level of safety. The final integrated dose is very low and is not the primary concern. Rather it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

This event will be escalated to an Alert when release is determined to be >200 x ODCM Limit for >15 minutes per IC RA1.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA2

INITIATING CONDITION

In-Plant Radiation Levels Impede Plant Operations

EAL THRESHOLD VALUES

1. Radiation readings > 15 mR/hr in **EITHER** of the following:
   - Main Control Room

     OR
   - Central Alarm Station

   AND

   Increase is **NOT** due to an anticipated increase from a planned event.

   OR
2. In-plant radiation readings > 5 R/hr

   AND

   Access is required to affected area(s) per SE-1, SE-6 or FSSG

   AND

   Increase is **NOT** due to an anticipated increase from a planned event.

MODE APPLICABILITY

ALL

BASIS (References)

This EAL addresses elevated radiation levels that impede necessary access to operator stations, or other areas containing equipment that must be operated manually in order to maintain safe operation or to perform a safe shutdown. The concern of the EAL is a loss of control of radioactive material causing high radiation levels. As such, this EAL is not intended to apply to anticipated temporary increases in radiation levels due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, controlled movement of radiological sources, or expected increases in area radiation levels due to normal operation of plant systems / components, etc.).

The impaired ability to operate the plant is to be considered as the actual or potential substantial degradation of the level of safety of the plant. The cause of the rise in radiation levels is not the major concern of this EAL. For example, a dose rate of 15 mR/hr in the control room or hi radiation monitor readings may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, the fission product barrier table may indicate a SAE or GE. This EAL could result in declaration of an Alert at one unit due to a radioactivity release or radiation shine resulting from a major accident at the other unit.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RA2 – Cont'd

BASIS (References) – Cont'd

Threshold Value 1 - The value of 15 mRem/hr is derived from the general design criteria (GDC) value of 5 REM in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737 "Clarification of TMI Action Plan Requirements" provides that the 15 mRem/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an ALERT.

Plant normal and emergency procedures may be implemented without requiring any areas except the Control Room and Central Alarm Station to be continuously occupied. The Radwaste Control Room is not required to be continuously occupied in order to maintain plant safety functions since inputs to radwaste will be isolated with secondary containment isolation and releases can only be performed manually.

Threshold Value 2 - Areas requiring infrequent access and dose rate values are based on those specified in the procedures SE-1, SE-6 or FSSG. EAL is applicable only when procedures SE-1, SE-6 or FSSG direct access, in other words, when you are in those procedures. Therefore, if you were in procedures SE-1, SE-6 or FSSG and you needed direct access to a particular area and at the time radiation levels were > 5 R/hr, classification under this EAL is appropriate. Just having radiation levels > 5 R/hr in those areas defined in SE-1, SE-6 or FSSG, when access is not directed per procedure, does not warrant classification under this EAL.

The single value of 5 R/hr was selected because it is based on radiation levels which result in exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits (i.e., 10 CFR 20), and in doing so, will impede necessary access. Stay times for levels up to that value are, generally several minutes, enough time to enter an area and manually operate the equipment. Dose rates > 5 R/hr will impede necessary access.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU2

INITIATING CONDITION

Rise in Plant Radiation Levels By a Factor of 1000

EAL THRESHOLD VALUE

1. Radiation monitor readings indicate an unplanned rise by a factor of 1000 over normal levels

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned - Not the result of an intended evolution and requiring corrective or mitigative actions.

Normal Levels – Normal radiation levels can be considered as the highest reading in the past 24-hour period, excluding the current peak value, as determined by recorder charts, surveys, logs, etc..

Classification of an UNUSUAL EVENT is warranted as a precursor to more serious events. The concern of this EAL is the loss of control of radioactive material representing a potential degradation of the level of safety of the plant. The Threshold Value tends to have a long lead-time relative to a radiological release and thus the threat to public health and safety is very low. In light of the elevated dose rates the Emergency Director should evaluate how these conditions will affect the other unit.

This event will be escalated to an Alert when in-plant radiation levels impede plant operations per IC RA2.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU3

INITIATING CONDITION:

High Main Steam Line (MSL) OR Off-gas Radiation Levels

EAL THRESHOLD VALUES

1. SJAE Radiation (Off-gas Monitor) > 2.1E+4 mR/hr

   OR

2. Main Steam Line Hi-Hi radiation Alarm (3xNFPB)

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

Threshold Value 1: The steam jet air ejector radiation monitor in the Control Room would be one of the first indicators of a degrading core. This instrument takes a sample before the recombiner. This indicator of elevated activity is equivalent to Technical Specification limit and is considered to be a precursor of more serious problems.

Threshold Value 2: Main Steam Line (MSL) Hi-Hi Radiation alarm > 3 times normal full power background (NFPB), may be indicative of minor fuel cladding degradation. The MSL Hi-Hi radiation condition requires a manual Main Steam Isolation Valve (MSIV) closure and a reactor scram. This transient may result in the introduction of fission product gases (previously contained in the gap area) to be suddenly released into the coolant due to the rapid down power transient and subsequent collapse of voids in the coolant.

This level of steam line activity is indicative of the release of gap activity to the coolant, rather than a major failure of the fuel clad. However, the mechanics that caused MSL radiation to rise to this level indicate there is a degradation of Fuel Clad integrity.

This EAL is NOT intended to apply to cases caused by resin intrusion or other known factors that are not directly indicative of fuel cladding degradation, but rather coolant chemistry issues.

This event will escalate to a Site Area Emergency based on a MSL break per the Fission Product Barrier Matrix (2.d.1 & 3.d.1) for an unisolable break, **OR** an Alert based on MA8 for an isolable MSL break scenario.

DEVIATION: The OPCON applicability [1,2,3] is a deviation from NUMARC [all] in that, the SJAE Radiation Monitor and Main Steam Line Radiation Monitors will only be a valid indication of Fuel Clad Degradation in those OPCON's. There are no other monitors, which can be used as an indicator of Fuel Clad Degradation.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / EFFLUENTS

RU4

INITIATING CONDITION

High Coolant Activity

EAL THRESHOLD VALUES

Reactor coolant activity > 4 µCi/gm I-131 dose equivalent

MODE APPLICABILITY:

ALL

BASIS: (References)

Coolant activity in excess of Technical Specifications (> 4 µCi/gm) is considered to be a precursor of more serious problems. The Technical Specification limit reflects a degrading or degraded core condition. This level is chosen to be above any possible short duration spikes under normal conditions. 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.

This event will be escalated to an Alert when Reactor Coolant activity exceeds 300 µCi/gm Dose Equivalent Iodine 131 per Fission Product Barrier Matrix 1.d.1.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FG1

INITIATING CONDITION

LOSS of Two Fission Product Barriers and POTENTIAL LOSS of the Third Barrier

EAL THRESHOLD VALUE

Comparison of conditions / values with those listed in Fission Product Barrier Matrix indicates:

LOSS of ANY Two Barriers

AND

POTENTIAL LOSS of Third Barrier

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Conditions / events required to cause the loss of 2 Fission Product Barriers with the potential loss of the third could reasonably be expected to cause a release beyond the immediate site area exceeding EPA Protective Action Guidelines.

Guidance for development of this EAL was taken from Recognition Category F in the NUMARC/NEI Methodology for Development of Emergency Action Levels.

A barrier LOSS shall also constitute a POTENTIAL LOSS for classification purposes.

NOTES:

1. Although the logic used for these initiating conditions appears overly complex, it is necessary to reflect the following considerations:

   - The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Primary Containment barrier. Unusual Event Initiating Conditions (ICs) associated with RCS and Fuel Clad barriers are addressed under the other plant condition EALs.

   - At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from General Emergency. For example, if the Fuel Clad barrier and RCS barrier "Loss" EALs existed, this would indicate to the Emergency Director that, in addition to offsite dose assessments, the ED must focus on continual assessments of radioactive inventory and containment integrity. If, on the other hand, both Fuel Clad barrier and RCS barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FG1 – Cont'd

BASIS (References) – Cont'd

- The ability to escalate to higher emergency classes as an event gets worse must be maintained. For example, RCS leakage steadily rising would represent an increasing risk to public health and safety.

2. Fission Product Barrier ICs must be capable of addressing event dynamics. Thus, the EAL Reference Table states that IMMINENT (i.e., within 1 to 2 hours) Loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

3. The Fuel Clad barrier is the cladding tubes that contain the fuel pellets.

4. The RCS Barrier is the reactor coolant system pressure boundary and includes the reactor vessel and all reactor coolant system piping up to the isolation valves.

5. The Primary Containment Barrier includes the drywell, the suppression pool, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves.

6. If a "Loss" condition is satisfied, the "Potential Loss" category can be considered satisfied. This is also applicable to conditions where there is a "Loss" indication with no corresponding "Potential Loss" condition.

7. For all conditions listed in Fission Product Barrier Table, the barrier failure column is only satisfied if it fails when called upon to mitigate an accident. For example, failure of both 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 operations, it will be an active Technical Specification Action Statement. However, during accident conditions, this will represent a breach of Primary Containment.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FS1

INITIATING CONDITION / EAL THRESHOLD VALUE

LOSS of **BOTH** Fuel Clad and RCS Barriers

OR

POTENTIAL LOSS of **BOTH** Fuel Clad and RCS Barriers

OR

POTENTIAL LOSS of **EITHER** the Fuel Clad or RCS Barrier, **AND** a LOSS of **ANY** Additional Barrier

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Loss of 2 Fission Product Barriers would be a major failure of plant systems needed for protection of the public.

Guidance for development of this EAL was taken from Recognition Category F in the NUMARC/NEI Methodology for Development of Emergency Action Levels.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FA1

INITIATING CONDITION

ANY LOSS or ANY POTENTIAL LOSS of EITHER the Fuel Cladding or Reactor Coolant System

EAL THRESHOLD VALUE

Comparison of conditions / values with those listed in Fission Product Barrier Matrix indicates:

LOSS or POTENTIAL LOSS of the Fuel Cladding Barrier

OR

LOSS or POTENTIAL LOSS of the Reactor Coolant System Barrier

MODE APPLICABILITY

1, 2, 3

BASIS (References)

The Fuel Cladding and the Reactor Coolant System are weighted more heavily than the Containment Barrier.

A LOSS or POTENTIAL LOSS of either the Fuel Cladding or the Reactor Coolant System would be a substantial degradation in the level of plant safety.

Guidance for development of this EAL was taken from Recognition Category F in the NUMARC/NEI Methodology for Development of Emergency Action Levels.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FU1

INITIATING CONDITION

ANY LOSS or ANY POTENTIAL LOSS of Containment

EAL THRESHOLD VALUE

Comparison of conditions / values with those listed in Fission Product Barrier Matrix indicates:

LOSS of the Containment Barrier

OR

POTENTIAL LOSS of the Containment Barrier

MODE APPLICABILITY

1, 2, 3

BASIS (References)

The Fuel Cladding and the Reactor Coolant System are weighted more heavily than the Containment Barrier.

Loss of the Containment would be a potential degradation in the level of plant safety.

Guidance for development of this EAL was taken from Recognition Category F in the NUMARC/NEI Methodology for Development of Emergency Action Levels.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.a

INITIATING CONDITION

Reactor Pressure Vessel (RPV) Water Level

THRESHOLD VALUE

LOSS:....................................1. RPV water level < -186 inches

POTENTIAL LOSS:............2. RPV water Level < -161 inches

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS – [Threshold Value #1] Value of –186 inches corresponds to the level, which is used in the TRIP guidance to indicate challenge of core cooling. This is the Minimum Steam Cooling RPV Water Level, and as such, is the lowest RPV level at which the submerged portion of the core will generate sufficient steam to prevent any clad in the uncovered portion of the core from exceeding 1500°F. This RPV level is utilized to preclude fuel damage when RPV level is below the Top of Active Fuel (TAF).

Core submergence is the preferred method of core cooling and as such, the failure to re-establish RPV water level above the top of active fuel for an extended period of time could lead to significant fuel damage. This condition, RPV level < -186 ", could be indicative of a large break Loss Of Coolant Accident (LOCA) where ECCS Systems are designed to maintain level at 2/3 core height, or a small LOCA with the inability of emergency core cooling systems to reflood the RPV.

POTENTIAL LOSS – [Threshold Value #2] Core submergence is the preferred method of core cooling, and as such, the failure to re-establish RPV water level above TAF for an extended period of time could lead to significant fuel damage.

A level of < -161 inches also corresponds to the EAL for a RCS Barrier LOSS (IC RCS 2.a.1). Thus, this EAL indicates a LOSS of RCS barrier and a POTENTIAL LOSS of the Fuel Clad Barrier.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.b

INITIATING CONDITION

Drywell (DW) Radiation Monitor

THRESHOLD VALUE

LOSS: .... 1. DW radiation monitor reading > 4.2E+4 R/hr

POTENTIAL LOSS: ....2. NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - The > 4.2E+4 R/hr reading on a containment high range radiation monitor RR-26-191, 291A, B, C, D, indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell. The intent is not to verify criteria used in calculation (e.g., release of reactor coolant into drywell), but rather to classify once EAL threshold value is reached or exceeded.

[Calculation Basis] The reading was calculated assuming an instantaneous release and dispersal of the Reactor Coolant noble gas and iodine inventory into the Primary Containment (direct reading not shine) at a coolant concentration of 300 μCi/gm Dose Equivalent Iodine 131.

This calculation is as follows:

Using Curve 3 [1%] of attached "Containment Radiation Monitor Dose Rate Curves":

Time after Shutdown = 1 hour (more conservative due to lower value for EAL)

1% fuel clad damage: dose rate = 15,000 R/hr

Extrapolating to 2.8%: (15,000 R/hr/1%)(2.8) = 42,000 R/hr

2.8% clad damage is based upon NUREG-1228 core damage analysis, and by virtue of its release into containment, the loss of the Reactor Coolant barrier (detailed calculations are contained in the Basis for Fission Product Barrier IC FC 1.d.1).

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for RCS barrier under 2.b.1. Thus, this EAL indicates a loss of both Fuel Clad barrier and RCS barrier.

POTENTIAL LOSS - NONE

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

Containment Radiation Monitor Dose Rate Curves

Percent of Fuel Inventory Airborne in the Containment vs. Approximate Source and Damage Estimate

Curve No.	% Fuel* Inventory Released	Approximate Source and Damage Estimate
1	100.	100% TID-14844, 100% fuel damage, potential core melt.
	50.	50% TID noble gases, TMI source.
2	10.	10% TID, 100% NRC gap activity, total clad failure, partial core uncovered.
	3.	3% TID, 100% WASH-1400 gap activity, major clad failure.
3	1.	1% TID, 10% NRC gap, Max. 10% clad failure.
4	0.1	0.1% TID, 10% NRC gap, 1% clad failure, local heating of 5-10 fuel assemblies.
5	0.01	0.01% TID, 0.1% NRC gap, clad failure of 3/4 of a fuel assembly (36 rods).
6	10^{-3}	0.01% NRC gap, clad failure of a few rods
	10^{-4}	100% coolant release with spiking.
7	$5X10^{-6}$	100% coolant inventory release.
	10^{-6}	Upper range of normal airborne noble gas activity in containment.

* 100% Fuel Inventory = 100% Noble Gases +25% Iodine + 1% particulates.

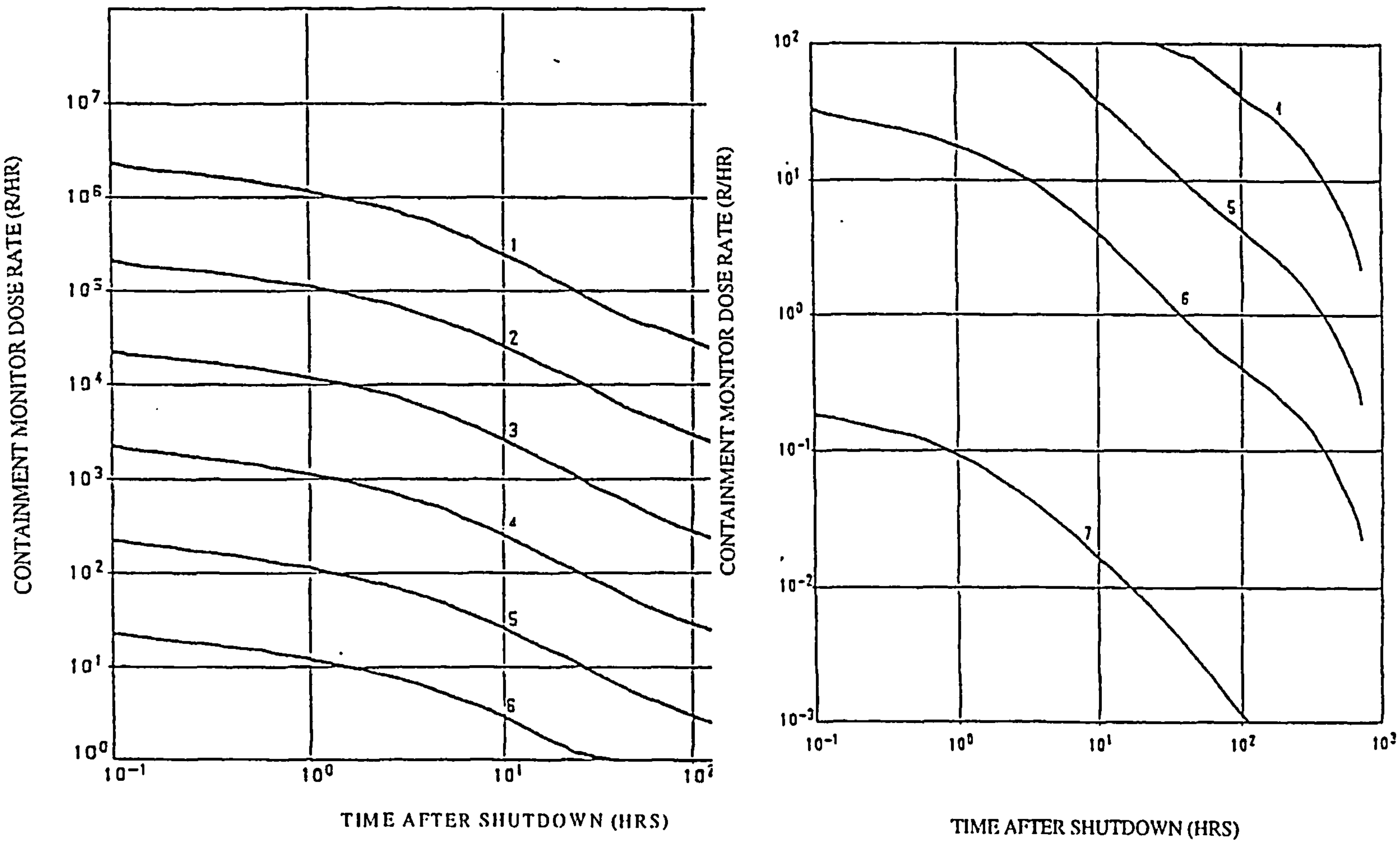
NOTES (1) These curves account for the finite containment volume and shield wall seen by the detector but do not account for any monitor physical or shielding characteristics or calibration uncertainties.

(2) The curves assume that both airborne noble gases and iodines are significant. Sprays (if used) would make the iodine and particulate contribution (presently about 50%) insignificant. However, particulate plate out on the monitor casing and direct shine doses from components may make the readings unreliable.

(3) Curve uncertainties are on the order of a factor of 5 to 10.

Table 3-2: LGS EAL Technical Basis
RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

LGS CONTAINMENT RADIATION MONITOR CURVES



1 - 100% TID
2 - 10% TID
3 - 1% TID
4 - 0.1% TID
5 - 0.01% TID
6 - 0.001% TID

4 - 0.1% TID
5 - 0.01% TID
6 - 0.001% TID
7 - 100% REACTOR COOLANT

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.c

INITIATING CONDITION

Drywell (DW) Pressure

THRESHOLD VALUE

Not Applicable

MODE APPLICABILITY

Not Applicable

BASIS: (References)

Not applicable

FUEL CLAD 1.d

INITIATING CONDITION

Breached / Bypassed

THRESHOLD VALUE

LOSS: .... 1. Coolant Activity > 300 µCi/gm I-131 dose equivalent [T04512]

OR

2. Core damage calculations indicate > 2.8% fuel clad damage

POTENTIAL LOSS: ....NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - A reactor coolant sample activity of greater than > 300 µCi/gm was determined to indicate significant clad heating and is indicative of the loss of the fuel clad barrier. This concentration is well above that expected for Iodine spikes and corresponds to 2.8% clad damage. 2.8% fuel clad damage is based upon NUREG-1228 core damage analysis.

Calculation of 300 µCi/cc equivalence to percent fuel clad damage is as follows: (For purposes of this calculation, cc and gm are considered equivalent.)

Iodine Isotope	Dose Factors (Reg Guide 1.109)	Ci/MWe Values (Time After Shutdown = 0) (NUREG-1228)
I-131	4.39E-3	85000
I-132	5.23E-5	120000
I-133	1.04E-3	170000
I-134	1.37E-5	190000
I-135	2.14E-4	150000

Time After Shutdown (T = 0) Ratios

$R_{132} = 120000/85000(\text{I-131}) = 1.41(\text{I-131})$

$R_{133} = 170000/85000(\text{I-131}) = 2.00(\text{I-131})$

$R_{134} = 190000/85000(\text{I-131}) = 2.24(\text{I-131})$

$R_{135} = 150000/85000(\text{I-131}) = 1.76(\text{I-131})$

Equation for Dose Equivalent Iodine (DEI_{131})

$$DEI_{131} = \frac{A_{131} DF_{131} + (R_{132}) A_{131} DF_{132} + (R_{133}) A_{131} DF_{133} + (R_{134}) A_{131} DF_{134} + (R_{135}) A_{131} DF_{135}}{DF_{131}}$$

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.d – Cont'd

BASIS: (References) – Cont'd

$$300 = \frac{A_{131}4.39E-3+1.41\,A_{131}5.23E-5+2.00\,A_{131}1.04E-3+2.24\,A_{131}1.37E-5+1.76\,A_{131}2.14E-4}{4.39E-3}$$

$$300 = \frac{6.95E-3\,A_{131}}{4.39E-3}$$

Solve for A_{131} assuming DEI_{131} = 300 μCi/cc

Therefore: A_{131} = 189 μCi/cc I-131

Clad damage fraction (NUREG-1228, Table 4.1) = .02

Full Power = 1150 MWe

Clad Activity I-131 = (Ci/MWe) (MWe) (Clad Damage Fraction)
= (85000Ci/MWe) (1150MWe) (.02)
= 1.96E6 Ci

Reactor Water Volume = 2.93E8 cc (ERP-C-1410)

Total Coolant Activity I-131 = (A_{131}) (Rx Water Volume) (Ci/μCi)
= (189 μCi/cc) (2.93E8cc) (1.0E-6Ci/ μCi)
= 5.54E4Ci

Percent Clad Damage = Total Coolant Activity/Clad Activity I-131
= (5.54E4) / (1.96E6)
= 2.8%

<u>POTENTIAL LOSS</u> - NONE

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.e

INITIATING CONDITION

Drywell Hydrogen Concentration

THRESHOLD VALUE

Not Applicable

MODE APPLICABILITY

Not Applicable

BASIS: (References)

Not applicable

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

FUEL CLAD 1.f

INITIATING CONDITION

Discretionary

THRESHOLD VALUE

LOSS: ....1. Any condition in the judgment of the Emergency Director that indicates a LOSS of the Fuel Clad barrier

POTENTIAL LOSS: ....2. Any condition in the judgment of the Emergency Director that indicates a POTENTIAL LOSS of the Fuel Clad barrier

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL, as a factor in Emergency Director judgment, that the barrier may be considered lost or potentially lost. (See also IC MG1, "Prolonged Loss of ALL Offsite AC Power AND Prolonged Loss of ALL Onsite AC Power" for additional information.)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.a

INITIATING CONDITION

Reactor Pressure Vessel (RPV) Water Level

THRESHOLD VALUE

LOSS:....................................1. Reactor water level < -161 inches

POTENTIAL LOSS:............2. Reactor water level CANNOT be determined

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS – [Threshold Value #1] This "Loss" EAL is the same as "Potential Loss" Fuel Clad Barrier IC FC 1.a.2. The -161 inches water level corresponds to the level, which is used in TRIPS to indicate challenge of core cooling. This EAL appropriately escalates the emergency class to a Site Area Emergency. Thus, this EAL indicates a loss of the RCS barrier and a Potential Loss of the Fuel Clad Barrier.
POTENTIAL LOSS - [Threshold Value #2] Inability to determine Reactor Pressure Vessel (RPV) level prevents assurance of adequate core cooling by methods that rely on being able to determine RPV water level (i.e., submergence or Minimum Steam Cooling RPV Water Level). TRIP procedures will provide criteria and strategies when RPV level CANNOT be determined

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.b

INITIATING CONDITION

Drywell (DW) Radiation Monitor

THRESHOLD VALUE

LOSS: ....1. DW radiation monitor reading > 15 R/hr

POTENTIAL LOSS: ....2. NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS – The 15 R/hr reading is a value, which indicates the release of reactor coolant to the drywell. The intent is not to verify criteria used in calculation (e.g., RCS breach), but rather to classify once EAL threshold value is reached or exceeded. This reading is less than that specified for a Fuel Clad Barrier LOSS (under IC Fuel Clad 1.b.1). Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading rises to that value specified under IC Fuel Clad 1.b.1, Fuel Clad damage would also be indicated.

[Calculation Basis] The value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with concentrations corresponding to 0.001% Total Isotopic Distribution (TID) into the drywell atmosphere.

Using Curve 6 [0.001%] of the "Containment Radiation Monitor Dose Rate Curves" under Fuel Clad 1.b:

Time after Shutdown = 0.1 hour

0.001% TID = 13 R/hr

This is rounded to 15 R/hr for human factors considerations

POTENTIAL LOSS - NONE

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.c

INITIATING CONDITION

Drywell (DW) Pressure

THRESHOLD VALUE

LOSS: ....1. Drywell pressure > 1.68 psig

AND

Indication of RCS leak inside Drywell

POTENTIAL LOSS: ....NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - The 1.68 psig drywell pressure is based on the drywell high pressure alarm set point, and along with indication of an RCS leak, indicates a LOCA. If drywell pressure exceeds 1.68 psig, and rise cannot be attributed to the loss of Drywell Cooling, there is indication that a leak of sufficient magnitude exists that prevents drywell pressure stabilization.

> **DEVIATION:** The EAL, as stated in NUMARC/NESP-007, contains only the drywell pressure. A qualifying: "AND Indication of a RCS leak inside drywell", was added as a human factor reminder to the Emergency Director that use of this EAL is for accident scenarios only. Thus, a Drywell Pressure rise due to the loss of Drywell Cooling will not require an emergency classification.

Cycling of safety relief valves to reduce primary system overpressure when no fuel damage is indicated, is NOT considered reactor coolant leakage.

Primary containment pressure rise due solely to loss of containment heat removal capability are also NOT considered to exceed this threshold.

POTENTIAL LOSS- NONE

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.d

INITIATING CONDITION

Breached / Bypassed

THRESHOLD VALUE

LOSS: ....1. Unisolable Main Steam Line (MSL) Break as indicated by the failure of **BOTH** MSIVs in **ANY** one line to close
AND
EITHER of the following:
- High MSL Flow and High Steam Tunnel Temperature annunciators
  OR
- Direct report of steam release

OR

2. SRV is stuck open or cycling
AND
Indication of a LOSS of the Fuel Clad Barrier per the Fission Product Barrier Matrix

POTENTIAL LOSS: ....3. RCS leakage > 50 gpm

OR

4. Unisolable primary system breach per T-103, as indicated by meeting conditions for **EITHER**:
   a. T-103, Step SCC/T-8
   OR
   b. T-103, Step SCC/RAD-10

MODE APPLICABILITY

1, 2, 3

RCS 2.d (Cont'd)

BASIS: (References)

LOSS – [Threshold Value #1] Hi Steam Flow and Hi Steam Tunnel Temperature Annunciators are both indicators of a Main Steam Line Break. Both parameters will cause an isolation of the MSIV's. Should both valves in any one line fail to isolate, this event would be also considered a LOSS of Primary Containment (per IC 3.d.1) and appropriately classified as a Site Area Emergency.

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 visual 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.

Refer to MA8 for classification of an Alert due to an isolable Main Steam Line Break break.

Design basis accident analyses of a Main Steam Line Break outside of secondary containment shows that even if MSIV closure occurs within design limits, dose consequences offsite from a "puff" release would be in excess of 10 mRem.

LOSS – [Threshold Value #2] Loss of the RCS Barrier based on an open safety relief valve (SRV) is dependent on other events. If an SRV is stuck open or cycling and no other emergency condition exists, an emergency declaration is not appropriate. However, if the fuel is damaged and the relief valve is allowing the fission products to escape into the Drywell (Containment), a LOSS of the RCS Barrier has occurred.

POTENTIAL LOSS – [Threshold Value #3] The potential loss of RCS based on leakage >50 gpm 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, a break propagation leading to a significantly larger loss of inventory is possible. RCS leakage is measured by the normal primary system leakage monitoring system and is leakage into the drywell. Under certain conditions, this system may be isolated due to elevated drywell pressure caused by the leak. In that case, a LOSS of RCS will be indicated and this "potential loss" of RCS would not impact the classification.

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 Value #4] Potential loss of RCS based on primary system leakage outside Containment is determined from site-specific area temperatures or radiation levels per T-103, "Secondary Containment Control", which indicate a direct path from the RCS to areas outside primary containment that cannot be isolated or the source of discharge cannot be determined.

T-103 Action Levels based on area water level (Table SC/L-2) are evaluated separately for an Alert classification under IC HA3 (Natural or Destructive Phenomena Affecting a Vital Area).

RCS 2.d (Cont'd)

BASIS:(References) – Cont'd

Terms:

Unisolable – Refers to a leak that cannot be isolated from the Control Room. When evaluating this EAL for unisolable primary system leakage, it is appropriate to attempt isolation from the Control Room prior to classification

Primary System – The pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the flow of steam or water being discharged through an unisolable break in the system.

Primary System Leakage - 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.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.e

INITIATING CONDITION

Drywell Hydrogen Concentration

THRESHOLD VALUE

Not Applicable

MODE APPLICABILITY

Not Applicable

BASIS: (References)

Not applicable

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.f

INITIATING CONDITION

Discretionary

THRESHOLD VALUE

LOSS: .... 1. Any condition in the judgment of the Emergency Director that indicates a LOSS of the Reactor Coolant System barrier

POTENTIAL LOSS: .... 2. Any condition in the judgment of the Emergency Director that indicates a POTENTIAL LOSS of the Reactor Coolant System barrier

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. (See also IC MG1, "Prolonged Loss of ALL Offsite AC Power AND Prolonged Loss of ALL Onsite AC Power" for additional information.)

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.a

INITIATING CONDITION

Reactor Pressure Vessel (RPV) Water Level

THRESHOLD VALUE

LOSS: ....NONE

POTENTIAL LOSS: ....1. ANY of the following direct entry into SAMP-1 and SAMP-2:

- T-111
- T-116
- T-117

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS- NONE

POTENTIAL LOSS – The entry into SAMP-1 and SAMP-2 (Primary Containment Flooding is required) indicates that the reactor core cannot be adequately cooled and the Primary Containment is required to be flooded to submerge the core and preserve Primary Containment integrity. Concurrent entry and execution of SAMP-2 with SAMP-1 properly coordinates Primary Containment control functions with RPV and Primary Containment injection.
Entry into Severe Accident Management Procedures (SAMP) is directed by the TRIP procedures when adequate core cooling requirements cannot be satisfied and core damage has or may occur.

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.b

INITIATING CONDITION

Drywell (DW) Radiation Monitor

THRESHOLD VALUE

LOSS: ....NONE

POTENTIAL LOSS: ....1. Drywell radiation monitor reading > 3.0E+5 R/hr

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - NONE

POTENTIAL LOSS - A containment high range radiation monitor RR-26-191/291A, B, C, D reading ≥ 3.0E+5 R/hr indicates significant fuel damage, well in excess of that required for the loss of the RCS and Fuel Clad. The intent is not to verify criteria used in calculation, but rather to classify once EAL threshold value is reached or exceeded.

[Calculation Basis] As stated in Section 3.8 of NUMARC/NESP-007, a major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

The reading was calculated assuming an instantaneous release of the Reactor Coolant volume into the Primary Containment (direct reading not shine) where the value corresponds to a release of approximately 20% of the gap region. This calculation is as follows:

Using Curve 3 [1%] of the "Containment Radiation Monitor Dose Rate Curves" under Fuel Clad 1.b:

Time after Shutdown = 1 hour (more conservative due to lower value for EAL)

1% fuel clad damage: dose rate = 15,000 R/hr

Extrapolating to 20%: (15,000 R/hr/1%)(20) = 300,000 R/hr

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.c

INITIATING CONDITION

Drywell (DW) Pressure

THRESHOLD VALUE

LOSS:.................................1. Rapid, unexplained drop in DW pressure following an initial rise

OR

2. DW pressure response not consistent with LOCA conditions indicating a Containment breach

POTENTIAL LOSS:............3. DW pressure > 44 psig

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - [Threshold Value #1] A rapid unexplained loss of Drywell pressure not due to use of containment sprays following an initial pressure rise indicates a loss of Primary Containment integrity.

LOSS - [Threshold Value #2] Drywell pressure should rise as a result of mass and energy release into the containment from a Loss of Coolant Accident (LOCA). Thus, Drywell pressure NOT rising under these conditions indicates a breach of containment integrity.

POTENTIAL LOSS - [Threshold Value #3] A Drywell pressure 44 psig is equal to the peak pressure expected from a Design Basis Accident (DBA) LOCA and 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.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.d

INITIATING CONDITION

Breached / Bypassed

THRESHOLD VALUE

LOSS: ....

1. Failure of **BOTH** Main Steam Isolation Valves (MSIVs) in **ANY** one line to close on an isolation signal
   AND
   Downstream pathway exists to the turbine, condenser or directly into the Turbine Enclosure
   OR
2. Intentional venting per T-200 or T-228 is required
   OR
3. Unisolable primary system breach per T-103, as indicated by meeting conditions for **EITHER**:
   a. T-103, Step SCC/T-8
      OR
   b. T-103, Step SCC/RAD-10

POTENTIAL LOSS: NONE

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - [Threshold #1] A failure of both Main Steam Isolation Valves (MSIVs) in any one line indicates a breach of the primary containment integrity as described in the primary containment Limiting Conditions for Operation. A down stream flow path to the turbine or condenser, or MSL releasing into the Turbine Enclosure, represents a pathway to the environment. 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 operations, it will be addressed by a Technical Specification Action Statement. However, during accident conditions, this will represent a breach of Primary Containment.

The breach is NOT isolable from the Control Room OR an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation from the Control Room should be made prior to the accident classification. If Operator actions from the Control Room are successful, then this IC is not applicable and REFER to IC MA8. Credit is NOT given for Operator actions taken in-plant (outside the Control Room) to isolate the leak.

CONTAINMENT 3.d

BASIS: (Cont'd)

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.

LOSS - [Threshold #2] Intentional venting of the primary containment as required by the TRIP or SAMP procedure guidance to the secondary containment and/or the environment is considered to be a breach of the primary containment for the purposes of accident classification.

LOSS - [Threshold #3] Loss of Primary Containment based on primary system leakage outside Containment is determined from site-specific area temperatures or radiation levels per T-103, "Secondary Containment Control", which indicates a direct path from the RCS to areas outside primary containment that cannot be isolated or the source of discharge cannot be determined.

T-103 Maximum Safe Operating Levels based on area water level are evaluated separately for an Alert classification under IC HA3 (Natural or Destructive Phenomena Affecting a Vital Area).

POTENTIAL LOSS – NONE

Terms:

Unisolable - A leak that cannot be isolated from the Control Room. When evaluating this EAL for unisolable primary system leakage, it is appropriate to attempt isolation from the Control Room prior to classification.

Primary System – The pipes, valves and other equipment which connect directly to the RPV such that a reduction in RPV pressure will cause a drop in the flowrate of steam or water being discharged through a break in the system.

Containment Bypassed – The unintentional opening of, or leakage through, penetration isolations (e.g., equipment / personnel access hatches / airlocks, dampers / valves, etc.), such that a path to the environment exists.

Primary System Leakage - 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.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.e

INITIATING CONDITION

Drywell Hydrogen Concentration

THRESHOLD VALUE

LOSS: ....NONE

POTENTIAL LOSS: ....1. Drywell Hydrogen (H_2) > 6%
AND
Drywell Oxygen (O_2) > 5%

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

LOSS - NONE

POTENTIAL LOSS - The specified value of 6% hydrogen and 5% oxygen concentration is the minimum, which can support a deflagration. Combustion of hydrogen in the deflagration concentration range creates a traveling flame causing a rapid rise in primary containment pressure. A deflagration may result in a peak primary containment pressure high enough to rupture the primary containment or damage the drywell-to-suppression pool boundary.

CONTAINMENT 3.f

INITIATING CONDITION

Discretionary

THRESHOLD VALUE

LOSS: ....1. Any condition in the judgment of the Emergency Director that indicates a LOSS of the Primary Containment barrier

POTENTIAL LOSS: ....2. Any condition in the judgment of the Emergency Director that indicates a POTENTIAL LOSS of the Primary Containment barrier

MODE APPLICABILITY

1, 2, 3

BASIS: (References)

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Primary Containment Barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in the Emergency Director's judgment that the barrier may be considered lost or potentially lost. See also IC MG1, "Prolonged Loss of ALL Offsite AC Power AND Prolonged Loss of ALL Onsite AC Power" for additional information.

MG1

INITIATING CONDITION

Prolonged Loss of **ALL** Offsite AC Power **AND** Prolonged Loss of **ALL** Onsite AC Power

EAL THRESHOLD VALUES

1. Loss of offsite power to **ALL** 4 KV Safeguard Busses

   AND

   ALL four of the 4 KV Safeguard Busses are de-energized for > 15 minutes

   AND

   ANY of the following:

   - Restoration of at least one 4 KV emergency bus in ≤ 2 hours is **NOT** likely

     OR

   - Reactor water level **CANNOT** be maintained > -161 inches

     OR

   - Suppression Pool temperature **CANNOT** be maintained on the **"SAFE"** side of the Heat Capacity Temperature Limit (HCTL) Curve (T-102, SP/T-1)

MODE APPLICABILITY

1, 2, 3

BASIS (References)

When evaluating this EAL for Suppression Pool level outside of the Heat Capacity Temperature Limit Curve, High or Low, it is appropriate to consider the operation to be on the "UNSAFE" side.

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment. The two hours to restore AC power is based on the site blackout coping analysis as described below. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

10 CFR 50.2 defines Station Blackout (SBO) as complete loss of AC power to essential and non-essential buses. SBO does not include loss of AC Power to busses fed by station batteries through inverters, nor does it assume a concurrent single failure or design basis accident.

MG1 – Cont'd

BASIS (References) – Cont'd

Successful SBO coping maintains the following key parameters within given acceptable limits:

1. Reactor water level > -161 inches (TAF)
2. Suppression Pool level low enough to prevent HPCI and/or RCIC steam exhaust line flooding
3. Reactor pressure >150 psig to maintain HPCI and RCIC operable
4. Containment pressure < 62.5 psig, design limit
5. Suppression Pool temperature < 170 degrees F, HPCI/RCIC lube oil temperature concern when suction aligned to Suppression Pool
6. Drywell temperature
   - 200 degrees F indefinitely
   - 250 degrees F 99 days
   - 320 degrees F 18 hours
   - 340 degrees F 3 hours

Successful extended SBO coping depends on ability to keep HPCI/RCIC available for injection, and ability to maintain RPV depressurized for low pressure injection should HPCI and RCIC become unavailable. 125V DC provides control power for HPCI, RCIC and SRVs. The parameters listed above can be maintained as long as the batteries are intact. Two hours is the earliest the batteries would fail, and thus is the basis for the time limit in this EAL.

The significance of a station blackout relative to the loss of fission product release barriers is that all three barriers will eventually be lost due to the inability to remove heat from the fuel and the containment. Although the RCS will be intact the longest, eventually SRVs will operate in the relief mode due to RPV over-pressurization and if the containment has already failed then there is a direct bypass of the RCS boundary.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS1

INITIATING CONDITION

Loss of **ALL** Offsite AC Power **AND** Loss of **ALL** Onsite AC Power to Essential Busses

EAL THRESHOLD VALUES

1. Loss of offsite power to **ALL** 4 KV Safeguard Busses

   AND

   ALL four of the 4 KV Safeguard Busses are de-energized for > 15 minutes

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Control Room annunciators would indicate that all offsite and onsite AC power feeds have been lost. Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal, RHR Service Water, and Emergency Service Water. Although instrumentation (supplied through instrument inverters) and DC power systems would be available, their operability would be limited to the amount of stored energy contained in their respective batteries. Instrumentation, communication equipment, and in-plant lighting and ventilation will be significantly hampered by the loss of all AC power.

Fifteen (15) minutes has been selected to allow adequate time to cross tie or address diesel generator failures and to exclude transient or momentary power losses. It is not necessary to wait for 15 minutes if it can be determined in less than 15 minutes that the power loss is not transient or momentary.

Escalation of this event would be based on the time that an Emergency Diesel Generator is unavailable.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA1

INITIATING CONDITION

AC Power to Essential Busses Reduced to a Single Source for > 15 minutes

EAL THRESHOLD VALUE

1. Loss of offsite power to ALL 4 KV Safeguard Busses

   AND

   Three of four of the 4 KV Safeguard Busses are de-energized for > 15 minutes

MODE APPLICABILITY

1, 2, 3

BASIS (References)

The reduction of available reliable power sources to a condition where ANY additional single failure will result in a station blackout is a substantial degradation in the level of safety of the plant. That is, the Unit is down to its last source of AC power. Loss of the single power supply would escalate to a SITE AREA EMERGENCY via IC MS1.

This EAL is intended to provide an escalation from "Loss of offsite Power for greater than 15 minutes." This condition is a degradation of the offsite and onsite power systems such that any additional failure would result in a station blackout. Fifteen (15) minutes has been selected to allow adequate time to cross tie or address diesel generator failures and to exclude transient or momentary power losses. However, an Alert should be declared in less than 15 minutes if it can be determined in less than 15 minutes that the power loss is not transient or momentary.

Depending on the 4 KV AC bus that remains energized there is a disparity in the systems that may be available. The ability to remove heat from the containment via Suppression Pool cooling may be lost due to the need to operate the remaining available RHR pump in other than Suppression Pool cooling (e.g., LPCI). As such there is a decrease in the systems available to remove heat transferred to the containment and there is an ongoing release of energy from the reactor to the containment (via SRVs, HPCI and/or RCIC operation). The ability to cool the nuclear fuel, remove decay heat, and control containment parameters is severely limited. Should equipment be unavailable prior to the loss of power, functions necessary to maintain the plant in a cold shutdown condition may be threatened.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU1

INITIATING CONDITION

Loss of ALL Offsite AC Power for > 15 minutes to Essential Busses

EAL THRESHOLD VALUE

1. Loss of offsite power to ALL 4 KV Safeguard Busses for >15 minutes

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned – Not the result of an intended evolution and requiring corrective or mitigative actions

This EAL addresses the loss of offsite AC power supplying the station. Offsite power is fed through 101 and 201 Safeguard Transformers. Loss of offsite power will cause a reactor scram and containment isolation. All four (4) emergency Diesel Generators will be available to carry the essential loads for each unit (the four Diesel Generators are shared between each unit). Balance of Plant systems that would assist in plant operations (i.e., condensate pumps, etc.) may be unavailable due the loss of power.

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (Station Blackout). Fifteen minutes was selected as a threshold to allow adequate time to cross tie or address diesel generator failures and to exclude transient or momentary power losses.

The Emergency Director must also consider the impact to the unaffected unit due to the loss of power to balance of plant equipment on common or shared systems.

Escalation of this event to an Alert would be based on having a loss of all offsite AC power coincident with onsite AC power being reduced to a single power source in Modes 1, 2, and 3 or having a loss of all offsite and onsite AC power in Modes 4 or 5.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA2

INITIATING CONDITION

Loss of ALL Offsite AC Power AND Loss of ALL Onsite AC Power to Essential Busses

EAL THRESHOLD VALUES

1. Loss of offsite power to ALL 4 KV Safeguard Busses

   AND

   ALL four of the 4 KV Safeguard Busses are de-energized for > 15 minutes

MODE APPLICABILITY

4, 5, Defueled

BASIS (References)

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode, the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, raising the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL.

Fifteen (15) minutes has been selected to allow adequate time to cross tie or address diesel generator failures and to exclude transient or momentary power losses. However, an Alert should be declared in less than 15 minutes if it can be determined in less than 15 minutes that the power loss is not transient or momentary.

Declaration of an Alert is not applicable when ECCS busses are de-energized for a planned evolution.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS3

INITIATING CONDITION

Loss of ALL Required T.S. Safety-Related 125 VDC Power Sources

EAL THRESHOLD VALUE

1. Loss of ALL required T.S. safety-related 125 VDC power sources for > 15 minutes as indicated by < 105 VDC on Panels 1(2)FA, B, C, D

MODE APPLICABILITY

1, 2, 3

BASIS (References)

A loss of all DC power compromises the ability to monitor and control plant functions. The 125 Volt DC system provides control power to engineered safety features valve actuation, diesel generator auxiliaries, plant alarm and indication circuits as well as the control power for the associated load group. If 125 Volt DC power is lost for an extended period of time (greater than 15 minutes) critical plant functions such as RPS Logic, Alternate Rod Insertion, Emergency Service Water Indication, 4KV Breaker Controls, HPCI, RCIC and RHR pump controls required to maintain safe plant conditions may not operate and core uncovery with subsequent reactor coolant system and primary containment failure might occur. The 125 volt DC Main Distribution Panel Busses are as follows:

1(2)FA, Division I Safeguard 125/250 DC Bus 1(2)FA
1(2)FB, Division II Safeguard 125/250 DC Bus 1(2)FB
1(2)FC, Division III Safeguard 125 DC Bus 1(2)FC
1(2)FD, Division IV Safeguard 125 DC Bus 1(2)FD

Loss of all DC Power causes the loss of the following equipment:

- Alternate Rod Insertion
- ADS
- HPCI
- RCIC
- Normal EDG Control
- Normal Recirculation Pump Trip
- Containment Instrument Gas Compressors
- Other 4KV Circuit Breakers (e.g., RHR, CS, CRD)

Loss of ADS creates a loss of low pressure ECCS due to the inability to depressurize the reactor. In addition, loss of these buses will eventually lead to MSIV closure and reactor trip due to the loss of the Containment Instrument Gas Compressor as a result of suction valve closure. Subsequent to MSIV closure, much of the equipment noted above will be required for plant stabilization and shutdown.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS3 – Cont'd

BASIS (References) – Cont'd

A sustained loss of DC power will threaten the ability to remove heat from the reactor core, resulting in eventual fuel clad damage. The loss of DC power will also result in the loss of the ability to remove heat from the containment. SRVs will remain operable in the relief mode and the heat addition to the containment could result in a loss of the primary containment as a fission product release barrier.

105 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is near the minimum voltage selected when battery sizing is performed.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU3

INITIATING CONDITION

Loss of BOTH Required T.S. Safety-Related 125 VDC Power Sources

EAL THRESHOLD VALUE

1. Loss of BOTH required T.S. safety-related 125 VDC power sources for > 15 minutes as indicated by < 105 VDC on DC Panels 1(2)FA, B, C, D

MODE APPLICABILITY

4, 5

BASIS (References)

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. The safety related 125 volt DC Distribution Panels are as follows:

1(2)FA, Division I Safeguard 125/250 DC Bus 1(2)FA
1(2)FB, Division II Safeguard 125/250 DC Bus 1(2)FB
1(2)FC, Division III Safeguard 125 DC Bus 1(2)FC
1(2)FD, Division IV Safeguard 125 DC Bus 1(2)FD

105 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is near the minimum voltage selected when battery sizing is performed.

Unplanned is included in this IC and EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, plants will perform maintenance on a Train related basis during shutdown periods. It is intended that the loss of the operating (operable) train be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will occur.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MG4

INITIATING CONDITION

Auto and Manual SCRAM **NOT** Successful, **AND** Loss of Core Cooling or Heat Sink

EAL THRESHOLD VALUES

1. Failure of automatic RPS, ARI and Manual SCRAM/ARI to shutdown the reactor as defined by **EITHER**:
   - Reactor power > 4%

     OR
   - Suppression Pool temperature is greater than 110°F.

   AND

   EITHER of the following criteria are met:
   - Suppression Pool Temperature **CANNOT** be maintained on the **"SAFE"** side of the Heat Capacity Temperature Limit (HCTL) curve (T-102, SP/T-1)

     OR
   - Reactor water level < -186 inches

MODE APPLICABILITY:

1, 2

BASIS: (References)

Manual SCRAM - Any set of actions by the reactor operators at the Reactor Control Panels which causes control rods to be inserted sufficiently to reduce reactor power to a condition where it will remain shutdown under all conditions without the use of boron injection (i.e., mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation.

Automatic actuation of the ARI system is a backup to the MANUAL SCRAM and, as a result, does not constitute a successful MANUAL SCRAM.

This EAL is not applicable if a manual scram is initiated and no RPS set points are exceeded. Taking the mode switch to shutdown is considered a manual scram action. Note that although placing the Mode Switch in "shutdown" is a manual scram action, when the Mode Switch passes through the "startup / hot standby" position the Nuclear Instrumentation Scram Setpoint is lowered. If reactor power is greater than the setpoint, an automatic scram will be initiated. If the RPS then fails to initiate a scram, then this should be evaluated as an automatic RPS set point being exceeded and a failure of the automatic scram.

MG4 – Cont'd

BASIS (References) – Cont'd

A valid automatic and/or manual scram signal is present as indicted by control room indications and/or alarms. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, reactor pressure, Suppression Pool temperature trend) can be used to determined if reactor power is greater than 4% power. The Reactor Protection System (RPS) is designed to function to shut down the reactor (either manually or automatically). The RPS system is "fail safe," that is, it de-energizes to function. An Anticipated Transient Without Scram (ATWS) event can be caused by either a failure of RPS (electrical ATWS) or the Control Rod Drive system to insert the control rods (hydraulic ATWS).

The TRIP procedures establish 4% power as a power level sufficient to challenge Primary Containment heat removal capabilities should this energy be directed to the Primary Containment. If Suppression Pool temperature is greater than Boron Injection Temperature (110°F) during ATWS, a precursor exists for a threat to Primary Containment.

In addition, control room instrumentation indicates that operation is on the "UNSAFE" side of the HCTL Curve (T-102, SP/T-1) or RPV level is < -186 inches. When Suppression Pool level is outside of the Heat Capacity Temperature Limit (HCTL) Curve, High or Low, it is appropriate to consider operation to be on the "UNSAFE" side. Failure of all automatic and manual trip functions coincident with a high Suppression Pool temperature will place the plant in a condition where reactivity control capability is jeopardized and heat removal capability is severely limited. RPV level <-186 inches indicates an extreme challenge to the ability to cool the core.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS4

INITIATING CONDITION

Auto and Manual SCRAM NOT Successful

EAL THRESHOLD VALUE:

1. Failure of automatic RPS, ARI and Manual SCRAM/ARI to shutdown the reactor as defined by **EITHER:**
   - Reactor power > 4%

     OR
   - Suppression Pool temperature is greater than 110°F.

MODE APPLICABILITY:

1, 2

BASIS: (References)

Manual Scram -- Any set of actions by the reactor operator(s) at the reactor control console which causes control rods to insert sufficiently to reduce reactor power to a condition where it will remain shutdown under all conditions without the use of boron injection (i.e., mode switch to shutdown, manual scram push buttons, or manual ARI initiation).

Automatic actuation of the ARI system is a backup to the MANUAL SCRAM and, as a result, does not constitute a successful MANUAL SCRAM.

This EAL is not applicable if a manual scram is initiated and no RPS set points are exceeded. Taking the mode switch to shutdown is considered a manual scram action. Note that although placing the Mode Switch in "shutdown" is a manual scram action, when the Mode Switch passes through the "startup / hot standby" position the Nuclear Instrumentation Scram Setpoint is lowered. This may cause an RPS Set point to be exceeded due to the change in Nuclear Instrumentation Scram set point when the mode switch is taken out of the Run positionIf reactor power is greater than the setpoint, an automatic scram will be initiated. If the RPS then fails to initiate a scram, then this should be evaluated as an automatic RPS set point being exceeded and a failure of the automatic scram.

A valid automatic and/or manual scram signal is present as indicted by control room indications and/or alarms. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, reactor pressure, Suppression Pool temperature trend) can be used to determined if reactor power is greater than 4% power. The Reactor Protection System (RPS) is designed to function to shut down the reactor (either manually or automatically). The RPS system is "fail safe," that is, it de-energizes to function. An Anticipated Transient Without Scram (ATWS) event can be caused by either a failure of RPS (electrical ATWS) or the Control Rod Drive system to insert the control rods (hydraulic ATWS).

MS4 – Cont'd

BASIS: (References) – Cont'd

The TRIP procedures establish 4% power as a power level sufficient to challenge Primary Containment heat removal capabilities should this energy be directed to the Primary Containment. If Suppression Pool temperature is greater than Boron Injection Temperature (110°F) during ATWS, a precursor exists for a threat to Primary Containment and thus a Site Area Emergency is warranted

This event escalation is based on rising Suppression Pool Temperature or lowering RPV water level that would result in the loss of containment integrity and the inability to remove the heat generated from the fuel per MG4.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA4

INITIATING CONDITION

Auto SCRAM NOT Successful

EAL THRESHOLD VALUE

1. RPS set point has been exceeded for an automatic SCRAM

   AND

   Failure of automatic RPS to achieve a state in which the reactor is shutdown under all conditions without boron injection

MODE APPLICABILITY

1, 2

BASIS (References)

This EAL is not applicable if a manual scram is initiated and no RPS set points are exceeded. Taking the mode switch to shutdown is considered a manual scram action. Note that although placing the Mode Switch in "shutdown" is a manual scram action, when the Mode Switch passes through the "startup / hot standby" position the Nuclear Instrumentation Scram set point is lowered. If reactor power is greater than the set point, an automatic scram will be initiated. If the RPS then fails to initiate a scram, then this should be evaluated as an automatic RPS set point being exceeded and a failure of the automatic scram.

Entry into this EAL is based on a reactor parameter actually exceeding a RPS set point and the reactor is not brought to a state in which the reactor is shutdown under all conditions without boron injection and maintained at that state with automatic RPS functions. The parameter must exceed the RPS set point by a significant margin eliminating minor set point drifts, which are accounted for in the Technical Specification Margin of Safety. Subsequent manual scram actions were successful in bringing the reactor to a state in which the reactor is shutdown under all conditions without boron injection. Confirmation indications include control room annunciators, APRM/IRM/SRM power level, SRM period, and Control rod position indication.

When partial control rod insertion occurs following a scram signal (either manual or automatic) judgment should be applied as to the Reactor will remain "Shutdown Under All Conditions Without Boron" and if classification should occur. Multiple control rods failing to insert beyond the Maximum Subcritical Banked Withdraw Position (MSBWP) may require actions to fully insert the control rods. However, the reactor has been made subcritical, and for all intent the reactor will remain subcritical. TRIP guidance will govern the insertion of these control rods.

MA4 – Cont'd

BASIS (References) – Cont'd

This condition is more than a potential degradation of a safety system in that a front line automatic protection system does 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 the Fuel Clad or RCS Barriers.

A scram is considered unsuccessful if it does not result in achieving a state in which the reactor will remain shutdown under all conditions without boron injection.

This EAL would be escalated to a Site Area Emergency with a failure of both manual and automatic scram signals and Reactor power > 4% or Suppression Pool temperature greater than Boron Injection Temperature.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS5

INITIATING CONDITION

Complete Loss of Functions Needed to Achieve **AND** Maintain Hot Shutdown

EAL THRESHOLD VALUES

1. Loss of function required for Hot Shutdown as evidenced by T-102 SP/T legs directing a T-112 Emergency Blowdown

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other EALs. T-102 SP/T legs requiring an Emergency Blowdown, which is directed when the Heat Capacity Temperature Limit (HCTL) curve is exceeded, indicate the loss of heat removal function.

The EAL is concerned with Suppression Pool temperature. It is not appropriate to make a Site Area Emergency classification for the condition where the T-102 Suppression Pool Level leg alone directs a T-112 Emergency Blowdown since the Emergency Blowdown is performed PRIOR to those Suppression Pool levels which may cause a loss of containment capability due to uncovering downcomers or excessive SRV tailpipe stresses.

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. Escalation to General Emergency would be via Effluent Release/In-Plant Radiation, Emergency Director Judgment, or Fission Product Barrier Degradation ICs.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA5

INITIATING CONDITION

Inability to Maintain Plant in Cold Shutdown

EAL THRESHOLD VALUE

1. Unplanned loss of **ALL** T.S. required decay heat removal systems

   AND

   EITHER of the following:

   - RCS temperature exceeding 200 °F for ≥ 15 minutes with a heat removal function restored

     OR

   - Uncontrolled RCS temperature rise approaching 200 °F with NO heat removal function restored

MODE APPLICABILITY

4, 5

BASIS (References)

Uncontrolled - A temperature rise that is not the result of a planned evolution

This EAL addresses complete loss of functions required for core cooling during refueling and cold shutdown modes. A loss of Technical Specifications components is paired with exceeding temperature limits to acknowledge additional plant capabilities to maintain plant cooling. Escalation to Site Area Emergency or General Emergency would be via Effluent Release/In-Plant Radiation or Emergency Director Judgment ICs.

The statement " Temporary Loss of ALL Tech Spec required Decay Heat Removal Systems" is intended to represent a complete loss of functions available, or an inadequate ability, to provide core cooling during the Cold Shutdown and Refueling Modes, including alternate decay heat removal methods. This EAL allows for actions taken in GP-6.2, "Shutdown Operations - Shutdown Condition Tech. Spec. Actions," to reestablish RHR in the Shutdown Cooling Mode or provide for alternate methods of decay heat removal, with the intent of maintaining RCS temperature below 200° F.

For loss of an in-service Decay Heat Removal system with other decay heat removal methods available, actions taken to provide for restoration of a decay heat removal function may require time to implement. If the event results in RCS temperature "momentarily" (for less than 15 minutes) rising above 200°F with heat removal capability restored, Emergency Director/Shift Management judgment will be required to determine whether heat removal systems are adequate to prevent boiling in the core and restoration of RCS temperature control.

MA5 – Cont'd

BASIS (References) – Cont'd

Momentary (not to exceed 15 minutes) unplanned excursions above 200° F, when alternate decay heat removal capabilities exist, should not be classified under this EAL.

The EAL guidance related to uncontrolled temperature rise is necessary to preserve the anticipatory philosophy of NUREG-0654 for events starting from temperatures much lower than the cold shutdown temperature limit.

This EAL is concerned with the ability to keep the reactor core temperature less than 200 °F. The criteria of uncontrolled Reactor Coolant temperature rise > 200 °F is met as soon as it becomes known that sufficient cooling cannot be restored in time to maintain the temperature < 200 °F, regardless of the current temperature. The inability to establish alternate methods of decay heat removal indicates that either alternate methods are unavailable to cool the core in the RPV or when the steam is transferred to the Suppression Pool, Suppression Pool cooling is unavailable. Loss of Suppression Pool cooling will result in a continuing, uncontrolled rise in reactor coolant temperature.

Special Test Exception 3.10.8 allows for temperature to rise above 200 °F during hydrostatic testing. The limit of 200 °F in this EAL does not apply under those conditions as that is not an "Uncontrolled Temperature rise."

Escalation to the Site Area Emergency is by EAL MS7, "Loss of Water Level in the Reactor Vessel that has or will uncover Fuel in the Reactor Vessel," or by Effluent Release/In-Plant Radiation RS1.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS6

INITIATING CONDITION

Inability to Monitor a Significant Transient in Progress

EAL THRESHOLD VALUE

1. A significant plant transient is in progress (Table M-1)

   AND

   ALL of the following are lost:

   - Safety system annunciators (Table M-2)
   - Safety function indicators (Table M-3)
   - Plant Monitoring System

Table M-1 Significant Plant Transients	Table M-2 Safety System Annunciators	Table M-3 Safety Function Indicators
• SCRAM • Recirc Runbacks (> 25% thermal power change) • Sustained Power Oscillations (25% peak to peak) • Stuck open relief valves • ECCS Injection	• ECCS • Containment Isolation • Reactor Trip • Process Radiation Monitoring	• Reactor Power • Decay Heat Removal • Containment Safety Functions

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

This EAL recognizes the difficulty associated in monitoring conditions without normal annunciators. In the opinion of the Shift Supervisor, this loss of annunciators requires increases surveillance to safely operate the plant. This EAL represents an increase in severity above MA6 in that the Plant Monitoring System can not provide compensatory indication, and that a significant transient is in progress.

Planned maintenance or testing activities are included in this EAL due to the significance of this event. Control Room panels with annunciators and the restoration is included in ON-122, Loss of Main Control Room Annunciators.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA6

INITIATING CONDITION

Loss of Annunciators OR Indicators Requiring Increased Surveillance

EAL THRESHOLD VALUES

1. Unplanned loss for > 15 minutes of **MOST** or **ALL** of **EITHER**:
   - Safety system annunciators (Table M-2)
     OR
   - Safety function indicators (Table M-3) for > 15 minutes

   AND
   Increased surveillance is required to safely operate the unit(s)

   AND
   EITHER of the following:
   - A significant plant transient is in progress (Table M-1)
     OR
   - Plant Monitoring System is unavailable

Table M-1 Significant Plant Transients	Table M-2 Safety System Annunciators	Table M-3 Safety Function Indicators
• SCRAM • Recirc Runbacks (> 25% thermal power change) • Sustained Power Oscillations (25% peak to peak) • Stuck open relief valves • ECCS Injection	• ECCS • Containment Isolation • Reactor Trip • Process Radiation Monitoring	• Reactor Power • Decay Heat Removal • Containment Safety Functions

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

MOST - 75% of safety system annunciators or indicators are lost or a significant risk that a degraded plant condition could go undetected exists. The use and definition of MOST is not intended to require a detailed count of lost annunciators or indicators but should be used as a guide to assess the ability to monitor the operation of the plant.

UNPLANNED - Loss of annunciators or indicators is not the result of scheduled maintenance or testing.

MA6 – Cont'd

BASIS: (References) – Cont'd

This EAL recognizes the difficulty associated in monitoring conditions without normal annunciators. It is not intended that a detailed count of instrumentation be performed, but by the use of judgment by the Shift Supervisor as the threshold for determining the severity of the plant conditions. This judgment is supported by the specific opinion of the Shift Supervisor that additional operating personnel will be required to provide increased monitoring of system operation to safely operate the plant.

This EAL represents an increase in severity above MU6 in that the Plant Monitoring System (PMS) cannot provide compensatory indication, or that a significant transient is in progress.

Fifteen minutes is used as a threshold to exclude transient or momentary power loses. Control Room panels with annunciators and direction for restoration is included in ON-122, Loss of Main Control Room Annunciators.

This EAL is not applicable in cold shutdown or refueling modes due to the limited number of safety systems required for operation.

This event will be escalated to a Site Area Emergency if a transient is in progress, the Plant Monitoring System is unavailable, and a loss of annunciators occurs.

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU6

INITIATING CONDITION

Unplanned Loss of Annunciators **OR** Indicators for > 15 minutes

EAL THRESHOLD VALUE

1. Unplanned loss for > 15 minutes of **MOST** or **ALL** of **EITHER**:
   - Safety system annunciators (Table M-2)
   OR
   - Safety function indicators (Table M-3)

   AND

   Increased surveillance is required to safely operate the unit(s)

Table M-2 Safety System Annunciators	Table M-3 Safety Function Indicators
• ECCS • Containment Isolation • Reactor Trip • Process Radiation Monitoring	• Reactor Power • Decay Heat Removal • Containment Safety Functions

MODE APPLICABILITY:

1, 2, 3

BASIS: (References)

MOST - 75% of safety system annunciators or indicators are lost **OR** a significant risk that a degraded plant condition could go undetected exists. The use and definition of MOST is not intended to require a detailed count of lost annunciators or indicators but should be used as a guide to assess the ability to monitor the operation of the plant.

UNPLANNED - Loss of annunciators or indicators is NOT the result of scheduled maintenance or testing.

This EAL recognizes the difficulty associated in monitoring conditions without normal annunciators. It is not intended that a detailed count of instrumentation be performed, but by the use of judgment by the Shift Supervisor as the threshold for determining the severity of the plant conditions. This judgment is supported by the specific opinion of the Shift Supervisor that additional operating personnel will be required to provide increased monitoring of system operation to safely operate the plant. The Plant Monitoring System (PMS) is available to provide compensatory indication. Fifteen minutes is used as a threshold to exclude transient or momentary power losses. Unplanned loss of annunciators excludes scheduled maintenance and testing activities. Control Room panels with annunciators and direction for response are included in ON-122, Loss of Main Control Room Annunciators.

This EAL is not applicable in cold shutdown or refueling modes due to the limited number of safety systems required for operation.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MS7

INITIATING CONDITION

Loss of Water Level in the Reactor Vessel That Has **OR** Will Uncover Fuel in the Reactor Vessel

EAL THRESHOLD VALUES

1. Reactor water level < -161 inches

MODE APPLICABILITY

4, 5

BASIS (References)

The indicator for "core is or will be uncovered" is Reactor Pressure Vessel Water level below the Top of Active Fuel (TAF), -161 inches as indicated on RPV Fuel Zone Level Instruments. Core submergence ensures adequate core cooling. When RPV level drops below the top of active fuel, the ability to remove the decay heat generated from the nuclear fuel becomes suspect and the Fuel Clad Fission Product barrier can no longer be considered intact. Sustained partial or total core uncovery can result in the release of a significant amount of fission products to the reactor coolant.

Under the conditions specified by this IC, severe core damage can occur and reactor coolant system pressure boundary integrity may not be assured. It is intended to address concerns raised by NRC Office for Analysis and Evaluation of Operational Data (AEOD) report AEOD/EG09, "BWR Operating Experience Involving Inadvertent Draining of the Reactor Vessel," dated August 8, 1986. This report states:

> In broadest terms, the dominant causes of inadvertent reactor vessel draining are related to the operational and design problems associated with the residual heat removal system when it is entering into or exiting from the shutdown cooling mode. During this transitional period, water is drawn from the reactor vessel, cooled by the residual heat removal system heat exchangers (from the cooling provided by the service water system), and returned to the reactor vessel. First, there are piping and valves in the residual heat removal system, which are common to both the shutdown cooling mode and other modes of operation such as low pressure coolant injection and suppression pool cooling. These valves, when improperly positioned, provide a drain path for reactor coolant to flow from the reactor vessel to the suppression pool or the radwaste system. Second, there is no comprehensive valve interlock arrangement for all shutdown cooling. Collectively, these factors have contributed to the inadvertent draining of the reactor vessel.

Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via effluent release EAL.

MS7 – Cont'd

BASIS (References) – Cont'd

DEVIATION: During EAL review and approval process, it was determined that the condition stated in NUMARC NESP-007, SS5, 1.a "Loss of all decay heat removal cooling as determined by (site-specific) procedure" is not necessary to conclude that the plant condition warrants a Site Area Emergency. Therefore, that sample NUMARC EAL was not included in this EAL.

MU7

INITIATING CONDITION

Reactor Coolant System leakage

EAL THRESHOLD VALUES

1. Unidentified primary system leakage > 10 gpm into the Drywell

   OR

2. Identified primary system leakage > 25 gpm into the Drywell

MODE APPLICABILITY

1, 2, 3, 4

BASIS (References)

Utilizing the leak before break methodology, it is anticipated that there will be indication(s) of minor reactor coolant system boundary integrity loss prior to this fault escalating to a major leak or rupture. Detection of low levels of leakage while pressurized is utilized to monitor for the potential of catastrophic failures. Leakage not associated with catastrophic failure potential such as SRV leakage, should not be considered in this EAL.

Identified and unidentified Primary System Leakage is measured by the normal primary system leakage monitoring system and is leakage into the drywell.

This EAL is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, it is considered to be a potential degradation of the level of safety of the plant. The value of 10 gpm unidentified leakage is significantly higher than the expected pressurized leak rate from the reactor coolant system. The 10 gpm value for the unidentified pressure boundary leakage was selected as it is twice the Technical Specification value, indicating an increase beyond that assumed in Safety Analysis. It also is observable with normal control room indications. The EAL for identified leakage is set at a higher value (25 gpm) due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.

Technical Specification LCO required actions would necessitate a plant shutdown and subsequent depressurization, unless the source of the leak can be isolated, identified, and/or stopped. Actions initiated by plant staff would include close monitoring of the calculated break size such that any sudden or gradual rise in leak rate would be identified. A slow power reduction and gradual depressurization would be necessitated due to the possibility that a sudden power and/or pressure surge could potentially worsen the break or cause a catastrophic failure.

The leak rate of 10 gpm may cause a high drywell pressure indication. Other indications of a leak of this magnitude would include a rise in drywell temperature or radiation.

MU7 – Cont'd

BASIS (References) – Cont'd

This event will escalate to an Alert based upon high Drywell pressure per Fission Product Barrier Matrix 2.c.1.

DEVIATION: NUMARC/NESP-007 Example EAL SU5.1.a identifies pressure boundary leakage. There is no Limerick EAL listed for pressure boundary leakage specifically since it is a subset of unidentified leakage. Limerick Tech. Specs. require a shutdown if any pressure boundary leakage is found.

MA8

INITIATING CONDITION

Main Steam Line Break

EAL THRESHOLD VALUES

1. MSL Break indicated by **EITHER** of the following:
   - High MSL Flow and High Steam Tunnel Temperature annunciators

     OR
   - Direct report of steam release

   AND

   MSL break is successfully isolated.

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Design basis accident analyses of a Main Steam Line Break outside of secondary containment shows that even if MSIV closure occurs within design limits, dose consequences offsite from a "puff" release would be in excess of 10 mRem.

Hi Steam Flow Annunciator and Hi Steam Tunnel Temperature Annunciator are both indicators of a Main Steam Line Break. Both parameters will cause an isolation of the MSIV's. Should both valves in any one line fail to isolate, this event would be considered a LOSS of Primary Containment (per IC 3.d.1) and a loss of RCS (per IC 2.d.1). This would then appropriately be classified as a Site Area Emergency.

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 visual 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.

Loss of the RCS Barrier due to an unisolable MSL break is covered under Fission Product Barrier Matrix (IC 2.d.1).

DEVIATION: NUMARC/NESP-007, Table 3 (RC Example EAL #1) EAL placed as a separate Alert threshold under previous NRC submittal to cover an isolable MSL break outside secondary containment. If the Main Steam Line (MSL) isolates as designed, this condition does not constitute a loss of RCS barrier. However, this condition was included as an event-based EAL due to the potential dose consequences associated with this event. This is consistent with the recommendations provided in the Industry-developed Questions and Answers on NUMARC/NESP-007 guidance, which was endorsed by the NRC in a letter dated June 10, 1993.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU9

INITIATING CONDITION

Unplanned loss of ALL onsite OR offsite communications capabilities

EAL THRESHOLD VALUES

1. ALL onsite communications equipment lost (Table M-4)

   OR

2. ALL offsite communications capability lost (Table M-5)

Table M-4 Onsite Communications Equipment	Table M-5 Offsite Communications Equipment
• Station Phones • PRELUDE System • Plant Public Address (PA) • Station Radio	• Station Phones • PRELUDE System • NRC (ENS) • County Police Radio • Load Dispatcher Radio • PA State Police Radio

MODE APPLICABILITY

ALL

BASIS (References)

Unplanned - The loss of communication is not a result of planned maintenance or surveillance activities

This EAL recognizes a loss of communication ability that significantly degrades the plant operations staff's ability to perform tasks necessary for plant operations or the ability to communicate with offsite authorities. This EAL is separated into two groups of communications, Onsite and Offsite. A complete loss of either group is so severe, that the Unusual Event declaration is warranted.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU10

INITIATING CONDITION

Inability to reach required operating mode within Technical Specification time limits

EAL THRESHOLD VALUE

1. Inability to reach required operating mode within Tech. Spec. LCO action completion time

MODE APPLICABILITY

1, 2, 3

BASIS (References)

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required operating mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate Notification of an Unusual Event is required when it is determined that there is an inability to bring the plant to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of an Unusual Event is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other various ICs address other required Technical Specification shutdowns that involve precursors to more serious events.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MA11

INITIATING CONDITION

Major Damage OR Uncovering of Spent Fuel

EAL THRESHOLD VALUES

1. Unplanned general area radiation > 500 mR/hr on the Refuel Floor (Table M-6)

   OR

2. Report or visual observation that irradiated fuel is uncovered

   OR

3. Water level < 22 ft. above seated irradiated fuel for the Spent Fuel Pool that will result in uncovery

Table M-6 Refuel Floor ARMs
• RIS29-M1-1(2)K600, Drywell Head Laydown • RIS30-M1-1(2)K600, Dryer/Separator Area • RIS31-M1-1(2)K600, Spent Fuel Pool • RIS32-M1-1(2)K600, New Fuel Storage Vault • RIS33-M1-1(2)K600, Pool Plug Laydown

MODE APPLICABILITY

ALL

BASIS: (References)

Offsite doses during these accidents would be well below the EPA Protective Action Guidelines and the classification as an Alert is therefore appropriate. This radiation level could also be caused by an inadvertent criticality and is included even though the probability of this event occurring is low. Radiation levels rise above 500 mR/hr, which were expected during a planned evolution, should not cause an Alert to be declared. Additionally, surveys, which identify "hot spots" greater than 500 mR/hr, should not cause an Alert to be declared.

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in NUMARC/NESP-007 IC AU2, "Unexpected Rise in Plant Radiation or Airborne Concentration."

NUREG-0818, "Emergency Action Levels for Light Water Reactors," forms the basis for this EAL. The areas where irradiated fuel is located forms the basis for Table 11-1. Unexpected radiation levels, which are at least 100 times higher than the normal background will generally indicate a fuel handling accident or loss of water covering the irradiated fuel. Readings may be from refuel floor Area Radiation Monitors or taken during a qualified radiological survey.

MA11 – Cont'd

BASIS: (References) – Cont'd

The value 22 feet above seated irradiated fuel is the Tech. Spec. Limit and an uncontrolled level drop that would uncover irradiated fuel is an indicator of a lowering in the level of safety of the plant.

There is time available to take corrective actions, and there is little potential for substantial fuel damage. In addition, 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 that risk of injury is low. In addition, NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel" presents the following in its discussion:

> In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of one mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85 in the event of an accident with decayed spent fuel.
>
> Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU11

INITIATING CONDITION

Potential Damage **OR** Uncovering of Spent Fuel

EAL THRESHOLD VALUE

1. Uncontrolled water level drop in the Spent Fuel Pool that cannot be quickly terminated with **ALL** irradiated fuel assemblies remaining covered by water

MODE APPLICABILITY

ALL

BASIS: (References)

Uncontrolled - An unexplained level drop that cannot be quickly terminated and is not the result of a planned evolution. The event should not be considered terminated if continuous make up is required and should not preclude classification of the Unusual Event.

This event tends to have a long lead time relative to potential for radiological release outside the site boundary, thus impact to public health and safety is very low.

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR all occurring since 1984, explicit coverage of these types of events via this EAL is appropriate given their potential for elevated doses to plant staff. Classification as an Unusual Event is warranted as a precursor to a more serious event.

This event will be escalated to an Alert as a result of uncovery of a fuel assembly and/or indication of high radiation levels on the refueling floor.

MA12

INITIATING CONDITION

Loss of Water Level That Has OR Will Uncover Irradiated Fuel

EAL THRESHOLD VALUES

1. Water level < 22 ft. above RPV flange for the Reactor Refueling Cavity (484 inches RPV water level)

   AND

   Loss of water level has or will result in irradiated fuel uncovering

MODE APPLICABILITY

5 (with the Reactor Cavity flooded)

BASIS: (References)

The value of 484 inches RPV water level, which equates to 22 feet above RPV flange, is the Tech. Spec. Limit and an uncontrolled level drop that would uncover irradiated fuel is an indicator of a lowering in the level of safety of the plant. Escalation would occur via Effluent Release, In-plant radiation, or Emergency Director Judgment.

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in NUMARC/NESP-007 IC AU2, "Unexpected Rise in Plant Radiation or Airborne Concentration."

NUREG-0818, "Emergency Action Levels for Light Water Reactors," forms the basis for this EAL.

There is time available to take corrective actions, and there is little potential for substantial fuel damage. In addition, 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 that risk of injury is low. In addition, NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel" presents the following in its discussion:

> In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of one mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85 in the event of an accident with decayed spent fuel.
>
> Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate.

Table 3-2: LGS EAL Technical Basis

RECOGNITION CATEGORY
SYSTEM MALFUNCTIONS

MU12

INITIATING CONDITION

Uncontrolled Water Level Decrease in Reactor Refueling Cavity

EAL THRESHOLD VALUE

1. Unexpected Fuel Pool Storage low level alarm

   AND

   Visual observation of an uncontrolled drop in water level below the fuel pool skimmer surge tank inlet that cannot be quickly terminated

MODE APPLICABILITY

ALL

BASIS (References)

Unexpected - An alarm that is not a result of a planned evolution

Uncontrolled - An unexplained level drop that cannot be quickly terminated and is not the result of a planned evolution. The event should not be considered terminated if continuous make up is required and should not preclude classification of the Unusual Event.

A drop in the Spent Fuel Pool level or the RPV [when in refueling and flooded up with the gates removed] will result in a control room annunciator Fuel Pool Storage Lo Level Alarm. This Control Room alarm directs an operator to be dispatched to a local alarm panel, which will identify the reason for the alarm. This alarm is validated with visual observation of a lowering Spent Fuel Pool level. If the spent fuel pool level drops below the inlet to the skimmer surge tank, without a planned event such as removing a large piece of equipment, there must be a leak in the spent fuel pool or the RPV. This event has a long lead time relative to potential for radiological release outside the site boundary, thus the impact to public health and safety is very low. Classification as an Unusual Event is warranted as a precursor to a more serious event.

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR all occurring since 1984, explicit coverage of these types of events via this EAL is appropriate given their potential for elevated doses to plant staff. Classification as an Unusual Event is warranted as a precursor to a more serious event.

This event will be escalated to an Alert as a result of uncovery of a fuel assembly and/or indication of high radiation levels on the refueling floor.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HG1

INITIATING CONDITION

Security Event Resulting in Loss of Ability to Reach **AND** Maintain Cold Shutdown

EAL THRESHOLD VALUES

1. Loss of physical control of the Control Room due to a security event.

   OR

2. Loss of physical control of the remote shutdown capability due to a security event.

MODE APPLICABILITY

ALL

BASIS (References)

This class of security event represents conditions under which a hostile force has taken physical control of areas required to reach and maintain cold shutdown. Loss of Remote Shutdown Capability would occur if the control function of the Remote Shutdown Panels were lost.

Security events, which meet the threshold for declaration of a General Emergency, are physical loss of the Control Room or the Remote and Alternate Shutdown Panels.

This situation leaves the plant in a very unstable condition with a high potential of multiple barrier failures.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS1

INITIATING CONDITION

Confirmed Security Event in a Vital Area

EAL THRESHOLD VALUES

1. Intrusion into plant Vital Area by a hostile force.

   OR

2. Confirmed bomb, sabotage or sabotage device discovered in a Vital Area

MODE APPLICABILITY

ALL

BASIS (References)

This class of security event represents an escalated threat to plant safety above that contained in an Alert in that a hostile intrusion or attack has progressed from the Protected Area to a Vital Area. The Vital Areas are within the Protected Area and are generally controlled by key card readers. These areas contain vital equipment, which includes any equipment, system, device or material, the failure, destruction or release of could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems, which would be required to function to protect health and safety following such failure, destruction or release, are also considered vital.

Identification of Vital Areas can be accomplished through discussions with security.

This event will be escalated to a General Emergency based upon the loss of physical control of the Control Room or Remote Shutdown Capability

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA1

INITIATING CONDITION

Confirmed Security Event in a Plant Protected Area

EAL THRESHOLD VALUES

1. Intrusion into the Protected Area by a hostile force.

   OR

2. Confirmed bomb, sabotage or sabotage device discovered in the Protected Area

MODE APPLICABILITY

ALL

BASIS (References)

This class of security event represents an escalated threat to the level of safety of the plant. This event is satisfied if physical evidence supporting the hostile intrusion or attack exists. The Emergency Director will declare an Alert subsequent after consulting with the on-shift Security representative to determine the validity of the entry conditions.

This event will be escalated to a Site Area Emergency based upon a hostile intrusion or act in-plant Vital Areas.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU1

INITIATING CONDITION

Confirmed Security Event That Indicates a Potential Degradation in the Level of Plant Safety

EAL THRESHOLD VALUES

1. A credible threat to the station reported by the NRC.

   OR

2. **BOTH** of the following criteria are met for an a credible threat reported by any other outside agency as determined per SY-AA-101-132, "Threat Assessment":
   - Is specifically directed towards the station.
   - Is imminent (≤ 2 hours)

   OR

3. Attempted intrusion and attack of the Protected Area

   OR

4. Attempted sabotage discovered within the Protected Area

   OR

5. Hostage/Extortion situation that threatens normal plant operations

MODE APPLICABILITY

ALL

BASIS (References)

A security threat that is identified as being directed towards the station and represents a potential degradation in the level of safety of the plant. A security threat is satisfied if physical evidence supporting the threat exists, if information independent from the actual threat exists, or if a specific group claims responsibility for the threat. The Shift Management will declare an Unusual Event subsequent to consulting with the on shift Security representative to determine the credibility of the security event per SY-AA-101-132 and the Physical Security Plan.

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 10 CFR 50.72 and will not cause an Unusual Event to be declared.

This event will be escalated to an Alert based upon a hostile intrusion or act within the Protected Area.

DEVIATION: A bomb device discovered within Plant Protected Area and outside the Plant Vital Areas is an Alert declaration as determined per the site Safeguards Contingency Plan and therefore is not included as an Unusual Event in the EAL scheme.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS2

INITIATING CONDITION

Control Room Evacuation Initiated **AND** Plant Control **CANNOT** be re-established in ≤ 15 minutes

EAL THRESHOLD VALUES

1. Control Room evacuation initiated

   AND

   Control of the plant **CANNOT** be re-established in ≤ 15 minutes per SE-1 or SE-6

MODE APPLICABILITY:

All

BASIS: (References)

The 15-minute time period starts when physical control of the plant is lost requiring Control Room evacuation OR when the required Control Room personnel have evacuated the Control Room.

Control - Placing all local control switches in local control necessary for operation from remote panels and the Shift Manager has determined that the systems for controlling reactivity, core cooling and heat sink functions are established.

Transfer of safety system control has not been performed in an expeditious manner but it is unknown if any damage has occurred to the fission product barriers. The 15-minute time limit for transfer of control is based on a reasonable time period for required Control Room personnel to leave the control room, arrive at the remote shutdown area, and re-establish plant control to preclude core uncovery and/or core damage. During this transitional period the function of monitoring and/or controlling parameters necessary for plant safety may not be occurring and as a result there may be a threat to plant safety.

This event will be escalated based upon system malfunctions or damage consequences.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA2

INITIATING CONDITION

Control Room Evacuation Initiated

EAL THRESHOLD VALUE

1. Entry into SE-1 or SE-6 for Control Room evacuation

MODE APPLICABILITY

All

BASIS (References)

Control Room evacuation requires establishment of plant control from outside the control room (e.g., local control and remote shutdown panel) and support from the Technical Support Center and/or other emergency facilities as necessary. Control Room evacuation represents a serious plant situation since the level of control is not as complete as it would be without evacuation. The establishment of system control outside of the Control Room will bypass many protective trips and interlocks. In addition, many of the instruments and assessment tools available in the Control Room will not be available.

This event will be escalated to a Site Area Emergency if control cannot be re-established within fifteen minutes.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA3

INITIATING CONDITION

Natural **OR** Destructive Phenomena Affecting a Vital Area

EAL THRESHOLD VALUE

1. Earthquake > 0.075g (Operating Basis Earthquake, OBE) as determined by procedure SE-5
   OR
2. Tornado or wind speeds > 75 mph causing damage to Plant Vital Structures (Table H-1)
   OR
3. Report of visible structural damage to **ANY** Plant Vital Structure (Table H-1)
   OR
4. Vehicle crash affecting a plant vital function contained in a Plant Vital Structure (Table H-1)
   OR
5. Turbine failure generated missiles result in visible structural damage to or penetration of **ANY** Plant Vital Structures (Table H-1)
   OR
6. Flooding in 2 or more areas designated in T-103, Table SCC-1 requiring a plant shutdown

Table H-1 Plant Vital Structures
• Reactor Enclosure • Control Enclosure • Turbine Enclosure • Diesel Generator Enclosure • Spray Pond Pump House / Spray Network

MODE APPLICABILITY

ALL

BASIS (References)

Each of these EALs is intended to address events that may have resulted in a plant vital 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 EAL to assess the actual magnitude of the damage.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA3 – Cont'd

BASIS (References)

Threshold Value 1 – This EAL addresses an earthquake that exceeds the Operating Basis Earthquake level of .075g and is beyond design basis limits . An earthquake of this magnitude may be sufficient to cause damage to safety related systems and functions. The Max Credible Earthquake for LGS is 0.15g per UFSAR Section 3.7; therefore, this EAL is conservative and warrants an Alert classification.

> Confirmed – As used in this EAL, a call to the National Earthquake Center is the primary confirmation source. Other confirmation includes reports from television or radio stations, or reports from university monitoring stations.

Threshold Value 2 – This EAL is based on FSAR design basis. Wind loads of this magnitude can cause damage to safety functions. This EAL addresses events where Plant Vital Structures have been struck with high winds, and thus damage may have occurred to a safe shutdown system. No attempt should be made to assess the magnitude of damage to Plant Vital Structures prior to classification.

Threshold Value 3 – Structural damage should be of sufficient force, that in the Emergency Director's judgment, the potential exists to affect the operation of systems and functions required for safe shutdown of the plant. This EAL specifies a Plant Vital Structure, which contain systems and functions required for safe shutdown of the plant.

Threshold Value 4 – The intent is to address such items as aircraft, train, barge or large motor vehicles (e.g. cranes, etc.). Automobiles, trucks and forklifts are also vehicles within the context of this EAL; however, the key is whether or not the vehicle can potentially affect a plant vital function, located within a designated Plant Vital Structure.

Threshold Value 5 – Missile impacts including rotating equipment or turbine failure causing casing penetration.

Threshold Value 6 – Flooding in vital areas that affect operability of safety-related systems or components. The source of the flooding need not be known.

Classification based on Schuylkill River level has been evaluated as not applicable. Design basis flood level at the site is 207 feet. Grade level is no lower than elevation 215 feet at any of the safety-related structures, and none of the safety-related structures has exterior openings below 217 feet. UFSAR Section 3.4.1 (Flood Protection) states, "...the safety-related structures are secure from Schuylkill River flooding and no special provisions for flood protection are necessary."

Per T.S. 3.7.1.3, the ultimate heat sink is the spray pond, which has a minimum LCO level of 250 ft. 10 inches Mean Sea Level. Spray Pond level is required to be verified once every 24 hours. No minimum river water level is specified in either the UFSAR or Technical Specifications. Restrictions for continued power operations are contained under S09.0.C (Compliance with Delaware River Basin Commission Docket Water Usage).

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU3

INITIATING CONDITION:

Natural OR Destructive Phenomena Affecting the Protected Area

EAL THRESHOLD VALUE:

1. Earthquake > 0.005g as determined by procedure SE-5
   OR
2. Report by plant personnel of a tornado strike within the Protected Area
   OR
3. Wind speeds > 75 mph as indicated on Site Meteorological instrumentation for > 15 minutes
   OR
4. Vehicle crash within the Protected Area Boundary that may potentially damage plant structures containing functions required for safe shutdown of the plant.
   OR
5. Report of turbine failure resulting in casing penetration or damage to generator seals.
   OR
6. Assessment by Control Room that a natural or destructive phenomena has occurred affecting Protected Areas

MODE APPLICABILITY:

ALL

BASIS: (References)

This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure or by the radiological releases and would be classified under the Fission Product Barrier Matrix or Event Category R, "Abnormal Radiological Conditions/Effluents".

The Emergency Director should consider how these Threshold Values may affect both units due to the affects of common or shared plant systems.

Threshold Value 1 - This EAL addresses a sensed earthquake. The magnitude of .005g is the lowest detectable earthquake measured on LGS seismic instrumentation per SE-5. An earthquake of this magnitude may be sufficient to cause minor damage to plant structures or equipment within the Protected Area. Damage is considered to be minor, as it would not affect physical or structural integrity. This event is not expected to affect the capabilities of plant safety functions. This event will be escalated to an Alert if the earthquake reaches an Operating Basis Earthquake (OBE).

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU3 – Cont'd

BASIS: (References) – Cont'd

Threshold Values 2 & 3 - A tornado touching down within the Protected Area or wind speeds > 75 mph within the Owner Controlled Area are of sufficient velocity to have the potential to cause damage to Plant Vital Structures. The value of 75 mph was selected to maintain consistency with plant value and to coincide with the Beaufort scale for Hurricane wind speed winds of 73-136 mph. These criteria are indicative of unstable weather conditions and represent a potential degradation in the level of safety of the plant. Verification of a tornado will be by direct observation and reporting by station personnel. Verification of wind speeds > 75 mph will be via meteorological data in the control room. This event will be escalated to an Alert if the tornado or high wind speeds strike a Plant Vital Structure.

Threshold Value 4 - This criterion is intended to address such items as plane, helicopter, or train crash that may potentially damage plant structures containing functions required for safe shutdown of the plant. Automobiles, trucks and forklifts are also vehicles within the context of this EAL; however, the key is whether or not the vehicle can potentially cause significant damage to plant structures. If the crash is confirmed to affect a plant vital function contained in a designated a plant vital structure, the event may be escalated to an Alert classification.

Threshold Value 5 - This criterion 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 leakage of combustible fluids (e.g., lubricating oils) and gases (e.g., hydrogen) to the plant environs. Actual fires and flammable gas build up are appropriately classified via other EALs. Turbine failure of sufficient magnitude to cause observable damage to the turbine casing or seals of the turbine generator raises the potential for leakage of combustible fluids and gases (Hydrogen cooling) to the Turbine Enclosure. The damage should be readily observable and should not require equipment disassembly to locate.

Threshold Value 6 – This criterion allows for the control room to determine that an event has occurred and take appropriate action based on personal assessment as opposed to verification (e.g., an earthquake is felt but does not register on any plant-specific instrumentation, etc.)

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA4

INITIATING CONDITION

Fire OR Explosion Affecting Operability of Safety Systems Required for Safe Shutdown

EAL THRESHOLD VALUES

1. ANY of the following are made potentially inoperable due to a fire or explosion:
   - 2 or more Safe Shutdown Systems (Table H-2)
   - 2 or more subsystems of a Safe Shutdown System (Table H-2), as defined by Tech. Specs.
   - 1 or more Plant Vital Structures containing Safe Shutdown Equipment (Table H-1)

   AND

   Safe Shutdown System or Plant Vital Structure is required for the present Operational Condition

Table H-1 **Plant Vital Structures**
• Reactor Enclosure • Control Enclosure • Turbine Enclosure • Diesel Generator Enclosure • Spray Pond Pump House / Spray Network

Table H-2: Safe Shutdown Systems		
• Diesel Generators • HPCI • Core Spray • SGTS • PCIS (Primary CNTMT Isolation System)	• 4 KV Safeguard Busses • RCIC • RHR Service Water • RERS • Control Room Ventilation	• RHR (all modes) • ESW • CAC

MODE APPLICABILITY:

ALL

BASIS: (References)

Explosion - A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to nearby structures or equipment.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA4 – Cont'd

BASIS: (References) – Cont'd

Fire - combustion characterized by the generation of heat and smoke. Sources of smoke such as overheated electrical equipment and slipping drive belts, for example, do not constitute fires. Observation of a flame is preferred, but is NOT required if large quantities of smoke and heat are observed.

The primary concern of this EAL is the magnitude of the fire and the effects on Safe Shutdown Systems required for the present Operational Condition. A Safe Shutdown System is defined as any system required to maintain safe operation or to establish or maintain Cold Shutdown. A system being "inoperable" means that it is incapable of performing the design function. For example, the LPCI System is intended to maintain adequate core cooling by covering the core to at least 2/3 core height following a DBA LOCA. In order for the system to be unable to maintain its intended function, multiple loops would need to be disabled by the fire. In addition to indication of degraded system performance, potential inoperability may be determined by visual observation and other control room indications such as loss of indicating lights.

Safe Shutdown Analysis is consulted to determine systems required for the applicable mode.

Two examples of applying this methodology are as follows:

- Diesel Generators and 4 KV Safeguard Buses: The fire disables multiple Diesel Generators or 4 KV Safeguard Buses so that the number of emergency power systems available would be lowered to below what would be required to mitigate an accident under the current operating conditions. For 100% power, this could be conservatively interpreted as at least two Diesel Generators or 4 KV Buses disabled.
- RHR - LPCI Mode: The fire disables multiple loops of LPCI so that adequate core submergence could not be assured following a DBA LOCA. For 100% power, this could also be conservatively interpreted as at least two loops disabled.

The EAL includes the condition that the fire must make "TWO OR MORE" subsystems (as defined by Tec. Specs.) or "TWO OR MORE" systems inoperable. In those cases where it is believed that the fire may have caused damage to *Safety Systems*, then an Alert declaration is warranted, since the full extent of the damage may not be known. For Plant Vital Structure damage, classification is required under this EAL if the structure houses or otherwise supports *Safety Systems* required for the present Operational Condition.

Degraded system performance or observation of damage that could degrade system performance is used as the indicator that the safe shutdown system was actually affected or made inoperable. A report of damage should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of damage. The occurrence of the fire or explosion with reports of damage (e.g., deformation, scorching) is sufficient for declaration.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU4

INITIATING CONDITION

Fire Within Protected Area Boundary **NOT** Extinguished in ≤ 15 minutes of Detection

EAL THRESHOLD VALUE

1. Fire within or impacting a Plant Vital Structure (Table H-1)

   AND

   Fire is **NOT** extinguished in ≤ 15 minutes of **EITHER**:
   - Control Room notification

     OR
   - Verification of alarm

   OR

2. Report by plant personnel of an explosion within the Protected Area Boundary resulting in visible damage to a permanent structure or equipment

Table H-1 Plant Vital Structures
• Reactor Enclosure • Control Enclosure • Turbine Enclosure • Diesel Generator Enclosure • Spray Pond Pump House / Spray Network

MODE APPLICABILITY

ALL

BASIS (References)

Verification - Determination is made that the fire alarm is not spurious.

Explosion - A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to nearby structures or equipment.

The purpose of this IC is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. This excludes such items as fires within administration buildings, wastebasket fires, and other small fires of no safety consequence. This IC applies to buildings and areas contiguous to plant vital areas or other significant buildings or areas. The intent of this IC is not to include buildings (e.g., warehouses) or areas that are not contiguous or immediately adjacent to plant vital areas.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU4 – Cont'd

BASIS (References) – Cont'd

Verification of the alarm in this context means those actions taken in the control room to determine that the control room alarm is not spurious.

This EAL addresses fires in Plant Vital Structures that house safety systems. These fires may be precursors to damage to safety systems contained in these structures. There are no areas/buildings contiguous to Plant Vital Structures, which could effect a safety system in one of the listed Plant Vital Structures except for those already on the list. Therefore, no additional areas/buildings are considered for this EAL.

Verification that a fire exists is by operator actions to confirm that fire alarms received in the Control Room are not spurious or by any verbal notification by plant personnel. Fifteen minutes has been established to allow plant staff to respond and control small fires or to verify that no fire exists.

This event will be escalated to an Alert if the fire damages redundant trains of plant safety systems required for the current operating condition.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA5

INITIATING CONDITION

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

EAL THRESHOLD VALUE

1. Report or detection of toxic gases within Plant Vital Structures (Table H-1) in concentrations that will be life threatening to plant personnel

   OR

2. Report or detection of flammable gases within Plant Vital Structures (Table H-1) in concentrations affecting the safe operation of the plant

Table H-1 Plant Vital Structures
• Reactor Enclosure • Control Enclosure • Turbine Enclosure • Diesel Generator Enclosure • Spray Pond Pump House / Spray Network

MODE APPLICABILITY

ALL

BASIS (References)

Gases within the site boundary that are above life-threatening or flammable concentrations, and have exceeded those concentrations within a Plant Vital Structure (as defined under Table H-1), should be declared as an Alert.

This IC is based on gases that have entered a plant structure affecting the safe operation of the plant. This IC applies to buildings and areas contiguous to plant Vital Areas or other significant buildings or areas. The intent of this IC is not to include buildings (e.g., warehouses) or other areas that are not contiguous or immediately adjacent to Plant Vital Areas. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred.

Concentrations above life-threatening or flammable concentrations that result from planned maintenance or repair activities on-site, where planned contingency measures are identified to monitor and control gas(es), do not require classification.

Threshold #1: Toxic gas concentration results in an atmosphere that is immediately harmful to unprotected personnel, and would preclude access to any such affected area. However, access into the affected area does not have to be required for classification purposes.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA5 – Cont'd

BASIS (References) – Cont'd

Threshold #2: Flammable gases, such as hydrogen and acetylene, are routinely used to maintain plant systems or to repair equipment / components. This EAL addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage / personnel injury.

This event will be escalated to higher classifications based on damage consequences covered under other various EAL Sections. 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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU5

INITIATING CONDITION

Release of Toxic OR Flammable Gases Deemed Detrimental to Safe Operation of the Plant

EAL THRESHOLD VALUES

1. Report or detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant

   OR

2. Report by Local, County, or State officials for a potential evacuation of site personnel based on an offsite event.

MODE APPLICABILITY

ALL

BASIS: (References)

Gases within the site boundary that are above life-threatening or flammable concentrations, and have not exceeded those concentrations within a Plant Vital Structure (as defined under Table H-1 in IC HA1), should be declared as an Unusual Event.

A toxic/flammable gas is considered to be any substance that is dangerous to life or limb by reason of inhalation or skin contact. It should not be construed to include confined spaces that must be ventilated prior to entry.

This IC is based on releases in concentrations within the site boundary that will affect the health of plant personnel or the safe operation of the plant with the plant being within the evacuation area of an offsite event (e.g., tanker truck accident releasing toxic gases, etc.). The evacuation is determined from the DOT Evacuation Tables for Selected Hazardous Materials, in the DOT Emergency Response Guide for Hazardous Materials.

Concentrations above life-threatening or flammable concentrations that result from planned maintenance or repair activities on-site, where planned contingency measures are identified to monitor and control gas(es), do not require classification.

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.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HG6

INITIATING CONDITION

Conditions Indicate Imminent Core Damage **OR** Release Affecting the Public

EAL THRESHOLD VALUES

1. Actual or imminent core degradation with potential loss of containment.

   OR

2. Potential uncontrolled radio nuclide release which can reasonably be expected to exceed 1 Rem TEDE or 5 Rem CDE Thyroid plume exposure levels at the Site Boundary

MODE APPLICABILITY

ALL

BASIS (References)

General Emergency - Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Imminent - Mitigation actions have been ineffective and trended information indicates that the event or condition will occur within 2 hours.

Potential - Mitigation actions are not effective and trended information indicates that the parameters are outside desirable bands and not stable or improving.

This EAL allows the Emergency Director to declare a General Emergency upon the determination of an actual or imminent substantial core degradation or melting with the potential for loss of containment integrity, but is not explicitly addressed by other EALs.

Releases may exceed the EPA Protective Action Guidelines for more than the immediate site area and will be classified under Event Category R, "Abnormal Radiological Levels/Effluents".

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HS6

INITIATING CONDITION

Conditions Indicate Actual OR Likely Failure of Plant Functions Needed for Public Protection

EAL THRESHOLD VALUE

1. Other conditions exist which in the judgment of the Emergency Director indicate actual or likely major failures of plant functions needed for protection of the public.

MODE APPLICABILITY

ALL

BASIS (References)

Site Area Emergency – Events are in process or have occurred which involve actual or likely failure of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels, which exceed EPA Protective Action Guideline exposure levels except near the site boundary.

This EAL allows the Emergency Director to declare a Site Area Emergency upon the determination of an actual or likely major failure of plant functions needed for protection of the public, but is not explicitly addressed by other EALs.

Releases are not expected to result in exposure levels, which exceed the EPA Protective Action Guidelines except within the site boundary and will be classified under Event Category R, "Abnormal Radiological Levels/Effluents".

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA6

INITIATING CONDITION

Conditions Indicate Actual OR Potential Substantial Degradation of the Level of Plant Safety

EAL THRESHOLD VALUE

1. Other conditions, exist which in the judgment of the Emergency Director indicate that plant safety systems may be degraded and that increased monitoring of plant functions is warranted.

MODE APPLICABILITY

ALL

BASIS (References)

Alert - Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

This EAL allows the Emergency Director to declare an Alert upon the determination that the level of safety of the plant has substantially degraded but is not explicitly addressed by other EALs. This includes a determination by Shift Management that the TSC and OSC should be activated and command and control functions should be transferred for the event to be effectively mitigated. Transfer of command and control functions is used as an initiator since an event significant to warrant transfer is a substantial reduction in the level of safety of the plant. Other examples are:

Internal flooding affects the operability of plant safety systems required to establish or maintain cold shutdown.

Releases that are expected will be limited to a small fraction of the EPA Protective Action Guidelines and will be classified under Event Category R, "Abnormal Radiological Levels/Effluents".

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HU6

INITIATING CONDITION

Conditions Indicate a Potential Degradation in the Level of Plant Safety

EAL THRESHOLD VALUE

1. ANY of the following occur, which in the judgment of the Emergency Director indicate a potential degradation in the level of safety of the plant:
   - Aircraft crash on-site
   - Train derailment on-site
   - Near-site explosion, which may adversely affect normal site activities

   OR

2. Other conditions exist, which in the judgment of the Emergency Director indicate a potential degradation in the level of safety of the plant.

MODE APPLICABILITY

ALL

BASIS (References)

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Unusual Event emergency class.

> Unusual Event – Events are in process of have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

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 response 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.

It is also intended that the Emergency Director's judgment not be limited by any list of events as defined here or as augmented by the site. This list is provided solely as examples for consideration and it is recognized that actual events may not always follow a pre-conceived description.

Section 4: Emergency Measures

4.1 Notification of the Emergency Organization

Notifications for the Limerick Generating Station are made to the following additional State and local agencies in accordance with Section E.3 of the Exelon Nuclear Standardized Radiological Emergency Plan:

- Pennsylvania Emergency Management Agency (PEMA)
- Berks County Emergency Management Agency
- Chester County Emergency Services
- Montgomery County Office of Emergency Preparedness

Notification of PEMA and the risk counties will be directed by the Emergency Director within 15 minutes of initial event classification, reclassification, or a change in a protective action recommendation (PAR) due to plant conditions or meteorological changes per Section E.3 of the Exelon Nuclear Standardized Radiological Emergency Plan. In addition, once the EOF is activated, the Corporate Emergency Director will contact the Senior Pennsylvania State Official as designated by PEMA following the decision to recommend a protective action for the general public.

Upon notification of an emergency at Limerick Generating Station, the Pennsylvania Bureau of Radiation Protection (BRP) will contact the appropriate station to verify that an emergency exists and to obtain technical information, and then makes recommendations to PEMA regarding protective actions for the public. The BRP Support Plan For Fixed Nuclear Facility Incidents utilizes the Protective Action Guidelines in the U.S. Environmental Protection Agency (EPA) 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents".

Exelon Nuclear will provide follow-up information to the BRP or other off-site authorities. The follow-up information will keep these authorities apprised of existing or potential radiological releases, meteorological conditions, projected doses and contamination levels, licensee actions, recommend protective actions and other information pertinent to the authorities responsibilities. The information may be provided over open communication paths or in person to BRP personnel.

4.2 Assessment Actions

The effluent radiation monitoring system provides indications of gross releases of gaseous and liquid radioactivity. By applying calibration factors, meteorological data, or river flow, the gross indications are used to calculate approximate release rates in μCi/sec and dose rates at specific distances along the release pathways. Particulate and iodine analysis depends on collecting installed filter papers and charcoal cartridges for analysis in the counting room. Similar calculation procedures are applied to approximate release rates and dose rates due to iodine.

Detectors are strategically located throughout the plant. These detectors indicate and alarm locally and in the Control Room. They serve the purpose of indicating current dose rates in those areas and are used for local evacuation action levels and re-entry operations.

Certain plant operating systems contain radiation monitors. These systems are described in the LGS UFSAR.

Portable monitoring instruments and sampling equipment consist of such items that are utilized and maintained on-site by the Chemistry and Health Physics sections for normal day-to-day plant operations and are thus available for emergency operations.

4.2.1 Core Damage Assessment Methodology

Core damage information is used to refine dose assessments and confirm or extend initial protective action recommendations. Limerick Generating Station utilizes NEDC-33045P, "Methods of Estimating Core Damage in BWRs" (Revision 0, July 2001), as the basis for the methodology for post-accident core damage assessment. This methodology utilizes real-time plant indications in addition to samples of plant fluids and atmospheres. Core damage is qualitatively evaluated per NRC Core Condition Categories (1-10) as shown in the table below:

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

4.3 Protective Actions for the Offsite Public

PEMA interface for incidents at Limerick Generating Station will be with Berks, Chester and Montgomery Counties. County and local governments have primary responsibilities for implementing protective measures for the public following a nuclear incident.

The BRP serves as lead Pennsylvania State agency for technical assistance to other state agencies, county, and local governments regarding radiological health and accident assessment. In the absence of communications with the state, recommendations for protective actions shall be made directly to county emergency operations centers from the station.

4.3.1 Alert and Notification System (ANS) Sirens

Annex E of the Commonwealth of Pennsylvania Emergency Operations Plan addresses notification to the general public and others regarding protective actions. An Alert Notification System, which is intended for use by the counties, in conjunction with the Emergency Alert System (EAS) to provide notification to the general public, has been installed.

Alerting of the EPZ population is provided by a siren system that was installed and is maintained by Exelon Nuclear. The system consists of high-powered rotating electro-mechanical sirens mounted on Class 1 utility poles throughout the Plume Exposure Pathway (10-Mile EPZ). Personnel at the risk county communication centers operate the sirens. The Pennsylvania Emergency Management Agency (PEMA) coordinates the activation of the siren system for Limerick Generating Station.

The siren system meets or exceeds the acoustic coverage requirements outlined in NUREG-0654/FEMA-REP-1 and FEMA-REP-10. A computer-based sound propagation model determined the location of each siren site.

The sirens are controlled by digitally encoded radio signals transmitted by a transceiver at the station. Each risk county has control of the sirens that are physically located in that county. The sirens can be activated on an individual, municipal, county, or EPZ-wide basis.

A controller located at the station serves as a backup to the county controllers. After the system is activated, each siren reports the result of its activation back to the respective county controller and the controller at the station. The siren system is tested regularly to ensure its operability.

Annex E delineates risk counties as responsible to:

- Develop a system for rapid notification (in priority order) of county and local government heads, key staff, emergency forces, volunteer organizations, schools, hospitals, nursing homes, business, and industry;
- Ensure that the alert and notification system is operable on an around-the-clock basis;
- Prepare and disseminate public information material on protective actions to provide clear instructions to the population at risk;
- Prepare and maintain material current for dissemination through the EAS; and
- Include provisions in the alert plan for notification of transients.

PEMA will notify other states within the Ingestion Pathway EPZ should such action be necessary.

Annex E calls for each risk county to promptly activate their alert notification system, when appropriate. EAS radio stations will be activated and instructed as to which prepared message to use. Detailed messages with specific instructions to the public will be provided to the EAS stations by state and county public information officers on a timely basis. Various state agencies will assist the counties in assuring notifications of transients.

4.3.2 Evacuation Time Estimates

The evacuation time estimates (ETE) were developed in coordination with the Commonwealth of Pennsylvania to assess the relative feasibility of an evacuation of the 10-Mile EPZ for the Limerick Generating Station. The evacuation times are based on a detailed consideration of the EPZ roadway network and population distribution. The ETE Study, maintained separately by Emergency Preparedness, presents representative evacuation times for daytime and nighttime scenarios under various weather conditions for the evacuation of various areas around the Limerick Generating Station, once a decision has been made to evacuate. Refer to Figure LGS 4-1, "10-Mile EPZ Permanent Resident Population Distribution," for an illustration of population location around LGS.

4.3.3 Potassium Iodide (KI)

The Department of Health, Commonwealth of Pennsylvania, is responsible for providing advice to PEMA on the planning for the use, stockpiling and distribution of Potassium Iodide (KI) or other thyroid blocking agents and such other radiological health materials as may be required for the protection of the general public. Their decision shall also be based on U.S. FDA guidance.

Based on the criteria established under the Appendix E of the Commonwealth of Pennsylvania Operations Plan, LGS will recommend to government officials that the general public be notified to take KI at a General Emergency classification in those sectors were an evacuation has been recommended. This notification will be approved by the Emergency Director in Command and Control of PAR decision-making and off-site notifications, and performed as part of the State / local notifications described under Sections II.B.4 and II.E.3 of the Exelon Nuclear Standardized Radiological Emergency Plan.

4.3.4 Public Information

a. Publications

Public information on protective actions is prepared and disseminated annually to provide clear instructions to the population-at-risk. Exelon Nuclear assists PEMA and risk counties in the preparation and distribution of their respective public information.

Pamphlets outlining public education response actions are readily available for transients in the 10-Mile EPZ. In addition, emergency information is provided to the operators of other recreational areas in the 10-Mile EPZ, as defined by the Commonwealth of Pennsylvania and risk counties.

These public information publications (including telephone book emergency information, etc.) instruct the public to go indoors and turn on their radios when they hear the ANS sirens operating. These publications identify the local radio stations to which the public should tune in for information related to the emergency. Additional materials (e.g., such as rumor control numbers, evacuation routes, information on inadvertent siren soundings, etc.) may also be included in these publications based on agreements with responsible State and risk county agencies.

b. News Media Education

Information kits are available to news media personnel. These kits include information on a variety of nuclear power plant related subjects.

4.3.5 Protective Action Recommendations (PARs) for the General Public

Figure LGS 4-2, "Plant-Based PAR Determination Flowchart", illustrates affected downwind sectors based on wind direction, using the generic plant-based event logic as outlined in Figure J-1 of the Exelon Nuclear Standardized Radiological Emergency Plan. Further evaluation of PAR based on dose assessments shall be performed in accordance with Section II.J.10.m.2 of the Exelon Nuclear Standardized Radiological Emergency Plan.

4.4 Protective Actions for Onsite Personnel

4.4.1 Plant Evacuation

Exelon Nuclear personnel and contractors filling emergency response organization positions are considered essential personnel. As such, they will report to their emergency response locations. They will not evacuate unless specifically directed by the Emergency Director. All other personnel are considered non-essential.

In-plant evacuation is initiated primarily by area radiation monitor alarms and continuous air monitor alarms, but is also applicable for fire alarms, explosions, toxic material conditions, as well as radiation, contamination, and airborne radioactivity surveys which indicate conditions above applicable limits. Notification for personnel to proceed with in-plant evacuation will be via a local alarm or an announcement on the plant PA system. The affected area and evacuation assembly areas (if appropriate) will be announced. The immediate response by individuals in the vicinity of such an alarm or announcement is evacuation to an unaffected area or designed assembly area. In the absence of readily available radiological survey information or other logical assessment of conditions, evacuation will be, at least, to a point where other area radiation monitors, continuous air monitors, or observation of local conditions show that the area is not affected.

Assigned plant personnel report to the scene to evaluate conditions, to provide information to the Control Room, and to perform other emergency functions such as personnel accountability, decontamination, medical assistance, and control of the hazard.

Notification of a Site Evacuation is accomplished by activating the Evacuation Alarm System followed by an announcement over the plant PA system. The evacuation assembly area(s) are announced. Evacuation assembly areas are illustrated in Figure LGS 4-3. Non-essential personnel will exit via the security exit points and will proceed to the parking lot for transportation. Evacuees are expected to use their personal vehicles in evacuating to the designated evacuation assembly area(s). Designated evacuation assembly areas are located outside the protected area. Plant access roads are maintained clear during the winter months, travel on these roads is expected to be possible at all times.

Plant visitors who have not completed the required training program are escorted at all times. This ensures proper response under emergency conditions. Visitors at the station shall follow the lead of their escorts to the assembly areas.

4.4.2 Personnel Accountability

The Security personnel shall follow security procedures for personnel accountability. For evacuations, information from evacuees is an important means of accounting for plant personnel. For Site Evacuations, non-essential personnel are accounted for at the security exit point. Emergency response personnel are accounted for by badging into their assembly areas.

4.4.3 Monitoring of Evacuees

Evacuees from the Limerick Site are checked for contamination. Necessary personnel and vehicle decontamination efforts are initiated at the evacuation assembly area using in-plant equipment or emergency kit supplies. Priority for decontamination shall be given to personnel found to have the highest levels of contamination. Any personnel suspected, or known, to have ingested or inhaled radioactive material shall be given a whole body count, as soon as conditions permit, to assess their internal exposure.

The registering and monitoring of the general public evacuating from the Plume Exposure Pathway EPZ, as described in Section II.J.12 of the Exelon Nuclear Standardized Radiological Emergency Plan, will occur at designated facilities per the respective State and County Radiological Emergency Response Plans.

4.5 Severe Accident Management

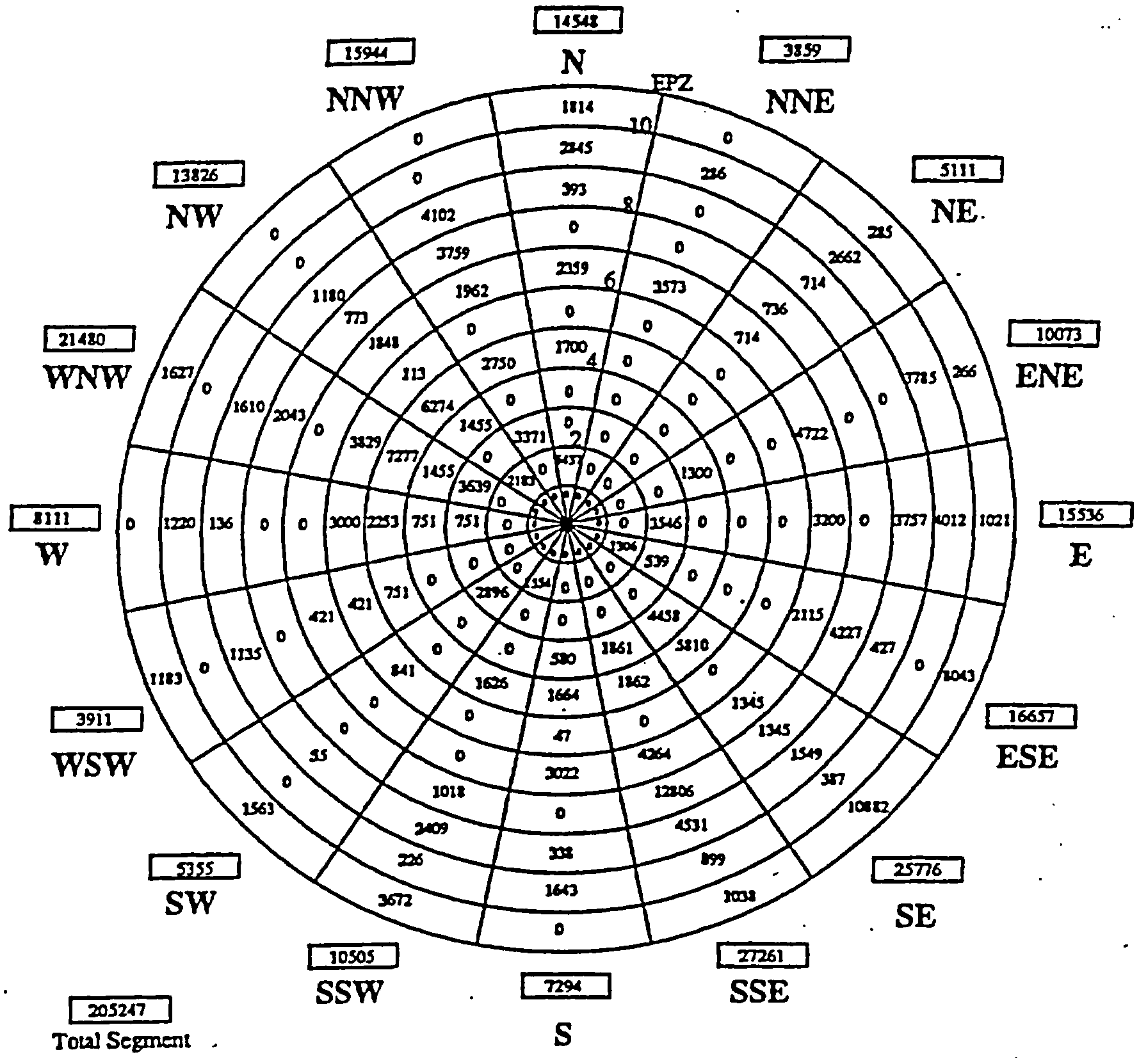
Accident management consists of those actions taken during the course of an accident, by the Emergency response Organization (ERO), specifically: plant operations, technical support, and plant management staff in order to:

- Prevent the accident from progressing to core damage;
- Terminate core damage once it begins;
- Maintain the capability of the containment as long as possible; and
- Minimize on-site and off-site releases and their effects.

The later three actions constitute a subset of accident management, referred to as Severe Accident Management (SAM) or severe accident mitigation. The Severe Accident Management Plan Procedures (SAMPs) provide sound technical strategies for maximizing the effectiveness of equipment and personnel in preventing, mitigating and terminating severe accidents.

Implementation of SAMP procedures is a collaborative effort between the Shift Manager and the Station Emergency Director in the TSC (once activated). The Station Emergency Director maintains ultimate responsibility for direction of mitigating strategies. Designated TSC Technical and Operations Support personnel are also trained to assist with decision-making by evaluating plant conditions using the SAM Technical Support Guidelines (TSG).

Figure LGS 4-1: 10-Mile EPZ Permanent Resident Population Distribution



205247
Total Segment Population 0 to EPZ Boundary

Population Totals			
Ring	Ring Population	Total	Cumulative Population
0-1 Mile	0	0-1 Mile	0
1-2 Mile	10480	0-2 Mile	10480
2-3 Mile	14742	0-3 Mile	25222
3-4 Mile	11860	0-4 Mile	37082
4-5 Mile	31967	0-5 Mile	69049
5-6 Mile	8251	0-6 Mile	77300
6-7 Mile	29545	0-7 Mile	106845
7-8 Mile	26707	0-8 Mile	133552
8-9 Mile	22336	0-9 Mile	155888
9-10 Mile	17965	0-10 Mile	173853
10-EPZ	31394	0-EPZ	205247

Figure LGS 4-2: Plant-Based PAR Determination Flowchart

General Emergency Declared? — No → No PARs Required

Yes ↓

LOSS of 'FUEL' Fission Product Barrier — No →

Yes ↓

LOSS of 'RCS' Fission Product Barrier — No → Evacuate 2 Mile Radius & 5 Miles Downwind; AND Recommend KI for general public in affected sectors.

Yes ↓

LOSS of 'Containment' Fission Product Barrier — No →

Yes ↓

(LOSS of ALL 3 barriers) Evacuate 5 Mile Radius & 10 Miles Downwind; AND Recommend KI for general public in affected sectors. →

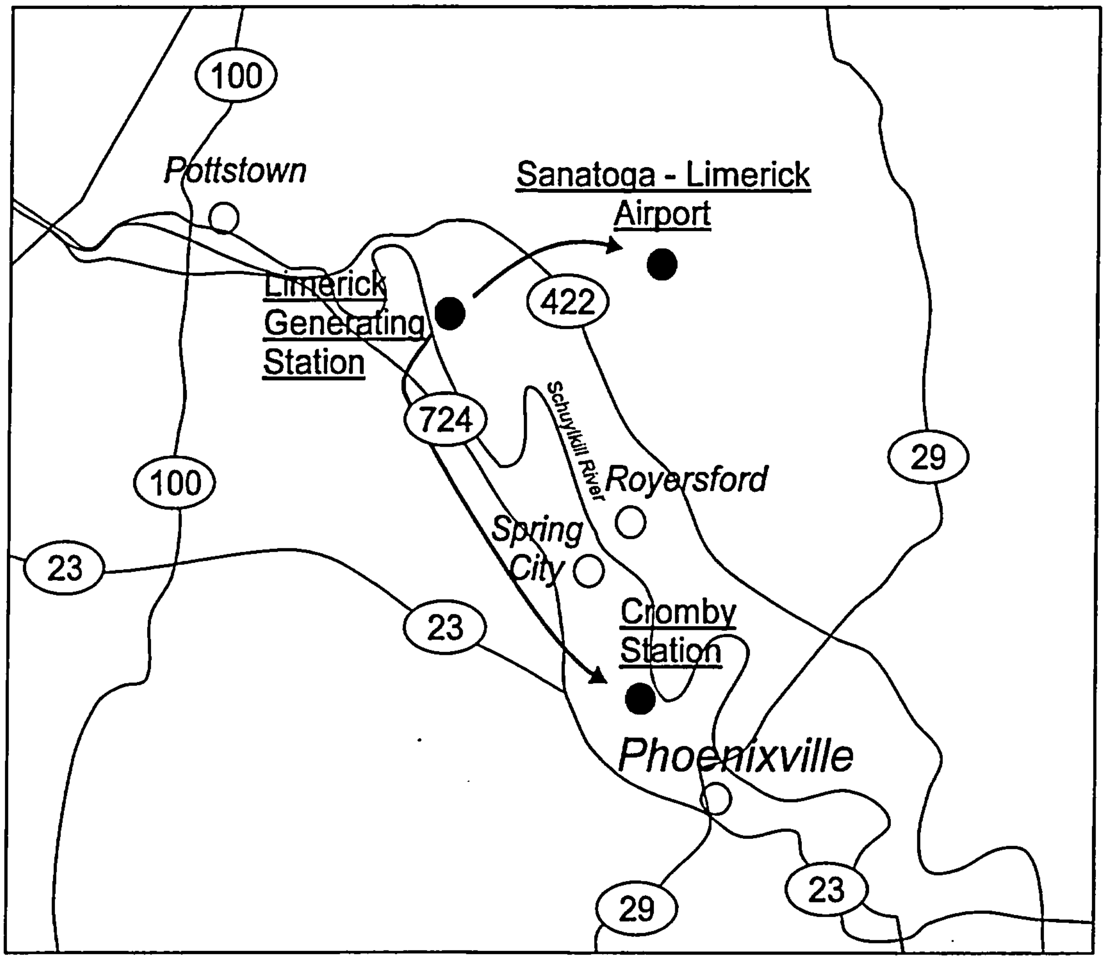
NOTE: Ensure dose-based PARs are evaluated when a release is in progress.*

WIND DIRECTION (FROM)	DOWNWIND SECTOR(S)*
355 to 005	SSE / S / SSW
006 to 017	SSE / S / SSW /SW
018 to 027	S / SSW /SW
028 to 039	S / SSW /SW / WSW
040 to 050	SSW /SW / WSW
051 to 062	SSW /SW / WSW / W
063 to 072	SW / WSW / W
073 to 084	SW / WSW / W / WNW
085 to 095	WSW / W / WNW
096 to 107	WSW / W / WNW / NW
108 to 117	W / WNW / NW
118 to 129	W / WNW / NW / NNW
130 to 140	WNW / NW / NNW
141 to 152	WNW / NW / NNW / N
153 to 162	NW / NNW / N
163 to 174	NW / NNW / N / NNE
175 to 185	NNW / N / NNE
186 to 197	NNW / N / NNE / NE
198 to 207	N / NNE / NE
208 to 219	N / NNE / NE / ENE
220 to 230	NNE / NE / ENE
231 to 242	NNE / NE / ENE / E
243 to 252	NE / ENE / E
253 to 264	NE / ENE / E / ESE
265 to 275	ENE / E / ESE
276 to 287	ENE / E / ESE / SE
288 to 297	E / ESE / SE
298 to 309	E / ESE / SE / SSE
310 to 320	ESE / SE / SSE
321 to 332	ESE / SE / SSE / S
333 to 342	SE / SSE / S
343 to 354	SE / SSE / S / SSW

*BOLD refers to affected sector(s)

* Dose projections are NOT required to support the decisions in the plant-based PAR flowcharts. However, it is expected that a dose projection be performed as soon as possible by the facility in Command and Control at a General Emergency with a release in progress.

Figure LGS 4-3: Off – Site Assembly Location



TYPE OF EVACUATION	EVACUATION ASSEMBLY AREAS
LOCAL EVACUATION	Announced on PA System
SITE EVACUATION	Limerick Generating Station Limerick Airport, Cromby Station, Other Designated Area

Section 5: Emergency Facilities and Equipment

5.1 Emergency Response Facilities

5.1.1 Station Control Room

The Limerick Generating Station Control Room shall be the initial onsite center of emergency control. The Control Room is located on the 269' elevation of the Control Structure. The ventilation system, shielding, and structural integrity are designed and built to permit continuous occupancy during the postulated design basis accident.

5.1.2 Technical Support Center (TSC)

Limerick Generating Station has established a Technical Support Center (TSC) adjacent to the Protected Area Main Access Facility. The TSC fully meets the requirements of Section H.1.b of the Exelon Nuclear Standardized Radiological Emergency Plan and conforms to Section 8.2.1 of Supp. 1, NUREG-0737.

5.1.3 Operational Support Center (OSC)

Limerick Generating Station has designated an Operational Support Center (OSC). The OSC is located on the 217' elevation in the Health Physics Field Office, adjacent to the Turbine Enclosure. The OSC conforms to the requirements of Section H.1.c of the Exelon Nuclear Standardized Radiological 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.

In the event the OSC is not habitable, personnel report to backup facilities that can be designated based upon specific event conditions.

5.1.4 Emergency Operations Facility (EOF)

The dedicated Emergency Operations Facility (EOF) is located on Exelon property at 175 North Caln Road, Coatesville, PA. The EOF supports both Peach Bottom and Limerick, and is located approximately 20 miles from Limerick Generating Station. Separate offices are provided for Exelon Nuclear, NRC, Maryland and Pennsylvania representatives and other emergency personnel.

Plant Monitoring System data is available through the Emergency Preparedness Data System (EPDS) at the EOF.

The EOF equipment includes:

a. Supplies and equipment for EOF personnel, and

b. Sanitary and food preparation facilities.

5.1.5 Joint Public Information Center (JPIC)

The Joint Public Information Center (JPIC) is the facility in which media personnel gather to receive information related to the emergency event. The JPIC is co-located with the EOF at 175 North Caln Road, Coatesville, Pennsylvania.

5.2 Assessment Resources

5.2.1 Geophysical Monitors

a. Onsite Meteorological Monitoring Program

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. Meteorological data is provided to the station Control Room from Meteorological Towers. Data include wind speed, wind direction, and temperature. Meteorological monitoring is described in the LGS UFSAR.

b. Seismic Monitoring

Seismic instrumentation includes time-history strong motion pressure triaxial seismic monitor accelerographs located in secondary containment. Peak recording accelerographs, and seismic switches are discussed in the LGS UFSAR.

5.2.2 Radiation Monitoring Equipment

For radiological assessments, instrumentation includes area radiation monitors (ARMs), ventilation effluent radiation monitors, liquid effluent radiation monitors, stack effluent monitors, primary containment radiation monitors and miscellaneous process radiation monitors (Refer to LGS UFSAR Sections 11, 12 and 15 for additional information). Data from these sources would be augmented by plant and field surveys for radiation and airborne levels.

a. Radiological Effluent Gaseous Monitoring

LGS has four monitored points of release of radioactive material to the atmosphere. These are South Stack 1 and 2, Hot Maintenance Shop, and the North Stack. Sample systems are installed for these four pathways. The sample system consists of isokinetic sample lines containing particulate/iodine filters and separate sample lines to shielded gas chambers. Detector output data associated with the gas chambers are available in the Control Room.

The stacks radiation monitoring system continuously monitors the noble gas being discharged from Unit 1 and Unit 2. Each unit has two independent monitoring skid stations for its North and South stacks and a common North stack Wide Range Accident Monitor (WRAM). The monitoring stations use scintillation detectors which readout in the Main Control Room in uCi/sec and uCi/cc.

Gas chamber detectors readouts are in uCi/cc. The WRAM readout is in units of uCi/sec. The uCi/sec Iodine and particulate concentrations are determined from the filter and charcoal cartridge samples.

b. Radiological Effluent Liquid Monitoring

Liquid releases are made on a batch basis from waste sample tanks. The contents of these tanks are circulated prior to sampling and analysis. Permits are prepared to authorize releases to the cooling tower blowdown line. Radiation monitors are located on certain process water systems, which indicate abnormal radioactivity levels. Procedures describe the technique for determining consequences of an abnormal release.

c. Laboratory Facilities

Chemical laboratories are located adjacent to the radwaste enclosure at LGS. A radiochemistry section is provided. The laboratories are adjacent to the counting room for convenience in transporting prepared samples for counting.

5.2.3 Data Acquisition Methods

a. Plant Monitoring System (PMS)

The LGS Main Control Room (MCR) and Technical Support Center (TSC) use an emergency facility data system to aid in assessing plant response and status during emergencies. PMS is a computer-based real-time data acquisition and display system, which gathers and records, selected plant parameters for display.

The system displays are designed to aid the Control Room operator in the performance of emergency response procedures. These displays provide information pertinent to reactor core cooling, reactor coolant system integrity, reactivity control, containment integrity and power system status. These displays are also available to personnel in the TSC.

PMS also provides concise displays of parameters selected for post-accident monitoring. These displays are designed to aid TSC personnel in assessing plant conditions and in assisting Main Control Room personnel in recovering from abnormal or accident conditions and in mitigating their consequences. The displays include parameter versus time and parameter versus parameter trending.

PMS utilizes high-speed data recording, long-term data storage and a transient analysis program package to aid the Technical Support Center staff in reconstructing the accident sequence as well as tracking the plant steady-state and dynamic behavior prior to and through the course of an event.

PMS displays are available in the Main Control Room and TSC, and EOF through EPDS interactive color graphic display consoles. Hardcopy output devices are available at each location. Provisions have been made to share data with State Liaisons located in the EOF.

b. Emergency Preparedness Data System (EPDS)

The Emergency Preparedness Data System (EPDS) is an emergency facility data system to aid in assessing plant response and status during emergencies. EPDS is a computer based real-time data acquisition and display system, which acquires, stores and re-packages data from PMS and RMMS plant parameters for display in the Technical Support Center and Emergency Operations Facility.

5.2.4 Onsite Fire Detection Instrumentation

LGS is afforded fire protection from various systems, selected for their applicability in coping with the several possible types of fires. These systems include an extensive fire water system, carbon dioxide system, air foam system, dry chemical system, heat and smoke detectors as well as portable fire extinguishers located throughout the plant. These systems have alarm outputs located in the Control Room. Fire protection systems are described in the LGS UFSAR.

5.2.5 Facilities and Equipment for Offsite Monitoring

Off-site Radiological Environmental Monitoring Program is described in the Offsite Dose Calculations Manual (ODCM). Installed radiological monitoring equipment and facilities, including process, area, and effluent, are described in the LGS UFSAR. Sets of instruments are available for emergency use by field survey teams. The field survey teams perform field surveys to locate and track the plume and to determine depositing of activity on the ground.

Emergency kits contain radiation survey equipment, which enables the Field Survey Teams to obtain dose rates, surface contamination, and airborne contamination including radio iodine measurements to supplement calculations based on effluent data. These emergency kits are located at facilities outside the plant for ready accessibility. The equipment in these kits is dedicated for emergency use only.

Concurrent field sampling and analysis for radio iodine provides the capability to detect 10^{-7} μCi/cc I-131, per NUREG-0654, FEMA-REP-1.

The services of Normandeau Associates Inc. (NAI) are contracted to provide for the collection of environmental media samples (e.g., water, grass vegetation, etc.) under emergency conditions and there transport to an offsite laboratory for analysis.

5.2.6 Site Hydrological Characteristics

A list of downstream users is maintained to ensure that they are notified. Should contamination of site drinking water sources be suspected, water samples shall be analyzed.

5.3 Protective Facilities and Equipment

5.3.1 Emergency Supplies

Refer to Table LGS 5-1 for a listing of Emergency Supplies and Equipment.

5.3.2 Maintenance Equipment

Maintenance equipment consists of normal and special purpose tools and devices utilized in the course of maintenance functions throughout the station. Maintenance and Radiation Protection personnel responding to the OSC are cognizant of the locations of equipment, which may normally be required in an emergency condition. The Maintenance supervision has access to keys for tool storage, shops, and other locations where maintenance equipment may be stored.

5.4 First Aid and Medical Facilities

EMT kits are located in designated areas and are checked and replenished as necessary. Stretchers are also provided at designated locations.

5.4.1 Decontamination and Medical Response

An on-site personnel decontamination facility for emergency conditions include showers and sinks that drain to the liquid radioactive waste processing system, at the primary health physics decontamination area in the plant. Special decontamination materials and personnel decontamination procedures are available in the area for use under the direction of health physics supervision. Provisions are made for medical decontamination when personnel are transported to hospitals.

5.4.2 Emergency Medical Assistance Program (EMAP)

An Emergency Medical Assistance Program plan has been established to provide for consultation and definitive care for radiation accident victims. The EMAP distinguishes three levels of medical care:

1. First aid, decontamination, and preliminary patient evaluation at the site
2. Emergency care and patient stabilization in a supporting hospital
3. When necessary, definitive evaluation and treatment.

The EMAP provides for a Radiation Emergency Medical Team (REM Team) to respond to accident 24 hours per day. The team consists of experienced physicians, board certified health physicists, and technicians. It has portable medical and health physics equipment to render emergency treatment at accident sites and conduct the initial evaluation of the radiation status of patients as well as the environment. For on-site medical assistance, REM Team capabilities include:

1. Consultation and actual assistance to site medical response personnel and the attending Physician
2. Assistance in personnel decontamination

The Emergency Medical Assistance Program (EMAP) consultant has access to extensive laboratory facilities, which provide the capability for radiochemical analysis of plant and environmental samples.

Bioassay including whole body counting, gamma spectroscopy and personnel dosimetry processing are among the capabilities of EMAP.

5.4.3 Medical Transportation

Transportation of injured personnel, who may or may not be radioactively contaminated, to medical treatment facilities is provided by local ambulance services. (Refer to Section 2.4 of the Limerick Annex)

5.5 Communications

Refer to Section F.1 of the Exelon Nuclear Standardized Radiological Emergency Plan for a description of dedicated communications lines to support both offsite and inter-facility communications.

5.5.1 Intra-Plant Public Address (PA) System

The LGS PA system is a six-channel system powered from a Class IE bus permitting simultaneous use of one page line and five party lines. Loudspeakers powered by individual amplifiers are located throughout the plant and in remote structures.

The LGS Public Address system has also been equipped with an advanced page line control system for the enhancement of page announcements throughout the site. This control system provides improved sound quality for emergency announcements made to and from the main control room. It is also capable of screening out page announcements that do not originate from designated page announcement control points such as the Control Room, TSC, OSC, security locations, etc.

Local area PA announcements can still be conducted by the use of the emergency page button, and the entire system can be reverted back to allow announcements from all locations as required during emergency conditions. The primary purpose of the screening function is to reduce the number of locations where site wide page announcements can originate.

The LGS Public Address stations within the plant are equipped with two page buttons. One is for normal plant pages, and the other is for emergency pages to the Control Room. When used, the emergency page button unlocks the PA speakers in the Control Room for the incoming message. The Control Room speakers are silent (muted) for all normal plant pages. This arrangement allows for a more orderly Control Room and emphasizes the emergency pages made to the Control Room. A PA station is located in the Main Control Room, Operations Support Center, and TSC. Capability exists to warn individuals in the vicinity of the river through the river warning system utilizing the plant PA system.

The Main Control Room has priority page abilities that allow the MCR announcements to override normal plant page announcements.

5.5.2 Private Branch Exchange (PBX) Telephone System

The LGS on-site commercial telephone system provides telephone communications capabilities throughout the plant, remote structures, and with off-site parties. Extensions are located in the Main Control Room, the TSC, and the OSC.

The power supply for this system consists of two separate on-site sources. Both sources are supplied from motor control centers. The primary source is backed-up by an emergency diesel generator, and the secondary source is backed up by a 2 -hour uninterruptible power supply (UPS). The 2-hour UPS is designed to allow sufficient time to restore the diesel-generator supplied power source, if necessary. This power configuration is designed to maintain this communication system during a total station blackout.

The PECO Energy Main Office and Exelon Nuclear headquarters are also served by separate commercial telephone systems (PBX's). All PECO Energy and Exelon Nuclear's PBX's are networked together to create a fully-integrated voice network, providing call management and network redundancy.

5.5.3 Dedicated Emergency PBX Telephone System

The LGS dedicated emergency (PBX) telephone system provides rapid and reliable communications in the event of an emergency. It is independent of the main PBX switch. The dedicated emergency PBX allows rapid dialing and conferencing of emergency response personnel. Extensions are located in the Control Room, the TSC, the OSC, the EOF, and the JPIC. Tie line access capability is provided both through the Limerick main PBX switch and the Peach Bottom dedicated emergency PBX switch.

The power supply for this system consists of two separate on-site sources, which are different than the sources for the main PBX switch. The primary source is backed-up by an emergency diesel generator. The secondary source backup is a 15-minute Uninterruptible Power Supply (UPS). The power configuration is designed to maintain this communication system during a total station blackout.

5.5.4 Intra-Plant Maintenance Telephone System

The intra-plant maintenance telephone system is a part of the PBX system and consists of telephone jacks into which telephone sets may be plugged. The telephone jacks are in various plant locations (predominantly in areas of high maintenance activity) and have the effect of expanding the PBX capability.

5.5.5 EOF/JPIC Private Branch Exchange (PBX)

A dedicated PBX is installed at the Coatesville EOF/JPIC. This switch will control telephone communications in and between the facility, other Exelon locations, and non-Exelon locations. In the event of a PBX failure, outside dial capability is available through trunk lines from the Coatesville Service Building. The EOF/JPIC PBX switch is powered by a source that is backed by a 4-hour uninterruptible power supply and an emergency diesel generator. The UPS is designed to allow sufficient time to bridge any power interruption caused by switching to diesel-supplied power.

5.5.6 Data and Facsimile Transmission Lines

Various data lines are provided to interface computer systems and facsimile machines located at Limerick, Peach Bottom, and EOF/JPIC.

5.5.7 Trunk Lines

Incoming and outgoing central office trunk lines are provided from the local telephone company. These lines are used to access the Public Switched Telephone Network.

5.5.8 Tie Lines

Two-way tie lines are provided between LGS, PBAPS, Corporate Main Office, Exelon Nuclear, and the EOF. The tie lines are available to emergency personnel to allow communications between the sites and Exelon Nuclear locations supporting the emergency.

Company tie lines are utilized to route NRC communications (e.g., ENS, HPN and counterpart circuits) from between Exelon Nuclear emergency response facilities for Limerick Generating Station.

5.5.9 Emergency PBX T-1 Circuit Lines

Two dedicated T-1 circuits between the Limerick Generating Station and Peach Bottom Atomic Power Station emergency PBX telephone systems are provided for calls within and outside the Exelon voice network. This linkage also allows the continuation of 2-way commercial telephone service in the event that one of the two main commercial telephone system PBX's becomes inoperable or unavailable.

5.5.10 Microwave Tie Lines

Dedicated microwave tie lines exist between LGS, PBAPS, Main Office, Exelon Nuclear, and the EOF/JPIC. The microwave system is backed up by eight hours of battery. In addition, communication lines exist between LGS, PBAPS, Main Office, the Nuclear Group Headquarters, and the EOF/JPIC.

5.5.11 Radio Equipment

A fixed base radio system with multiple channels provides primary/backup outside communication capability as shown in Figure LGS 5-1, "Emergency Radio Links".

A separate group of fixed radio channels provides primary/backup communications between in-plant user groups. These channels function through a distributed antenna system located on-site to ensure proper coverage of the area.

The fixed base radio repeaters, antenna system, and radio consoles are powered from a variety of emergency AC buses (diesel backup) and dedicated alternate battery supplies.

The LGS radio system was designed to maintain communications between facilities as described in the Fire Protection Evaluation Report, UFSAR.

5.5.12 Evacuation Alarm System

The Evacuation Alarm System consists of a siren tone generator, PA system speakers, a roof siren, and evacuation alarm beacons. The siren tone generator injects an audible evacuation alarm in the PA system, which is broadcast over the PA system speakers. The evacuation alarm beacons provide an audible and visual alarm through two mechanical sirens and flashing red beacon on each beacon unit. The evacuation alarm beacons are installed in all high noise areas of the plant and in areas not covered by the PA system. A selector switch in the Control Room manually initiates the evacuation alarm.

Table LGS 5-1: Emergency Supplies and Equipment

The following is a listing of typical equipment available for use during emergencies. While specific equipment designations and items may be subject to change, equivalent emergency activity capabilities will be maintained. Procedures define the specific locations, types, and amounts of equipment for emergency use and define requirements for applicable surveillance, testing, maintenance, and inventory activities to ensure that the equipment is in a state of readiness.

I. PROTECTIVE	LOCATIONS STORED OR AVAILABLE
Anti-C Clothing	2, 5
Dosimeters	2, 3, 6
Dosimeter Charging Unit	2, 3, 6
Thermoluminescent Dosimeters	2
Respirator/Filters	2
Self-Contained Breathing Apparatus	1, 2, 3
Radiation Signs, Rope & Tape	2, 3, 7, 8
Potassium Iodide	3

NOTE: Equipment from the above list utilized by field survey personnel is stored in Field Survey Kits in the Site Management Building.

II. RADIATION MONITORING	LOCATIONS STORED OR AVAILABLE
Air Sampler2, 3, 5	
G. M. Counter	2, 3, 5, 6, 7, 8
Ion Chamber	2, 3, 5
Frisker	2, 3, 5, 6, 7, 8, 10
Radiation Survey Forms	2, 3, 5, 7, 8, 9, 10
Smears	2, 3, 5, 7, 8, 9, 10
CAM	3
Area Monitors	2, 3, 10

NOTE: Equipment from the above list utilized by field survey personnel is stored in Field Survey Kits in the Site Management Building.

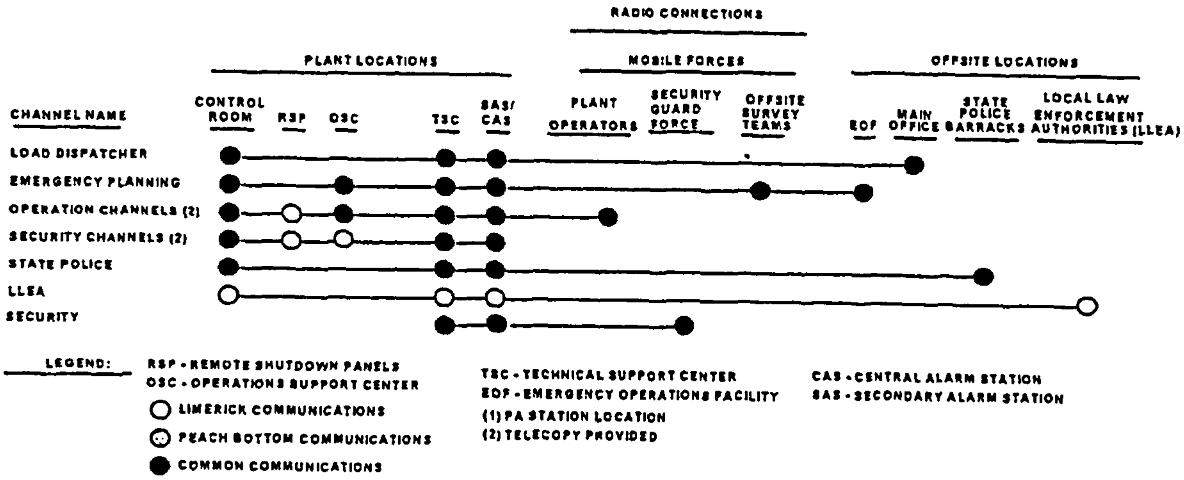
Table LGS 5-1: Emergency Supplies and Equipment (Cont'd)

III.	SEARCH AND RESCUE	LOCATIONS STORED OR AVAILABLE
	Flashlight	10
	Blanket	10
	Stretcher	10
	Rope	10
	EMT Kits	10
IV.	DECISION AIDS	LOCATIONS STORED OR AVAILABLE
	Nuclear Emergency Plan	1, 3, 4, 11
	LGS EP Procedures	1, 2, 3, 4, 11
	EP-Corporate Procedures	3, 4, 11
	Maps & Overlays	3, 4
	Prints (Aperture Cards)	3, 4
	Drawings (Aperture Cards)	3, 4
	P&ID	1, 2, 3, 4
V.	COMMUNICATIONS	LOCATIONS STORED OR AVAILABLE
	Base Stations	1, 2, 3, 4,
	Mobile Radios	1, 2, 3
VI.	DECONTAMINATION	LOCATIONS STORED OR AVAILABLE
	Soap/Detergent	3, 6, 7, 8, 9
	Brushes or Sponges	3, 6, 7, 8, 9

LOCATION KEY

1. Control Room Area
2. Operations Support Center/Health Physics Office
3. Technical Support Center
4. Emergency Operations Facility
5. Field Monitoring Kits
6. Support Hospitals
7. Cromby Station Kit
8. Limerick Airport Kit
9. Decontamination Room
10. Strategically located throughout Station
11. Joint Public Information Center

Figure LGS 5-1: Emergency Radio Links



APPENDIX 1: NUREG-0654 CROSS-REFERENCE

Annex Section	NUREG-0654
1.0	Part I, Section A
1.1	Part I, Section B
1.2	Part I, Section D
1.3	Part I, Section F
Table LGS 1-1	Part I, Section F
Figure LGS 1-1	Part II, Section J.10
Figure LGS 1-2	Part II, Section J.11
2.0	Part II, Section B.1
2.1	Part II, Section B.5
2.2	Part II, Section A.3
2.3	Part II, Section C.3
2.4	Part II, Section C.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.f
4.3.1	Part II, Section E.6
4.3.2	Part II, Section J.8
4.3.3	Part II, Section J.6.c
4.3.4a	Part II, Section G.1 & 2
4.3.4b	Part II, Section G.5
4.3.5	Part II, Section J.7
4.4.1	Part II, Sections I.2 & 3.a
4.4.2	Part II, Section J.5
4.4.3	Part II, Section J.3
Figure LGS 4-1	Part II, Section J.10.1 & Appendix 4
Figure LGS 4-2	Part II, Section I. J.7
Figure LGS 4-3	Part II, Section I. J.4
5.1	Part II, Section H.1-2, & G.3.a
5.2.1	Part II, Section H.5.a & 8
5.2.2	Part II, Section H.5.b, H.6.c &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, I.9-10
5.2.6	Part II, Section H.5.a & 6.a
5.3	Part II, Section H.9-10
5.4	Part II, Section L.1 & 2
5.5	Part II, Section F.1
Table LGS 5-1	Part II, Section H.11
Figure LGS 5-1	Part II, Section F.1.d
Appendix 1	Part II, Section P.8
Appendix 2	Part II, Section J.8

APPENDIX 2: SITE-SPECIFIC LETTERS OF AGREEMENT

The following is a listing of letters of agreement and contracts specific to emergency response activities in support of Limerick Generating Station. Letters of agreement and contracts common to the multiple Exelon Nuclear stations are listed under Appendix 3 to the Exelon Nuclear Standardized Radiological Emergency Plan.

- Pennsylvania Emergency Management Agency (Letter on File)

  NOTE: Documentation of agreement for Berks, Chester and Montgomery Counties are contained as part agreement with PEMA.

- Pennsylvania Department of Environmental Resources / Bureau of Radiation Protection (Letter on File)
- Pennsylvania State Police#
- Porter Consultants, Inc. (P.O.)
- Goodwill Ambulance Service (Letter on File)
- Linfield Fire Company (Letter on File)
- Limerick Fire Company (Letter on File)
- Montgomery County Communications (Letter on File)
- Montgomery Hospital (Letter on File)
- Pottstown Memorial Medical Center (Letter on File)
- Trappe Fire Company Ambulance (Letter on File)
- Affidavit, PECo Bus Driver Pool* [T04510]

Agreements with State and local law enforcement agencies maintained by Station Security under the Nuclear Station Security Plan

* Refers to "Affidavit of Joseph W. Gallagher (VP, PECo Nuclear Operations) in Response to the Request in ALAB-857 for Confirmation of the Status of Licensee's Volunteer Employee Bus Driver Pool", dated January 12, 1987, to augment bus driver staffs for Spring-Ford Area School District and Owen J. Roberts School District. (NOTE: Bus driver pool was reduced from 200 to 100 under separate 10 CFR 50.54(q) and 10 CFR 50.59 evaluations approved by the LGS Plant Manager on 04/11/96 (PORC Mtg. #96-034, 04/04/96).

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, with the exception of Oyster Creek Generating Station (OCGS). 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 ≥ 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

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 increase in gamma detection.
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. START UP THE CDAM PROGRAM

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 **C:\CDAM\BWR 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 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.

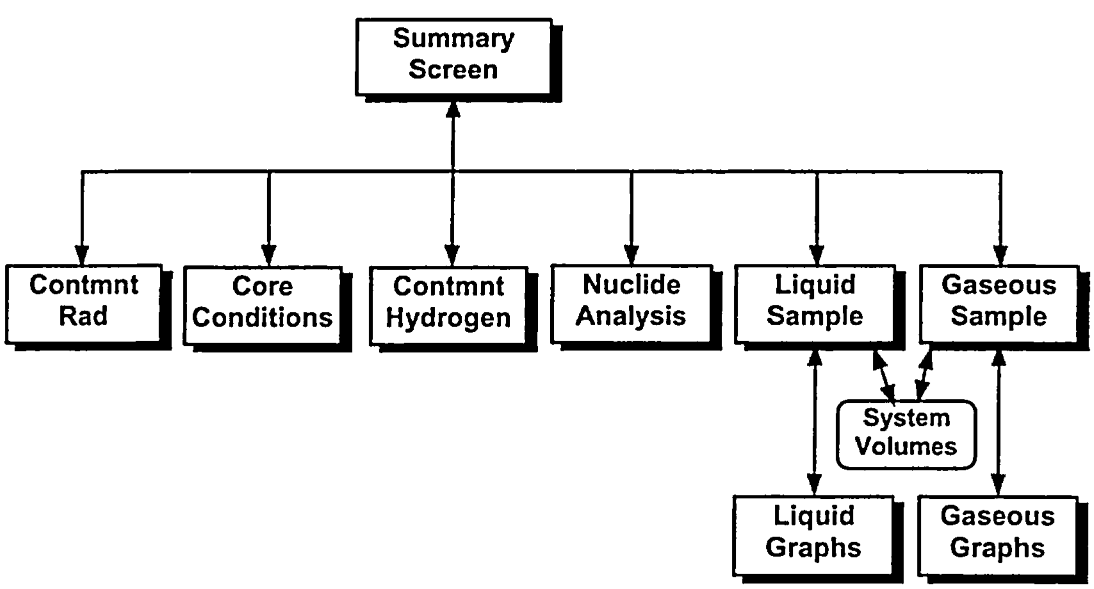
2. **UPDATE** the "properties" of the file by deselecting write protection.

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

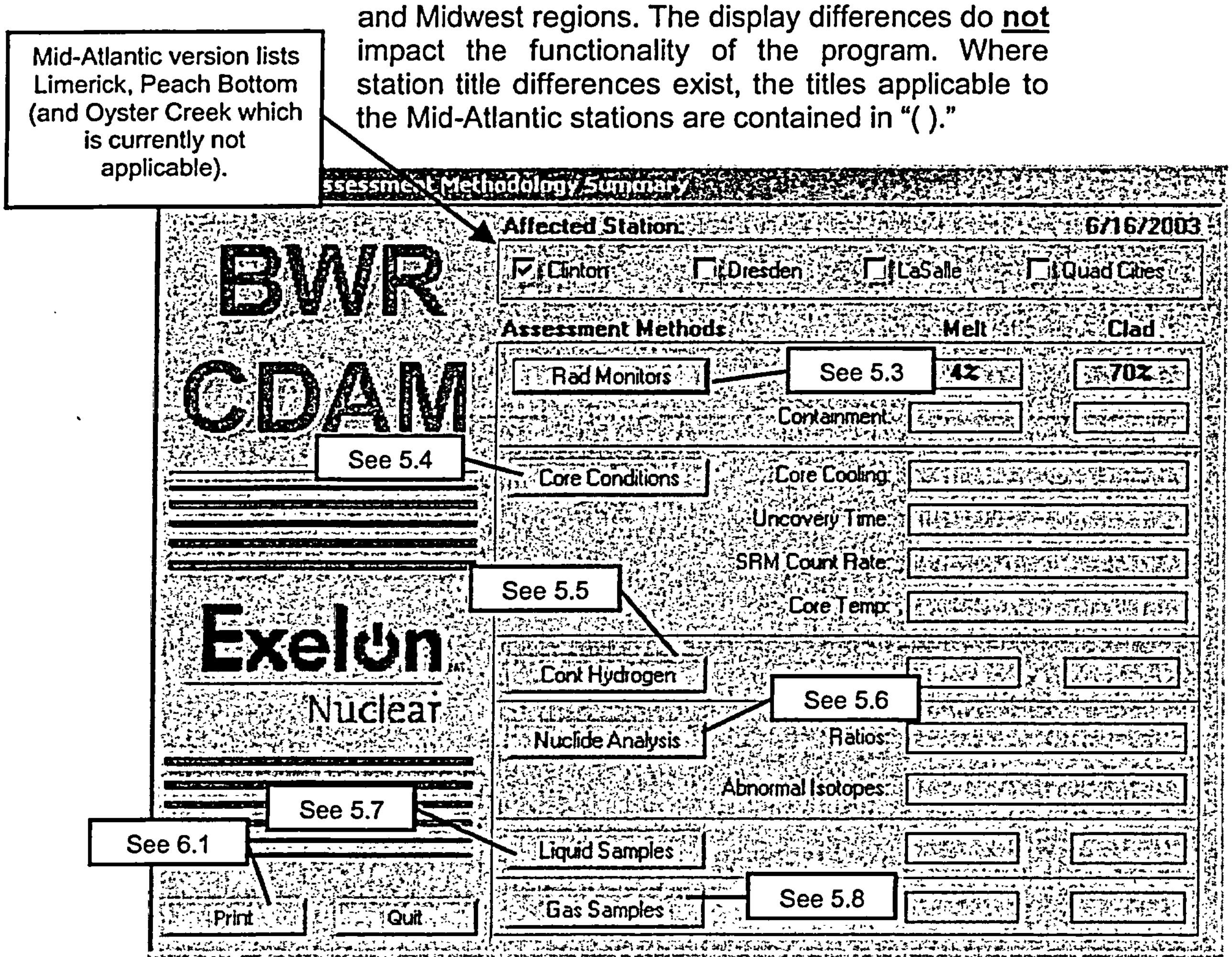


Attachment 1
BWR CDAM User Guide
Page 4 of 20

5. **PROGRAM SCREENS AND INPUTS**

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 not impact the functionality of the program. Where station title differences exist, the titles applicable to the Mid-Atlantic stations are contained in "( )."



Attachment 1
BWR CDAM User Guide
Page 5 of 20

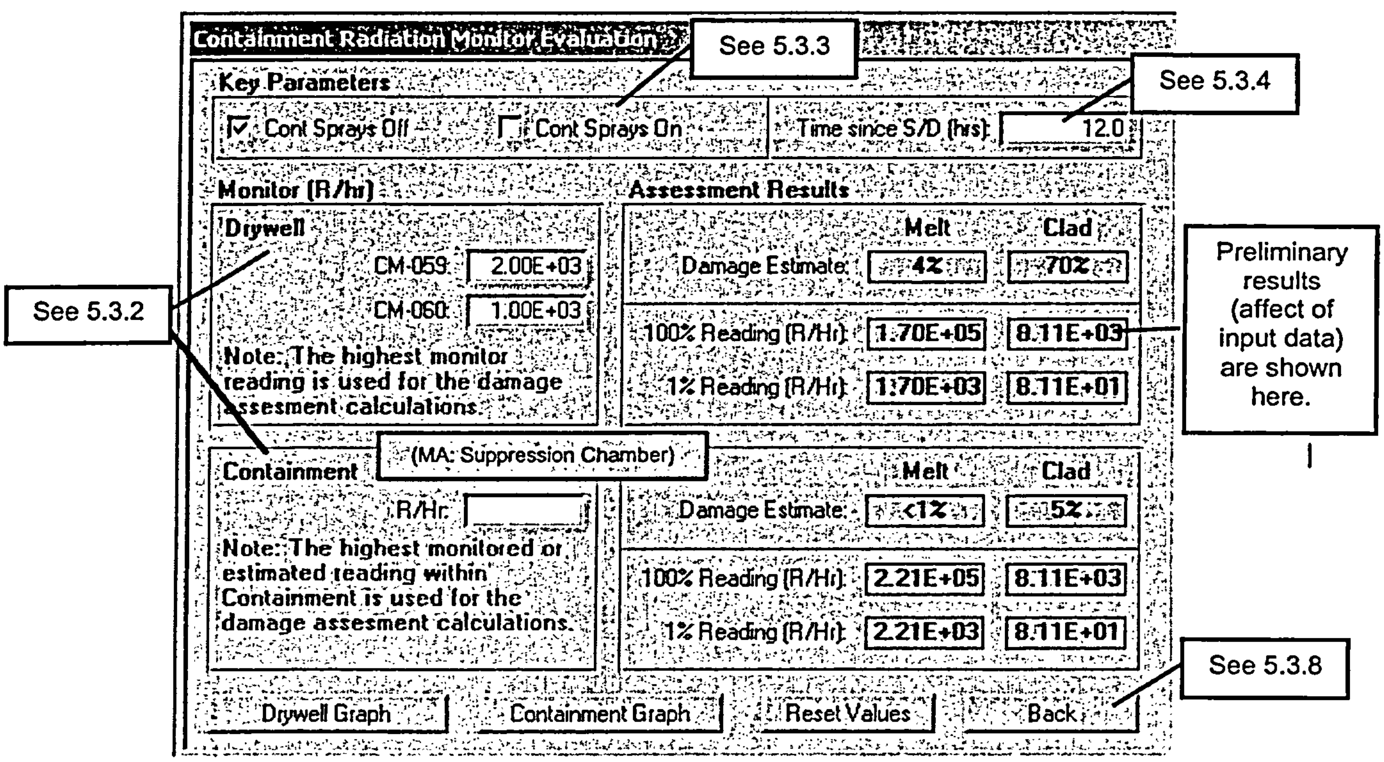
CAUTION
Selecting an "Affected Station" resets all inputs to default values.

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:



Attachment 1
BWR CDAM User Guide
Page 6 of 20

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. **If** the Drywell/Containment Spray system was operated for the majority of the time since the estimated time of the onset of core damage **then** choose "Drywell Spray On."

2. **If** the Drywell/Containment Spray system was <u>not</u> operated or only operated briefly (e.g., <10% of time since the estimated time of the onset of core damage) **then** choose "Drywell Spray Off."

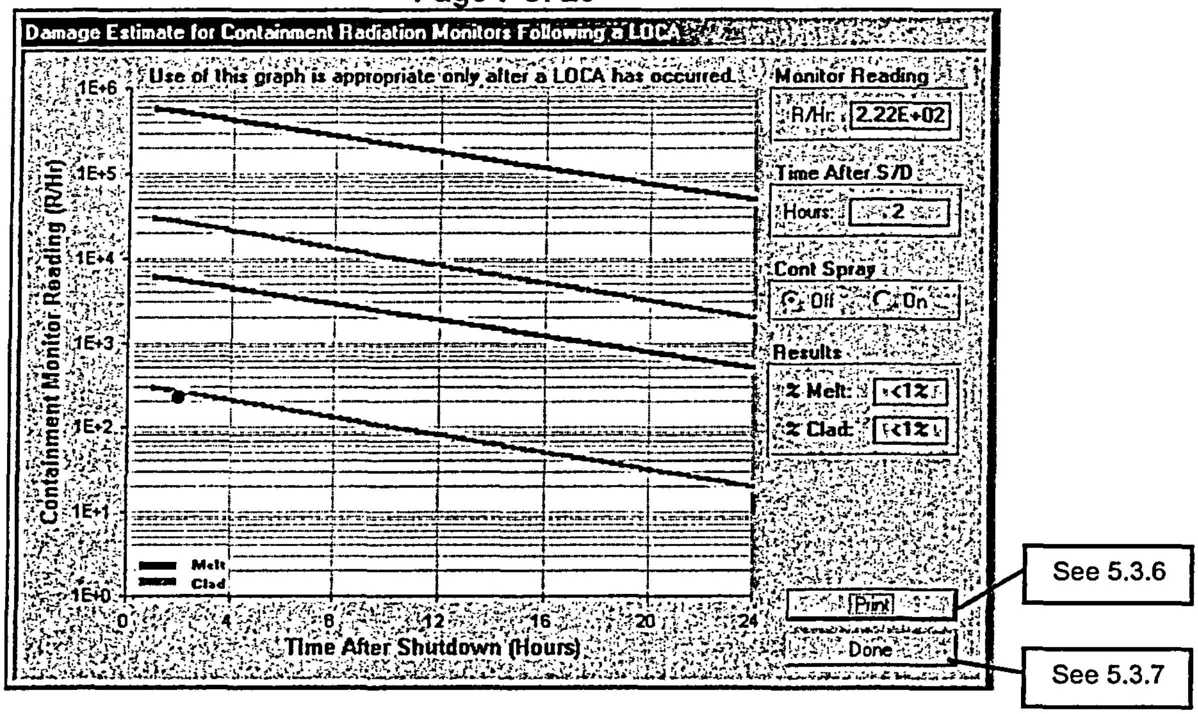
5.3.4. **ENTER** the time after reactor shutdown, which corresponds the time the containment radiation reading was taken. Value must be between 1 hour and 24 hours after shutdown, which corresponds to the time period in which this method is considered effective.

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.)

Attachment 1
BWR CDAM User Guide
Page 7 of 20



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.

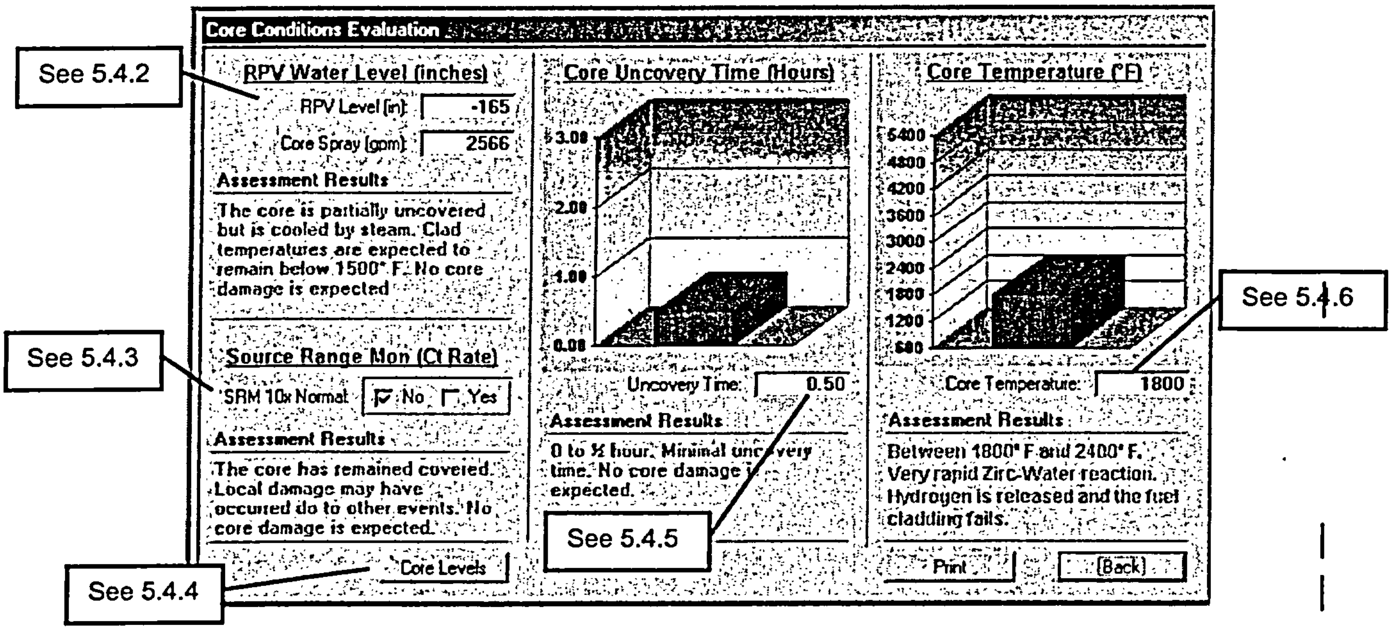
5.4. Core Conditions Methods

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.)

Attachment 1
BWR CDAM User Guide
Page 8 of 20



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

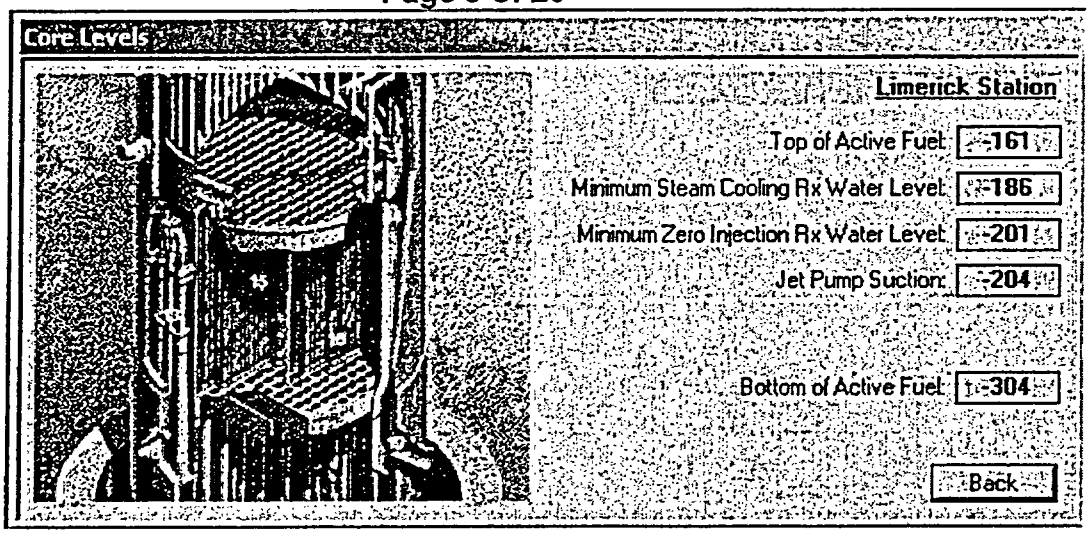
NOTE: Steps 5.4.3 through 5.4.6 are based on inputs from Reactor Operators, TSC Staff and other engineering personnel (including outside sources such as General Electric personnel).

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 10 times 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.)

Attachment 1
BWR CDAM User Guide
Page 9 of 20



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.

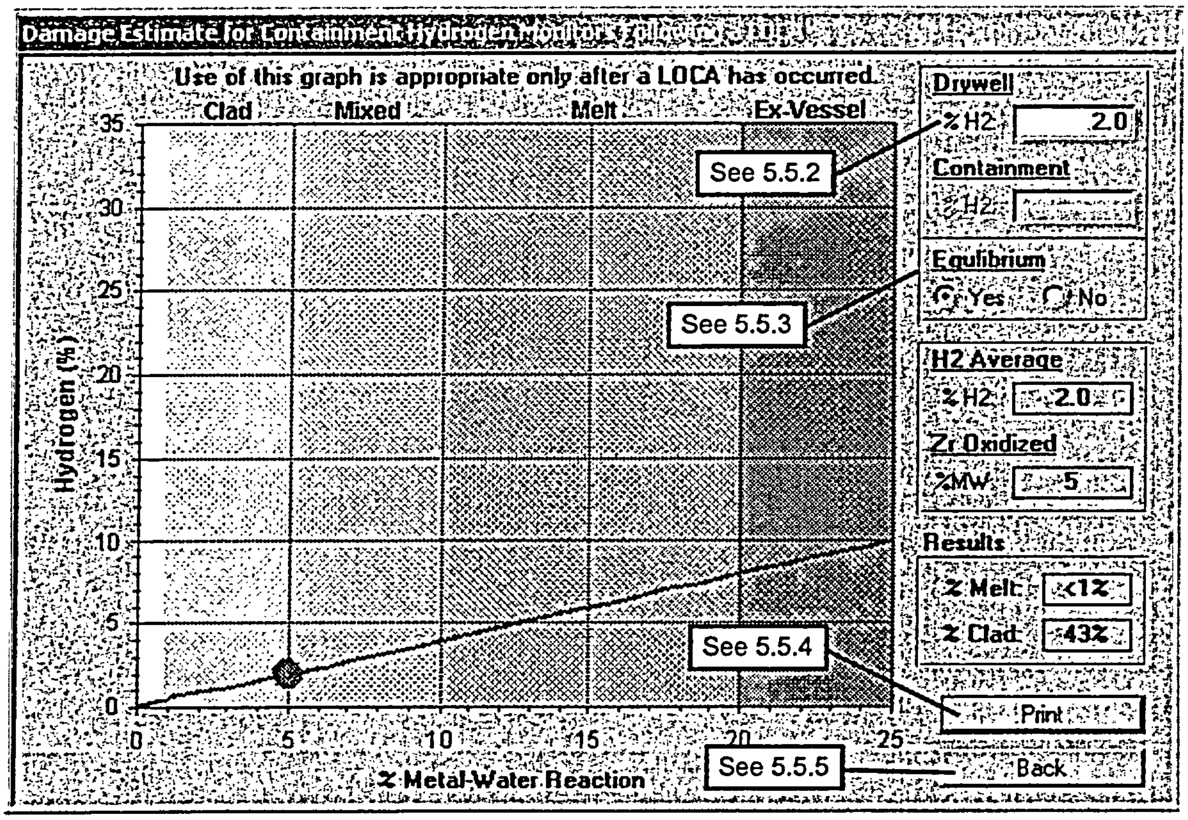
5.5. Containment Hydrogen Evaluations

CAUTION

This CDAM assumes no ignitor operation. Ignitor use limits containment hydrogen concentration affecting the reliability of this method.

Attachment 1
BWR CDAM User Guide
Page 10 of 20

5.5.1. **PRESS** the "Cont Hydrogen" button on the Summary Screen to open the following form:



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:

1. **If** Containment and Suppression Chamber monitors read the same or only atmospheres are assumed equalized, **then SELECT** "Yes" for equilibrium.

2. **If** containment and suppression chamber atmospheres are not in equilibrium or only containment H2 reading is available, **then SELECT** "No" for equilibrium.

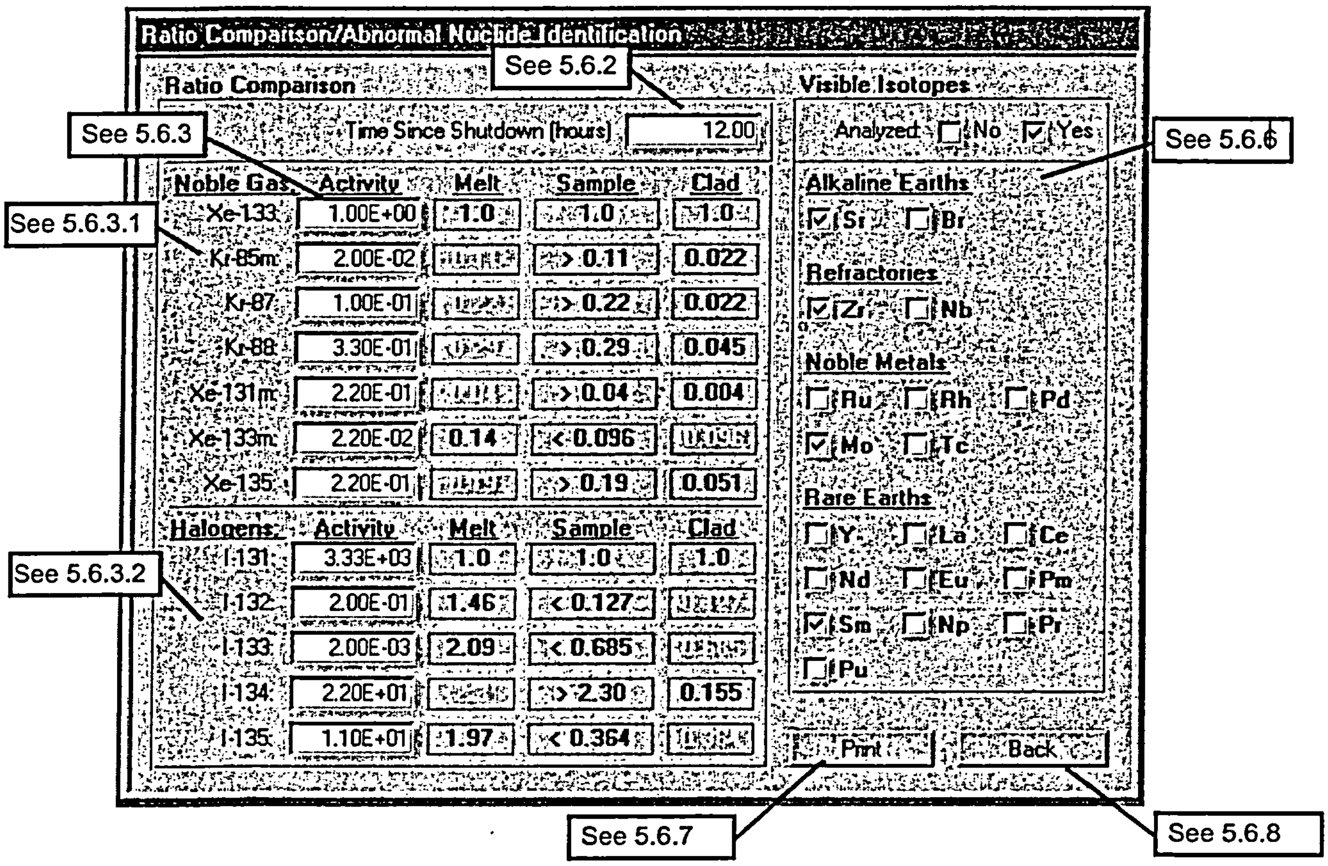
5.5.4. **PRESS** the "Print" button to print a report of inputs and results of core level methods of core damage assessment.

Attachment 1
BWR CDAM User Guide
Page 11 of 20

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:



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.

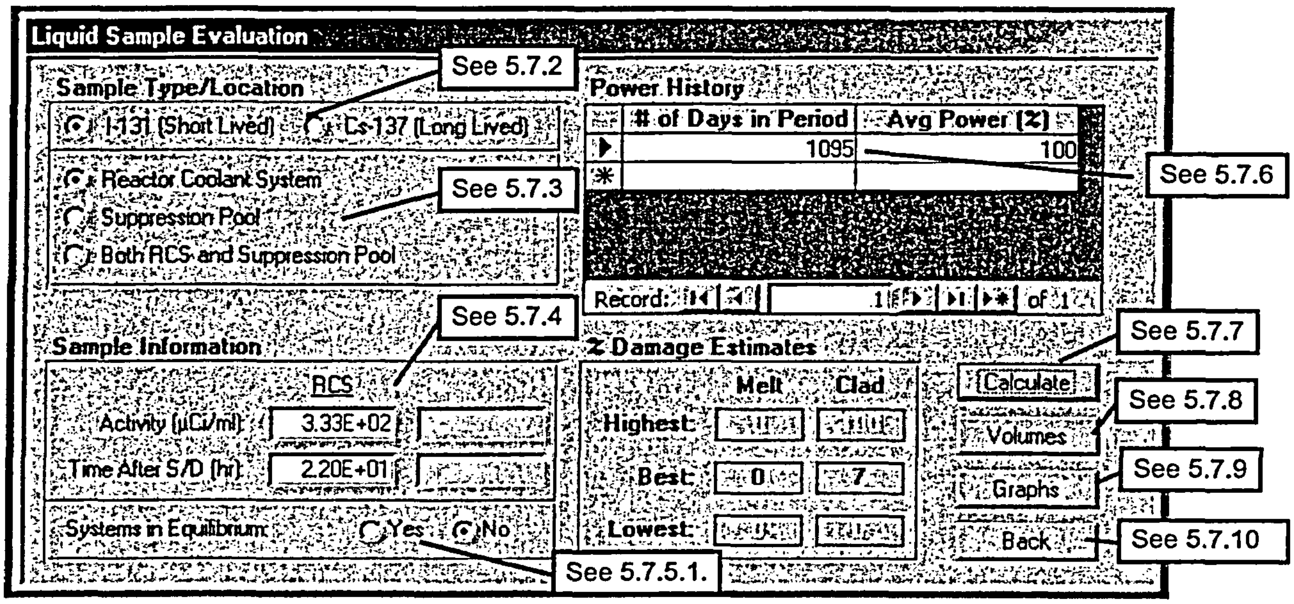
5.6.6. **If** abnormal levels of rare isotopes are present **then** check "Yes" **and** check which isotopes are present.

5.6.7. **PRESS** the "Print" button to print a report of inputs and results of core level methods of core damage assessment.

5.6.8. **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:



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:

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.

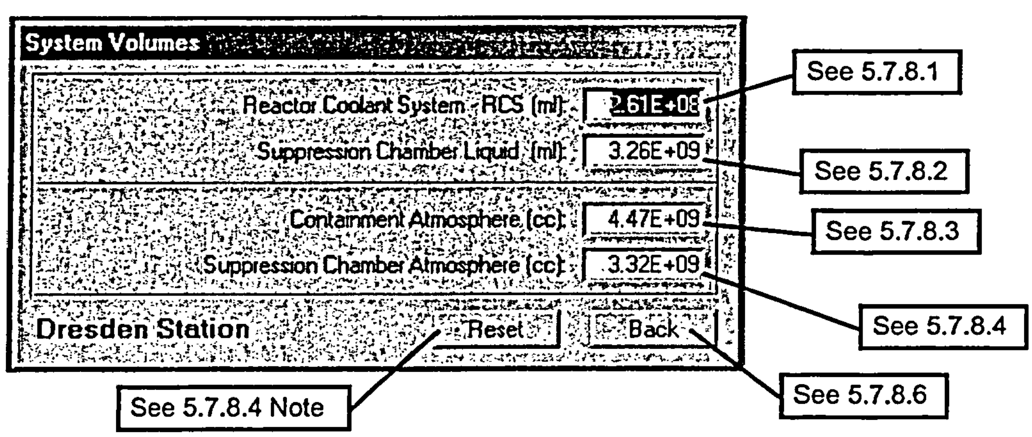
1. For short-lived isotopes, **EXTEND** Power History at least 30 days.

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:



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.

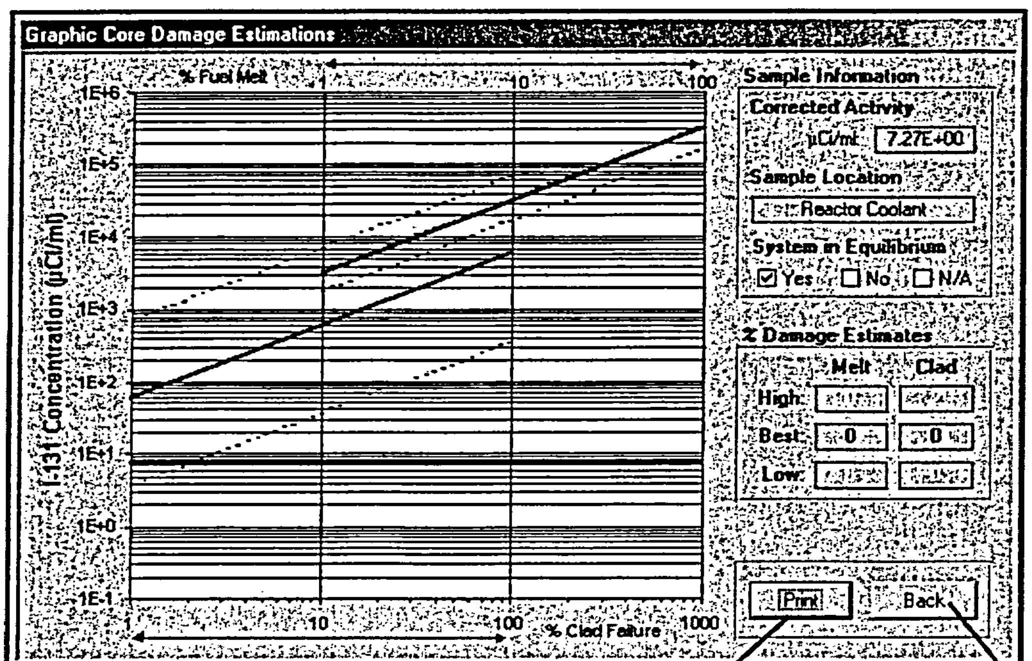
Attachment 1
BWR CDAM User Guide
Page 14 of 20

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:



See 5.7.9.1

See 5.7.9.2

(See Note on next page.)

Attachment 1
BWR CDAM User Guide
Page 15 of 20

NOTE: Graph on previous page shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating entered corrected sample activity.

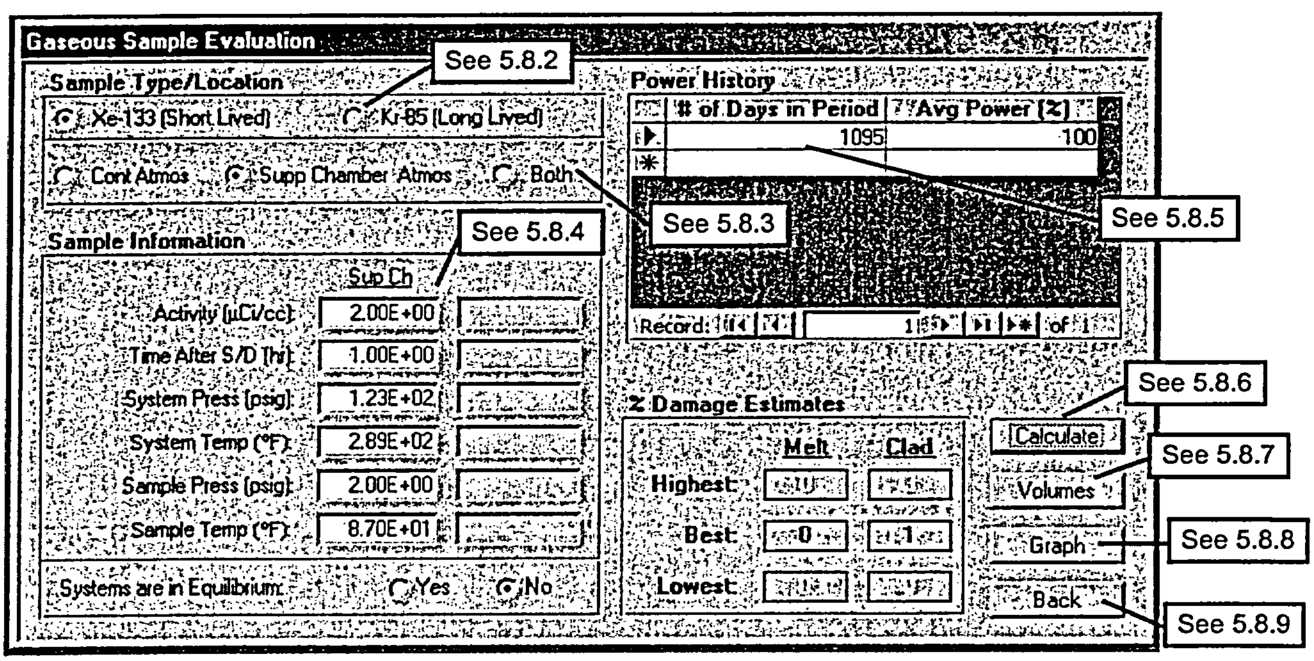
1. **PRESS** the "Print" button to print a graph and summary of inputs.

2. **PRESS** the "Back" button to go back to liquid or gaseous form which called this form.

5.7.10. **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:



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. **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.

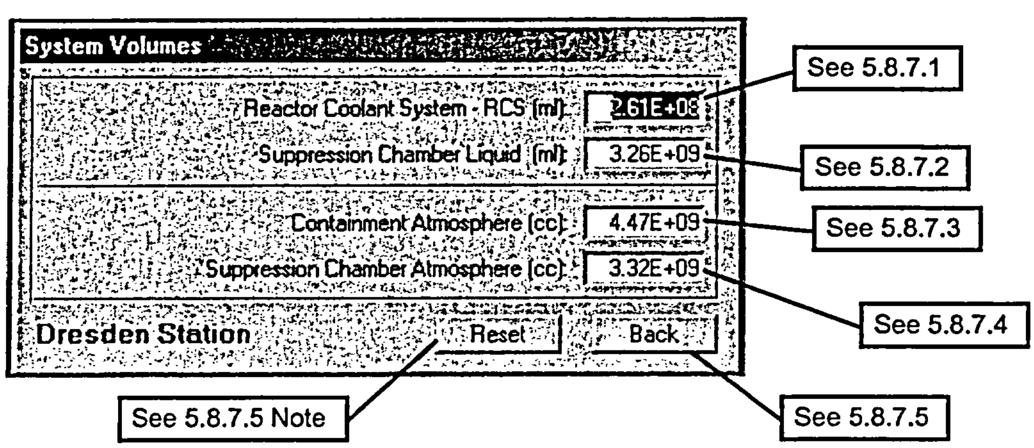
1. For short-lived isotopes, **EXTEND** Power History at least 30 days.

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.

5.8.6. Once all data has been entered **PRESS** the "Calculate" button to display the % Damage Estimates.

5.8.7. **PRESS** the "Volumes" button to display the following screen (same as 5.7.8):



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.

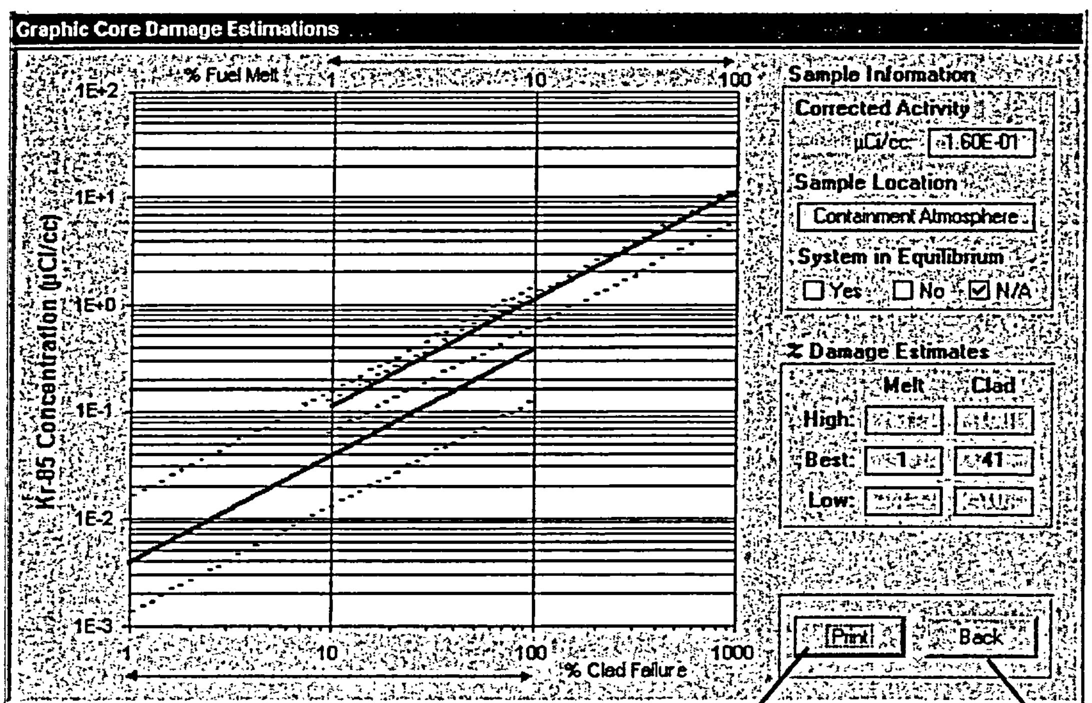
3. Program enters default Containment free air volume which user may change based on conditions at time of sample. Unless there has been significant flooding of drywell this value will not change.

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.

5.8.8. **PRESS** the "Graph" button to display the following screen:



See 5.8.8.1

See 5.8.8.2

NOTE: Graph shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating entered.

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.

5.8.9. **PRESS** the "Back" button to return to the Summary Screen.

6. <u>CORE DAMAGE SUMMARY REPORT</u>

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)..

(Sample report on next page.)

CDAM Method: **Core Damage Summary**

Station: ☐ Clinton ☐ Dresden ☐ LaSalle ☒ Quad Cities

Assessment Methods:

		Melt	Clad
Containment Radiation Monitors*	Containment:	29%	79%
	Suppersion Chamber:	<1%	23%
Core Conditions	Core Cooling:	Clad Damage	
	Core Uncovery Time:	No Core Damage	
	SRM Count Rate:	No Core Damage	
	Core Temp:	Clad Failure	
Containment Hydrogen*		<1	20.8
Sample Analysis	Ratios:	Fuel Melt	
	Abnormal Isotopes:	6 of 19 Present	
	RCS: Liquid Samples:	0%	0%
	Chamber: Gas Samples:	23%	100%

* These methods should NOT be used for qualitative or quantitative assessment except in the case of a LOCA.

Analyst's Estimate:

☐ No Core Damage ☐ Cladding Failure ☐ Fuel Melt Amount:

NRC Core Condition Category:

Degree of Degradation	Minor (<10%)	Intermediate (10%-50%)	Major (>50%)
No Core Damage	1	1	1
Cladding Failure	2	3	4
Fuel Overheat	5	6	7
Fuel Melt	8	9	10

Generated By:

Name: ______________________ Date: 12/05/02 Time: 8:29 AM

Attachment 1
BWR CDAM User Guide
Page 20 of 20

6.3. The Individual tasked with assessing core damage shall then **ANALYZE** the report to determine best estimate of type and amount of damage.

NOTE: The CDAM program does not use the Fuel Overheat Condition Category

6.4. Based on estimated type and amount of damage and following table (table also printed on summary report) **ASSIGN** NRC Core Condition Category (1-4 or 8 -10).

NRC Core Condition Categories

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. <u>QUITING, OR EXITING, THE PROGRAM</u>

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.

ASSEMBLY AND SITE EVACUATION

1. PURPOSE

1.1 This procedure provides the necessary guidance for the Accountability of personnel within the Station Protected Area by means of assembly in designated areas and the Evacuation of non-essential personnel from the Site.

Accountability Guidelines	**REFER** to Section 4.1
Initiation of Personnel Accountability • Station Emergency Director	**REFER** to Section 4.2
Remote Assembly Areas • TSC Radiation Protection Manager • Vehicle and Evacuee Control Group Leader • Vehicle and Evacuee Control Group Members	**REFER** to Section 4.3

2. TERMS AND DEFINITIONS

2.1 Accountability - Accountability is the process of verifying the location of personnel who are inside the Protected Area. That is, any unaccounted for person that has keyed into the Protected Area will be identified as missing. Accountability is accomplished through the on-site assembly of essential (on-site and ERO personnel) in designated emergency response facilities and evacuation from the site of non-essential personnel.

2.2 Assembly –Onsite assembly is used to account for on-shift and ERO personnel within the Protected Area by assembling in the emergency response facilities, and occurs in conjunction with evacuation of non-essential personnel from the site.

2.3 Evacuation - A site evacuation of non-essential personnel is performed as part of Accountability actions. Evacuation can involve the movement of large numbers of personnel outside of the Protected Area by keying out of the turnstiles. Evacuation may warrant station egress control by Security. Specific instructions may be provided to personnel leaving the Protected Area. Evacuees may be directed to an offsite assembly area for monitoring and decontamination, or sent home. Other situations that involve the evacuation of personnel from occupied localized onsite areas are controlled on a case-by-case basis.

2.4 Owner Controlled Area - Company owned property on which a nuclear station is located and may include Exelon Nuclear / AmerGen leased-lands adjacent to that nuclear station.

3. RESPONSIBILITIES

3.1 The *Station Emergency Director* is responsible for implementing personnel accountability and/or the evacuation of non-essential personnel

3.2 The *Shift Manager, as Shift Emergency Director*, will assume the responsibilities of the Station Emergency Director prior to TSC activation and the transfer responsibility.

3.3 The *TSC Security Coordinator* is responsible for coordinating activities between the TSC and the Station Security force.

4. MAIN BODY

4.1 Personnel Accountability Guidelines

4.1.1 **Accountability** is required to be conducted at a Site Area or General Emergency, if not previously initiated and maintained. Accountability may be conducted at the Alert level at the discretion of the Station Emergency Director, or Shift Manager (Shift Emergency Director) prior to TSC activation.

1. Accountability and evacuation of non-essential site personnel should also be considered when a security-related Unusual Event or Alert has been declared.
2. Accountability shall be initiated expeditiously, but only after evaluating the need for offsite assembly and appropriate evacuation route based on radiological release and meteorological conditions.
3. Once initiated, accountability is required to be completed (i.e., the names of any missing persons identified by security and the number of missing provided to the Station Emergency Director) within 30 minutes of initial PA announcement for site evacuation.

4.1.2 A **Site Evacuation** may be delayed if the health and safety of the plant personnel may be in jeopardy, such as severe weather or due to a security-related Unusual Event or Alert.

4.1.3 **Non-essential personnel evacuating the site**, contractors and visitors, shall report to a Remote (Off-Site) Assembly Area, if designated, for radiological monitoring and decontamination as warranted.

4.1.4 **Protected Area access** is halted during personnel accountability, except for the following:

- Direct approval from the TSC Security Coordinator or Shift Manager
- Key ("on-call") ERO responders requiring access to staff the Operations Support Center (OSC) and/or Technical Support Center (TSC)

4.1.5 **Accountability**, once achieved, will be maintained by restricting Protected Area access and controlling/tracking the movement of on-shift personnel or ERO personnel on site in or out of their respective emergency response facility.

4.2 Initiation of Personnel Accountability

4.2.1 The *Station Emergency Director* shall:

1. In coordination with the Security Coordinator and Radiation Protection Manager, **DETERMINE** the following prior to initiating accountability based on on-going or potential security threats.

   A. **If** an on-going Security Threat exists, **then DETERMINE** whether personnel accountability should be initiated early or delayed, or remote assembly areas and/or evacuation routes designated.

   B. **DETERMINE** the need for the remote assembly of non-essential personnel to be evacuated from the site:

      1. **If** a radiological release has NOT occurred, **then DIRECT** non-essential personnel to report home and await instructions. **PROCEED** directly to Step 3.

      2. **If** a significant radiological release has or is occurring requiring further monitoring and potential decontamination of personnel and vehicles, **then DIRECT** non-essential to designated remote assembly areas. **PROCEED** directly to Step 2.

2. **DESIGNATE** a remote assembly area and evacuation route(s) for non-essential personnel, while taking into account the following:
   - Radiological release status
   - Weather conditions, including wind direction (WD)
   - Security threats

   A. Prior to designating evacuation route other than normal entrance / egress road(s), **DETERMINE** what physical security measures (i.e., jersey barriers, etc.) are in place and must be removed to allow vehicle egress from site.

      – **REFER** to Table 1 for Limerick Generating Station and Peach Bottom Atomic Power Station.

      – **REFER** to Table 2 for TMI Station.

TABLE 1

Guidelines for Selection of Evacuation Route(s) and Assembly Areas

Designated Assembly Areas:

LIMERICK (Attachment 1B)	**PEACH BOTTOM** (Attachment 1A)
• If WD (from) is 210° to 260°, **then USE** the Cromby Generating Station • If WD (from) is 261° to 209°, **then USE** the Limerick Airport (Pottstown)	• If WD (from) is North to WEST, **then USE** the North Sub-Station • If WD (from) is South to East, **then USE** Unit 1

Designated Site Evacuation Routes

LIMERICK (Attachment 1B)	**PEACH BOTTOM** (Attachment 1A)
• If WD (from) is 165° to 215°, **then USE** the Back Gate only • If WD (from) is 305° to 350°, **then USE** the Main Gate only • If ANY other WD, **then USE** both gates based on ED's judgment	• ED judgment. No designated evacuation route options.

TABLE 2 (TMI) – Refer Attachment 1C

Guidelines for Selection of Evacuation Route(s) and Assembly Areas

Wind Direction (from)	Site Evacuation Route	Offsite Assembly Area
1° to 80°	North Gate	Training Center
81° to 170°	North Gate	Training Center
171° to 240°	South Gate	Training Center
241° to 320°	North Gate	Training Center or EOF (1)
321° to 360°	North Gate	Training Center

NOTE (1): **USE** the Training Center as the offsite Assembly Area unless projected dose between the Site Boundary and 1 mile is greater than 5 mRem CDE thyroid or 1 mRem TEDE.

3. **NOTIFY** the TSC Radiation Protection Manager and Security Coordinator (or on-shift RP Supervisor and Nuclear Security Supervisor, if TSC is not yet activated) of the impending evacuation and location of assembly area(s) and designated route(s), if applicable.

   A. If evacuation route contains physical security measures (i.e., jersey barriers, etc.) prohibiting vehicle egress from site, **then DIRECT** the Security Coordinator to coordinate barrier removal with the Maintenance Manager prior to initiating Site Evacuation announcement.

B. **If** the Security Computer and/or accountability card readers are NOT operational, **then INSTRUCT** affected facility(ies) to complete Attachment 6, "Facility Accountability List (Within the Protected Area)".

4. **When** prepared, **PERFORM** the following using announcement instructions in Attachment 2:

   A. **COMPLETE** appropriate announcement instructions:

Peach Bottom Station....................................	**REFER** to Attachment 2A
Limerick Station...	**REFER** to Attachment 2B
TMI Station..	**REFER** to Attachment 2C

   B. **DIRECT** the TSC Director to initiate a Site Evacuation per Attachment 2 over the Station PA, or **REQUEST** the Shift Manager (thru the Operations Manager) to make announcements from Control Room, using completed Attachment 2.

5. **VERIFY** that the TSC Security Coordinator has performed the following:

   A. Sweeps have been initiated to ensure that non-essential personnel have been evacuated from the Owner Controlled Area and that access is being controlled.

   – **REFER** to Attachment 3 for an illustration or listing of potentially occupied areas outside the Protected Area within of the Owner Controlled Area (OCA).

Peach Bottom Station..........	**REFER** to Attachment 3A
Limerick Station..................	**REFER** to Attachment 3B
TMI Station........................	**REFER** to Attachment 3C

   1. **If** sufficient Security personnel are NOT available, **then REQUEST** support from the TSC Maintenance Manager to use OSC personnel to perform OCA notifications.

   B. Appropriate local law enforcement agencies (LLEAs) have been contacted an informed of site evacuation and support requested, if needed, to perform sweeps of and control access to the Owner Controlled Area and to provide traffic control at Remote Assembly Area, as required.

6. **DIRECT** TSC Director to communicate to State and county agencies over the NARS Circuit that a site evacuation has been initiated.

7. **OBTAIN** the number of unaccounted for personnel from the TSC Security Coordinator within 30 minutes of initiating Accountability.

8. **VERIFY** that appropriate actions have been implemented to ensure continuous Accountability of all onsite personnel through limiting ingress and egress for the Protected Area.

9. **PROVIDE** further direction to assembled ERO members <u>NOT</u> considered Essential to support imminent event mitigation activities. Consideration shall be given to relief staffing in support of continuing accident and recovery activities.

4.3 <u>Offsite Assembly Areas</u>

4.3.1 The *TSC Radiation Protection Manager* shall:

1. **DETERMINE** available RP resources onsite through the TSC Maintenance Manager, or directly with the OSC RP Group Lead if present.

   A. If sufficient RP staff IS present onsite, **then REQUEST** that 3 RP personnel be designated as the Vehicle and Evacuee Control Team, using an OSC Team Request Form (EP-AA-112-200, Attachment 2).

   B. If adequate RP staff is NOT present onsite, **then INITIATE** callouts or contact the EOF Radiation Protection Manager for additional resources from unaffected stations. When responding from offsite, RP personnel should be directed to assembly area.

2. **APPOINT** an RP Supervisor or Technician as Vehicle and Evacuee Control Group Leader, and **PROVIDE** a briefing to team members.

   A. If responding from offsite, **then PERFORM** this action upon team's arrival at assembly area.

   NOTE: Alpha contamination should be considered whenever there is reactor fuel degradation.

3. **VERIFY** with the TSC Security Coordinator that local law enforcement has been contacted to provide traffic control at designated remote assembly area.

4. **COORDINATE** with the TSC Logistics Coordinator and the EOF to arrange for the transportation of personnel whose vehicles cannot be immediately decontaminated.

5. If an evacuation is recommended or implemented by offsite authorities, which encompasses the remote assembly area, **then OBTAIN** approval from the Station Emergency Director and **PERFORM** the following:

   A. **REQUEST** via the EOF Radiation Protection Manager that the local a County Emergency Operations Center (EOC) be notified to designate a reception center for the relocation of personnel from the remote assembly area(s).

TMI
Williams Valley High School has been pre-designated. **REFER** to Attachment 1, Table 1-3 for directions.

   B. **DIRECT** non-essential personnel, who have not yet been monitored to the designated County reception center

4.3.2 The *Vehicle and Evacuee Control Group Leader* shall:

1. **OBTAIN** remote assembly area kits.

   A. **If** reporting from offsite, **then REQUEST** that the TSC RP Manager arrange for delivery of kits or survey instrumentation, if not prestaged at remote assembly area.

2. **OBTAIN** portable radios for communication with the TSC and group members.

3. Upon arrival at the Remote Assembly Area, **ESTABLISH** communications with the TSC RP Manager via portable radio or available telephone and **REQUEST** the following:

   A. Instructions on whether assembled non-essential personnel should be held at remote assembly areas, relocated or released. Unless instructed otherwise, non-essential personnel shall be released to their homes and await further instructions, once monitoring is completed.

   B. Assistance with traffic control if local law enforcement has not yet arrived at assembly area

4. **INSTRUCT** the team members to implement Section 4.3.3 and **SET UP** for the monitoring and possible decontamination of non-essential personnel.

5. **LOG** personnel arriving at the remote assembly area using Attachment 4.

6. **CONTACT** the Radiation Protection Manager to arrange for further decontamination of personnel at an unaffected station, as necessary.

NOTE: Personnel at the remote assembly area who are identified as being contaminated should not be returned to the site for decontamination if a release is ongoing or expected to occur.

7. When all personnel and vehicles have been monitored, and the team is no longer needed for decontamination, **INSTRUCT** them to survey the monitoring areas, themselves, their equipment and vehicle, and return to the TSC, or to the EOF if the TSC is inaccessible.

8. Upon return of the group, **COLLECT** personnel and vehicle survey and decontamination records and **FORWARD** them to the TSC RP Manager.

9. Using completed Vehicle Survey and Decontamination Reports (Attachment 5), **CONTACT** owners and inform them where and when they may pick up their vehicle.

10. **ARRANGE** for return of all contaminated material to the site for disposition in accordance with RP procedures.

4.3.3 The *Vehicle and Evacuee Control Group Members* shall:

1. Upon the direction of the Group Leader, **PROCEED** to the designated remote assembly area.

2. **OBTAIN** an Assembly Area Kit (and radiation monitoring instruments).

3. **PERFORM** an inventory of the equipment in the kits by comparing contents with the inventory lists contained in the respective kits.

    A. **PERFORM** battery, calibration and source checks on all instrumentation.

    B. **REPORT** any missing items or inoperable equipment to the Group Leader and request replacements.

4. **COORDINATE** the set-up of a Personnel Contamination Monitoring Area, a Vehicle Contamination Monitoring Area, and a Vehicle Decontamination Area.

NOTE: Background radiation is to be less than 300 cpm.

PERSONNEL/VEHICLE MONITORING

Priority should be given to the monitoring of personnel. Vehicle monitoring, if necessary, should only be performed when personnel monitoring has been completed.

5. **PERFORM** whole body frisk of each individual in accordance Exelon procedures.

   A. If individuals are found to be contaminated, **then DECONTAMINATE** using approved techniques for personnel decontamination, and COMPLETE the documentation described therein.

   B. **PLACE** all materials used to perform decontamination or other materials, which are found to be contaminated, in plastic bags for disposal and seal with radioactive material tape.

   NOTE: While stored at the assembly area, radioactive material must be stored in areas posted in accordance with RP procedures.

   C. If any individual cannot be decontaminated below the release limits specified in RP procedures, **then CONTACT** the Vehicle Evacuee and Control Group Leader for further instructions.

6. **PERFORM** vehicle surveys in accordance with RP procedures, surveying both the exterior and interior of the vehicle.

   A. If contamination levels are less than 100 cpm above background, **then RELEASE** the vehicle or equipment by checking the appropriate block on the report.

   B. If the vehicle is contaminated, **then RECORD** readings on the Vehicle Survey and Decontamination Report (Attachment 5) and **NOTE** contaminated areas and levels on the illustration.

   C. **MOVE** contaminated vehicles to designated area.

   D. **HANDLE** all materials used to perform decontamination or other materials, that are found to be contaminated, in accordance with RP procedures.

   E. **SET UP** the following vehicle areas:

      1. For vehicles that cannot be decontaminated after several attempts; and

      2. Designated clean area for checking vehicles once decontamination is completed.

7. **PERFORM** vehicle decontamination as follows upon arrival at the decontamination area using the following techniques:

   NOTE: Dry methods of decontamination shall be the method of choice. First priority of decontamination shall be to those vehicles needed to support the emergency response for which dry methods will be effective.

   A. **WIPE** down hard, smooth surfaces with dry masslin cloth.

   B. **WIPE** down vehicles with damp cloth.

8. After the initial decontamination, **RESURVEY** the vehicle and record post-decontamination survey results on the copy of Vehicle Survey and Decontamination Report accompanying the vehicle.

   A. **RELEASE** vehicles meeting the release criteria the owner. **If** owner is NOT present, **then MOVE** vehicle to clean holding area.

   B. **If** vehicle is still contaminated, **then RETURN** it to the contaminated holding area for further decontamination at a later time.

9. When decontamination operations are complete, **RETURN** completed forms to the Vehicle and Evacuee Control Group Leader and **REMAIN** on-station until released.

5. DOCUMENTATION – NONE

6. REFERENCES – NONE

7. ATTACHMENTS

7.1 Attachment 1, Remote Assembly Areas (Layouts / Directions)
- Attachment 1A, PBAPS Remote Assembly Areas (Layouts / Directions)
- Attachment 1B, LGS Remote Assembly Areas (Layouts / Directions)
- Attachment 1C, TMI Remote Assembly Areas (Layouts / Directions)

7.2 Attachment 2, Site Evacuation Alarm and Announcement Instructions
- Attachment 2A, PBAPS Site Evacuation Alarm / Announcement Instructions
- Attachment 2B, LGS Site Evacuation Alarm / Announcement Instructions
- Attachment 2C, TMI Site Evacuation Alarm / Announcement Instructions

7.3 Attachment 3, Owner Controlled Area

- Attachment 3A, PBAPS Potential Occupied Areas Outside the Protected Area
- Attachment 3B, LGS Potential Occupied Areas Outside the Protected Area
- Attachment 3C, TMI Potential Occupied Areas Outside the Protected Area

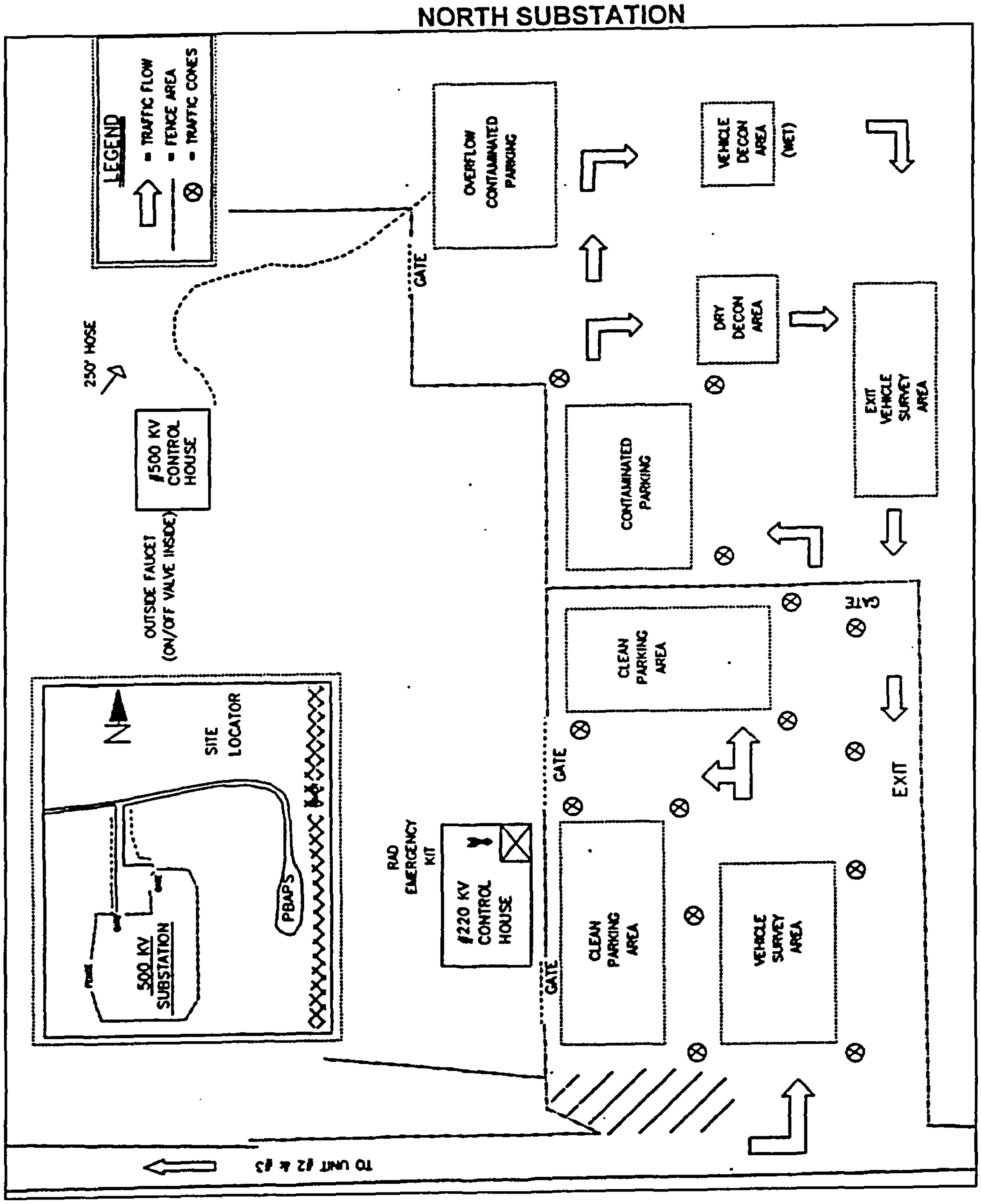
7.4 Attachment 4, Remote Assembly Area Muster List

7.5 Attachment 5, Vehicle Survey and Decontamination Report

7.6 Attachment 6, Facility Accountability List (Within Protected Area)

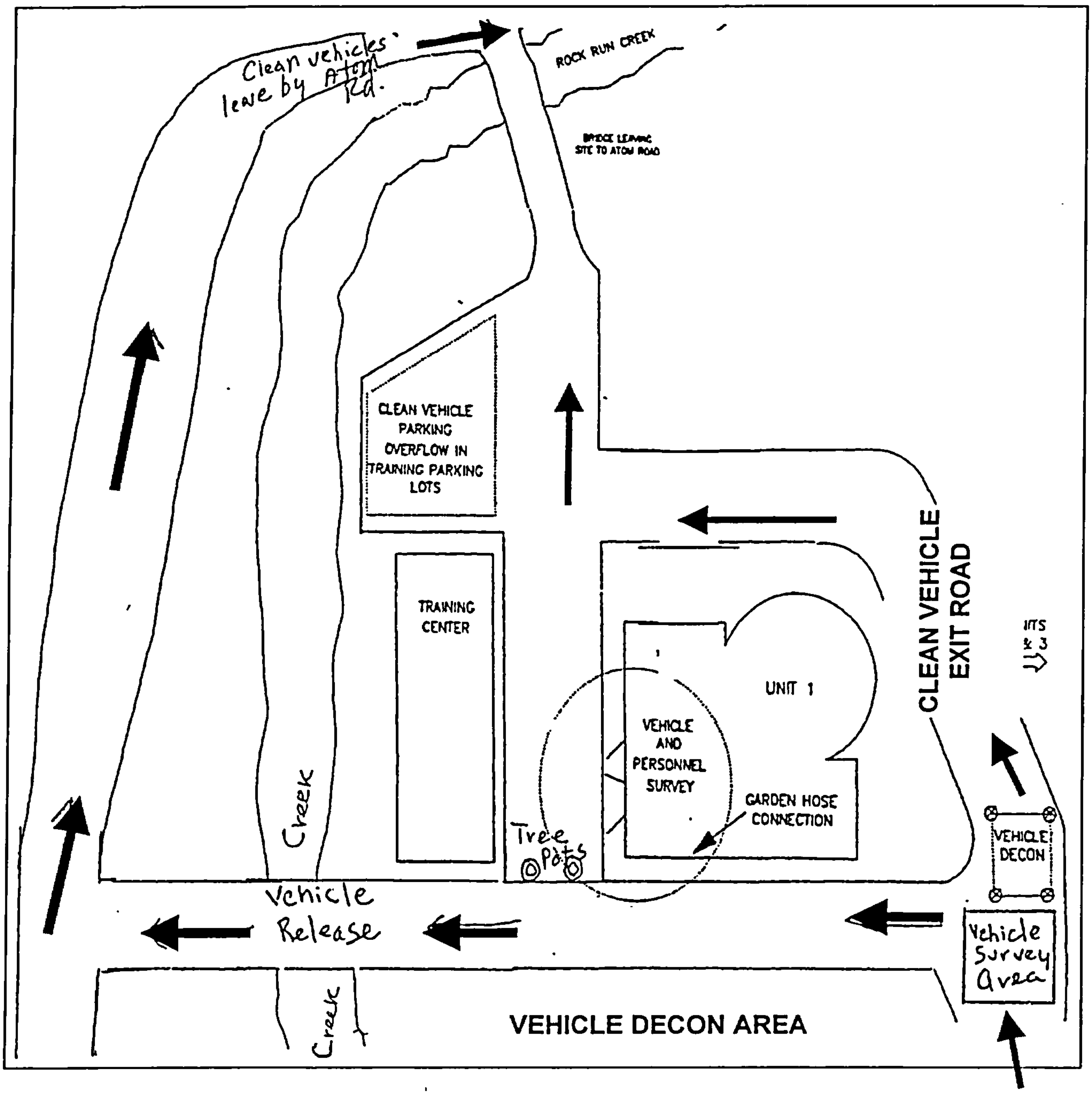
ATTACHMENT 1A
PBAPS REMOTE ASSEMBLY AREAS (LAYOUT / DIRECTIONS)
Page 1 of 2

FIGURE 1-1
NORTH SUBSTATION



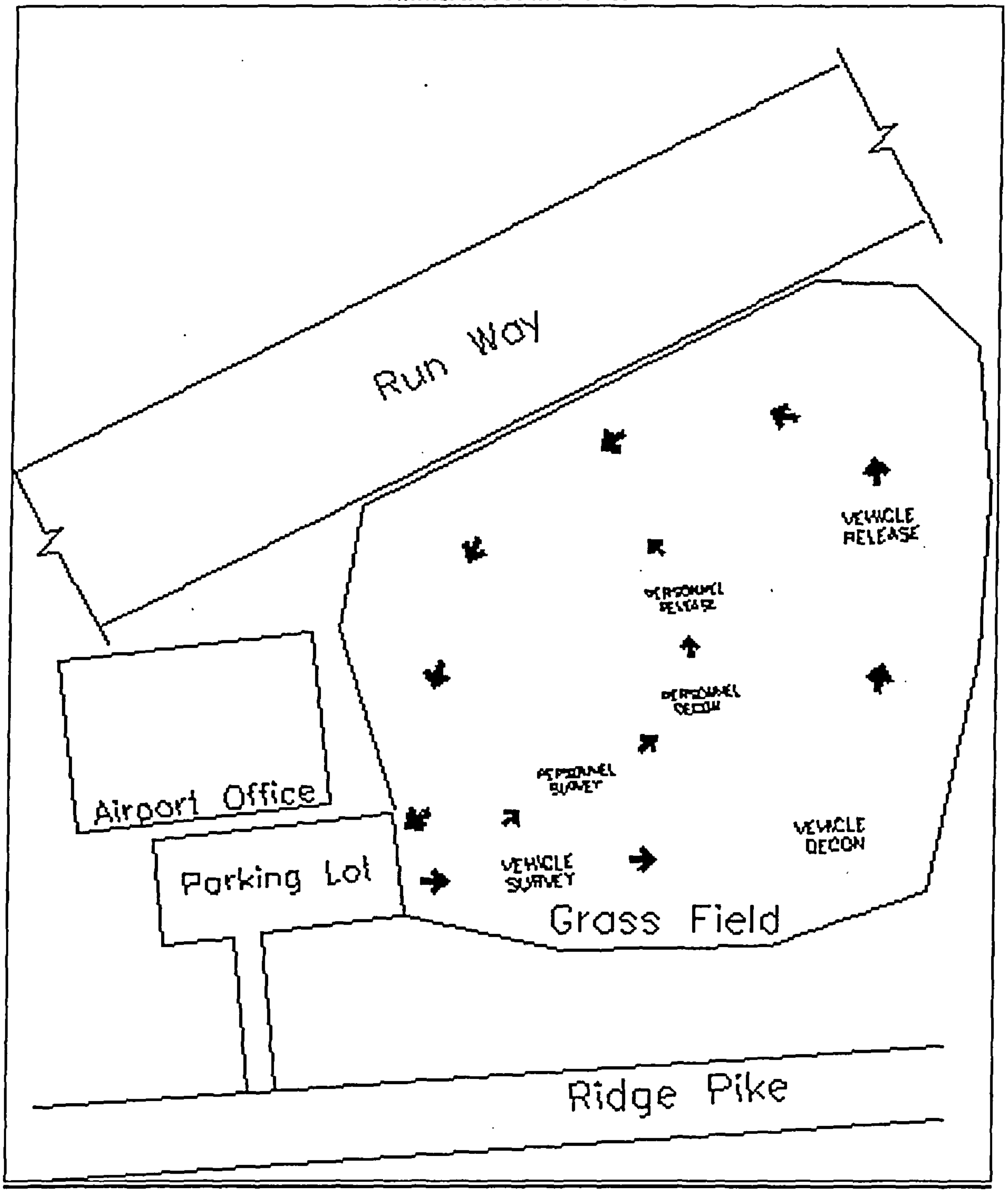
ATTACHMENT 1A
PBAPS REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)
Page 2 of 2

FIGURE 1-2
UNIT 1



ATTACHMENT 1B
LGS REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)
Page 1 of 4

FIGURE 2-1
LIMERICK AIRPORT



ATTACHMENT 1B
LGS REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)
Page 2 of 4

FIGURE 2-1 (Cont'd)

If the Limerick Airport is to be used as the RAA, **then PERFORM** the following:

1. **OBTAIN** keys for the storage shed from the HPFO (H-14)

2. **INFORM** airport office of situation and need to use hanger and field, using telephone no. contained in Section 4.0 (Outside Support Groups) of the Emergency Response Facility (ERF) Directory.

3. **SET UP** of airport area, as follows:

   NOTE: Persistent contamination or other problems – contact the TSC Radiation Protection Manager (using Section 8.1.3 of the ERF Telephone Directory).

   a. **SET UP** cones to direct traffic to Northwest grass area (behind shed)

   b. **DIRECT** personnel to hanger

   c. **SET UP** frisking area in hanger

   d. **POST** areas, as appropriate.

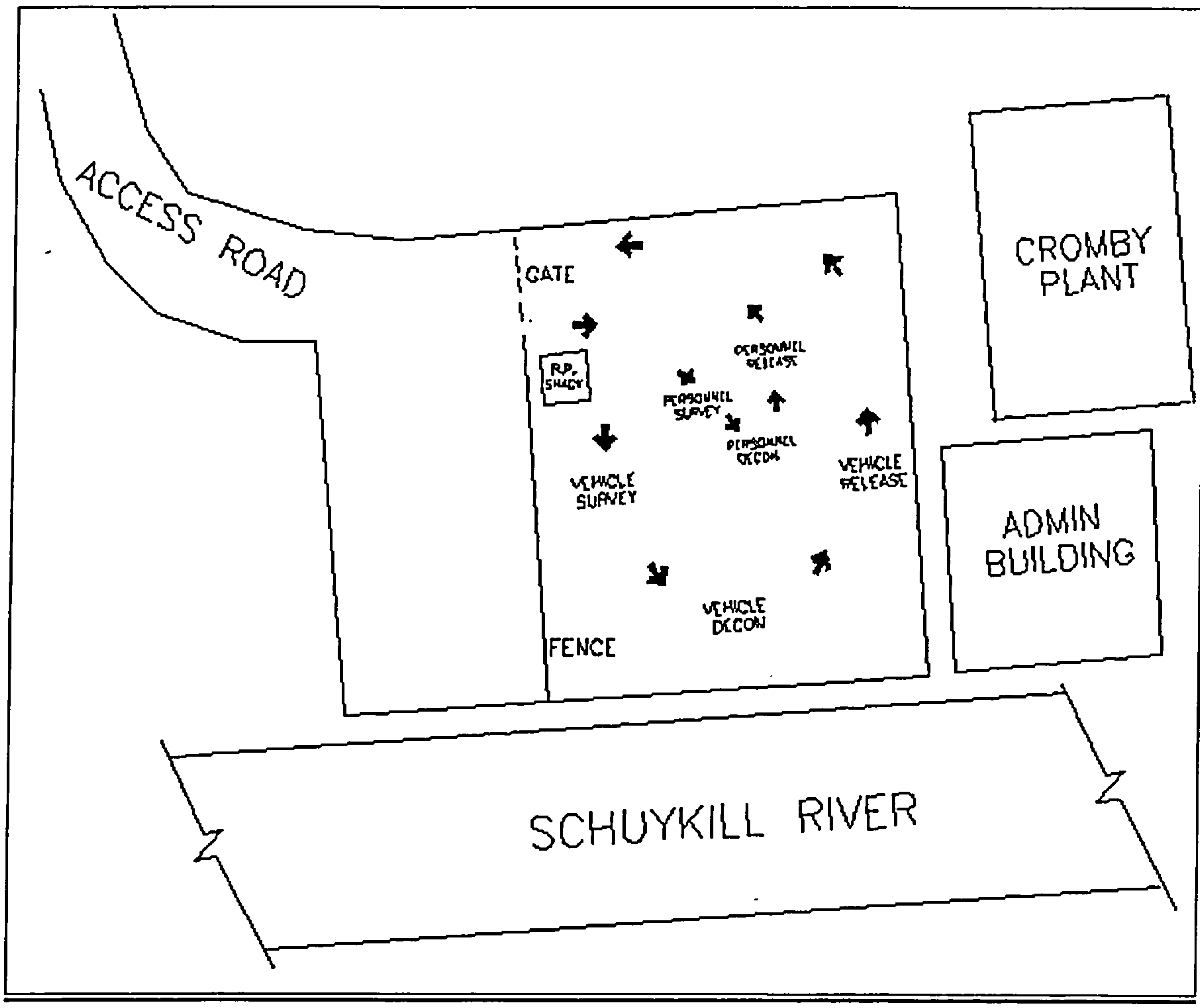
      NOTE: Posting and decontamination supplies are located in the storage shed West of the airport – (There are 2 sides to the shed.)

4. **Personnel decon has priority**. After all personnel have been evaluated, **ATTEMPT** to release monitor and vehicles

5. **CONTROL** all radioactive material for disposition by RMSC. Any water used (secondary method) may be disposed of down normal drains with no special precautions.

ATTACHMENT 1B
LGS REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)
Page 3 of 4

FIGURE 2-2
CROMBY GENERATING STATION



ATTACHMENT 1B
LGS REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)

FIGURE 2-2 (Cont'd)

If Cromby Station is to be used as the RAA, **then PERFORM** the following:

1. **OBTAIN** key (H-14) and gate badge from HPFO
2. **SET UP** the Cromby Station area, as follows:

   NOTE: For persistent contamination or other problems, **CONTACT** the TSC Radiation Protection Manager (using Section 8.1.3 of the ERF Telephone Directory).

   a. **SET UP** cones to direct traffic through gate to the parking lot West of the storage shack.

   b. **DIRECT** personnel to the plant entrance guard shack.

   c. **SET UP** frisking area in or near guard shack.

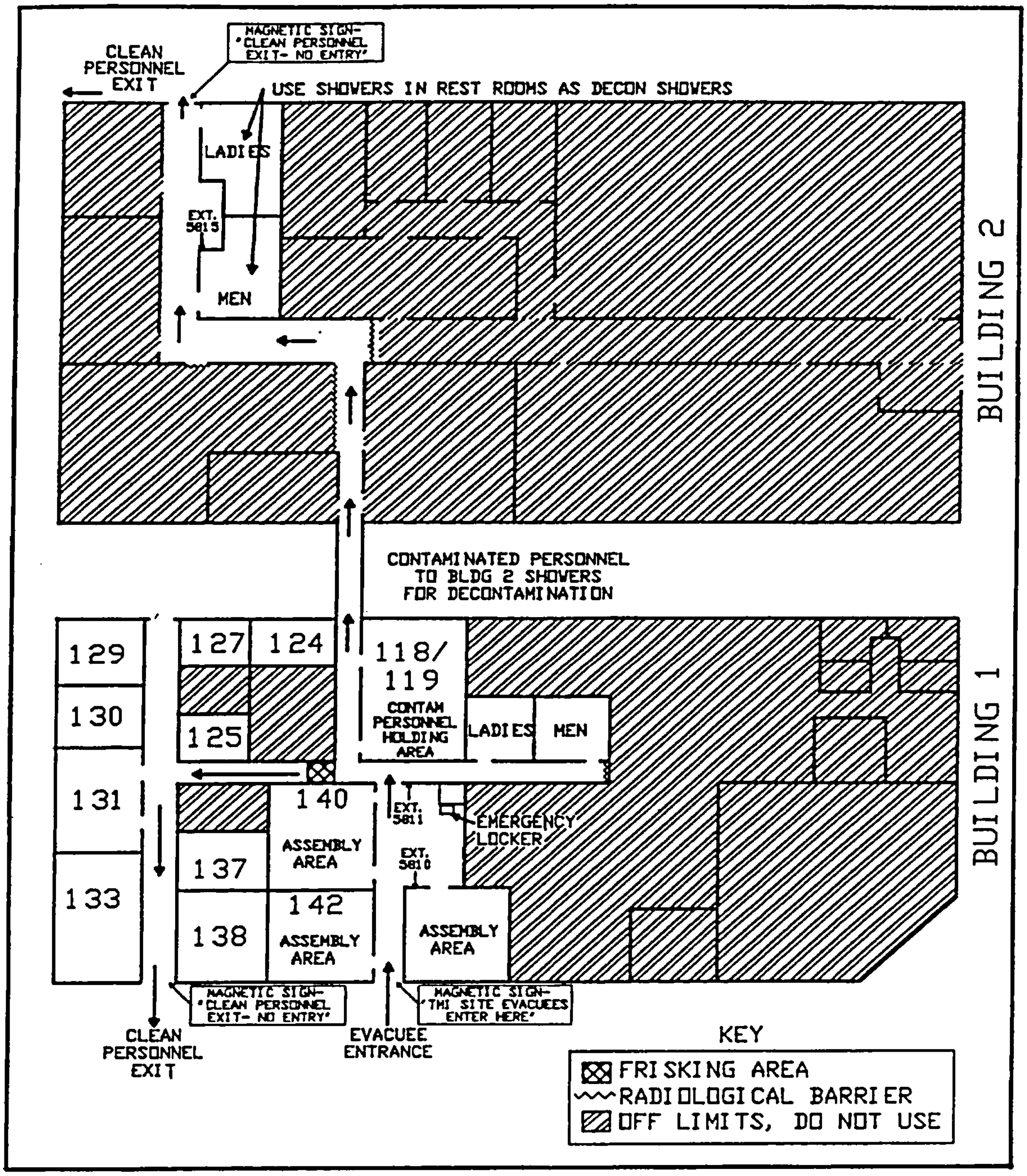
   d. **POST** areas, as appropriate.

   NOTE: Posting and decontamination supplies are located in the storage shack just inside the gate.

3. **Personnel decon has priority.** After all personnel have been evaluated, **ATTEMPT** to release monitor and vehicles
4. **CONTROL** all radioactive material for disposition by RMSC. Any water used (secondary method) may be disposed of down normal drains with no special precautions.

ATTACHMENT 1C
TMI REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)
Page 1 of 8

FIGURE 1-3
TRAINING CENTER



ATTACHMENT 1C
TMI REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)
Page 2 of 8

FIGURE 1-3 (CONT'D)
TRAINING CENTER

1. If the Training Center is to be used as the RAA:

   a. During normal working hours - Obtain access to the needed areas/classrooms by requesting assistance from Training Department Personnel upon arrival at the RAA.

   b. After normal working hours - Obtain keys/keycards for the Training Center (Buildings 1 and 2 and classrooms) from the Site Protection Department.

2. Set up the Training Center Set Up for Personnel Monitoring and Decon, as follows:

   a. Obtain the magnetic signs from the emergency locker and post them as follows (Figure 1-3):

      - Post a magnetic "TMI Site Evacuees Enter Here" sign on the outside of the exterior door nearest the Training Center vending machine area.
      - Post a magnetic "Clean Personnel Exit - No Entry" sign outside the exterior door near Room 138.
      - Post a magnetic "Clean Personnel Exit - No Entry" outside the exterior door near the ladies room in Building 2.

   b. Set up a frisking area in the hallway of Building 1 near Room 140 as shown on Figure 1-3.

      - Place a step off pad on the floor.
      - Obtain a small table or a chair and place the frisker on it.
      - Place RAA muster / sign-in sheets (Attachment 4) and pens on the table.
      - Set up additional frisking areas in the hallway according to the number of friskers available.

   c. Unlock and open the necessary classrooms for use as assembly areas and holding areas for contaminated personnel awaiting decontamination.

ATTACHMENT 1C
TMI REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)

FIGURE 1-3 (Cont'd)
TRAINING CENTER

d. Erect radiological barriers in Building 2 as shown on Figure 1-3.

- The barriers should have radiation warning signs attached to them.
- The signs should face the <u>outside</u> of the area being controlled and should state:
  - "No Entry" or "Keep Out",
  - "Contamination Area" and
  - "Radioactive Materials Area".

e. Take the following personnel decon supplies from the emergency locker to the decon area:

- Paper towels
- Waterless hand cleaner
- Wash basin
- Bath soap
- Shampoo
- Scrub brushes
- Nail clippers
- Barber scissors
- Nasal swabs
- Disposable PCs
- Masking tape
- Poly bags
- Frisker
- Step off pad

f. Personnel decontamination can be performed at the Training Center using the showers and sinks in the rest rooms in Building 2 (Simulator Bldg.).

- Wastewater from personnel decontamination can be disposed of down the normal sink and shower drains with no special precautions.

3. Vehicles should be monitored as they enter the rear parking lot. Contaminated vehicles should be directed to one side of the lot and clean vehicles to the other side of lot.

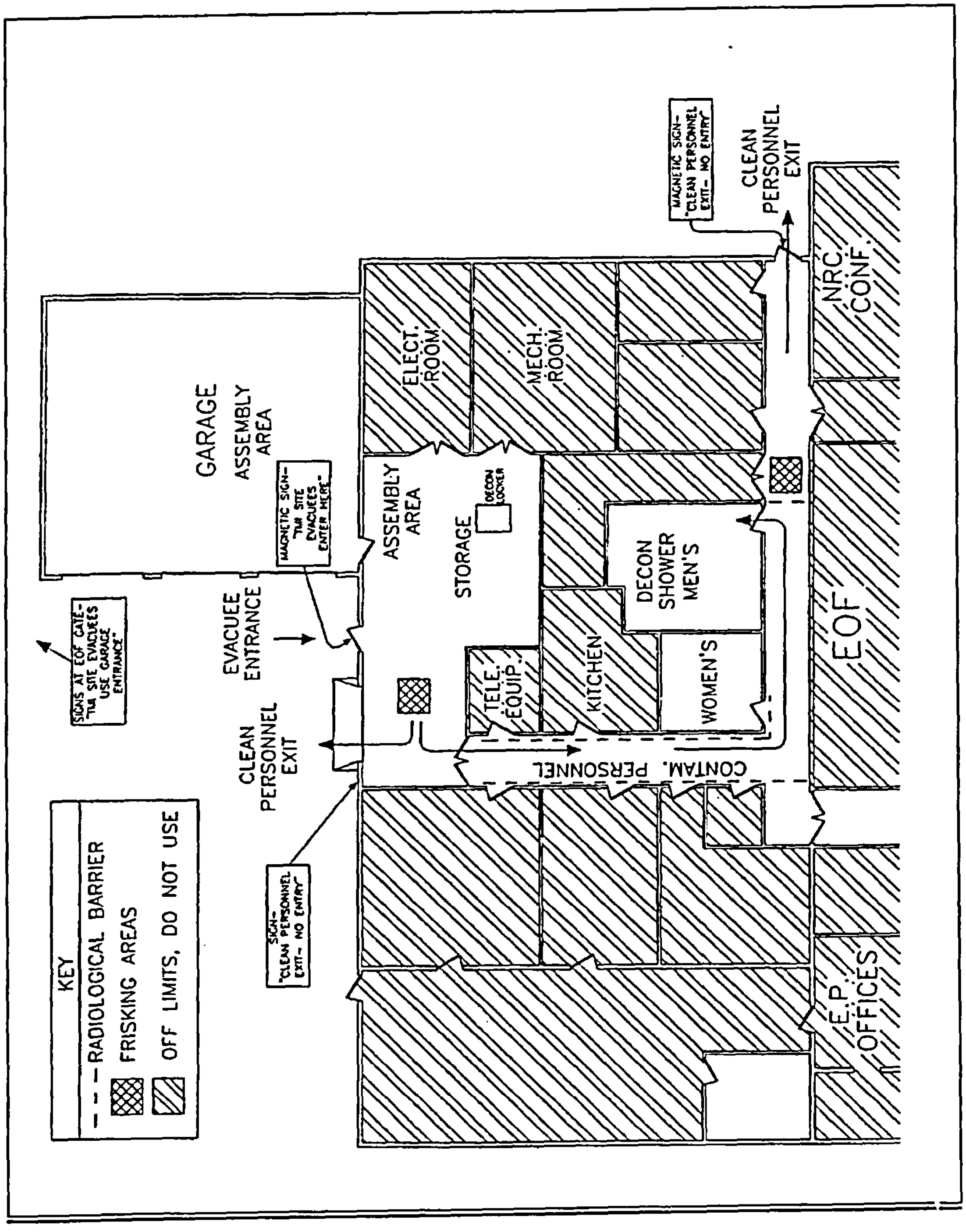
ATTACHMENT 1C
TMI REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)

TABLE 1-3
INSTRUCTIONS TO WILLIAMS VALLEY HIGH SCHOOL

1. **PROCEED** north on Route 441 to Geyers Church Road, **then TURN** right.
2. **FOLLOW** Geyers Church Road to Route 230, **then TURN** right.
3. **TURN** left at the connector road to Route 283, and **FOLLOW** the signs to Route 283 East.
4. **FOLLOW** Route 283 East to Route 743.
5. **TAKE** Route 743 North to Interstate 81.
6. **TAKE** Interstate 81 North to Route 209 South (Exit 33).
7. **TAKE** Route 209 South to the Reception Center at Williams Valley High School in Williamstown, PA.

ATTACHMENT 1C
TMI REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)
Page 5 of 8

FIGURE 1-4
JOINT PUBLIC INFORMATION CENTER LAYOUT



ATTACHMENT 1C
TMI REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)

FIGURE 1-4 (CONT'D)
JOINT PUBLIC INFORMATION CENTER

1. Set Up for Personnel Monitoring and Decon:

   a. Obtain the signs from the decon locker and post them as follows (see Figure 1-4):

      - Post a magnetic "TMI Site Evacuees Enter Here" sign on the outside of the door to the sample storage area.
      - Tape a "Clean Personnel Exit - No Entry" sign to the outside of the EOF next to the small roll-up door by the sample storage area.
      - Place the 2 "TMI Site Evacuees - Use Garage Entrance" signs that are on safety cones at the EOF gate. Make sure the signs point toward the EOF garage entrance.
      - Post a magnetic "Clean Personnel Exit - No Entry" sign outside the door at the front of the EOF near the NRC Conference Room.

   b. Establish positive access control at the following doors (see Figure 1-4A for exact locations):

      - Evacuee entrance door in the sample storage area,
      - Clean personnel exit door (roll-up door) in the sample storage area and
      - Clean personnel exit door near the NRC conference room.

      NOTE: Limit access to the facility to identifiable site personnel (i.e., those presenting a company photo badge or employees that you recognize).

   c. Set up two frisking areas as shown on Figure 1-4:

      - One in the sample storage area for personnel arriving from the site and
      - One in the hallway near the Count Room for frisking personnel who have been decontaminated.
      - Place a step off pad on the floor in each location.
      - Obtain small tables or chairs on which to place friskers.
      - Place RAA muster / sign-in sheets (Attachment 4) and pens with the frisker in the sample storage area.

ATTACHMENT 1C
TMI REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)
Page 7 of 8

FIGURE 1-4
JOINT PUBLIC INFORMATION CENTER LAYOUT

d. Erect radiological barriers as shown on Figure 1-4.

- Use stanchions and/or tape to support the barrier ropes.
- The barriers should have radiation warning signs attached to them.
- The signs should face the outside of the area being controlled and should state:
  - "No Entry" or "Keep Out",
  - "Contamination Area" and
  - "Radioactive Materials Area".

e. Take the following personnel decon supplies from the decon locker to the men's' room:

- Paper towels
- Waterless hand cleaner
- Wash basin
- Bath soap
- Shampoo
- Scrub brushes
- Nail clippers
- Barber scissors
- Nasal swabs
- Disposable PCs
- Masking tape
- Poly bags
- Frisker
- Step off pad

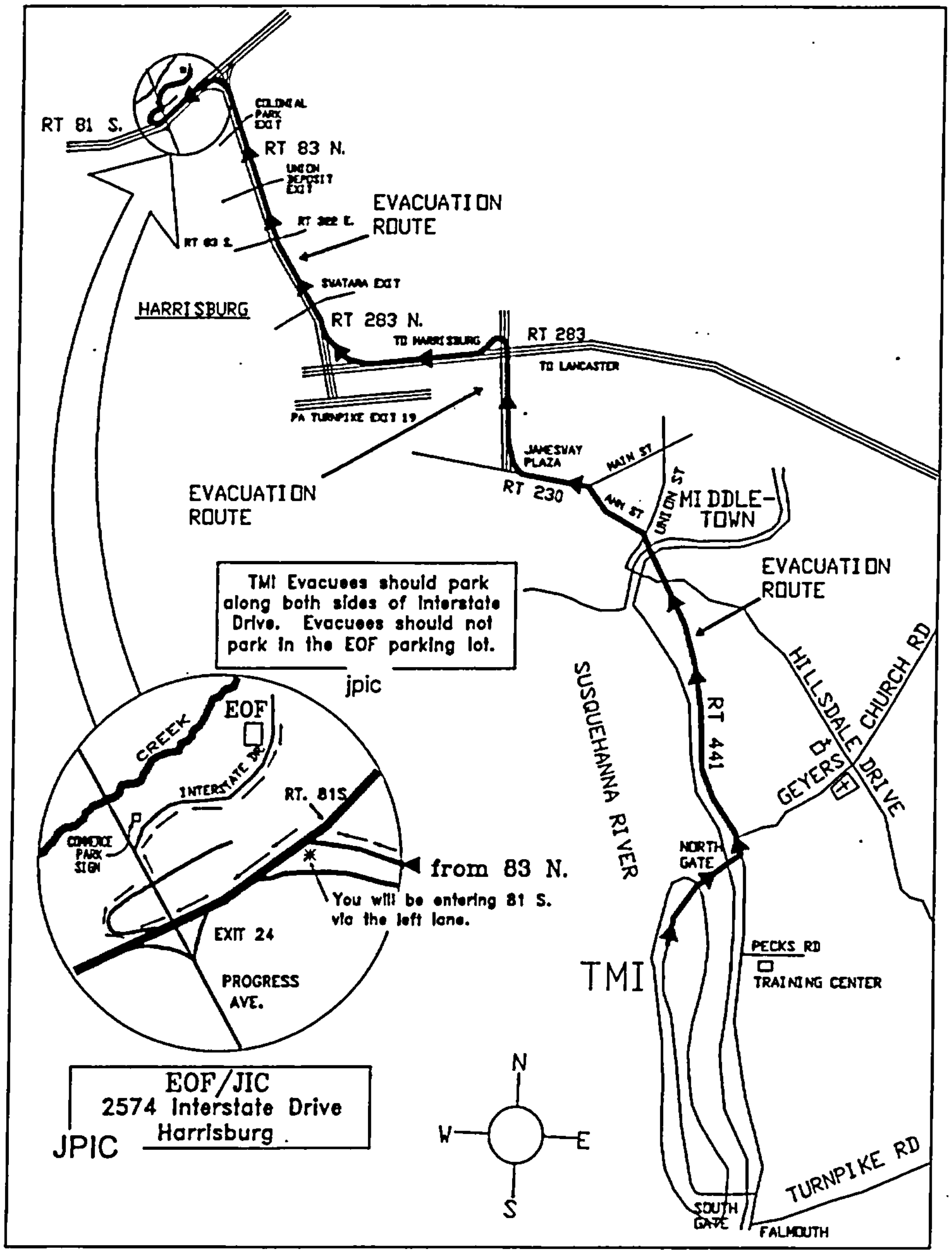
f. Personnel decontamination can be performed at the EOF using the shower and sinks in the men's rest room.

- Wastewater from personnel decontamination can be disposed of down the normal sink and shower drains with no special precautions.

2. Evacuees should use on-street parking. Vehicles should be monitored where they are parked and note made of any contaminated vehicles.

ATTACHMENT 1C
TMI REMOTE ASSEMBLY AREAS (LAYOUT/ DIRECTIONS)
Page 8 of 8

FIGURE 1-5
DIRECTIONS TO JOINT PUBLIC INFORMATION CENTER



ATTACHMENT 2A
PBAPS SITE EVACUATION ALARM / ANNOUNCEMENT INSTRUCTIONS
Page 1 of 2

Step 1: **ACTIVATE** the Page Alert Tone and **MAKE** the following announcement over the Plant Public Address (PA) System, twice, in a clear and distinct voice:

"ATTENTION ALL STATION PERSONNEL. THIS [IS / IS NOT] A DRILL.
I REPEAT, THIS [IS / IS NOT] A DRILL.

A SITE EVACUATION HAS BEEN ORDERED. ALL EMERGENCY RESPONSE ORGANIZATION MEMBERS SHALL REPORT TO YOUR RESPECTIVE EMERGENCY FACILITY OR ASSEMBLY AREA. ALL OTHER NON-ESSENTIAL PERSONNEL SHALL EVACUATE THE SITE IMMEDIATELY.

AS APPLICABLE: EVACUATING PERSONNEL SHALL: [A + (B.1 or B.2)]

[ ] A. EXIT THE PROTECTED AREA VIA ____________________

- [ ] Option 1: Using normal existing procedures; or
- [ ] Option 2: Designated route based on direction from Security

[ ] B.1 ASSEMBLE AT THE ______________________

- [ ] Option 1: North Sub-Station; or
- [ ] Option 2: Unit 1 Complex
- [ ] Option 3: As designated by the Station Emergency Director

OR

[ ] B.2 PROCEED HOME AND AWAIT FURTHER INSTRUCTIONS

Step 2: **SOUND** the Evacuation Alarm

- **ROTATE** the "Evacuation Alarm / MIC Selector" Switch #43, on Diesel Generator Panel 00C026B, to switch Position 6 (PLANT).
- **ACTIVATE** the Evacuation Alarm by pulling the handle OUT.
- **SOUND** alarm for approximately 60 seconds.
- **SECURE** alarm by pushing Switch #43 IN.
- **RETURN** Switch #43 to Position 3 (Normal).

Step 3: **ANNOUNCE** message, as state in Step 1, over the Plant Radio System (all channels known to be in use), twice.

ATTACHMENT 2A
PBAPS SITE EVACUATION ALARM / ANNOUNCEMENT INSTRUCTIONS
Page 2 of 2

Step 4: **ANNOUNCE** message, as state in Step 1, over the Pond Paging System:

- **ROTATE** the "Evacuation Alarm / MIC Selector" Switch #43, on the Diesel Generator Panel 00C026B, while in the IN mode to Position 2 (microphone river speakers).
- **ACTIVATE** the microphone by pulling the handle OUT.
- **REPEAT** the evacuation announcement twice.
- **SECURE** by pushing Switch #43 IN.

Step 5: **REPEAT** Steps 1 through 4 approximately every 10-15 minutes until accountability is complete.

PEACH BOTTOM STATION

ATTACHMENT 2B
LGS SITE EVACUATION ALARM / ANNOUNCEMENT INSTRUCTIONS
Page 1 of 2

Step 1: **ACTIVATE** the Evacuation Alarm

- **SET** the "Siren Tone Generator" selector switch, located on MCR Panel 00C650, to the SIREN (harmonic tone) position.
- **PULL OUT** the "Evacuation Alarm and River Warning Select" switch, located on MCR Panel 00C650.
- **ROTATE** the "Evacuation Alarm and River Warning Select" switch to the PLANT ALARM position, and **PUSH** selector switch IN.
- **SOUND** alarm for approximately 30 seconds.
- **RETURN** the "Evacuation Alarm and River Warning Select" switch to the OFF position to silence alarm.

Step 2: ANNOUNCE the following over the Priority Page System:

"ATTENTION ALL STATION PERSONNEL. THIS [IS / IS NOT] A DRILL.
I REPEAT, THIS [IS / IS NOT] A DRILL.

A SITE EVACUATION HAS BEEN ORDERED. ALL EMERGENCY RESPONSE ORGANIZATION MEMBERS SHALL REPORT TO YOUR RESPECTIVE EMERGENCY FACILITY OR ASSEMBLY AREA. ALL OTHER NON-ESSENTIAL PERSONNEL SHALL EVACUATE THE SITE IMMEDIATELY.

AS APPLICABLE: EVACUATING PERSONNEL SHALL: [A + B + (C.1 or C.2)]

[ ] A. EXIT THE PROTECTED AREA VIA ____________________

- [ ] Option 1: TSC GUARD STATION EXIT LANES; <u>or</u>
- [ ] Option 2: Designated route based on direction from Security

[ ] B. EXIT THE SITE VIA ______________________

- [ ] Option 1: Front Gate using Evergreen Road; <u>and/or</u>
- [ ] Option 2: Back Gate using Longview Road

[ ] C.1 ASSEMBLE AT THE ______________________

- [ ] Option 1: Cromby Generating Station; <u>or</u>
- [ ] Option 2: Pottstown – Limerick Airport
- [ ] Option 3: As designated by the Station Emergency Director

OR

[ ] C.2 PROCEED HOME AND AWAIT FURTHER INSTRUCTIONS

Step 3: **REPEAT** Steps 1& 2 approximately every 5-10 minutes until accountability is completed.

ATTACHMENT 2B
LGS SITE EVACUATION ALARM AND ANNOUNCEMENT INSTRUCTIONS
Page 2 of 2

Step 4: **ACTIVATE** the River Warning System

- **PULL OUT** the "Evacuation Alarm and River Warning Select" switch, located on MCR Panel 00C650.
- **ROTATE** the "Evacuation Alarm and River Warning Select" switch to the MIKE position, and **PUSH** selector switch IN.
- **DEPRESS** the "River Broadcast Microphone" pushbutton, on the grey handset, and **BROADCAST** message from Step 2, twice.
- **VERIFY** that the message is being broadcasted by observing response on the "River Broadcast Speaker Volume Monitor."
- **RETURN** the "Evacuation Alarm and River Warning Select" switch to the OFF position to end broadcast.

ATTACHMENT 2C
TMI SITE EVACUATION ALARM / ANNOUNCEMENT INSTRUCTIONS
Page 1 of 1

Step 1: **TURN ON** the Whelan speakers

Step 2: **ANNOUNCE** the following over the Plant PA System:

"ATTENTION ALL STATION PERSONNEL. THIS [IS / IS NOT] A DRILL.
I REPEAT, THIS [IS / IS NOT] A DRILL.

A SITE EVACUATION HAS BEEN ORDERED. ALL EMERGENCY RESPONSE ORGANIZATION MEMBERS SHALL REPORT TO YOUR RESPECTIVE EMERGENCY FACILITY OR ASSEMBLY AREA. ALL OTHER NON-ESSENTIAL PERSONNEL SHALL EVACUATE THE SITE IMMEDIATELY.

AS APPLICABLE: EVACUATING PERSONNEL SHALL: [A + B + (C.1 or C.2)]

[ ] A. EXIT THE PROTECTED AREA VIA ____________________
- [ ] Option 1: PROCESSING CENTER EXIT LANES; <u>or</u>
- [ ] Option 2: Designated route based on direction from Security

[ ] B. EXIT THE SITE VIA ____________________
- [ ] Option 1: North Gate; <u>and/or</u>
- [ ] Option 2: South Gate

[ ] C.1 ASSEMBLE AT THE ____________________
- [ ] Option 1: Training Center; <u>or</u>
- [ ] Option 2: Joint Public Information Center
- [ ] Option 3: As designated by the Station Emergency Director

OR

[ ] C.2 PROCEED HOME AND AWAIT FURTHER INSTRUCTIONS

Step 3: **REPEAT** Steps 1& 2 approximately every 5-10 minutes until accountability is completed.

Step 4: **TURN OFF** the Whelan speakers.

ATTACHMENT 3A
PBAPS POTENTIAL OCCUPIED AREAS OUTSIDE THE PROTECTED AREA
Page 1 of 1

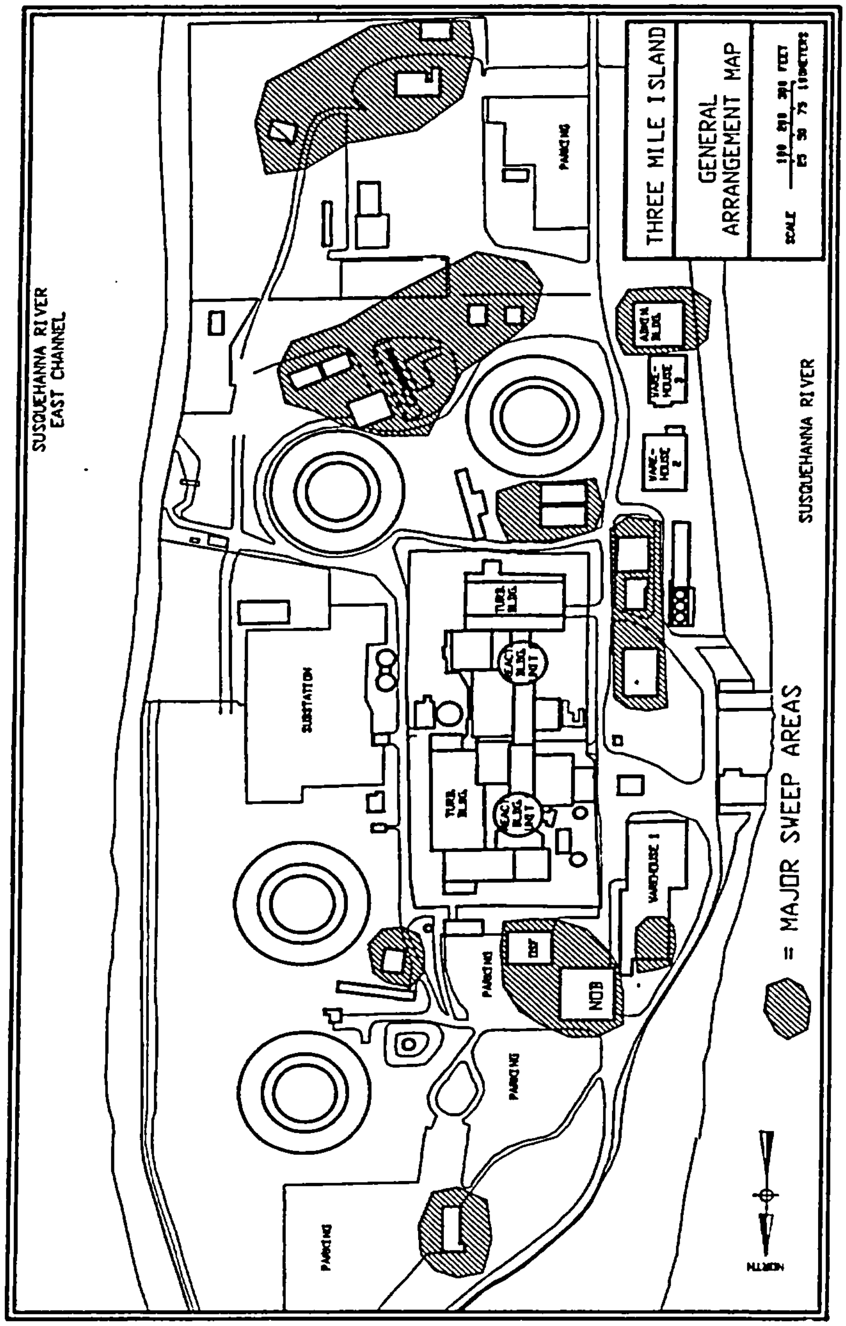
LOCATION	TYPE OF OCCUPANT
1. Mason-Dixon trail	Open to Public
2. Dorsey Park & Boat Launch	Open to Public
3. Site Management Building	Full Time
4. Unit #1	Full Time
5. Training Center	Full Time
6. Spent Fuel Storage	Part Time
7. Radwaste Shipping Facility	Part Time
8. North Sub-Station	Part Time
9. South Sub-Station	Part Time
10. Cooling Tower Area	Part Time
11. Outer Screen House	Part Time

ATTACHMENT 3B
LGS POTENTIAL OCCUPIED AREAS OUTSIDE THE PROTECTED AREA

LOCATION	TYPE OF OCCUPANT
1. Materials Management Building	Site personnel
2. Site Management Building	Site personnel
3. Personnel Processing Center	Site personnel
4. Facilities Shop	Site personnel and contractors
5. Limerick Information Center	Site personnel and public
6. 500 KV Yard	Site and outage personnel
7. 220 KV Yard	Site and outage personnel
8. Schuylkill River Pump house	Site personnel
9. Cooling towers, acid houses, trailers, and sampling station	Site, contractors, and outage personnel
10. Cooling tower trailers	Site, contractors, and outage personnel
11. Spray Pond Pump house	Site and outage personnel
12. Radwaste Storage Pad	Site, contractors, and outage personnel
13. Holding pond	Site and contract personnel
14. Chemical drum storage area	Site and contract personnel

ATTACHMENT 3C
TMI POTENTIAL OCCUPIED AREAS OUTSIDE THE PROTECTED AREA
Page 1 of 1

NOTE: **OBTAIN** a vehicle and **SWEEP** the designated areas, ensuring that personnel in the Purchasing Area of Warehouse 1 are notified.



ATTACHMENT 4
REMOTE ASSEMBLY AREA MUSTER LIST
Page 1 of 1

LOCATION: ______________________________

DATE: _______________ TIME: ____________ COMPLETED: ______________

LAST NAME	FIRST NAME	M.I.	SUPERVISOR OR GROUP	SOCIAL SECURITY NUMBER

ATTACHMENT 5
VEHICLE SURVEY AND DECONTAMINATION REPORT
Page 1 of 1

LICENSE #______________________________ STATE ____________________

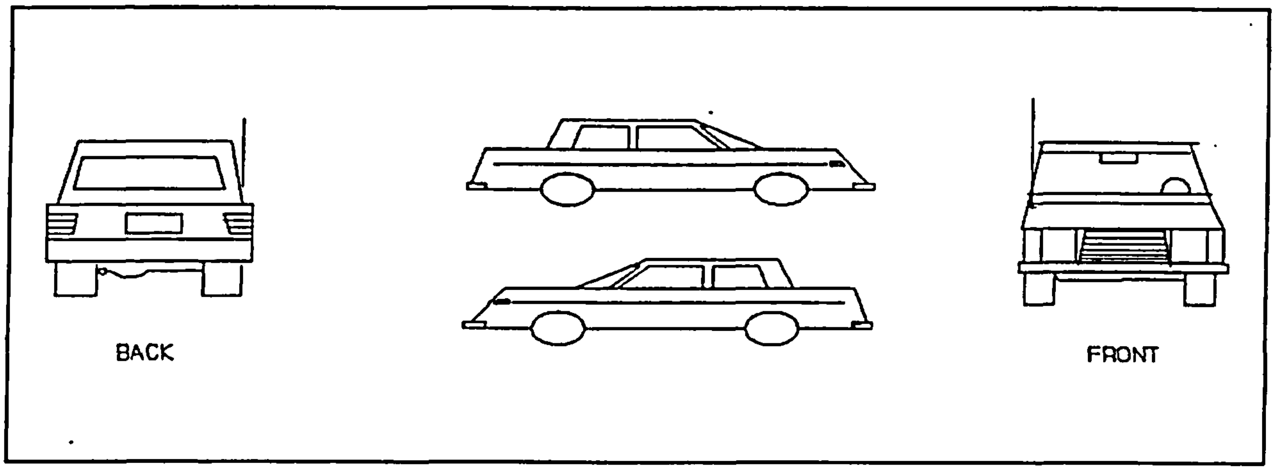
NAME OF OWNER ______________________ PHONE ( )____________________

ADDRESS (IF NOT EXELON)____
NUMBER - STREET CITY STATE ZIP

INITIAL SURVEY RESULTS

INSTRUMENT USED_________ DATE____________TIME______ SURVEYED BY: ________________

MODEL NO. (S/N & PROBE TYPE: __________________________



_____CLEAN-AUTHORIZED FOR RELEASE _____ CONTAMINATION DETECTED - RELEASE DENIED

COMMENTS

____

POST-DECONTAMINATION SURVEY RESULTS

DECONED BY:____
NAME DATE TIME

DECON METHOD USED: ____

POST-DECON SURVEY RESULTS: ____

FOLLOWUP ACTION REQUIRED: ____NONE ____SPECIAL FOLLOWUP (SPECIFY):

ATTACHMENT 6
FACILITY ACCOUNTABILITY LIST (WITHIN THE PROTECTED AREA)
Page 1 of 1

FACILITY: ______________________________

DATE: _______________ TIME: ____________ COMPLETED: ______________

NAME (Last, First, MI)	SOCIAL SECURITY NUMBER	BADGE #	TIME IN	TIME OUT

ENCLOSURE 2

LIMERICK GENERATING STATION, UNITS 1 & 2
PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 & 3
THREE MILE ISLAND, UNIT 1

Docket Nos. 50-352
50-353
50-277
50-278
50-289

License Nos. NPF-39
NPF-85
DPR-44
DPR-56
DPR-50

EMERGENCY RESPONSE PROCEDURES

REPORT INDICES

LIMERICK GENERATING STATION

PROCEDURE INDEX REPORT:

FAC	DOC TYPE	PROC TYPE	PROCEDURE NUMBER	CURR REV NBR	TITLE	EFFECTIVE DATE	RESP GROUP	SYSTEM NBR
LG	PROC	EP	EP-AA-1	0000	EMERGENCY PREPAREDNESS	10/20/00		
LG	PROC	EP	EP-AA-10	0001	EMERGENCY PREPAREDNESS PROCESS DESCRIPTION	12/12/02		
LG	PROC	EP	EP-AA-11	0001	OPERATING STATIONS EMERGENCY PREPAREDNESS PROCESS DESCRIPTION	12/12/02		
LG	PROC	EP	EP-AA-1101	0001	EP FUNDAMENTALS	12/20/02		
LG	PROC	EP	EP-AA-1102	0000	ERO FUNDAMENTALS	12/20/02		
LG	PROC	EP	EP-AA-110	0004	ASSESSMENT OF EMERGENCIES	02/20/03		
LG	PROC	EP	EP-AA-110-301	0002	CORE DAMAGE ASSESSMENT (BWR)	10/31/03		
LG	PROC	EP	EP-AA-110-302	0001	CORE DAMAGE ASSESSMENT (PWR) ****NO HARDCOPY DIST AT LGS SEE P4****	12/03/02		
LG	PROC	EP	EP-AA-111	0007	EMERGENCY CLASSIFICATION AND PROTECTIVE ACTION RECOMMENDATIONS	09/18/03		
LG	PROC	EP	EP-AA-112	0008	EMERGENCY RESPONSE ORGANIZATION (ERO)/EMERGENCY RESPONSE FACILITY (ERF) ACTIVATION AND OPERATION	05/23/03		
LG	PROC	EP	EP-AA-112-100	0005	CONTROL ROOM OPERATIONS	02/20/03		
LG	PROC	EP	EP-AA-112-200	0004	TSC ACTIVATION AND OPERATION	02/20/03		
LG	PROC	EP	EP-AA-112-201	0001	TSC COMMAND AND CONTROL	02/20/03		
LG	PROC	EP	EP-AA-112-202	0001	TSC FACILITY SUPPORT GROUP	02/20/03		
LG	PROC	EP	EP-AA-112-203	0001	TSC OPERATION GROUP	02/20/03		
LG	PROC	EP	EP-AA-112-204	0001	TSC TECHNICAL SUPPORT GROUP	02/20/03		
LG	PROC	EP	EP-AA-112-205	0001	TSC MAINTENANCE GROUP	02/20/03		
LG	PROC	EP	EP-AA-112-206	0001	TSC RADIATION PROTECTION/CHEMISTRY GROUP	02/20/03		
LG	PROC	EP	EP-AA-112-300	0004	OPERATIONS SUPPORT CENTER ACTIVATION AND OPERATION	02/20/03		
LG	PROC	EP	EP-AA-112-400	0004	EMERGENCY OPERATIONS FACILITY ACTIVATION AND OPERATION	02/20/03		
LG	PROC	EP	EP-AA-112-401	0002	NUCLEAR DUTY OFFICER (NDO)	10/13/03		
LG	PROC	EP	EP-AA-112-402	0001	EOF COMMAND AND CONTROL	02/20/03		
LG	PROC	EP	EP-AA-112-403	0002	EOF LOGISTICS SUPPORT GROUP	09/18/03		
LG	PROC	EP	EP-AA-112-404	0001	EOF TECHNICAL SUPPORT GROUP	02/20/03		
LG	PROC	EP	EP-AA-112-405	0001	EOF PROTECTIVE MEASURES GROUP	02/20/03		
LG	PROC	EP	EP-AA-112-600	0006	JOINT PUBLIC INFORMATION CENTER (JPIC) ACTIVATION	05/23/03		
LG	PROC	EP	EP-AA-112-601	0001	EMERGENCY NEWS CENTER (ENC) OPERATIONS	02/20/03		
LG	PROC	EP	EP-AA-112-602	0002	JPIC ACTIVATION AND OPERATION	05/23/03		
LG	PROC	EP	EP-AA-113	0004	PERSONNEL PROTECTIVE ACTIONS	08/30/02		
LG	PROC	EP	EP-AA-114	0004	NOTIFICATIONS	02/20/03		
LG	PROC	EP	EP-AA-115	0002	TERMINATION AND RECOVERY	09/18/03		
LG	PROC	EP	EP-AA-120	0003	EMERGENCY PLAN ADMINISTRATION	12/20/02		
LG	PROC	EP	EP-AA-120-1001	0003	10 CFR 50.54(Q) CHANGE EVALUATION	05/21/03		
LG	PROC	EP	EP-AA-120-1002	0000	STORM/EVENT RESTORATION	10/09/02		
LG	PROC	EP	EP-AA-121	0003	EMERGENCY RESPONSE FACILITIES AND EQUIPMENT READINESS	12/20/02		
LG	PROC	EP	EP-AA-121-1001	0003	AUTOMATED CALL-OUT SYSTEM MAINTENANCE	05/21/03		
LG	PROC	EP	EP-AA-122	0004	DRILLS AND EXERCISES	09/05/03		
LG	PROC	EP	EP-AA-122-1001	0003	DRILL DEVELOPMENT, CONDUCT AND EVALUATION	09/05/03		
LG	PROC	EP	EP-AA-122-1002	0003	EXERCISE DEVELOPMENT, CONDUCT AND EVALUATION	09/05/03		
LG	PROC	EP	EP-AA-122-1003	0003	SCHEDULING OF DRILLS AND EXERCISES	09/05/03		
LG	PROC	EP	EP-AA-122-1004	0002	DEMONSTRATION CRITERIA	09/05/03		
LG	PROC	EP	EP-AA-123	0002	COMPUTER PROGRAMS	11/05/02		
LG	PROC	EP	EP-AA-123-1003	0000	CORE DAMAGE ASSESSMENT METHODOLOGY (CDAM) PROGRAM TECHNICAL BASIS	10/31/03		
LG	PROC	EP	EP-AA-124	0004	INVENTORIES AND SURVEILLANCES	12/20/02		
LG	PROC	EP	EP-AA-125	0002	EMERGENCY PREPAREDNESS SELF EVALUATION PROCESS	12/20/02		
LG	PROC	EP	EP-AA-125-1001	0002	EP PERFORMANCE INDICATOR GUIDANCE	12/20/02		
LG	PROC	EP	EP-AA-125-1002	0002	ERO PERFORMANCE - PERFORMANCE INDICATORS GUIDANCE	12/20/02		

LIMERICK GENERATING STATION

PROCEDURE INDEX REPORT:

FAC	DOC TYPE	PROC TYPE	PROCEDURE NUMBER	CURR REV NBR	TITLE	EFFECTIVE DATE	RESP GROUP	SYSTEM NBR
LG	PROC	EP	EP-AA-125-1003	0002	ERO READINESS - PERFORMANCE INDICATORS GUIDANCE	12/20/02		
LG	PROC	EP	EP-AA-125-1004	0002	EMERGENCY RESPONSE FACILITIES & EQUIPMENT PERFORMANCE INDICATORS GUIDANCE	12/20/02		
LG	PROC	EP	EP-AA-125-1005	0000	PROBLEM INDENTIFICATION & RESOLUTION PERFORMANCE INDICATOR GUIDANCE	12/20/02		
LG	PROC	EP	EP-LG-112-500	0000	EMERGENCY ENVIRONMENTAL MONITORING	11/03/03		
LG	PROC	EP	EP-MA-110-100	0002	ERO COMPUTER APPLICATIONS	07/01/03		
LG	PROC	EP	EP-MA-110-200	0003	DOSE ASSESSMENT	08/08/08		
LG	PROC	EP	EP-MA-112-406	0001	MAROG OFFSITE LIAISONS	02/20/03		
LG	PROC	EP	EP-MA-113-100	0002	ASSEMBLY AND SITE EVACUATION	10/31/03		
LG	PROC	EP	EP-MA-114-100	0005	MID-ATLANTIC STATE/LOCAL NOTIFICATIONS	09/18/03		
LG	PROC	EP	EP-MA-121-1002	0000	ALERT NOTIFICATION SYSTEM (ANS) DESCRIPTION, TESTING, MAINTENANCE AND PERFORMANCE TRENDING PROGRAM	12/20/02		
LG	PROC	EP	EP-MA-121-1004	0000	EMERGENCY PREPAREDNESS ALERT NOTIFICATION SYSTEM (ANS) CONTROL OF EQUIPMENT & OUTAGES	12/20/02		
LG	PROC	EP	EP-MA-123-1001	0000	KI ASSESSMENT SPREADSHEET TECHNICAL BASIS	07/01/03		
LG	PROC	EP	EP-MA-123-1004	0000	DOSE ASSESSMENT AND PROTECTIVE ACTION RECOMMENDATION (DAPAR) PROGRAM TECHNICAL BASIS FOR LIMERICK GENERATING STATION	08/08/03		
LG	PROC	EP	EP-MA-124-1001	0003	FACILITY INVENTORIES AND EQUIPMENT TESTS	10/13/03		
LG	MANL	EPPL	EP-AA-1000	0014	STANDARDIZED RADIOLOGICAL EMERGENCY PLAN	02/20/03		
LG	MANL	EPPL	EP-AA-1008	0005	RADIOLOGICAL EMERGENCY PLAN ANNEX FOR LIMERICK GENERATING STATION	10/31/03		

** END OF REPORT **

PEACH BOTTOM ATOMIC POWER STATION

PROCEDURE INDEX REPORT:

FAC	DOC TYPE	PROC TYPE	PROCEDURE NUMBER	CURR REV NBR	TITLE	EFFECTIVE DATE	RESP GROUP	SYSTEM NBR
PB	PROC	EP	EP-AA-1000	0014	STANDARIZED RADIOLOGICAL EMERGENCY PLAN	02/20/03	PWE	
PB	PROC	EP	EP-AA-1007	0009	RADIOLOGICAL EMERGENCY PLAN ANNEX FOR PEACH BOTTOM ATOMIC POWER STATION	10/31/03	PWE	
PB	PROC	EP	EP-AA-1101	0001	EP FUNDAMENTALS	12/20/02	PWE	
PB	PROC	EP	EP-AA-1102	0000	ERO FUNDAMENTALS	12/20/02	PWE	
PB	PROC	EP	EP-AA-110	0004	ASSESSMENT OF EMERGENCIES	02/20/03	PWE	
PB	PROC	EP	EP-AA-110-301	0002	CORE DAMAGE ASSESSMENT (BWR)	10/22/03	PWE	
PB	PROC	EP	EP-AA-110-302	0001	CORE DAMAGE ASSESSMENT (PWR)	12/17/02	PWE	
PB	PROC	EP	EP-AA-111	0007	EMERGENCY CLASSIFICATION AND PROTECTIVE ACTION RECOMMENDATIONS	09/18/03	PWE	
PB	PROC	EP	EP-AA-112	0008	EMERGENCY RESPONSE ORGANIZATION (ERO)/EMERGENCY RESPONSE FACILITY (ERF) ACTIVATION AND OPERATION	05/23/03	PWE	
PB	PROC	EP	EP-AA-112-100	0005	CONTROL ROOM OPERATIONS	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-200	0004	TSC ACTIVIATION AND OPERATION	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-201	0001	TSC COMMAND AND CONTROL	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-202	0001	TSC FACILITY SUPPORT GROUP	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-203	0001	TSC OPERATION GROUP	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-204	0001	TSC TECHNICAL SUPPORT GROUP	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-205	0001	TSC MAINTENANCE GROUP	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-206	0001	TSC RADIATION PROTECTION/CHEMISTRY GROUP	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-300	0004	OPERATIONS SUPPORT CENTER ACTIVIATION AND OPERATION	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-400	0004	EMERGENCY OPERATIONS FACILITY ACTIVATION AND OPERATION	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-401	0002	NUCLEAR DUTY OFFICER (NDO)	10/10/03	PWE	
PB	PROC	EP	EP-AA-112-402	0001	EOF COMMAND AND CONTROL	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-403	0002	EOF LOGISTICS SUPPORT GROUP	09/18/03	PWE	
PB	PROC	EP	EP-AA-112-404	0001	EOF TECHNICAL SUPPORT GROUP	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-405	0001	EOF PROTECTIVE MEASURES GROUP	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-500	0005	EMERGENCY ENVIRONMENTAL MONITORING	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-600	0006	PUBLIC INFORMATION ORGANIZATION ACTIVATION AND OPERATIONS	05/23/03	PWE	
PB	PROC	EP	EP-AA-112-601	0001	EMERGENCY NEWS CENTER (ENC) OPERATIONS	02/20/03	PWE	
PB	PROC	EP	EP-AA-112-602	0002	JPIC ACTIVATION AND OPERATION	05/23/03	PWE	
PB	PROC	EP	EP-AA-113	0004	PERSONNEL PROTECTIVE ACTIONS	08/30/02	PWE	
PB	PROC	EP	EP-AA-114	0004	NOTIFICATIONS	02/20/03	PWE	
PB	PROC	EP	EP-AA-115	0002	TERMINATION AND RECOVERY	09/18/03	PWE	
PB	PROC	EP	EP-AA-120	0003	EMERGENCY PLAN ADMINISTRATION	12/20/02	PWE	
PB	PROC	EP	EP-AA-120-1001	0003	10 CFR 50.54(Q) CHANGE EVALUATION	04/30/03	PWE	
PB	PROC	EP	EP-AA-120-1002	0000	STORM/EVENT RESTORATION	10/18/02	PWE	
PB	PROC	EP	EP-AA-121	0003	EMERGENCY RESPONSE FACILITIES AND EQUIPMENT READINESS	12/20/02	PWE	
PB	PROC	EP	EP-AA-121-1001	0003	AUTOMATED CALL-OUT SYSTEM MAINTENANCE	04/30/03	PWE	
PB	PROC	EP	EP-AA-122	0004	DRILLS AND EXERCISES	09/12/03	PWE	
PB	PROC	EP	EP-AA-122-1001	0003	DRILL DEVELOPMENT, CONDUCT AND EVALUATION	09/12/03	PWE	
PB	PROC	EP	EP-AA-122-1002	0003	EXERCISE DEVELOPMENT, CONDUCT AND EVALUATION	09/12/03	PWE	
PB	PROC	EP	EP-AA-122-1003	0003	SCHEDULING OF DRILLS AND EXERCISES	09/12/03	PWE	
PB	PROC	EP	EP-AA-122-1004	0002	OBJECTIVES AND DEMONSTRATION CRITERIA	09/12/03	PWE	
PB	PROC	EP	EP-AA-123	0002	COMPUTER PROGRAMS	11/12/02	PWE	
PB	PROC	EP	EP-AA-123-1003	0000	CORE DAMAGE ASSESSMENT METHODOLOGY (CDAM) PROGRAM TECHNICAL BASIS	10/22/03	PWE	
PB	PROC	EP	EP-AA-124	0004	INVENTORIES AND SURVEILLANCES	12/20/02	PWE	
PB	PROC	EP	EP-AA-125	0002	EMERGENCY PREPAREDNESS SELF EVALUATION PROCESS	12/20/02	PWE	
PB	PROC	EP	EP-AA-125-1001	0002	EP PERFORMANCE INDICATOR GUIDANCE	12/20/02	PWE	
PB	PROC	EP	EP-AA-125-1002	0002	ERO PERFORMANCE - PERFORMANCE INDICATORS GUIDANCE	12/20/02	PWE	

PEACH BOTTOM ATOMIC POWER STATION

PROCEDURE INDEX REPORT:

FAC	DOC TYPE	PROC TYPE	PROCEDURE NUMBER	CURR REV NBR	TITLE	EFFECTIVE DATE	RESP GROUP	SYSTEM NBR
PB	PROC	EP	EP-AA-125-1003	0002	ERP READINESS - PERFORMANCE INDICATORS GUIDANCE	12/20/02	PWE	
PB	PROC	EP	EP-AA-125-1004	0002	EMERGENCY RESPONSE FACILITIES & EQUIPMENT PERFORMANCE INDICATORS GUIDANCE	12/20/02	PWE	
PB	PROC	EP	EP-AA-125-1005	0000	PROBLEM IDENTIFICATION & RESOULTION PERFORMANCE INDICATOR GUIDANCE	12/20/02	PWE	
PB	PROC	EP	EP-MA-110-100	0002	ERO COMPUTER APPLICATIONS	07/01/03	PWE	
PB	PROC	EP	EP-MA-110-200	0003	DOSE ASSESSMENT	08/08/03	PWE	
PB	PROC	EP	EP-MA-112-406	0001	MAROG OFFSITE LIASONS	02/20/03	PWE	
PB	PROC	EP	EP-MA-113-100	0002	ASSEMBLY AND SITE EVACUATION	10/31/03	PWE	
PB	PROC	EP	EP-MA-114-100	0005	MID-ATLANTIC STATE/LOCAL NOTIFICATIONS	09/18/03	PWE	
PB	PROC	EP	EP-MA-121-1002	0000	ALERT NOTIFICATION SYSTEM (ANS) DESCRIPTION, TESTING, MAINTENANCE AND PERFORMANCE TRENDING PROGRAM	12/20/02	PWE	
PB	PROC	EP	EP-MA-121-1004	0000	EMERGENCY PREPAREDNESS ALERT NOTIFICATION SYSTEM (ANS) CONTROL OF EQUIPMENT & OUTAGES	12/20/02	PWE	
PB	PROC	EP	EP-MA-123-1001	0000	KI ASSESSMENT SPREADSHEET TECHNICAL BASIS	07/01/03	PWE	
PB	PROC	EP	EP-MA-123-1005	0000	DOSE ASSESSMENT AND PROTECTIVE ACTION RECOMMENDATION (DAPAR) PROGRAM TECHNICAL BASIS FOR PEACH BOTTOM ATOMIC POWER STATION	08/08/03	PWE	
PB	PROC	EP	EP-MA-124-1001	0003	FACILITY INVENTORIES AND EQUIPMENT TESTS	10/10/03	PWE	
PB	PROC	EP	EP-UG-01	0005	CONTROL OF EP GUIDELINES	12/07/98		

** END OF REPORT **

SERIES: 0000 EXELON POLICIES AND DIRECTIVES

PROCEDURE NUMBER	REV	EFFDATE	SITE	PROCEDURE TITLE	TC NUMBER	LEVEL
EP-AA-1	0	2000-10-20	TMI1	EMERGENCY PREPAREDNESS		N/A

SERIES: 0016 EMERGENCY PLAN IMPLEMENTING PROCEDURE / DOCUMENT

PROCEDURE NUMBER	REV	EFFDATE	SITE	PROCEDURE TITLE	TC NUMBER	LEVEL
EP-AA-110	4	2003-03-28	TMI1	ASSESSMENT OF EMERGENCIES		2
EP-AA-110-301	1	2003-09-12	TMI1	CORE DAMAGE ASSESSMENT (BWR)		2
EP-AA-110-302	1	2003-03-28	TMI1	CORE DAMAGE ASSESSMENT (PWR)		2
EP-AA-111	7	2003-09-18	TMI1	EMERGENCY CLASSIFICATION AND PROTECTIVE ACTION RECOMMENDATIONS		2
EP-AA-112	8	2003-05-23	TMI1	EMERGENCY RESPONSE ORGANIZATION (ERO) - EMERGENCY RESPONSE FACILITY (ERF) ACTIVATION AND OPERATION		2
EP-AA-112-100	5	2003-03-28	TMI1	CONTROL ROOM OPERATIONS		2
EP-AA-112-200	4	2003-03-28	TMI1	TSC ACTIVATION AND OPERATION		2
EP-AA-112-201	1	2003-03-28	TMI1	TSC COMMAND AND CONTROL		2
EP-AA-112-202	1	2003-03-28	TMI1	TSC FACILITY SUPPORT GROUP		2
EP-AA-112-203	1	2003-03-28	TMI1	TSC OPERATION GROUP		2
EP-AA-112-204	1	2003-03-28	TMI1	TSC TECHNICAL SUPPORT GROUP		2
EP-AA-112-205	1	2003-03-28	TMI1	TSC MAINTENANCE GROUP		2
EP-AA-112-206	1	2003-03-28	TMI1	TSC RADIATION PROTECTION / CHEMISTRY GROUP		2
EP-AA-112-300	4	2003-03-28	TMI1	OPERATIONS SUPPORT CENTER ACTIVATION AND OPERATION		2
EP-AA-112-400	4	2003-03-28	TMI1	EMERGENCY OPERATIONS FACILITY ACTIVATION AND OPERATION		2
EP-AA-112-401	2	2003-10-10	TMI1	NUCLEAR DUTY OFFICER (NDO)		2
EP-AA-112-402	1	2003-03-28	TMI1	EOF COMMAND AND CONTROL		2
EP-AA-112-403	2	2003-09-18	TMI1	EOF LOGISTICS SUPPORT GROUP		2
EP-AA-112-404	1	2003-03-28	TMI1	EOF TECHINICAL SUPPORT GROUP		2

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SERIES: 0016 EMERGENCY PLAN IMPLEMENTING PROCEDURE / DOCUMENT

PROCEDURE NUMBER	REV	EFFDATE	SITE	PROCEDURE TITLE	TC NUMBER	LEVEL
EP-AA-112-405	1	2003-03-28	TMI1	EOF PROTECTIVE MEASURES GROUP		2
EP-AA-112-500	5	2003-03-28	TMI1	EMERGENCY ENVIRONMENTAL MONITORING		2
EP-AA-112-600	6	2003-05-23	TMI1	PUBLIC INFORMATION ORGANIZATION ACTIVATION AND OPERATIONS		2
EP-AA-112-601	1	2003-03-28	TMI1	EMERGENCY NEWS CENTER (ENC) OPERATIONS		2
EP-AA-112-602	2	2003-05-23	TMI1	JPIC ACTIVATION AND OPERATION		2
EP-AA-113	4	2003-03-28	TMI1	PERSONNEL PROTECTIVE ACTIONS		2
EP-AA-114	4	2003-03-28	TMI1	NOTIFICATIONS		2
EP-AA-115	2	2003-09-18	TMI1	TERMINATION AND RECOVERY		2
EP-AA-120	3	2003-03-28	TMI1	EMERGENCY PLAN ADMINISTRATION		2
EP-AA-121	3	2003-03-28	TMI1	EMERGENCY RESPONSE FACILITIES AND EQUIPMENT READINESS		2
EP-AA-122	4	2003-09-12	TMI1	DRILLS AND EXERCISES		2
EP-AA-123	2	2003-03-28	TMI1	COMPUTER PROGRAMS		2
EP-AA-124	4	2003-03-28	TMI1	INVENTORIES AND SURVEILLANCES		2
EP-AA-125	2	2002-12-20	TMI1	EMERGENCY PREPAREDNESS SELF EVALUATION PROCESS		2
EP-MA-110-100	2	2003-07-01	TMI1	ERO COMPUTER APPLICATIONS		2
EP-MA-110-200	3	2003-08-08	TMI1	DOSE ASSESSMENT		2
EP-MA-112-406	1	2003-03-28	TMI1	MAROG OFFSITE LIAISONS		2
EP-MA-113-100	2	2003-10-31	TMI1	ASSEMBLY AND SITE EVACUATION		2
EP-MA-114-100	5	2003-09-18	TMI1	MID-ATLANTIC STATE / LOCAL NOTIFICATIONS		2
EPIP-TMI-.16	11	2002-07-12	TMI1	CONTAMINATED INJURIES		2

SERIES: 0016 EMERGENCY PLAN IMPLEMENTING PROCEDURE / DOCUMENT

PROCEDURE NUMBER	REV	EFFDATE	SITE	PROCEDURE TITLE	TC NUMBER	LEVEL
EPIP-TMI-.19	10	2000-10-20	TMI1	EMERGENCY DOSIMETRY / SECURITY BADGE ISSUANCE		2
TEP-ADM-1300.01	11	2003-03-28	TMI1	MAINTAINING EMERGENCY PREPAREDNESS		2
TEP-ADM-1300.05	13	2003-08-08	TMI1	EMERGENCY EQUIPMENT READINESS		2

SERIES: 0054 EMERGENCY PREPAREDNESS PROCEDURE

PROCEDURE NUMBER	REV	EFFDATE	SITE	PROCEDURE TITLE	TC NUMBER	LEVEL
TEP-SUR-1310.01	11	2003-03-28	TMI1	EMERGENCY COMMUNICATIONS TEST PROCEDURE		2
TEP-SUR-1310.05	5	2003-03-28	TMI1	VERIFICATION OF EMERGENCY PREPAREDNESS AIDS		3
TEP-SUR-1310.10	5	2001-11-13	TMI1	PROCEDURE CHANGE NOTIFICATION		3

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SERIES: 0149 EXELON TRAINING AND REFERENCE MATERIAL

PROCEDURE NUMBER	REV	EFFDATE	SITE	PROCEDURE TITLE	TC NUMBER	LEVEL
EP-AA-1000	14	2003-03-28	TMI1	STANDARDIZED RADIOLOGICAL EMERGENCY PLAN		N/A
EP-AA-1009	2	2003-09-18	TMI1	EXELON NUCLEAR RADIOLOGICAL EMERGENCY PLAN ANNEX FOR THREE MILE ISLAND (TMI) STATION		2
EP-AA-1101	1	2003-03-28	TMI1	EP FUNDAMENTALS		N/A
EP-AA-1102	0	2003-03-28	TMI1	ERO FUNDAMENTALS		N/A
EP-AA-120-1001	3	2003-05-09	TMI1	10 CFR 50.54(Q) CHANGE EVALUATION		N/A
EP-AA-120-1002	0	2003-03-28	TMI1	STORM / EVENT RESTORATION		N/A
EP-AA-121-1001	3	2003-05-09	TMI1	AUTOMATED CALL-OUT SYSTEM MAINTENANCE		N/A
EP-AA-122-1001	3	2003-09-12	TMI1	DRILL DEVELOPMENT CONDUCT AND EVALUATION		N/A
EP-AA-122-1002	3	2003-09-12	TMI1	EXERCISE DEVELOPMENT CONDUCT AND EVALUATION		N/A
EP-AA-122-1003	3	2003-09-12	TMI1	SCHEDULING OF DRILLS AND EXERCISES		N/A
EP-AA-122-1004	2	2003-09-12	TMI1	OBJECTIVES AND DEMONSTRATION CRITERIA		2
EP-AA-125-1001	2	2002-12-20	TMI1	EP PERFORMANCE INDICATOR GUIDANCE		2
EP-AA-125-1002	2	2002-12-20	TMI1	ERO PERFORMANCE - PERFORMANCE INDICATORS GUIDANCE		2
EP-AA-125-1003	2	2003-03-28	TMI1	ERO READINESS - PERFORMANCE INDICATORS GUIDANCE		N/A
EP-AA-125-1004	2	2002-12-20	TMI1	EMERGENCY RESPONSE FACILITIES & EQUIPMENT PERFORMANCE INDICATORS GUIDANCE		N/A
EP-AA-125-1005	0	2002-12-20	TMI1	PROBLEM IDENTIFICATION AND RESOLUTION PERFORMANCE INDICATOR GUIDANCE		2
EP-MA-121-1002	0	2003-03-28	TMI1	ALERT NOTIFICATION SYSTEM (ANS) DESCRIPTION TESTING MAINTENANCE AND PERFORMANCE TRENDING PROGRAM		N/A
EP-MA-121-1004	0	2003-03-28	TMI1	EMERGENCY PREPAREDNESS ALERT NOTIFICATION SYSTEM (ANS) CONTROL OF EQUIPMENT & OUTAGES		N/A

SERIES: 0149 EXELON TRAINING AND REFERENCE MATERIAL

PROCEDURE NUMBER	REV	EFFDATE	SITE	PROCEDURE TITLE	TC NUMBER	LEVEL
EP-MA-123-1001	0	2003-07-01	TMI1	KI ASSESSMENT SPREADSHEET TECHNICAL BASIS		N/A
EP-MA-123-1002	0	2003-08-08	TMI1	DOSE ASSESSMENT AND PROTECTIVE ACTION RECOMMENDATION (DAPAR) PROGRAM TECHNICAL BASIS FOR TMI STATION		N/A
EP-MA-124-1001	3	2003-10-10	TMI1	FACILITY INVENTORIES AND EQUIPMENT TESTS		N/A
EP-MA-125-1002	N/A	2001-06-21	TMI1	COLLECTION AND EVALUATION OF DATA FOR INDICATOR R.EP.01 DRILL AND EXERCISE PERFORMANCE		N/A
EP-TM-120-1001	0	2003-09-05	TMI1	MAINTENANCE OF THE TMI DUTY ROSTER		N/A

SERIES: 0151 EXELON PROCESS DESCRIPTIONS

PROCEDURE NUMBER	REV	EFFDATE	SITE	PROCEDURE TITLE	TC NUMBER	LEVEL
EP-AA-10	1	2002-12-06	TMI1	EMERGENCY PREPAREDNESS PROCESS DESCRIPTION		N/A
EP-AA-11	1	2002-12-06	TMI1	OPERATING STATIONS EMERGENCY PREPAREDNESS PROCESS DESCRIPTION		N/A