DUKE ENERGY

McGUIRE NUCLEAR SITE

EMERGENCY PLAN

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SITE VICE PRESIDENT

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McGUIRE NUCLEAR SITE
EMERGENCY PLAN
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B. On-Site Emergency Organization

B.1 Plant Staff Under Emergency Conditions

Figures B-2 - B-5 shows the emergency organization of plant staff personnel for all shifts. The relationship of these personnel to their normal responsibilities and duties is unchanged during an emergency condition.

B.2 Emergency Coordinator

Initial activities at McGuire during any emergency condition are directed by the Shift Manager from the Control Room. The Shift Manager shall assume the functions of the Emergency Coordinator until the arrival of the Station Manager/Designee at which time the Station Manager/Designee will assume the functions of the Emergency Coordinator. The Emergency Coordinator will have the authority and the responsibility to immediately and unilaterally initiate any emergency actions including:

a. Provide protective action recommendations to authorities responsible for implementing off-site emergency measures, implement event classification, notification, and event escalation/de-escalation/termination. THIS AUTHORITY SHALL NOT BE DELEGATED TO OTHER ELEMENTS OF THE EMERGENCY ORGANIZATION EXCEPT FOR EVENT CLASSIFICATIONS WHICH ARE PERFORMED AT THE STATION BY AN EMERGENCY COORDINATOR WHEN EOF IS ACTIVATED.

b. Notification and activation of the Site, Corporate, County/City, North Carolina and the Nuclear Regulatory Commission emergency organizations having a response role.

c. Continued assessment of actual or potential consequences both on-site and off-site throughout the evolution of the emergency condition.

d. Effective implementation of emergency measures in the environs including protective actions for affected areas, implementation of emergency monitoring teams and facilities to evaluate the environmental consequences of the emergency condition, prompt notification and communications with off-site authorities.

e. Continued maintenance of an adequate state of emergency preparedness until the emergency situation has been effectively managed and the site is returned to a normal or safe operating condition.

B.3 Emergency Coordinator (Line of Succession)

The Emergency Coordinator function as described above in paragraph B.2 will later be assumed by the TSC Emergency Coordinator and/or EOF Director at the Emergency Operations Facility as this organization is staffed and ready to take over its functions.

This assumption of the Emergency Coordinator functions will take place for the Alert, Site Area Emergency and General Emergency categories.

IF AT ANY TIME the EOF is Activated,
THEN the following applies:

- Classification of events are performed by either the TSC or Control Room
- Immediate communication to the EOF is required upon upgrade of a classification of an event by either the TSC or Control Room
- Notifications to Offsite Agencies are performed by the EOF
- Protective Action Recommendations (PAR) are performed by the EOF

B.4 Functional Responsibilities of the Emergency Coordinator

The functional responsibilities of the Emergency Coordinator are described in paragraph B.2. Protective Action recommendations to state and local authorities is initially vested with the Shift Manager/ Emergency Coordinator. As the Emergency Operations Facility (EOF) becomes activated, the EOF Director is the person who is responsible for making protective action recommendations.

B.5 Minimum Staffing Requirements

The positions, title and major tasks to be performed by the persons assigned to the functional areas of emergency activity at the site are described in the Emergency Planning Group Manual, Section 1.1. These assignments shall cover the emergency functions in Figure B-1(a/b). The minimum on-shift staffing reflective of 2 Units in operation is as indicated in Figure B-1a. The capability to augment on-shift resources after declaration of an emergency is as indicated in Figure B-1b. The functional tasks to be performed by persons assigned to the areas of emergency activity are as designated in Emergency Planning Group Manual, Section 1.1.

A detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions as specified in Figure B-1.a is located in MNS-OSSA-12212012 Rev: 0.

B.6 On-site Functional Area Interfaces

Figures B-5 - B-7 describe and specify the interfaces between and among the functional areas of emergency activity, licensee headquarters support, local services support, and state/local government response organizations. Figure B-6 is for use prior to activation of the EOF. Figure B-7 is for use after the EOF is established.

B.7 Augmented Support of On-site Emergency Organization

Upon declaration of an Alert, Site Area Emergency or General Emergency, the EOF organization will be alerted and personnel will report to the EOF as soon as possible. The EOF organization is described in Emergency Planning Implementation Procedures. The Communications organization is described in Section G.3.a. Figure B-4 shows the minimum staff required to declare the EOF operational. The EOF will be staffed using 75 minutes as a goal for the minimum staff to be in place and operational.

In addition to the minimum staff shown in Figure B-4, other EOF personnel shown are expected to report to the EOF to augment the minimum staff. This augmentation would occur gradually and would range from a few minutes to a few hours depending on the proximity of the personnel to the EOF.

The organization identified in this section is capable of continuous (24 hours) operations for a protracted period. The individual responsible for assuring continuity of resources is the EOF Director. Each group's operational plan is specified in the Emergency Plan.

B.8 Contractor and Private Organizations

The Institute of Nuclear Power Operations (INPO) serves as a clearinghouse for industry wide support during an emergency. When notified of an emergency situation at a nuclear plant, INPO will provide emergency response as requested. Contact will be made with INPO through the EOF. INPO will be able to provide the following emergency support functions:

a. Assistance to the affected utility in locating sources of emergency manpower and equipment.

b. Analysis of the operational aspects of the incident.

c. Dissemination to member utilities of information concerning the incident.

d. Organization of industry experts who could advise on technical matters.

If requested, one or more suitably qualified members of the INPO staff will report to the EOF Director and will assist in coordinating INPO's response to the emergency.

The State of North Carolina

The response by the State of North Carolina to an emergency development is described in the North Carolina Radiological Emergency Response Plan in Support of McGuire Nuclear Site and in their plan for Catawba Nuclear Site.

The principal state agency for mobilization of State resources to cope with an emergency is the Division of Emergency Management. This agency is supported by the Division of Radiation Protection for radiological assessment and protection functions, and by other State agencies.

The state organization, when it is mobilized as the State Emergency Response Team (SERT), becomes the primary response authority. For an emergency at McGuire, the SERT organization is established in the Emergency Operations Center in Raleigh, N.C.

Nuclear Regulatory Commission

The response provided by the NRC to an emergency developing at a Duke nuclear site is described in the NRC Region II Emergency Plan. The representative of the NRC who would provide input to the EOF Director is the Director of Region II. He is provided work space and a telephone in the EOF.

The role of the NRC in an emergency situation is to provide oversight and recommendations on licensee actions.

County Governments

In an emergency situation at a nuclear site, county governments are immediately notified of the accident. They have the primary responsibility for the protection of the citizens within the county boundaries. The principal Duke Energy contact with county government is through the Emergency Management Director or designee. This contact will be maintained by the TSC until relieved by EOF Offsite Agency Communicator.

It is recognized that the county council, the chief executive of the county, and mayors of local communities have responsibilities in an emergency situation as well. The Government Liaison on the staff of the Joint Information Center serves as the primary Duke Energy contact with these people.

American Nuclear Insurers (ANI)

ANI will be notified of emergency conditions by the EOF Services. ANI's response group would set up claims payments and other such capabilities at facilities appropriate to the emergency.

Contractor

The contractor who may be requested to respond is Westinghouse. Westinghouse will operate from Pittsburgh, Pennsylvania, with a small contingent at the plant.

B.9 Local Agency Support Service

State, local and county agencies responsible for public health and safety work through the Emergency Management Agency Emergency Operations Center in the affected county until the State Emergency Response Team assumes control. The EOF coordinates with the agencies necessary to support the emergency condition. Agencies that have agreed to provide support, as necessary to McGuire Nuclear Site and surrounding areas, are listed below:

B.9.a Law Enforcement, Emergency Traffic Control, Related Police Matters

1. Charlotte-Mecklenburg Police

2. North Carolina Highway Patrol

B.9.b Early Warning or Evacuation of the Populace

1. Department of Emergency Management, Catawba County, (Newton, NC)

2. Department of Emergency Management, Gaston County (Gastonia, NC)

3. Department of Emergency Management, Mecklenburg County (Charlotte, NC)

4. Civil Preparedness Agency of Iredell County (Statesville, NC)

5. Department of Emergency Management, Cabarrus County (Concord, NC)

6. Department of Emergency Management, Lincoln County (Lincolnton NC)

7. North Carolina Department of Public Safety (Raleigh, NC)

B.9.c Radiological Emergency Monitoring Assistance

1. US/DOE Radiological Assistance Team, Savannah River Operations Office (Aiken, SC)

2. North Carolina Department of Environmental Natural Resources, Division of Radiation Protection (Raleigh, NC)

3. Civil Air Patrol, North Carolina Wing (Charlotte, NC)

B.9.d Hospitals, Medical Support

1. Carolinas Medical Center (Charlotte, NC)

2. REACTS Facility, DOE (Oak Ridge, TN)

B.9.e Ambulance Service

1. MEDIC (Cornelius, NC)

B.9.f Fire-Fighting

1. Cornelius Volunteer Fire Department (Cornelius, NC)

2. Huntersville Fire Department (Huntersville, NC)

3. Mecklenburg County Fire Marshall

B.9.g Public Health and Safety, Evaluation of the Radiological Situation.

1. North Carolina Department of Environmental Natural Resources, Division of Radiation Protection (Raleigh, NC)

B.9.h Local, State and Federal Support Responsibilities

Agreements have been made with local, state and federal agencies to provide fire protection, medical support, ambulance service, rescue service and hostile action response. Implementation of the emergency plans of the Emergency Management Agencies of six adjacent counties will provide assistance and logistics support if evacuation of portions of the ten mile EPZ becomes necessary. The emergency plans of the Emergency Management Agencies in Mecklenburg County where the site is located, and in Gaston, Catawba, Iredell, Lincoln and Cabarrus Counties, North Carolina, as they relate to the protection of the public who may be affected by an emergency at McGuire, all address the following aspects:

1. Notification of their own personnel and other agencies involved, including the Sheriff's Department, the Highway Patrol, police, rescue squads, fire departments and the Red Cross.

2. Law enforcement and traffic control.

3. Notification or warning of persons in affected areas

4. Evacuation, as necessary, to designated schools or other public buildings out of the affected area, where shelter, food, overnight accommodations, communications, medical care, etc. would be made available.

5. Assistance and cooperation with related agencies in other counties, Duke Energy, and other state and federal agencies.

NOTE: Summary written agreements with the agencies that have various responsibilities for emergency preparedness support and for emergency response in the public domain are included in the Appendix 5.

FIGURE B-1a
MCGUIRE NUCLEAR SITE
MINIMUM ON-SHIFT ERO STAFFING REQUIREMENTS FOR EMERGENCIES

Functional Area	Major Tasks	Emergency Positions	Shift Staffing
1. Plant Operations and Assessment of Operational Aspects (a)	--	CR Supervisor (SRO) Control Room Operator (RO) Non-Licensed Operator (NLO) WCC SRO	1 4 3 1
2. Emergency Direction and Control	Command and Control	Ops Shift Manager (SRO)	1
3. Notification & Communication	Licensee	Operator (SRO/RO/NLO)	1(b)
	Local/ State	Operator (SRO/RO/NLO)	1(b)
	Federal	Operator (SRO/RO/NLO)	1(b)
4. Radiological Assessment	Dose Assessment	RP Qualified Individual	1
	In-plant Surveys	RP Qualified Individual	1
	Onsite Surveys	RP Qualified Individual	1
	Chemistry	Chemistry Technician	1
5. Plant System Engineering, Repair, and Corrective Actions	Tech Support – OPs	Shift Technical Advisor	1
	– Core Damage	Shift Technical Advisor	1(b)
	Repair and Corrective Actions	Mechanical Maintenance IAE Maintenance	1 2
6. In-Plant PAs	Radiation Protection (such as access control, job coverage and personnel monitoring)	RP Qualified Individual	2(b)
7. Fire Fighting (c)	--	Fire Brigade Lead (RO/SRO) Fire Brigade Member (NLO) Fire Brigade Member (SPOC)	1 3 1(b)
8. 1st Aid and Rescue	--	MERT (d)	2
9. Site Access Control and Accountability	Security & Accountability	SAS Operator Security Personnel	1 (e)
		Minimum # of Personnel:	**25**

(a) The Control Room staff complement is reflective of 2 Units in operation.

(b) May be performed by an individual filling another position provided they are qualified to do the collateral function.

(c) The Fire Brigade requirement of five members is met by using four personnel from Operations (including the Fire Brigade Leader) and one person from SPOC (SLC 16.13.1).

(d) The Medical Emergency Response Team (MERT) is filled by Security Officers.

(e) Per Duke Energy MNS Security Plan.

FIGURE B-1b
MCGUIRE NUCLEAR SITE
MINIMUM AUGMENTED ERO STAFFING REQUIREMENTS FOR EMERGENCIES

Major Functional Area	**Major Task**	**Position, Title or Expertise**	**Capability for Additions****** 45 Min.	75 Min.
Emergency Direction and Control (Emergency Coordinator)**		TSC Emergency Coordinator		1
Notification/Communication	Notify Company Personnel, State, County, Federal Agencies and Maintain Communication	Off-site Agency Communicator		2
Emergency Operations Facility (EOF) Radiological Accident Assessment and Support	EOF Director Dose Assessment Plant Status Access Control Communications Off-site Surveys	Senior Manager Radiological Assessment Manager Accident Assessment Manager Electronic Card Reader Off-site Agency Communicators FMT Members (2 Teams)		1 1 1*** # 2 4*****
Radiological Support and Protective Actions	RP Coverage for Repair/ Corrective Actions, Access Control, Search & Rescue, Radiochemistry, Contaminated Injury Medical Response, Personnel Monitoring, Dosimetry, Firefighting	RP Qualified Individuals		6
	Out of Plant Surveys		1	1
	In-Plant Surveys		1	1
	Dose Assessment	TSC Off-site Dose Assessor		1
	Chem/Radwaste Operations	Radwaste Operator		1
Plant System Engineering, Repair and Corrective Actions	Technical Support Repair and Corrective Actions	Core/Thermal Hydraulics Electrical Mechanical Mechanical Maint. Tech. IAE Technician		1*** 1 1 1 2
Firefighting		Fire Brigade		Local Support
Rescue Operations and First Aid		MERT		Local Support

** Management of the off-site Emergency Response will be assumed by the EOF Director when the Emergency Operations Facility is activated. Management of the on-site Emergency Response is assumed by the Station Manager/alternate acting as the Emergency Coordinator when the Technical Support Center and Operations Support Center are activated.

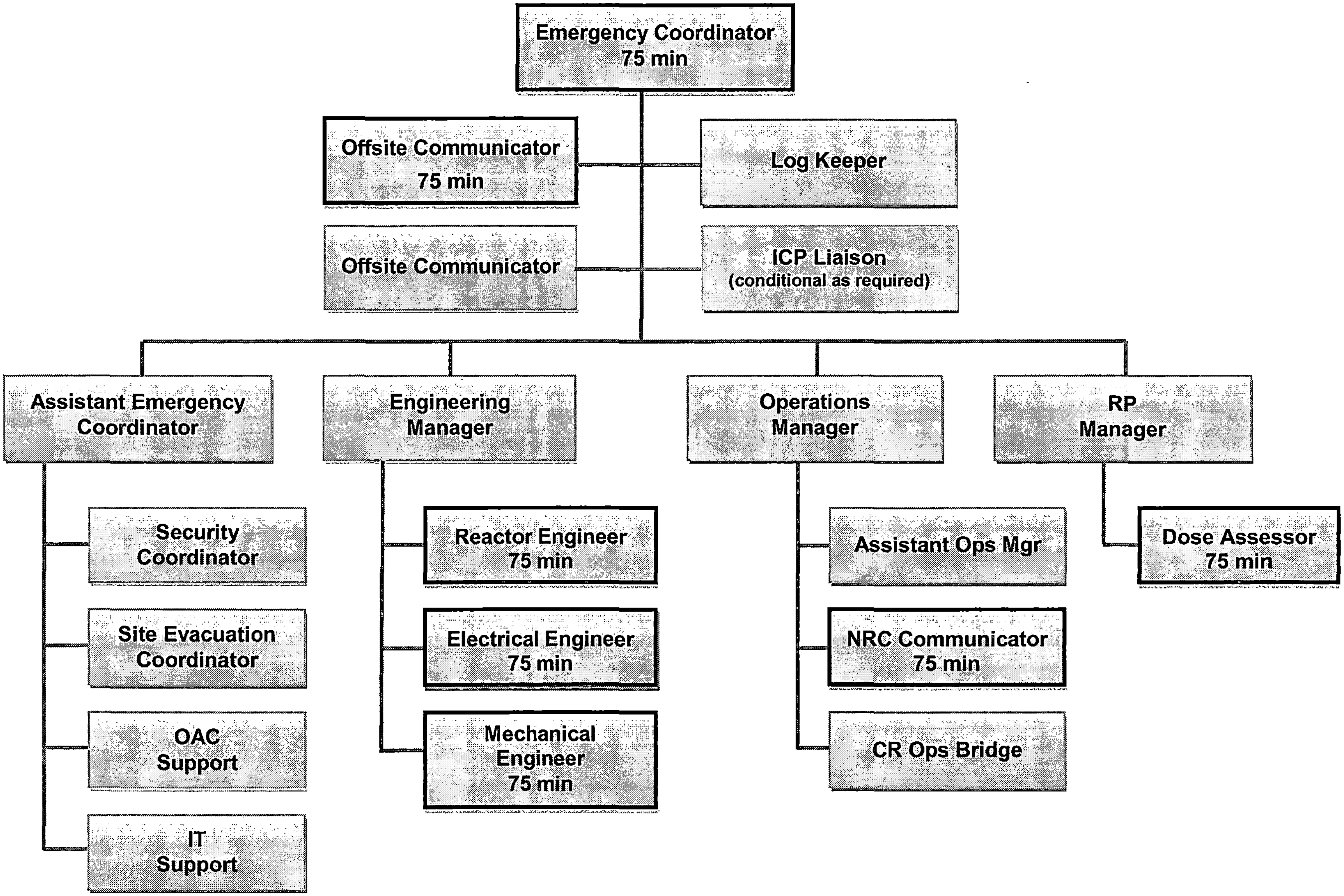
*** The TSC Reactor Engineer and the Accident Assessment Manager in the EOF will provide additional support in the area of core/thermal hydraulics within 75 minutes.

**** Consideration is given to the fact that most McGuire Site staff and support personnel do not choose or are unable to live within a radius of the site which will allow a response time of 30 minutes or less under ideal conditions. Factors such as weather conditions, road capacity and traffic density, limited housing (near site) and the distance to travel from residence to plant site indicate a realistic response time of from a few minutes to 1 hour and 15 minutes for most employees. Consideration is also given to personnel on shift who are qualified and sufficient in number to handle any emergency condition until response personnel begin to arrive onsite in from a few minutes to one (1) hour and 15 minutes.

***** The Field Monitoring Teams will initially report to the Operations Support Center (OSC). If needed, the Field Monitoring Teams will be dispatched from the Operations Support Center (OSC). Once the Emergency Operations Facility (EOF) Field Monitoring Coordinator is ready he/she will assume control of the Field Monitoring Teams. An FMT consists of one RP qualified individual and one vehicle driver.

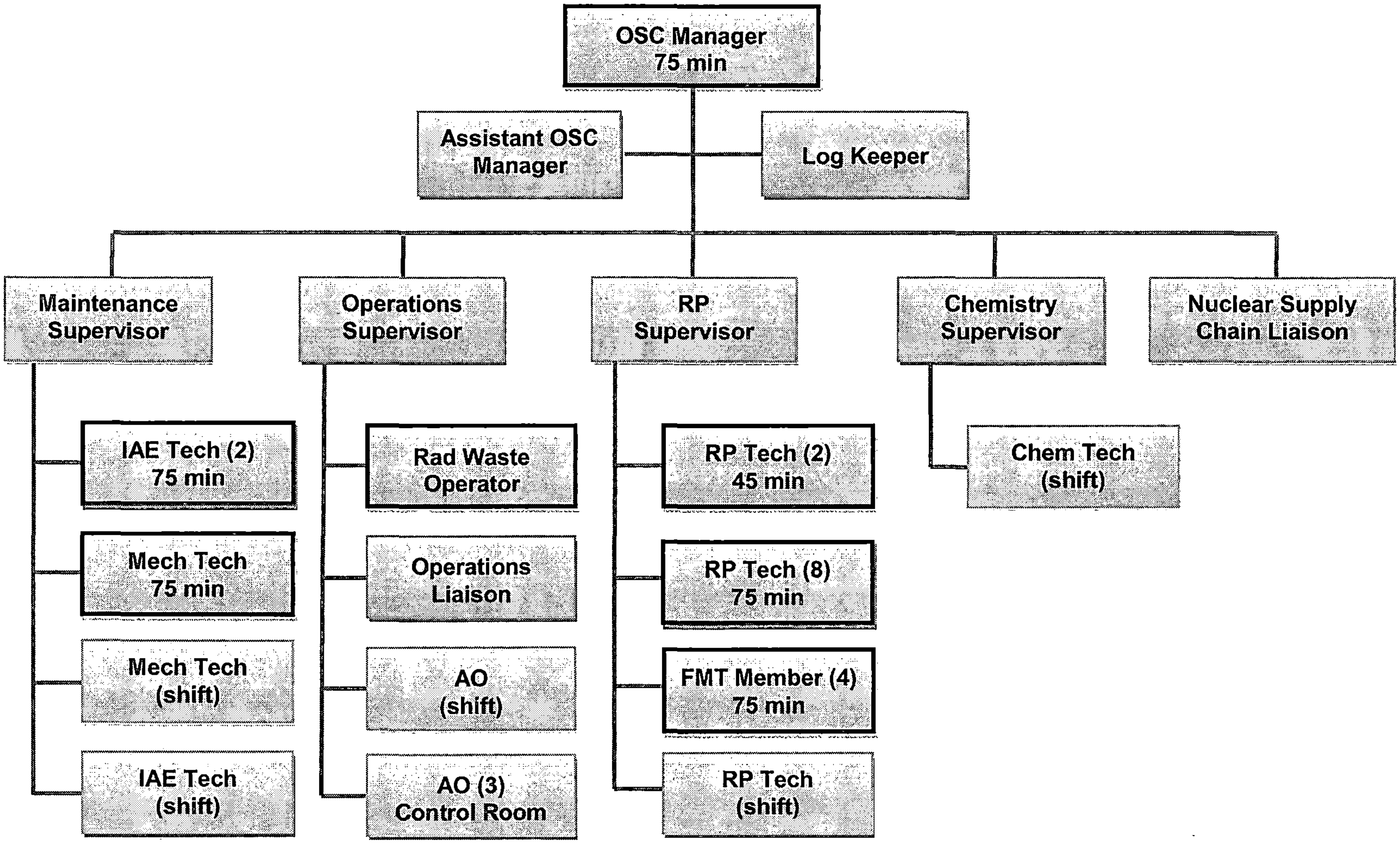
An electronic card reader in conjunction with a posted building security officer fulfills the function for controlling access to the EOF during emergencies.

FIGURE B-2
MCGUIRE NUCLEAR SITE
SITE EMERGENCY ORGANIZATION (TSC)



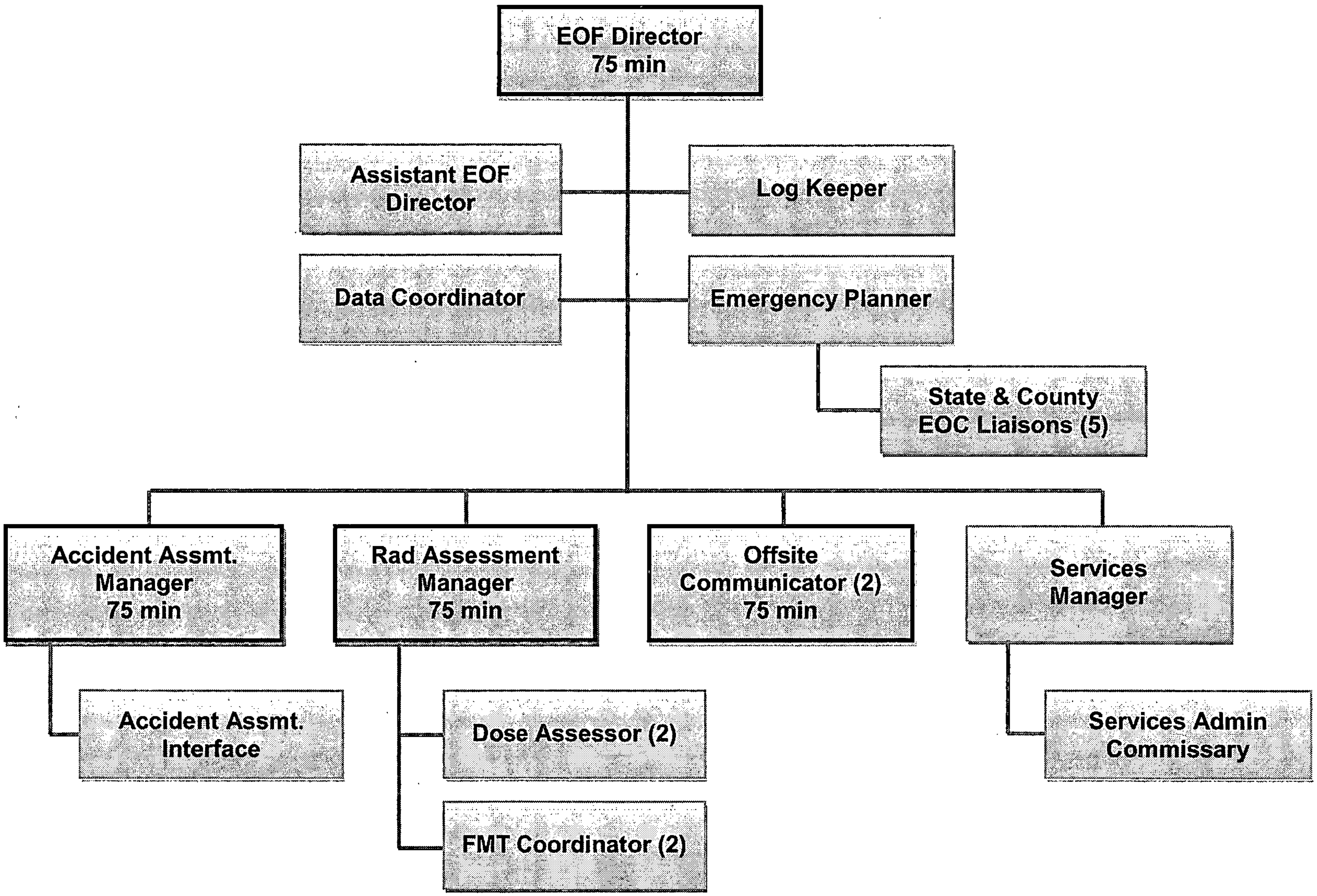
- Bold Boxes indicates minimum staff position

FIGURE B-3
MCGUIRE NUCLEAR SITE
SITE EMERGENCY ORGANIZATION (OSC)



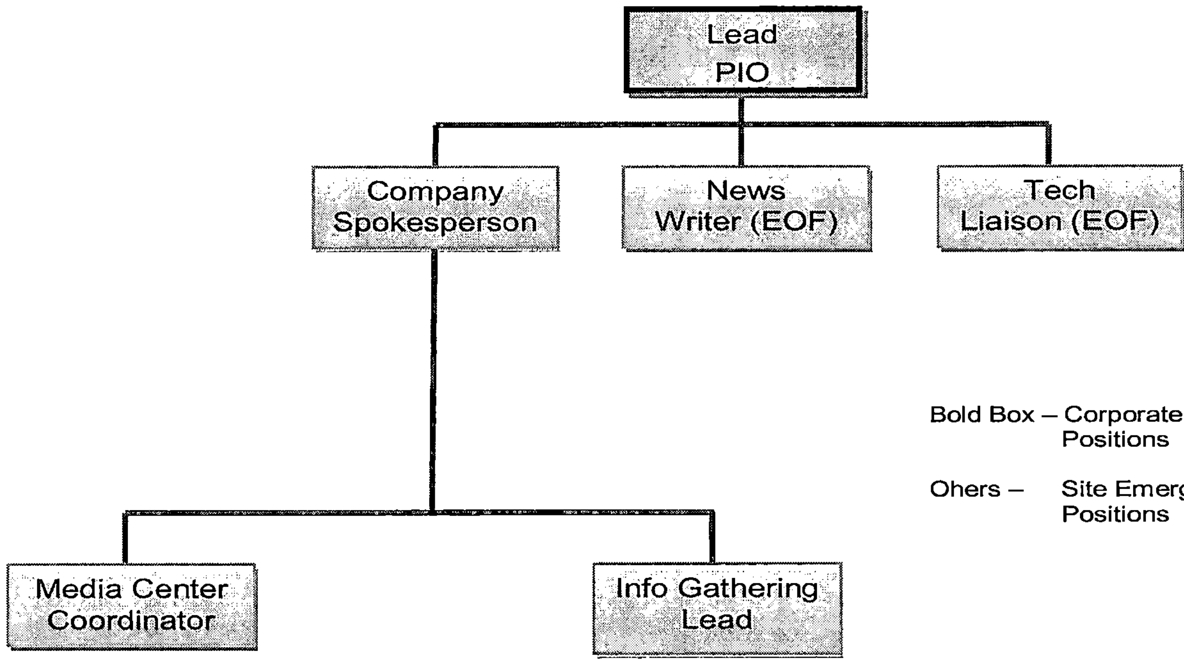
- Bold Boxes indicates minimum staff position

FIGURE B-4
MCGUIRE NUCLEAR SITE
EMERGENCY ORGANIZATION (EOF)



- Bold Boxes indicates minimum staff position

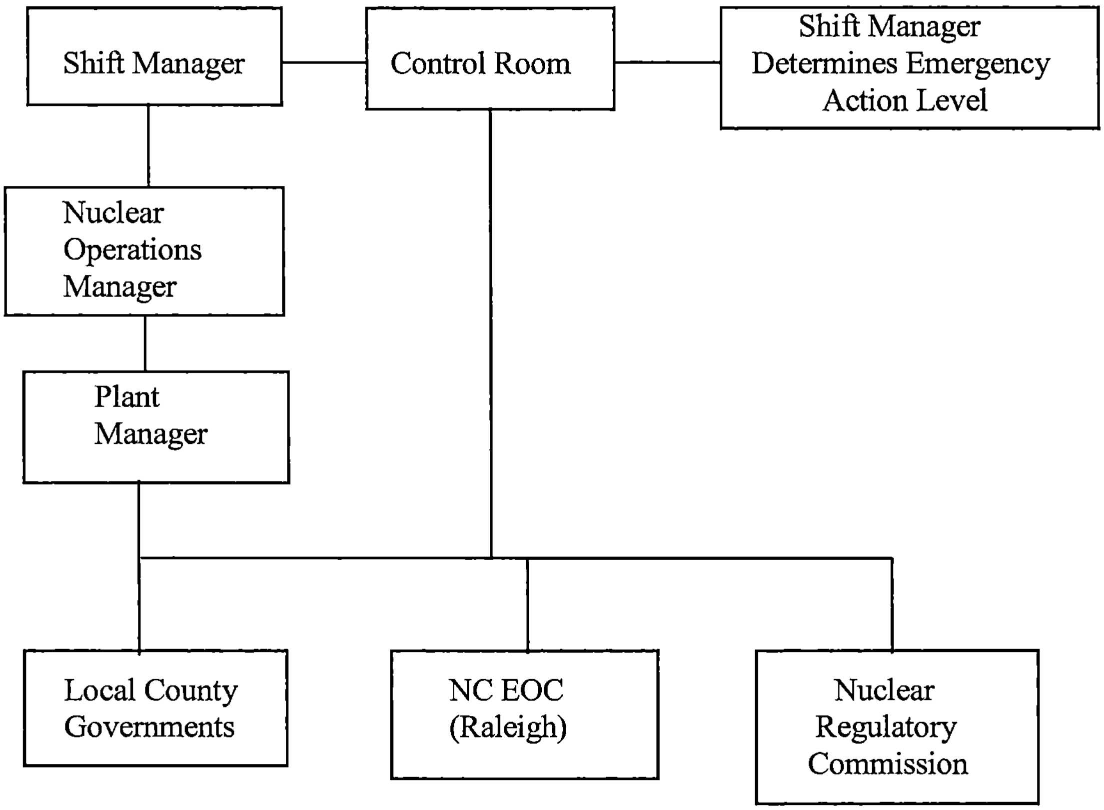
FIGURE B-5
MCGUIRE NUCLEAR SITE
JOINT INFORMATION CENTER (JIC)



Bold Box – Corporate Support Positions

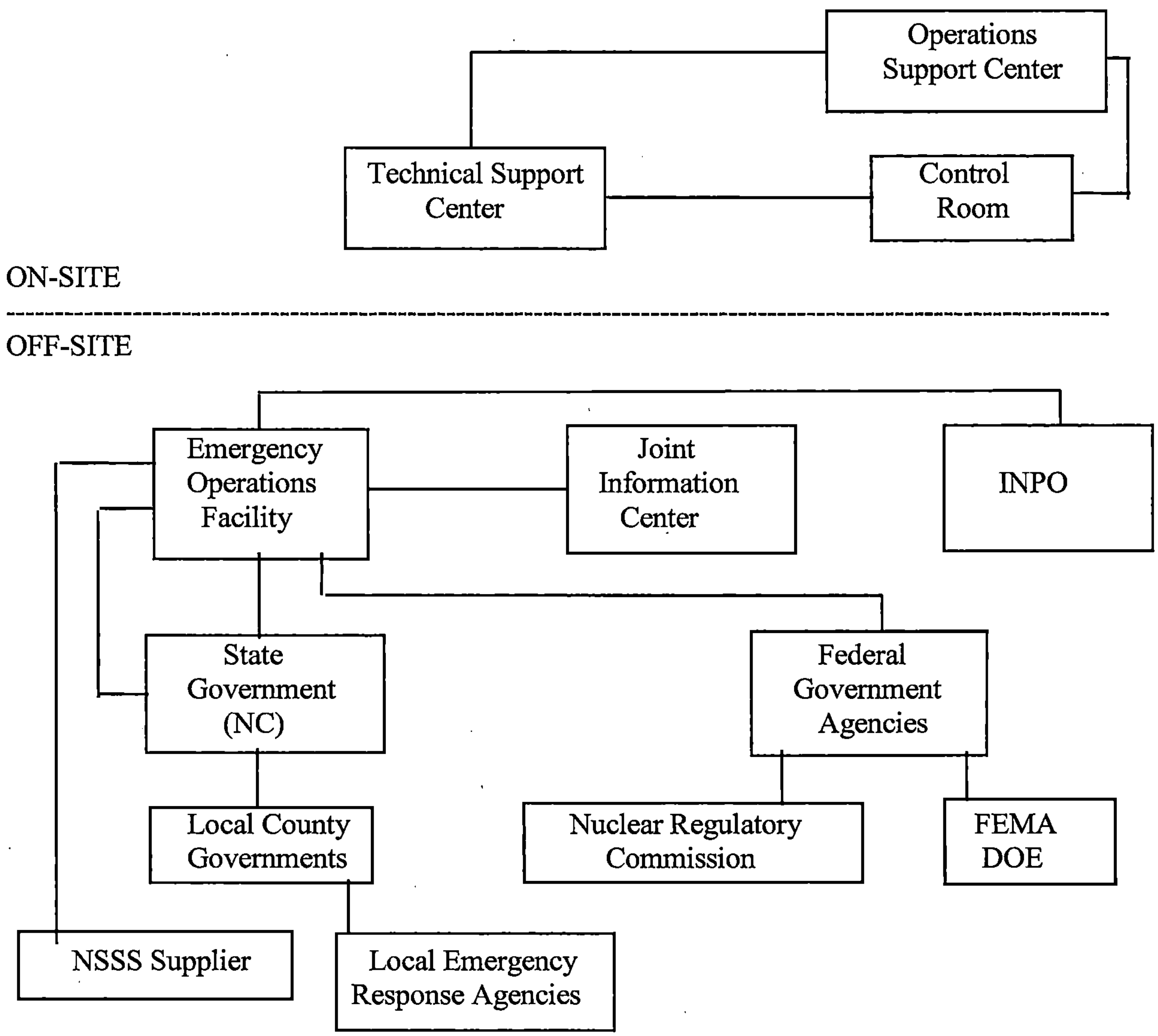
Ohers – Site Emergency Plan Positions

FIGURE B-6
MCGUIRE NUCLEAR SITE
INTER-RELATIONSHIPS OF RESPONSE ORGANIZATIONS
UNUSUAL EVENT*



* Does not require activation of any Emergency Response Organization

FIGURE B-7
MCGUIRE NUCLEAR SITE
INTER-RELATIONSHIPS OF RESPONSE ORGANIZATIONS
ALERT
SITE AREA EMERGENCY
GENERAL EMERGENCY



C. Emergency Response Support and Resources

C.1.a. Individuals Authorized to Request Federal Assistance

Environmental radiological measurements are made in the field by radiological survey teams. This information is used by the Radiological Assessment Manager to confirm environmental projections of doses and dose rates. If necessary to relieve Duke personnel, environmental surveillance support personnel from the DOE Radiological Assistance Plan may be requested by the Radiological Assessment Manager or the EOF Director.

C.1.b. Federal Resources - Arrival Time

The Agreement letter between Duke Energy and DOE - Savannah River is found in Appendix 5. DOE emergency radiological assistance is expected within 3 to 4 hours from a call for these services at McGuire. (Driving and set up time - does not consider use of helicopter or other aerial means of transport).

NRC's full team from Region 2 would be on-site within 7-8 hours from declaration of an emergency at McGuire. Some portions of their team could arrive on-site much earlier by the use of helicopter transport from Atlanta.

C.1.c. Emergency Operations Facility Resources Available to Federal Response Organizations

The following Duke Energy resources are available to support Federal emergency response from DOE - Savannah River.

McGuire Nuclear Site

Airfield - Charlotte/Douglas International airport (~30 - 40 minutes from site)

C.2.a State and County Representation at the Emergency Operations Facility (EOF)

The State and counties in the EPZ around Duke Energy nuclear facilities are provided for as described in Section H.

C.2.b Licensee Representation at the Off-Site EOC's

Provisions have been made to dispatch representatives to principal off-site governmental Emergency Operations Centers (EOC's). The representatives act as liaisons to clarify the information contained in emergency notifications and provide an additional on-site link to the nuclear facility emergency response staffs.

C.3 Radiological Laboratories - Availability and Capability

Laboratory facilities include mobile emergency monitoring capabilities available through the NC Department of Environmental Health and Natural Resources, Division of Radiation Protection; and the DOE Radiological Assistance Team. In addition, the site has 2 emergency vehicles set up to provide monitoring capability. Fixed facilities are available for gross counting and spectral analysis in the site counting laboratory and at the nearby Duke Energy Applied Sciences Center (1/2 mile). Other facilities within the Duke System at Catawba Nuclear Site (45 miles) and at Oconee Nuclear Site (160 miles) could provide further analysis support within a short period of time (1-4 hours). The above radiological laboratories are available on a 24 hour a day basis and could provide their services and equipment on demand.

C.4 Emergency Support From Other Organizations

Other support can be provided by:

- INPO Fixed Nuclear Facility, Voluntary Assistance Agreement Signatories
- DOE Savannah River
- Area Hospitals (see section B.9)
- Volunteer Fire Departments (see section B.9)
- Radiation Emergency Assistance Center/Training Site (REAC/TS)

EMERGENCY ACTION LEVEL TECHNICAL BASES

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SECTION PAGE

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for McGuire Nuclear Station (MNS). It should be used to facilitate review of the MNS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of RP/0/A/5700/000 Classification of Emergency, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Coordinator/EOF Director in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Coordinator refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q).

2.0 DISCUSSION

2.1 Background

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

In 1992, the NRC endorsed NUMARC/NESP-007 "Methodology for Development of Emergency Action Levels" as an alternative to NUREG-0654 EAL guidance.

NEI 99-01 (NUMARC/NESP-007) Revisions 4 and 5 were subsequently issued for industry implementation. Enhancements over earlier revisions included:

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

- Initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations (ISFSIs).

- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Subsequently, Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805) (ref.4.1.1), MNS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

Fission product barrier thresholds represent threats to the defense in depth design concept that precludes the release of radioactive fission products to the environment. This concept relies on multiple physical barriers, any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment.

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

The primary fission product barriers are:

A. Fuel Clad (FC): The Fuel Clad Barrier is the zircalloy tubes that contain the fuel pellets.

B. Reactor Coolant System (NCS): The NCS Barrier includes the NCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. Containment (CMT): The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency

2.3 Fission Product Barrier Classification Criteria

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

Alert:

Any loss or any potential loss of either Fuel Clad or NCS barrier *Site*

Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

The MNS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby, Startup, or Power Operation mode.
  - EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or No Mode.

The purpose of the groups is to avoid review of hot condition EALs when the plant is in a cold condition and avoid review of cold condition EALs when the plant is in a hot condition. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL-user for a given plant condition, reduces EAL-user reading burden and, thereby, speeds identification of the EAL that applies to the emergency.

- Within each group, assignment of EALs to categories and subcategories:

Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. The MNS EAL categories are aligned to and represent the NEI 99-01"Recognition Categories." Subcategories are used in the MNS scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The MNS EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – Emergency Coordinator Judgment
E – Independent Spent Fuel Storage Installation (ISFSI)	1 – Confinement Boundary
Hot Conditions:	
S – **S**ystem Malfunction	1 – Loss of Essential AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – NCS Activity 5 – NCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Containment Failure 9 – Hazardous Event Affecting Safety Systems
F – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – NCS Level 2 – Loss of Essential AC Power 3 – NCS Temperature 4 – Loss of Vital DC Power 5 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

The primary tool for determining the emergency classification level is the EAL Classification Matrix. The user of the EAL Classification Matrix may (but is not required to) consult the EAL Technical Bases Document in order to obtain additional information concerning the EALs under classification consideration. The user should consult Section 3.0 and Attachments 1 & 2 of this document for such information.

2.5 Technical Bases Information

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

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

Site-specific description of the generic IC given in NEI 99-01 Rev. 6.

EAL Identifier (enclosed in rectangle)

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

1. First character (letter): Corresponds to the EAL category as described above (R, C, H, S, F or E)
2. Second character (letter): The emergency classification (G, S, A or U)

   G = General Emergency
   S = Site Area Emergency
   A = Alert
   U = Unusual Event

3. Third character (number): Subcategory number within the given category. Subcategories are sequentially numbered beginning with the number one (1). If a category does not have a subcategory, this character is assigned the number one (1).
4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

Exact wording of the EAL as it appears in the EAL Classification Matrix

Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operations, 2 - Startup, 3 – Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, NM - No Mode, or All. (See Section 2.6 for operating mode definitions)

Definitions:

If the EAL wording contains a defined term, the definition of the term is included in this section. These definitions can also be found in Section 5.1.

Basis:

A basis section that provides MNS-relevant information concerning the EAL as well as a description of the rationale for the EAL as provided in NEI 99-01 Rev. 6.

MNS Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.7)

1 Power Operation

$K_{eff} \geq 0.99$ and reactor thermal power > 5%

2 Startup

$K_{eff} \geq 0.99$ and reactor thermal power $\leq$ 5%

3 Hot Standby

$K_{eff} < 0.99$ and average coolant temperature $\geq$ 350°F

4 Hot Shutdown

$K_{eff} < 0.99$ and average coolant temperature 350°F > T_{avg} > 200 °F

5 Cold Shutdown

$K_{eff} < 0.99$ and average coolant temperature $\leq$ 200°F

6 Refueling

One or more reactor vessel head closure bolts are less than fully tensioned

NM No mode

Reactor vessel contains no irradiated fuel

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Coordinator/EOF Director must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.12).

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, verification could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel.

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

3.1.3 Imminent Conditions

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency Coordinator/EOF Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. If an ongoing radiological release is detected and the release start time is unknown, it should be assumed that the release duration specified in the IC/EAL has been exceeded, absent data to the contrary.

3.1.4 Planned vs. Unplanned Events

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that: 1) the activity proceeds as planned, and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 § CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, NCS leak rate calculation, etc.). For these EALs, the EAL wording or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

3.1.6 Emergency Coordinator Judgment

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. The NEI 99-01 EAL scheme provides the Emergency Coordinator/EOF Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Emergency Coordinator/EOF Director will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, the associated IC is likewise met, the emergency classification process "clock" starts, and the ECL must be declared in accordance with plant procedures no later than fifteen minutes after the process "clock" started.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.14).

3.2.1 Classification of Multiple Events and Conditions

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. For example:

- If an Alert EAL and a Site Area Emergency EAL are met, whether at one unit or at two different units, a Site Area Emergency should be declared.

There is no "additive" effect from multiple EALs meeting the same ECL. For example:

- If two Alert EALs are met, whether at one unit or at two different units, an Alert should be declared.

Related guidance concerning classification of rapidly escalating events or conditions is provided in Regulatory Issue Summary (RIS) 2007-02, *Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events* (ref. 4.1.2).

3.2.2 Consideration of Mode Changes During Classification

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the Emergency Coordinator/EOF Director must remain alert to events or conditions that could lead to meeting or exceeding an EAL within a relatively short period of time (i.e., a change in the ECL is IMMINENT). If, in the judgment of the Emergency Coordinator/EOF Director, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

3.2.4 Emergency Classification Level Upgrading and Downgrading

An ECL may be downgraded when the event or condition that meets the highest IC and EAL no longer exists, and other site-specific downgrading requirements are met. If downgrading the ECL is deemed appropriate, the new ECL would then be based on a lower applicable IC(s) and EAL(s). The ECL may also simply be terminated.

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02 (ref. 4.1.2).

3.2.5 Classification of Short-Lived Events

Event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. By their nature, some of these events may be short-lived and, thus, over before the emergency classification assessment can be completed. If an event occurs that meets or exceeds an EAL, the associated ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include an earthquake or a failure of the reactor protection system to automatically trip the reactor followed by a successful manual trip.

3.2.6 Classification of Transient Conditions

Many of the ICs and/or EALs employ time-based criteria. These criteria will require that the IC/EAL conditions be present for a defined period of time before an emergency declaration is warranted. In cases where no time-based criterion is specified, it is recognized that some transient conditions may cause an EAL to be met for a brief period of time (e.g., a few seconds to a few minutes). The following guidance should be applied to the classification of these conditions.

<u>EAL momentarily met during expected plant response</u> - In instances where an EAL is briefly met during an expected (normal) plant response, an emergency declaration is not warranted provided that associated systems and components are operating as expected, and operator actions are performed in accordance with procedures.

<u>EAL momentarily met but the condition is corrected prior to an emergency declaration</u> – If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example:

An ATWS occurs and the high pressure ECCS systems fail to automatically start. RPV level rapidly decreases and the plant enters an inadequate core cooling condition (a potential loss of both the fuel clad and NCS barriers). If an operator manually starts a high pressure ECCS system in accordance with an EOP step and clears the inadequate core cooling condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period (process clock) is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event.

Emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations when an operator is able to take a successful corrective action prior to the Emergency Coordinator/EOF Director completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

3.2.7 After-the-Fact Discovery of an Emergency Event or Condition

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event orcondition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

3.2.8 Retraction of an Emergency Declaration

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022 (ref. 4.1.3).

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805

4.1.2 RIS 2007-02 Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events, February 2, 2007.

4.1.3 NUREG-1022 Event Reporting Guidelines: 10CFR50.72 and 50.73

4.1.4 10 § CFR 50.72 Immediate Notification Requirements for Operating Nuclear Power Reactors

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 MNS UFSAR Figure 2-4 Plot Plan and Site Area

4.1.7 Technical Specifications Table 1.1-1 Modes

4.1.8 PT/1(2)/A/4200/002 C (Containment Closure)

4.1.9 PRO-NGGC-0201 NGG Procedure Writers Guide

4.1.10 NSIR/DPR-ISG-01 Interim Staff Guidance, Emergency Planning for Nuclear Power Plants

4.1.11 MNS ISFSI Certificate of Compliance

4.1.12 MNS Emergency Plan

4.1.13 MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary

4.2 Implementing

4.2.1 RP/0/A/5700/000 Classification of Emergency

4.2.2 NEI 99-01 Rev. 6 to MNS EAL Comparison Matrix

4.2.3 MNS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

Selected terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

Alert

Events are in progress, or have occurred, which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action. Any releases are expected to be small fractions of the EPA Protective Action Guideline exposure levels.

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the MNS ISFSI, Confinement Boundary is defined as the Transportable Storage Cask (TSC) for TN, UMS and MAGNASTOR storage systems.

Containment Closure

The procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to MNS, Containment Closure is established when the requirements of PT/1(2)/A/4200/002 C are met (ref. 4.1.8).

Emergency Action Level (EAL)

A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Emergency Classification Level (ECL)

One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are:

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

EPA PAGs

Environment Protection Agency Protective Action Guidelines. The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires MNS to recommend protective actions for the general public to offsite planning agencies.

Explosion

A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

Faulted

The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

Fire

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

Fission Product Barrier Threshold

A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

Flooding

A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

General Emergency

Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or hostile actions that result in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Hostage

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

Hostile Action

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

Hostile Force

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

Imminent

The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Impede(d)

Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Initiating Condition (IC)

An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

Intrusion

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

Maintain

Take appropriate action to hold the value of an identified parameter within specified limits.

Normal Levels

As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.

Owner Controlled Area

Area outside the PROTECTED AREA fence that immediately surrounds the plant. The site property owned by, or otherwise under the control of, Duke Energy.

Projectile
An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

Protected Area

An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in MNS UFSAR Figure 2-4 Plot Plan and Site Area (ref. 4.1.6).

NCS Intact

The NCS should be considered intact when the NCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).

Refueling Pathway

The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway.

Ruptured

The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Restore

Take the appropriate action required to return the value of an identified parameter to the applicable limits

Safety System

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Security Condition

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

Site Area Emergency

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the SITE BOUNDARY.

Site Boundary

Area as depicted in MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary (ref. 4.1.13).

Unisolable

An open or breached system line that cannot be isolated, remotely or locally.

Unplanned

A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Unusual Event

Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Valid

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

Visible Damage

Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
° .... Degrees
AC .... Alternating Current
AP .... Abnormal Operating Procedure
ATWS .... Anticipated Transient Without Scram
MNS .... McGuire Nuclear Station
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CSFST .... Critical Safety Function Status Tree
DBA .... Design Basis Accident
DC .... Direct Current
EAL .... Emergency Action Level
EC .... Emergency Coordinator
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
ERG .... Emergency Response Guideline
EPIP .... Emergency Plan Implementing Procedure
ESF .... Engineered Safety Feature
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
GE .... General Emergency
IC .... Initiating Condition
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI .... Independent Spent Fuel Storage Installation
K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident

LWR...Light Water Reactor
MPC................................ Maximum Permissible Concentration/Multi-Purpose Canister
MSIV...Main Steam Isolation Valve
MSL ....Main Steam Line
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MW .... Megawatt
NCS ....Reactor Coolant System
NEI ....Nuclear Energy Institute
NESP.. National Environmental Studies Project
NPP ....Nuclear Power Plant
NRC..Nuclear Regulatory Commission
NSSS...Nuclear Steam Supply System
NORAD.. North American Aerospace Defense Command
(NO)UE...Notification of Unusual Event
OBE ....Operating Basis Earthquake
OCA...Owner Controlled Area
ODCM...Off-site Dose Calculation Manual
ORO ....Offsite Response Organization
PA... Protected Area
PAG ....Protective Action Guideline
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR...Pressurized Water Reactor
PSIG ....Pounds per Square Inch Gauge
R .... Roentgen
Rem, rem, REM ....Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RPS ....Reactor Protection System
RV ....Reactor Vessel
RVLIS .... Reactor Vessel Level Indicating System
SAR...Safety Analysis Report
SBGTS ....Stand-By Gas Treatment System
SBO .... Station Blackout
SCBA.. Self-Contained Breathing Apparatus
SG ....Steam Generator

SI .... Safety Injection
SLC .... Selected Licensee Commitment
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
SSF .... Standby Shutdown Facility
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
WOG .... Westinghouse Owners Group

6.0 MNS-TO-NEI 99-01 Rev. 6 EAL CROSS-REFERENCE

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

MNS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
RU1.1	AU1	1, 2
RU1.2	AU1	3
RU2.1	AU2	1
RA1.1	AA1	1
RA1.2	AA1	2
RA1.3	AA1	3
RA1.4	AA1	4
RA2.1	AA2	1
RA2.2	AA2	2
RA2.3	AA2	3
RA3.1	AA3	1
RA3.2	AA3	2
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	3
RS2.1	AS2	1
RG1.1	AG1	1
RG1.2	AG1	2
RG1.3	AG1	3
RG2.1	AG2	1

MNS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
CU1.1	CU1	1
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	3
CG1.1	CG1	2
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4

MNS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1
HA1.2	HA1	2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG1.1	HG1	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
SU8.1	SU7	1, 2
SA1.1	SA1	1

MNS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
SA3.1	SA2	1
SA6.1	SA5	1
SA9.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1
EU1.1	E-HU1	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier Matrix and Basis

Category R – Abnormal Rad Release / Rad Effluent

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

Conditions indicative of a loss of adequate shielding or damage to irradiated fuel may preclude access to vital plant areas or result in radiological releases that warrant emergency classification.

3. Area Radiation Levels

Sustained general area radiation levels which may preclude access to areas requiring continuous occupancy also warrant emergency classification.

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the SLC limits for 60 minutes or longer

EAL:

RU1.1	**Unusual Event**
Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent Noble Gas Low	1(2)EMF36L	---	---	4.85E+6 cpm	3.10E+3 cpm
Gaseous	Unit Vent Noble Gas High	1(2)EMF36H	2.61E+4 cpm	2.61E+3 cpm	2.70E+2 cpm	---
Liquid	Liquid Waste Effluent Line High	EMF49H	---	---	---	2.15E+2 cpm
Liquid	CVUCDT High	1(2)EMF44H	---	---	---	4.29E+2 cpm

Mode Applicability:
All

Definition(s):
None

Basis:

The column "UE" gaseous and liquid release values in Table R-1 represent two times the appropriate SLC release rate limits associated with the specified monitors (ref. 2, 3, 4).

Gaseous Releases

Instrumentation that may be used to assess this EAL is listed below (ref. 1):

- Unit Vent Noble Gas Low Monitor – 1(2)EMF36L

Liquid Releases

Instrumentation that may be used to assess this EAL is listed below (ref. 1):

- Liquid Waste Effluent Line High Monitor – EMF49H (batch release)
- CVUCDT High Monitor – 1(2)EMF44H

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

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

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

Escalation of the emergency classification level would be via IC RA1.

MNS Basis Reference(s):

1. MNS ODCM Section 3.0 Setpoint Calculations
2. MNS-SLC 16.11.1 Liquid Effluents - Concentration
3. MNS-SLC 16.11.6 Dose Rate - Gaseous Effluents
4. EP-EALCALC-MNS-1401 MNS Radiological Effluent EAL Values, Rev. 0
5. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the SLC limits for 60 minutes or longer.

EAL:

RU1.2	**Unusual Event**
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x SLC limits for ≥ 60 min. (Notes 1, 2)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All **Definition(s):**

None

Basis:

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

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

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

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

Escalation of the emergency classification level would be via IC RA1.

MNS Basis Reference(s):

1. MNS Offsite Dose Calculation Manual
2. MNS-SLC 16.11.1 Liquid Effluents - Concentration
3. MNS-SLC 16.11.6 Dose Rate - Gaseous Effluents
4. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.1 Alert

Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4 The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent Noble Gas Low	1(2)EMF36L	---	---	4.85E+6 cpm	3.10E+3 cpm
Gaseous	Unit Vent Noble Gas High	1(2)EMF36H	2.61E+4 cpm	2.61E+3 cpm	2.70E+2 cpm	---
Liquid	Liquid Waste Effluent Line High	EMF49H	---	---	---	2.15E+2 cpm
Liquid	CVUCDT High	1(2)EMF44H	---	---	---	4.29E+2 cpm

Mode Applicability:

All **Definition(s):**

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 10 mRem TEDE
- 50 mRem CDE Thyroid

The column "ALERT" gaseous effluent release values in Table R-1 correspond to calculated doses of 1% (10% of the SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 2).

Instrumentation that may be used to assess this EAL is Unit Vent Noble Gas Low Monitor – 1(2)EMF36L and Unit Vent Noble Gas High Monitor – 1(2)EMF36H (ref. 1).

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

MNS Basis Reference(s):

1. MNS ODCM Section 3.0 Setpoint Calculations
2. EP-EALCALC-MNS-1401 MNS Radiological Effluent EAL Values, Rev. 0
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.2	**Alert**
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary.

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2)

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have

stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

MNS Basis Reference(s):

1. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.3	**Alert**
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary.

Basis:

Dose assessments based on liquid releases are performed per Offsite Dose Calculation Manual (ref. 1).

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

MNS Basis Reference(s):

1. MNS Offsite Dose Calculation Manual
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.4 Alert

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary.

Basis:

AD-EP-MNS-0203 MNS Site Specific Field Monitoring and AD-EP-ALL-0203 Field Monitoring During Declared Emergency provide guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions

alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Escalation of the emergency classification level would be via IC RS1.

MNS Basis Reference(s):

1. AD-EP-MNS-0203 MNS Site Specific Field Monitoring
2. AD-EP-ALL-0203 Field Monitoring During Declared Emergency.
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.1 Site Area Emergency

Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent Noble Gas Low	1(2)EMF36L	---	---	4.85E+6 cpm	3.10E+3 cpm
Gaseous	Unit Vent Noble Gas High	1(2)EMF36H	2.61E+4 cpm	2.61E+3 cpm	2.70E+2 cpm	---
Liquid	Liquid Waste Effluent Line High	EMF49H	---	---	---	2.15E+2 cpm
Liquid	CVUCDT High	1(2)EMF44H	---	---	---	4.29E+2 cpm

Mode Applicability:

All **Definition(s):**

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 100 mRem TEDE
- 500 mRem CDE Thyroid

The column "SAE" gaseous effluent release value in Table R-1 corresponds to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

Instrumentation that may be used to assess this EAL is Unit Vent Noble Gas High Monitor – 1(2)EMF36H (ref 2).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RG1.

MNS Basis Reference(s):

1. EP-EALCALC-MNS-1401 MNS Radiological Effluent EAL Values, Rev. 0
2. MNS ODCM Section 3.0 Setpoint Calculations
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary.

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RG1.

MNS Basis Reference(s):

1. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.3 Site Area Emergency

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary.

Basis:

AD-EP-MNS-0203 MNS Site Specific Field Monitoring and AD-EP-ALL-0203 Field Monitoring During Declared Emergency provide guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Escalation of the emergency classification level would be via IC RG1.

MNS Basis Reference(s):

1. AD-EP-MNS-0203 MNS Site Specific Field Monitoring Information
2. AD-EP-ALL-0203 Field Monitoring During Declared Emergency.
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.1 General Emergency

Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent Noble Gas Low	1(2)EMF36L	----	----	4.85E+6 cpm	3.10E+3 cpm
Gaseous	Unit Vent Noble Gas High	1(2)EMF36H	2.61E+4 cpm	2.61E+3 cpm	2.70E+2 cpm	----
Liquid	Liquid Waste Effluent Line High	EMF49H	----	----	----	2.15E+2 cpm
Liquid	CVUCDT High	1(2)EMF44H	----	----	----	4.29E+2 cpm

Mode Applicability:

All **Definition(s):**

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 1000 mRem TEDE
- 5000 mRem CDE Thyroid

The column "GE" gaseous effluent release values in Table R-1 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

Instrumentation that may be used to assess this EAL is the Unit Vent Noble Gas High Monitor 1(2)EMF36H (ref 2).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

MNS Basis Reference(s):

1. EP-EALCALC-MNS-1401 MNS Radiological Effluent EAL Values, Rev. 0
2. MNS ODCM Section 3.0 Setpoint Calculations
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary.

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

MNS Basis Reference(s):

1. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.3 General Emergency

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 1,000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary.

Basis:

AD-EP-MNS-0203 MNS Site Specific Field Monitoring and AD-EP-ALL-0203 Field Monitoring During Declared Emergency provide guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

MNS Basis Reference(s):

1. AD-EP-MNS-0203 MNS Site Specific Field Monitoring
2. AD-EP-ALL-0203 Field Monitoring During Declared Emergency.
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication

AND

UNPLANNED rise in corresponding area radiation levels as indicated by **EITHER** of the following radiation monitors:

- 1EMF17 (2EMF4) Spent Fuel Building Refueling Bridge
- 1EMF16 (2EMF3) Containment Building Refueling Bridge (Mode 6)

Mode Applicability:

All

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY-. The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway.

Basis:

The spent fuel pool low water level alarm setpoint is OAC point M1(2)D2937 (ref. 1). Water level restoration instructions are performed in accordance with AOPs (ref. 2, 3).

The specified radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 2, 3). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING PATHWAY level are not classifiable under this EAL. 1EMF16 (2EMF3) Containment Building Refueling Bridge monitors are only operable in Mode 6 (Refueling).

When the spent fuel pool and reactor cavity are connected, there could exist the possibility of uncovering irradiated fuel. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the reactor vessel and spent fuel pool.

This IC addresses a decrease in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an unplanned loss of water level.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RA2.

MNS Basis Reference(s):

1. OP/1(2)/A/6102/001
2. AP/1(2)/A/5500/40 Loss of Refueling Cavity Level
3. AP/1(2)/A/5500/41 Loss of Spent Fuel Cooling or Level
4. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.1	**Alert**
Uncovery of irradiated fuel in the REFUELING PATHWAY	

Mode Applicability:

All

Definition(s):

REFUELING PATHWAY-. The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway.

Basis:

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This EAL escalates from RU2.1 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil- off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RS1 or RS2.

MNS Basis Reference(s):

1. AP/1(2)/A/5500/040 Loss of Refueling Cavity Level
2. AP/1(2)/A/5500/041 Loss of Spent Fuel Cooling or Level
3. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

A Trip 2 radiation alarm on **any** of the following radiation monitor indications:

- 1EMF17 (2EMF4) Spent Fuel Building Refueling Bridge
- 1EMF16 (2EMF3) Containment Building Refueling Bridge (Mode 6)
- 1EMF42 (2EMF42) Fuel Building Ventilation
- 1EMF39 (2EMF39) Containment Gas

Mode Applicability:

All **Definition(s):**

None

Basis:

The specified radiation monitors are those expected to see increased area radiation levels as a result of damage to irradiated fuel (ref. 1). 1EMF16 (2EMF3) Containment Building Refueling Bridge monitors are only operable in Mode 6 (Refueling).

The Trip 2 alarm setpoints for the radiation monitors are set to be indicative of significant increases in area and/or airborne radiation (ref. 2).

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This EAL applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EAL EU1.1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

Escalation of the emergency classification level would be via IC RS1 or RS2.

MNS Basis Reference(s):

1. AP/1(2)/A/5500/25 Spent Fuel Damage
2. HP/0/B/1003/008 Determination of Radiation Monitor Setpoints (EMFs)
3. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.3	**Alert**
Spent fuel pool level ≤ -15 ft. (756 ft. ele.) (KFP5350 or NVPG6530)	

Mode Applicability:

All **Definition(s):**

None

Basis:

Post-Fukushima order EA-12-051 (ref.1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3).

The SFP level instruments consist of a primary channel (1(2)KFP5350) and back-up channel (1(2)NVPG6530) each spanning approximately 30 ft. (-25 ft. – +5 ft.) (745 ft. ele. – 775 ft. ele.). Level 2 is a SFP level of -15 ft. (756' ft. ele.) or approximately 10 ft. above the top of the SFP racks (ref. 2, 3).

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the emergency classification level would be via IC RS1.

MNS Basis Reference(s):

1. NRC EA-12-051 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. MNS-14-023 Second Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)
3. Engineering Change Packages #109073 and #109074
4. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level at the top of the fuel racks

EAL:

RS2.1	**Site Area Emergency**
Spent fuel pool level ≤ -25 ft. (746 ft. ele.) (KFP5350 or NVPG6530)	

Mode Applicability:

All **Definition(s):**

None

Basis:

Post-Fukushima order EA-12-051 (ref.1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3).

The SFP level instruments consist of a primary channel (1(2)KFP5350) and back-up channel (1(2)NVPG6530) each spanning approximately 30 ft. (-25 ft. – +5 ft.) (745 ft. ele. – 775 ft. ele.). Level 3 is a SFP level of -25 ft. (746' ft. ele.) or approximately the top of the SFP racks (ref. 2, 3).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

It is recognized that this IC would likely not be met until well after another Site Area Emergency IC was met; however, it is included to provide classification diversity.

Escalation of the emergency classification level would be via IC RG1 or RG2.

MNS Basis Reference(s):

1. NRC EA-12-051 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. MNS-14-023 Second Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)
3. Engineering Change Packages #109073 and #109074
4. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer

EAL:

RG2.1	**General Emergency**
Spent fuel pool level cannot be restored to > -25 ft. (746 ft. ele.) (KFP5350 or NVPG6530) for ≥ 60 min. (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 (ref.1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3).

The SFP level instruments consist of a primary channel (1(2)KFP5350) and back-up channel (1(2)NVPG6530) each spanning approximately 30 ft. (-25 ft. – +5 ft.) (745 ft. ele. – 775 ft. ele.). Level 3 is a SFP level of -25 ft. (746' ft. ele.) or approximately the top of the SFP racks (ref. 2, 3).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another General Emergency IC was met; however, it is included to provide classification diversity.

MNS Basis Reference(s):

1. NRC EA-12-051 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. MNS-14-023 Second Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)
3. Engineering Change Packages #109073 and #109074
4. NEI 99-01 AG2

ATTACHMENT 1
EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.1 Alert

Dose rates > 15 mR/hr in **EITHER** of the following areas:

Control Room (1EMF12)

OR

Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Basis:

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). 1EMF Channel 12 monitors the Control room for area radiation (ref. 1). The CAS is included in this EAL because of its' importance to permitting access to areas required to assure safe plant operations.

There is no permanently installed CAS area radiation monitors that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS.

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable.

An emergency declaration is not warranted if the following condition applies.

- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

MNS Basis Reference(s):

1. UFSAR Table 12-11 Area Radiation Monitors
2. NEI 99-01 AA3

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Table R-2	**Safe Operation & Shutdown Rooms/Areas**		
Bldg. Elevation	**Unit 1 Room/Area**	**Unit 2 Room/Area**	**Modes**
Auxiliary 716'	P/C, RHole, near 1NI-185, Outside CAD 212	ABPC thru CAD Door, FF59	4
Auxiliary 750'	800 (1EMXA)	820 (2EMXA)	3, 4
	803 (1ETA)	805 (2ETA)	3, 4
Auxiliary 733'	702 (Elec. Pene.)	713 (Elec. Pene.)	3
	722 (1EMXB-1)	724 (2EMXB-1)	3, 4
	705 (1ETB)	716 (2ETB)	3, 4

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.

- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).

- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

MNS Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases
2. NEI 99-01 AA3

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions (NCS temperature ≤ 200°F); EALs in this category are applicable only in one or more cold operating modes.

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to NCS integrity, containment closure, and fuel clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refueling, NM – No Mode).

The events of this category pertain to the following subcategories: 1.

1. NCS Level

Reactor Pressure Vessel water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity.

2. Loss of Essential AC Power

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

3. NCS Temperature

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

4. Loss of Vital DC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to or degraded voltage on the 125 VDC vital buses.

5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

Certain hazardous natural and technological events may result in visible damage to or degraded performance of safety systems warranting classification.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – NCS Level

Initiating Condition: UNPLANNED loss of NCS inventory for 15 minutes or longer

EAL:

CU1.1	**Unusual Event**
UNPLANNED loss of reactor coolant results in NCS water level less than a required lower limit for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

NCS water level less than a required lower limit is meant to be less than the lower end of the level control band being procedurally maintained for the current condition or evolution.

With the plant in Cold Shutdown, NCS water level is normally maintained above the pressurizer low level setpoint of 17% (ref. 1). However, if NCS level is being controlled below the pressurizer low level setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the NCS that is the concern.

With the plant in Refueling mode, NCS water level is normally maintained at or above the reactor vessel flange (Technical Specification LCO 3.9.7 requires at least 23 ft of water above the top of the reactor vessel flange in the refueling cavity during refueling operations) (ref. 2).

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor NCS level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease NCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required

limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL recognizes that the minimum required NCS level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

Continued loss of NCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/F-0 Critical Safety Function Status Tree - Inventory
2. MNS Technical Specifications Section 3.9.7 Refueling Cavity Water Level
3. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – NCS Level

Initiating Condition: UNPLANNED loss of NCS inventory for 15 minutes or longer

EAL:

CU1.2 Unusual Event

NCS water level cannot be monitored

AND EITHER

- UNPLANNED increase in **any** Table C-6 sump or tank level due to a loss of NCS inventory
- Visual observation of unisolable NCS leakage

Table C-6 Sumps/Tanks
• NCDT • PRT • CFAE sump • ND/NS sump • RHT • WDT • WEFT • SRST

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

In Cold Shutdown mode, the NCS will normally be intact and standard NCS level monitoring means are available. NCS level in the Refueling mode is normally monitored using the sight glass.

In this EAL, all water level indication is unavailable and the NCS inventory loss must be detected by indirect leakage indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of NCS leakage. If the make-up rate to the NCS unexplainably rises above

the pre-established rate, a loss of NCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the NCS that cannot be isolated could also be indicative of a loss of NCS inventory (ref. 1, 2).

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RPV level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease NCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL addresses a condition where all means to determine RPV level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the NCS.

Continued loss of NCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

MNS Basis Reference(s):

1. AP/1(2)/A/5500/10 NC System Leakage Within the Capacity of Both NV Pumps
2. PT/1(2)/A/4150/001D Identifying NC System Leakage
3. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – NCS Level

Initiating Condition: Loss of NCS inventory

EAL:

CA1.1	**Alert**
Loss of NCS inventory as indicated by NCS water level < 5 in. above hotleg centerline	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

None

Basis:

5.1 in. above hotleg centerline (reounded to 5 in.) NCS level indication is the lowest level to assure adequate net positive suction head and prevent ND pump cavitation and air entrainment for all flow rates (ref. 1).

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, a lowering of NCS water level below 5 in. above hotleg centerline indicates that operator actions have not been successful in restoring and maintaining NCS water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

Although related, this EAL is concerned with the loss of NCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Decay Heat Removal suction point). An increase in NCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

If NCS water level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

MNS Basis Reference(s):

1. EP Calculation File MCC-1552.08-00-0208
2. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – NCS Level

Initiating Condition: Loss of NCS inventory

EAL:

CA1.2 Alert

NCS water level cannot be monitored for ≥ 15 min. (Note 1)

AND EITHER

- UNPLANNED increase in **any** Table C-6 sump or tank level due to a loss of NCS inventory
- Visual observation of unisolable NCS leakage

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-6 Sumps/Tanks
• NCDT • PRT • CFAE sump • ND/NS sump • RHT • WDT • WEFT • SRST

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

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

In this EAL, all NCS water level indication would be unavailable for greater than 15 minutes, and the NCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the NCS unexplainably rises above the pre-established rate, a loss of NCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the NCS that cannot be isolated could also be indicative of a loss of NCS inventory (ref. 1, 2).

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, the inability to monitor NCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the NCS.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1.

If the NCS inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

MNS Basis Reference(s):

1. AP/1(2)/A/5500/10 NC System Leakage Within the Capacity of Both NV Pumps
2. PT/1(2)/A/4150/001D Identifying NC System Leakage
3. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – NCS Level

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

EAL:

CS1.1 Site Area Emergency

NCS water level cannot be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **any** of the following:

- UNPLANNED increase in **any** Table C-6 sump or tank level due to a loss of NCS inventory
- Visual observation of unisolable NCS leakage
- Reactor Building Refueling Bridge Monitor 1EMF16 (2EMF3) reading > 9000 mR/hr (Mode 6)
- Erratic Source Range or Wide Range Flux Monitor indication

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-6 Sumps/Tanks
• NCDT • PRT • CFAE sump • ND/NS sump • RHT • WDT • WEFT • SRST

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

The lowest measurable NCS level is the elevation of the NCS hot leg mid-loop. Therefore, NCS inventory loss relative to the NCS level elevation corresponding to the top of active fuel must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the NCS unexplainably rises above the pre-established rate, a loss of NCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the NCS in areas outside the containment that cannot be isolated could also be indicative of a loss of NCS inventory (ref. 1, 2).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors. 1EMF16 (2EMF3), Reactor Building Refueling Bridge Monitor is located in the containment in proximity to the reactor cavity and is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds 9,000 mR/hr (90% of instrument scale), a loss of inventory with potential to uncover the core is likely to have occurred.
Radiation monitors 1EMF16 and 2EMF3 are only required to be operable in Mode 6.

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

This IC addresses a significant and prolonged loss of reactor vessel/NCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a NCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If NCS level cannot be restored, fuel damage is probable.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor NCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the NCS .

This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown

and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or RG1

MNS Basis Reference(s):

1. AP/1(2)/A/5500/10 NC System Leakage Within the Capacity of Both NV Pumps
2. PT/1(2)/A/4150/001D Identifying NC System Leakage 3. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – NCS Level

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

EAL:

CG1.1 General Emergency

NCS level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **any** of the following:

- UNPLANNED increase in **any** Table C-6 sump or tank level due to a loss of NCS inventory
- Visual observation of UNISOLABLE NCS leakage
- Reactor Building Refueling Bridge Monitor 1EMF16 (2EMF3) reading > 9,000 mR/hr
- Erratic Source Range or Wide Range Flux Monitor indication

AND

Any Containment Challenge indication, Table C-1

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

Table C-6	Sumps/Tanks
• NCDT • PRT • CFAE sump • ND/NS sump • RHT • WDT • WEFT • SRST	

Table C-1	Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Containment hydrogen concentration > 6% • UNPLANNED rise in containment pressure	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to MNS, Containment Closure is established when the requirements of PT/1(2)/A/4200/002 C are met.

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

The lowest measurable NCS level is the elevation of the NCS hot leg mid-loop. Therefore, NCS inventory loss relative to the NCS level elevation corresponding to the top of active fuel must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the NCS unexplainably rises above the pre-established rate, a loss of NCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems

connected to the NCS in areas outside the containment that cannot be isolated could also be indicative of a loss of NCS inventory (ref. 1, 2).

1EMF16 (2EMF3), Reactor Building Refueling Bridge Monitor is located in the containment in proximity to the reactor cavity and is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds 9,000 mR/hr (90% of instrument scale), a loss of inventory with potential to uncover the core is likely to have occurred. Radiation monitors 1EMF16 and 2EMF3 are only required to be operable in Mode 6.

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

Three conditions are associated with a challenge to containment integrity:

- CONTAINMENT CLOSURE is not established (ref. 3).
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the containment atmosphere is greater than 6% (upper limit of for operability of hydrogen recombiners) by volume in the presence of oxygen (>5%) (ref. 4).
- Any unplanned increase in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned containment pressure increases indicates containment closure cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If NCS level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment

damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor NCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from theNCS.

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

MNS Basis Reference(s):

1. AP/1(2)/A/5500/10 NC System Leakage Within the Capacity of Both NV Pumps
2. PT/1(2)/A/4150/001D Identifying NC System Leakage
3. PT/1(2)/A/4200/002 C Containment Closure
4. CALC MCC-1552-08-00-0208 Emergency Procedure Setpoints
5. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential AC Power

Initiating Condition: Loss of all but one AC power source to essential buses for 15 minutes or longer

EAL:

CU2.1 Unusual Event

AC power capability, Table C-2, to essential 4160V buses 1(2)ETA and 1(2)ETB reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-2	AC Power Sources
Offsite:	
• ATC (Train A)	
• SATA (Train A)	
• ATD (Train B)	
• SATB (Train B)	
Onsite:	
• D/G 1(2) A (Train A)	
• D/G 1(2) B (Train B)	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling, NM - No Mode

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses ETA (Train A) and ETB (Train B) (ref. 1).

The essential buses are normally powered from the 6.9KV offsite power system through their respective 6.9KV/4160V Station Auxiliary Transformers (1ATC & 1ATD). Additionally, a standby source of power to each 4160V essential bus is provided from the 6.9KV offsite power system via two separate and independent 6.9KV/4160V transformers (SATA & SATB). These transformers are shared between the two units (ref. 1, 2).

Each essential bus has a dedicated diesel generator (D/G 1(2) A & D/G 1(2) B) to supply an onsite essential source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power. The D/Gs will automatically start and tie onto the essential buses if the normal power source or off-site power is lost (ref. 1).

An Alternate AC power source, the Standby Shutdown Diesel Generator, which provides power to the Standby Shutdown System, is located in the Standby Shutdown Facility (SSF). This AC power source must be started locally from the SSF Control Room. The SSF Diesel Generator has sufficient capability to operate equipment necessary to maintain a safe shutdown condition for the 4 hour SBO event but is not credited as an AC power source by Technical Specifications (ref. 1).

This cold condition EAL is equivalent to the hot condition EAL SA1.1.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

When in the cold shutdown, refueling, or no mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional

time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an essential bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being fed from an offsite power source.

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

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

MNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/07 Loss of Electrical Power
3. ECA-0.0 EP/1(2)/A/5000/ECA-0.0 Loss of All AC Power
4. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential AC Power

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

EAL:

CA2.1	**Alert**
Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, NM - No Mode

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses ETA (Train A) and ETB (Train B) (ref. 1).

The essential buses are normally powered from the 6.9KV offsite power system through their respective 6.9KV/4160V Station Auxiliary Transformers (1ATC & 1ATD). Additionally, a standby source of power to each 4160V essential bus is provided from the 6.9KV offsite power system via two separate and independent 6.9KV/4160V transformers (SATA & SATB). These transformers are shared between the two units (ref. 1, 2).

Each essential bus has a dedicated diesel generator (D/G 1(2) A & D/G 1(2) B) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power. The D/Gs will automatically start and tie onto the essential buses if the normal power source or off-site power is lost (ref. 1).

An Alternate AC power source, the Standby Shutdown Diesel Generator, which provides power to the Standby Shutdown System, is located in the Standby Shutdown Facility (SSF). This AC power source must be started locally from the SSF Control Room. The SSF Diesel Generator has sufficient capability to operate equipment necessary to maintain a safe shutdown condition for the 4 hour SBO event. Although it is not credited as an AC power source by Technical Specifications, it is a credited source with regards to this EAL provided it is aligned within the 15 classification criteria (ref. 1).

This cold condition EAL is equivalent to the hot condition loss of all offsite AC power EAL SS1.1.

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or no mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an essential bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

Escalation of the emergency classification level would be via IC CS1 or RS1.

MNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/07 Loss of Electrical Power
3. ECA-0.0 EP/1(2)/A/5000/ECA-0.0 Loss of All AC Power
4. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – NCS Temperature

Initiating Condition: UNPLANNED increase in NCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED increase in NCS temperature to > 200°F due to loss of decay heat removal capability	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Several instruments are capable of providing indication of NCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1) including both hot leg and cold leg RTDs and core exit T/Cs (ref. 2, 3).

In the absence of reliable NCS temperature indication caused by a loss of decay heat removal capability, classification should be based on EAL CU3.2 should NCS level indication be subsequently lost.

This IC addresses an UNPLANNED increase in NCS temperature above the Technical Specification cold shutdown temperature limitand represents a potential degradation of the level of safety of the plant. If the NCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Coordinator should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

This EAL involves a loss of decay heat removal capability, or an addition of heat to the NCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel

flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

MNS Basis Reference(s):

1. MNS Technical Specifications Table 1.1-1
2. MNS UFSAR Section 7.0 Instrumentation and Controls
3. AP/1(2)/A/5500/19 Loss of ND or ND System Leakage System
4. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – NCS Temperature

Initiating Condition: UNPLANNED increase in NCS temperature

EAL:

CU3.2	**Unusual Event**
Loss of **all** NCS temperature and NCS level indication for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Definition(s):

None

Basis:

Several instruments are capable of providing indication of NCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1) including both hot leg and cold leg RTDs and core exit T/Cs (ref. 2, 3).

NCS water level is normally monitored using various instruments including NC System narrow range and wide range monitors, RVLIS, NC System sightglass, tygon tube and Pressurizer level instruments (ref. 4).

This EAL addresses the inability to determine NCS temperature and level, and represents a potential degradation of the level of safety of the plant. If the NCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Coordinator should also refer to IC CA3.

This EAL reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor NCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

MNS Basis Reference(s):

1. MNS Technical Specifications Table 1.1-1
2. MNS UFSAR Section 7.0 Instrumentation and Controls
3. AP/1(2)/A/5500/19 Loss of ND or ND System Leakage System
4. OP/1(2)/A/6100/SD-20 Draining the NC System
5. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – NCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in NCS temperature to > 200°F for > Table C-3 duration (Notes 1, 9)

OR

UNPLANNED NCS pressure increase > 20 psig due to a loss of NCS cooling (this does **not** apply during water-solid plant conditions)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Note 9: In the absence of reliable NCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the NCS pressure increase criteria when in Mode 5 or based on time to boil data when in Mode 6.

Table C-3: NCS Heat-up Duration Thresholds

NCS Status	Containment Closure Status	Heat-up Duration
Intact (but **not** reduced inventory)	N/A	60 min.*
Not intact **OR** At reduced inventory	established	20 min.*
	not established	0 min.

* If an NCS heat removal system is in operation within this time frame and NCS temperature is being reduced, the EAL is **not** applicable.

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to MNS, Containment Closure is established when the requirements of PT/1(2)/A/4200/002 C are met.

UNPLANNED -. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Several instruments are capable of providing indication of NCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1) including both hot leg and cold leg RTDs and core exit T/Cs (ref. 2, 3).

A 20 psig RPV pressure increase can be read on various instruments during outage (1NCLP5122 and 5142) (ref. 4).

In the absence of reliable NCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the NCS pressure increase criteria when in Mode 5 or based on time to boil data when in Mode 6.

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the NCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

The NCS Heat-up Duration Thresholds table addresses an increase in NCS temperature when CONTAINMENT CLOSURE is established but the NCS is not intact, or NCS inventory is reduced (e.g., mid-loop operation). The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The NCS Heat-up Duration Thresholds table also addresses an increase in NCS temperature with the NCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact NCS is providing a high pressure barrier to a fission product release. The 60- minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in NCS temperature, the NCS is not intact or is at reduced inventory, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the containment atmosphere and subsequently to the environment, and
2) there is reduced reactor coolant inventory above the top of irradiated fuel.

The NCS pressure increase threshold provides a pressure-based indication of NCS heat-up in the absence of NCS temperature monitoring capability.

Escalation of the emergency classification level would be via IC CS1 or RS1.

MNS Basis Reference(s):

1. MNS Technical Specifications Table 1.1-1

2. MNS UFSAR Section 7.0 Instrumentation and Controls
3. AP/1(2)/A/5500/19 Loss of ND or ND System Leakage System 4. MCC-1210.04-00-0040
5. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

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

EAL:

CU4.1	**Unusual Event**
< 105 VDC bus voltage indications on Technical Specification **required** 125 VDC buses for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

The 125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Train A or EVDA, and Train B or EVDD). Each subsystem consists of two channels of 125 VDC batteries (each battery 100% capacity), the associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling. (ref. 1).

The Train A and Train B DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, 4.16 kV switchgear, and 600 V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC vital buses. (ref. 1).

The minimum battery discharge voltage (requiring opening the degraded battery output breaker) is 105 VDC (ref. 1, 2).

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS7.1.

This IC addresses a loss of vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, “required” means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

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

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category R.

MNS Basis Reference(s):

1. MNS Technical Specification 3.8.4 DC Sources – Operating Bases
2. AP/1/A/5500/15 Loss of Vital or Aux Control Power
3. MNS UFSAR Section 8.0 Electrical Power
4. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-4 onsite communication methods

OR

Loss of **all** Table C-4 ORO communication methods

OR

Loss of **all** Table C-4 NRC communication methods

Table C-4	Communication Methods		
System	**Onsite**	**ORO**	**NRC**
Public Address	X		
Internal Telephones	X		
Onsite Radios	X		
DEMNET		X	
Offsite Radio System		X	
Commercial Telephones		X	X
NRC Emergency Telephone System (ETS)			X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, NM – No Mode

Definition(s):

None

Basis:

Onsite/offsite communications include one or more of the systems listed in Table C-4 (ref. 1).

Public Address System

The McGuire Nuclear Station public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

Internal Telephone System

The McGuire Nuclear Station PBX telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, TSC and EOF.

DEMNET

DEMNET is the primary means of offsite communication. This circuit allows intercommunication among the EOF, TSC, control room, counties, and states. DEMNET operates as an internet based (VoIP) communications system with a satellite back-up. Should the internet transfer rate become slow or unavailable, the DEMNET will automatically transfer to satellite mode.

Offsite Radio System

A dedicated radio network can be used for communication with county and state warning points.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by Duke Energy. The local service provider provides primary and secondary power for their lines at the Central Office.

NRC Emergency Telephone System

The NRC uses a Duke Energy dedicated telephone line which allows direct telephone communications from the plant to NRC regional and national offices. The Duke Energy communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the McGuire Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the cold condition equivalent of the hot condition EAL SU7.1.

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Gaston, Catawba, Iredell, Lincoln, Cabarrus and Mecklenburg County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

MNS Basis Reference(s):

1. MNS Emergency Plan Section F Emergency Communications
2. MNS Emergency Plan Section B On-Site Emergency Organization.
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

CA6.1 Alert

The occurrence of **any** Table C-5 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table C-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post- event inspection to determine if the attributes of an explosion are present.

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

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).
- External flooding may be due to high lake level. MNS powerhouse yard elevation is 760 ft MSL. The administration building and yard are elevation 747 ft MSL. The maximum water level elevation at the site is 760.375 ft MFL (ref. 3, 4).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of 95 mph. (ref. 5).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area in the fire response procedure (ref. 5).

- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first conditional addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second conditional addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC CS1 or RS1.

MNS Basis Reference(s):

1. RP/0/A/5000/007 Earthquake
2. AP/0/A/5500/030 Plant Flooding
3. UFSAR Section 2.1 Site Location
4 UFSAR Section 3.4 Water Level (Flood) Design
5. UFSAR Section 3.3.1 Wind Loadings
6. AP/0/A/5500/45 Plant Fire
7. NEI 99-01 CA6

Category H – Hazards and Other Conditions Affecting Plant Safety

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

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

1.Security

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

2.Seismic Event

Natural events such as earthquakes have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety.

3.Natural or Technology Hazard

Other natural and non-naturally occurring events that can cause damage to plant facilities include tornados, FLOODING, hazardous material releases and events restricting site access warranting classification.

4.Fire

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

5.Hazardous Gas

Toxic, corrosive, asphyxiant or flammable gas leaks can affect normal plant operations or preclude access to plant areas required to safely shutdown the plant.

6.Control Room Evacuation

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

7.Emergency Coordinator Judgment

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

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1	**Unusual Event**
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

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

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

Basis:

Security Shift Supervision is the Shift Security Supervisor or Response Team Leader. These individuals are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Duke Energy Physical Security Plan for MNS (Safeguards) information (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4).
Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program.*

This threshold references the Security Shift Supervison because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for MNS (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

MNS Basis Reference(s):

1. Duke Energy Physical Security Plan for MNS
2. AP/0/A/5500/47 Security Events
3. AP/0/A/5500/48 Extensive Damage Mitigation
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.2	**Unusual Event**
Notification of a credible security threat directed at the site	

Mode Applicability:

All

Definition(s):

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

Basis:

Security Shift Supervision is the Shift Security Supervisor or Response Team Leader. These individuals are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Duke Energy Physical Security Plan for MNS (Safeguards) information (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4). Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program*.

This threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the MNS Security Contingency Plan (ref. 1).

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for MNS (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

MNS Basis Reference(s):

1. Duke Energy Physical Security Plan for MNS
2. AP/0/A/5500/47 Security Events
3. AP/0/A/5500/48 Extensive Damage Mitigation
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.3	**Unusual Event**
A validated notification from the NRC providing information of an aircraft threat	

Mode Applicability:

All

Definition(s):

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

Basis:

Security Shift Supervision is the Shift Security Supervisor or Response Team Leader. These individuals are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Duke Energy Physical Security Plan for MNS (Safeguards) information (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4). Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program.*

This threshold addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the

NRC. Validation of the threat is performed in accordance with the MNS Security Contingency Plan (ref. 1).

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for MNS (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

MNS Basis Reference(s):

1. Duke Energy Physical Security Plan for MNS
2. AP/0/A/5500/47 Security Events
3. AP/0/A/5500/48 Extensive Damage Mitigation
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

EAL:

HA1.1	**Alert**
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

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

OWNER CONTROLLED AREA - Area outside the PROTECTED AREA fence that immediately surrounds the plant. Access to this area is generally restricted to those entering on official business.

Basis:

Security Shift Supervision is the Shift Security Supervisor or Response Team Leader. These individuals are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Duke Energy Physical Security Plan for MNS (Safeguards) information (ref. 1).

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

This threshold is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located outside the plant PROTECTED AREA.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for MNS (ref. 1).

MNS Basis Reference(s):

1. Duke Energy Physical Security Plan for MNS
2. AP/0/A/5500/47 Security Events
3. AP/0/A/5500/48 Extensive Damage Mitigation
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

EAL:

HA1.2 **Alert**
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Mode Applicability:

All

Definition(s):

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

OWNER CONTROLLED AREA - Area outside the PROTECTED AREA fence that immediately surrounds the plant. Access to this area is generally restricted to those entering on official business.

Basis:

Security Shift Supervision is the Shift Security Supervisor or Response Team Leader. These individuals are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Duke Energy Physical Security Plan for MNS (Safeguards) information (ref. 1).

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

This threshold addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with site-specific security procedures (ref. 2).

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for MNS (ref. 1).

MNS Basis Reference(s):

1. Duke Energy Physical Security Plan for MNS
2. AP/0/A/5500/47 Security Events
3. AP/0/A/5500/48 Extensive Damage Mitigation
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile Action within the Protected Area

EAL:

HS1.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in MNS UFSAR Figure 2-4 Plot Plan and Site Area.

Basis:

Security Shift Supervision is the Shift Security Supervisor or Response Team Leader. These individuals are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Duke Energy Physical Security Plan for MNS (Safeguards) information (ref. 1).

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program.*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for MNS (ref. 1).

Escalation of the emergency classification level would be via IC HG1.

MNS Basis Reference(s):

1. Duke Energy Physical Security Plan for MNS
2. AP/0/A/5500/47 Security Events
3. AP/0/A/5500/48 Extensive Damage Mitigation
4. NEI 99-01 HS1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile Action resulting in loss of physical control of the facility

EAL:

HG1.1 General Emergency

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

AND EITHER of the following has occurred:

Any of the following safety functions cannot be controlled or maintained

- Reactivity control
- Core cooling
- NCS heat removal

OR

Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

All

Definition(s):

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

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in MNS UFSAR Figure 2-4 Plot Plan and Site Area.

Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program.*

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for MNS (ref.1).

MNS Basis Reference(s):

1. Duke Energy Physical Security Plan for MNS
2. AP/0/A/5500/47 Security Events
3. AP/0/A/5500/48 Extensive Damage Mitigation
4. AP/1(2)/A/5500/17 Loss of Control Room
5. NEI 99-01 HG1

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1	**Unusual Event**
Seismic event > OBE as indicated by OBE EXCEEDED alarm on 1AD-13, E7	

Mode Applicability:

All **Definition(s):**

None

Basis:

Ground motion acceleration of 0.08g horizontal or 0.0533g vertical is the Operating Basis Earthquake for MNS (ref. 1, 3).

Five strong motion triaxial accelerographs are used to obtain seismic event data at the station site. The seismic instrumentation system also consists of a network control center (NCC), which is used for rapid interrogation of the accelerograph data and for data transfer to a dedicated system computer for subsequent data processing and analysis. The time-history recorded at each accelerograph location can be analyzed to determine its corresponding peak acceleration values and to verify that site Operating Basis Earthquake (OBE) limits have not been exceeded. Immediate control room alarm indication of an earthquake of 0.08 g horizontal or 0.533 g vertical or greater is annunciated through the system's network control center (NCC), following seismic trigger actuation by at least two accelerographs (ref. 2).

RP/0/A/5700/007 Earthquake provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions. (ref. 4)

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center) can confirm that an earthquake has occurred in the area of the plant. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the OBE alarm. The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of MNS. Provide the analyst with the following MNS coordinates: **35° 25' 59" north latitude, 80° 56' 55" west longitude** (ref. 5). Alternatively, near real-time seismic activity can be accessed via the NEIC website:

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

An additional method to rule out spurious activation of the seismic instrumentation is to download seismic recorders stored memory on the dedicated laptop computer located in the Control Room, Elevation 767 ft., behind 1MC9. Such validation should not, however, preclude a timely emergency declaration based on receipt of OBE alarm.

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk- downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or Emergency Coordinator may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

MNS Basis Reference(s):

1. UFSAR Section 3.1 Conformance with General Design Criteria
2. UFSAR Section 3.7.4.2 Location and Description of Instrumentation
3. OP/1/A/6100/010N Annunciator Response for Panel 1AD-13
4. RP/0/A/5700/007 Earthquake
5. UFSAR section 2.1.1 Site Location
6. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1	**Unusual Event**
A tornado strike within the PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in MNS UFSAR Figure 2-4 Plot Plan and Site Area.

Basis:

Response actions associated with a tornado onsite is provided in RP/0/A/5700/006 Natural Disasters (ref. 1).

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA9.1.

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

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL HU3.1 addresses a tornado striking (touching down) within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

MNS Basis Reference(s):

1. RP/0/A/5700/006 Natural Disasters
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2	**Unusual Event**

Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

Mode Applicability:

All

Definition(s):

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Areas susceptible to internal flooding are Turbine Building, Service Building and Auxiliary from the following systems: Condenser Circulating Water, Fire Protection, Nuclear and Conventional Service Water and Condensate Storage (ref.1). Refer to EAL CA6.1 for internal flooding affecting one or more SAFETY SYSTEM trains.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns.

Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

MNS Basis Reference(s):

1. AP/0/A/5500/44 Plant Flooding
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3	**Unusual Event**
Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in MNS UFSAR Figure 2-4 Plot Plan and Site Area.

Basis:

As used here, the term "offsite" is meant to be areas external to the MNS PROTECTED AREA.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

MNS Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4	**Unusual Event**
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)	

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Mode Applicability:

All **Definition(s):**

None

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended to apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

MNS Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.1 Unusual Event

A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

The FIRE is located within **any** Table H-1 area

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Containment • Auxiliary Building • Diesel Generator Rooms • FWST • Dog Houses • Standby Shutdown Facility (SSF)

Mode Applicability:

All

Definition(s):

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

Basis:

The 15 minute requirement begins with a credible notification that a fire is occurring, or receipt of multiple valid fire detection system alarms or field validation of a single fire alarm. The alarm

is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Those actions could include visual observation or evaluation of thermal detector or pressure indicator data.

Table H-1 Fire Areas are based on MCS-1465.00-00-0022 Design Basis Specification for the Appendix R Safe Shutdown Analysis and AP/0/A/5500/45 Plant Fire. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

For EAL HU4.1 the intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

MNS Basis Reference(s):

1. MCS-1465.00-00-0022 Design Basis Specification for the Appendix R Safe Shutdown Analysis
2. AP/0/A/5500/45 Plant Fire
3. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.2	**Unusual Event**
Receipt of a single fire alarm (i.e., no other indications of a FIRE)	
	AND
The fire alarm is indicating a FIRE within **any** Table H-1 area	
	AND
The existence of a FIRE is not verified within 30 min. of alarm receipt (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Containment • Auxiliary Building • Diesel Generator Rooms • FWST • Dog Houses • Standby Shutdown Facility (SSF)

Mode Applicability:

All

Definition(s):

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

Basis:

The 30 minute requirement begins upon receipt of a single valid fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it

is not spurious, or by reports from the field. Those actions could include visual observation or evaluation of thermal detector or pressure indicator data. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1.

Table H-1 Fire Areas are based on MCS-1465.00-00-0022 Design Basis Specification for the Appendix R Safe Shutdown Analysis and AP/0/A/5500/45 Plant Fire. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to

limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in this EAL, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

MNS Basis Reference(s):

1. MCS-1465.00-00-0022 Design Basis Specification for the Appendix R Safe Shutdown Analysis
2. AP/0/A/5500/45 Plant Fire
3. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.3	**Unusual Event**
A FIRE within the plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in MNS UFSAR Figure 2-4 Plot Plan and Site Area.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

MNS Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.4	**Unusual Event**
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish	

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in MNS UFSAR Figure 2-4 Plot Plan and Site Area.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

If a FIRE within the plant PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

MNS Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted

Table H-2	**Safe Operation & Shutdown Rooms/Areas**		
Bldg. Elevation	**Unit 1 Room/Area**	**Unit 2 Room/Area**	**Modes**
Auxiliary 716'	P/C, RHole, near 1NI-185, Outside CAD 212	ABPC thru CAD Door, FF59	4
Auxiliary 750'	800 (1EMXA)	820 (2EMXA)	3, 4
	803 (1ETA)	805 (2ETA)	3, 4
Auxiliary 733'	702 (Elec. Pene.)	713 (Elec. Pene.)	3
	722 (1EMXB-1)	724 (2EMXB-1)	3, 4
	705 (1ETB)	716 (2ETB)	3, 4

Mode Applicability:

3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those

rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL does not require atmospheric sampling; it only requires the Emergency Coordinator's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This

reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

NOTE: *IC HA5 mode applicability has been limited to the applicable modes identified in Table H-2 Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table H-2 are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to IC HA5 mode applicability is required.*

MNS Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations

EAL:

HA6.1	**Alert**
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panels or Standby Shutdown Facility (SSF)	

Mode Applicability:

All **Definition(s):**

None

Basis:

The Shift Manager (SM) determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

Inability to establish plant control from outside the Control Room escalates this event to a Site Area Emergency per EAL HS6.1.

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS6.

MNS Basis Reference(s):

1. AP/1(2)/A/5500/17 Loss of Control Room
2. MCS-1465.00-00-0022 Appendix R Safe Shutdown Analysis
3. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Inability to control a key safety function from outside the Control Room

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panels or Standby Shutdown Facility (SSF)

AND

Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1):

- Reactivity (Modes 1, 2 and 3 **only**)
- Core Cooling
- NCS heat removal

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

The Shift Manager determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Emergency Coordinator judgment. The Emergency Coordinator is expected to make a reasonable, informed judgment within 15 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

Escalation of the emergency classification level would be via IC FG1 or CG1

MNS Basis Reference(s):

1. AP/1(2)/A/5500/17 Loss of Control Room
2. MCS-1465.00-00-0022 Appendix R Safe Shutdown Analysis
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE

EAL:

HU7.1	**Unusual Event**

Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Mode Applicability:

All

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

The Emergency Coordinator/EOF Director are the designated onsite individuals having the responsibility and authority for implementing the MNS Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator/EOF Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator/EOF Director, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response,

but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under the emergency classification level description for an Unusual Event.

MNS Basis Reference(s):

1. MNS Emergency Plan section B On-Site Emergency Organization Section B.2 Emergency Coordinator
2. NEI 99-01 HU7

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions exist that in the judgment of the Emergency Coordinator/EOF Director warrant declaration of an Alert

EAL:

HA7.1	**Alert**
Other conditions exist which, in the judgment of the Emergency Coordinator/EOF Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	

Mode Applicability:

All

Definition(s):

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

Basis:

The Emergency Coordinator/EOF Director are the designated onsite individuals having the responsibility and authority for implementing the MNS Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref.1).

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under the emergency classification level description for an Alert.

MNS Basis Reference(s):

1. MNS Emergency Plan section B On-Site Emergency Organization Section B.2 Emergency Coordinator
2. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Coordinator/EOF Director warrant declaration of a Site Area Emergency

EAL:

HS7.1	**Site Area Emergency**
Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.	

Mode Applicability:

All

Definition(s):

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

SITE BOUNDARY - Area as depicted in MNS-SLC-16.11.1 Figure 16.11.1-1 Site Boundary/Exclusion Area Boundary

Basis:

The Emergency Coordinator/EOF Director are the designated onsite individuals having the responsibility and authority for implementing the MNS Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is

responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under the emergency classification level description for a Site Area Emergency.

MNS Basis Reference(s):

1. MNS Emergency Plan section B On-Site Emergency Organization Section B.2 Emergency Coordinator
2. NEI 99-01 HS7

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director warrant declaration of a General Emergency

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area	

Mode Applicability:

All

Definition(s):

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

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Basis:

The Emergency Coordinator/EOF Director are the designated onsite individuals having the responsibility and authority for implementing the MNS Emergency Response Plan. The Operations Shift Manager(SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the Site Boundary.

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under the emergency classification level description for a General Emergency.

MNS Basis Reference(s):

1. MNS Emergency Plan section B On-Site Emergency Organization Section B.2 Emergency Coordinator
2. NEI 99-01 HG7

Category S – System Malfunction

EAL Group: Hot Conditions (NCS temperature > 200ºF); EALs in this category are applicable only in one or more hot operating modes.

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of Essential AC Power

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

2. Loss of Vital DC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of vital plant 125 VDC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. NCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. NCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive NCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, NCS and containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any trip failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, NCS and containment integrity.

7. Loss of Communications

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

8. Containment Isolation Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification.

9. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant safety system performance or significant visible damage warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **all** offsite AC power capability to essential buses for 15 minutes or longer

EAL:

SU1.1	**Unusual Event**
Loss of **all** offsite AC power capability, Table S-1, to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1	**AC Power Sources**
Offsite:	
• ATC (Train A)	
• SATA (Train A)	
• ATD (Train B)	
• SATB (Train B)	
Onsite:	
• D/G 1(2) A (Train A)	
• D/G 1(2) B (Train B)	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None **Basis:**

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses ETA (Train A) and ETB (Train B) (ref. 1).

The essential buses are normally powered from the 6.9KV offsite power system through their respective 6.9KV/4160V Station Auxiliary Transformers (1ATC & 1ATD). Additionally, a

standby source of power to each 4160V essential bus is provided from the 6.9KV offsite power system via two separate and independent 6.9KV/4160V transformers (SATA & SATB). These transformers are shared between the two units (ref. 1, 2).

Each essential bus has a dedicated diesel generator (D/G 1(2) A & D/G 1(2) B) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power. The D/Gs will automatically start and tie onto the essential buses if the normal power source or off-site power is lost (ref. 1).

An Alternate AC power source, the Standby Shutdown Diesel Generator, which provides power to the Standby Shutdown System, is located in the Standby Shutdown Facility (SSF). This AC power source must be started locally from the SSF Control Room. The SSF Diesel Generator has sufficient capability to operate equipment necessary to maintain a safe shutdown condition for the 4 hour SBO event but is not credited as an AC power source by Technical Specifications (ref. 1).

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC essential buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the essential buses, whether or not the buses are powered from it.

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

Escalation of the emergency classification level would be via IC SA1.

MNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/07 Loss of Electrical Power
3. ECA-0.0 EP/1(2)/A/5000/ECA-0.0 Loss of All AC Power
4. NEI 99-01 SU1

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all but one** AC power source to essential buses for 15 minutes or longer

EAL:

SA1.1 Alert

AC power capability, Table S-1, to essential 4160V buses 1(2)ETA and 1(2)ETB reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 AC Power Sources
Offsite:
• ATC (Train A)
• SATA (Train A)
• ATD (Train B)
• SATB (Train B)
Onsite:
• D/G 1(2) A (Train A)
• D/G 1(2) B (Train B)

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

For emergency classification purposes, "capability" means that an AC power source is available to the essential buses, whether or not the buses are powered from it.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses ETA (Train A) and ETB (Train B) (ref. 1).

The essential buses are normally powered from the 6.9KV offsite power system through their respective 6.9KV/4160V Station Auxiliary Transformers (1ATC & 1ATD). Additionally, a standby source of power to each 4160V essential bus is provided from the 6.9KV offsite power system via two separate and independent 6.9KV/4160V transformers (SATA & SATB). These transformers are shared between the two units (ref. 1, 2).

Each essential bus has a dedicated diesel generator (D/G 1(2) A & D/G 1(2) B) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power. The D/Gs will automatically start and tie onto the essential buses if the normal power source or off-site power is lost (ref. 1).

An Alternate AC power source, the Standby Shutdown Diesel Generator, which provides power to the Standby Shutdown System, is located in the Standby Shutdown Facility (SSF). This AC power source must be started locally from the SSF Control Room. The SSF Diesel Generator has sufficient capability to operate equipment necessary to maintain a safe shutdown condition for the 4 hour SBO event but is not credited as an AC power source by Technical Specifications (ref. 1).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).

- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being fed from an offsite power source.

Escalation of the emergency classification level would be via IC SS1.

MNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/07 Loss of Electrical Power
3. ECA-0.0 EP/1(2)/A/5000/ECA-0.0 Loss of All AC Power
4. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SS1.1	**Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the loss of all offsite and onsite AC power capability (Table C-2) to 4160V essential buses ETA and ETB. The essential switchgear are buses ETA (Train A) and ETB (Train B) (ref. 1). For emergency classification purposes, "capability" means that an AC power source is available to the essential buses, whether or not the buses are powered from it.

The essential buses are normally powered from the 6.9KV offsite power system through their respective 6.9KV/4160V Station Auxiliary Transformers (1ATC & 1ATD). Additionally, a standby source of power to each 4160V essential bus is provided from the 6.9KV offsite power system via two separate and independent 6.9KV/4160V transformers (SATA & SATB). These transformers are shared between the two units (ref. 1, 2).

Each essential bus has a dedicated diesel generator (D/G 1(2) A & D/G 1(2) B) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power. The D/Gs will automatically start and tie onto the essential buses if the normal power source or off-site power is lost (ref. 1).

An Alternate AC power source, the Standby Shutdown Diesel Generator, which provides power to the Standby Shutdown System, is located in the Standby Shutdown Facility (SSF). This AC power source must be started locally from the SSF Control Room. The SSF Diesel Generator has sufficient capability to operate equipment necessary to maintain a safe shutdown condition for the 4 hour SBO event. Although it is not credited as an AC power source by Technical Specifications, it is a credited source with regards to this EAL provided it is aligned within the 15 minute classification criteria (ref. 1).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when both offsite and onsite AC power capability are lost.

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

MNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/07 Loss of Electrical Power
3. ECA-0.0 EP/1(2)/A/5000/ECA-0.0 Loss of All AC Power
4. NEI 99-01 SS1

ATTACHMENT 1
EAL Bases

Category: S –System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to essential buses

EAL:

SG1.1 General Emergency
Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB
AND EITHER:
• Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1)
• Core Cooling RED PATH conditions met

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4160V emergency buses ETA and ETB either for greater then the MNS Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 1) or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. (ref. 2).

The essential buses are normally powered from the 6.9KV offsite power system through their respective 6.9KV/4160V Station Auxiliary Transformers (1ATC & 1ATD). Additionally, a standby source of power to each 4160V essential bus is provided from the 6.9KV offsite power system via two separate and independent 6.9KV/4160V transformers (SATA & SATB). These transformers are shared between the two units (ref. 1, 2).

Each essential bus has a dedicated diesel generator (D/G 1(2) A & D/G 1(2) B) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power. The D/Gs will automatically start and tie onto the essential buses if the normal power source or off-site power is lost (ref. 1).

An Alternate AC power source, the Standby Shutdown Diesel Generator, which provides power to the Standby Shutdown System, is located in the Standby Shutdown Facility (SSF). This AC power source must be started locally from the SSF Control Room. The SSF Diesel Generator has sufficient capability to operate equipment necessary to maintain a safe shutdown condition for the 4 hour SBO event (ref. 3).

Four hours is the station blackout coping time (ref 2).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Coordinator judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met (ref. 2). Specifically, Core Cooling RED PATH conditions exist if either core exit T/Cs are reading greater than or equal to 1200°F or subcooling is 0°F AND no NC pumps are on AND core exit T/Cs are reading greater than or equal to 700°F AND Reactor Vessel Lower Range level less than or equal to 39% (ref. 2).

This IC addresses a prolonged loss of all power sources to AC essential buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC essential bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one essential bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

MNS Basis Reference(s):

1. UFSAR Section 8.4.2 Station Blackout Duration
2. EP/1(2)/A/5000/F-0 Critical Safety Function Status Tress – Core Cooling
3. UFSAR Section 8.0 Electric Power
4. AP/1(2)/A/5500/07 Loss of Electrical Power
5. ECA-0.0 EP/1(2)/A/5000/ECA-0.0 Loss of All AC Power
6. NEI 99-01 SG1

ATTACHMENT 1
EAL Bases

Category: S –System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **all** essential AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2	**General Emergency**

Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min.

AND

Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **both** vital DC buses EVDA and EVDD for ≥ 15 min.

(Note 1)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 4160V emergency buses ETA and ETB for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The essential buses are normally powered from the 6.9KV offsite power system through their respective 6.9KV/4160V Station Auxiliary Transformers (1ATC & 1ATD). Additionally, a standby source of power to each 4160V essential bus is provided from the 6.9KV offsite power system via two separate and independent 6.9KV/4160V transformers (SATA & SATB). These transformers are shared between the two units (ref. 1, 2).

Each essential bus has a dedicated diesel generator (D/G 1(2) A & D/G 1(2) B) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power. The D/Gs will automatically start and tie onto the essential buses if the normal power source or off-site power is lost (ref. 1).

An Alternate AC power source, the Standby Shutdown Diesel Generator, which provides power to the Standby Shutdown System, is located in the Standby Shutdown Facility (SSF). This AC power source must be started locally from the SSF Control Room. The SSF Diesel Generator has sufficient capability to operate equipment necessary to maintain a safe shutdown condition for the 4 hour SBO event (ref. 1).

The 125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Train A or EVDA, and Train B or EVDD). Each subsystem consists of two channels of 125 VDC batteries (each battery 100% capacity), the associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling. (ref. 1).

The Train A and Train B DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, 4.16 kV switchgear, and 600 V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC vital buses. (ref. 1, 3).

The minimum battery discharge voltage (requiring opening the degraded battery output breaker) is 105 VDC (ref. 1, 3).

This IC addresses a concurrent and prolonged loss of both essential AC and Vital DC power. A loss of all essential AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A

sustained loss of both essential AC and vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

MNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/07 Loss of Electrical Power
3. AP/1(2)/A/5500/15 Loss of Vital or Aux Control Power
4. ECA-0.0 EP/1(2)/A/5000/ECA-0.0 Loss of All AC Power
5. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of all vital DC power for 15 minutes or longer

EAL:

SS2.1	**Site Area Emergency**
Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **both** vital DC buses EVDA and EVDD for ≥ 15 min (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

The 125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Train A or EVDA, and Train B or EVDD). Each subsystem consists of two channels of 125 VDC batteries (each battery 100% capacity), the associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling. (ref. 1).

The Train A and Train B DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, 4.16 kV switchgear, and 600 V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC vital buses. (ref. 1, 2).

The minimum battery discharge voltage (requiring opening the degraded battery output breaker) is 105 VDC (ref. 1, 2).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

MNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/15 Loss of Vital or Aux Control Power
3. NEI 99-01 SS8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

SU3.1	**Unusual Event**
An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-2 Safety System Parameters
• Reactor power
• NCS level
• NCS pressure
• Core exit T/C temperature
• Level in at least one S/G
• Auxiliary feed flow in at least one S/G

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Operator Aid Computer (OAC), which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and NCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC SA3.

MNS Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. OP/1(2)/A/6100/SD-2 Cooldown to 400 Degrees F
3. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-3

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-2 Safety System Parameters
• Reactor power
• NCS level
• NCS pressure
• Core exit T/C temperature
• Level in at least one S/G
• Auxiliary or emergency feed flow

Table S-3 Significant Transients
• Reactor trip
• Runback > 25% thermal power
• Electrical load rejection > 25% electrical load
• Safety injection actuation

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Operator Aid Computer (OAC), which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Significant transients are listed in Table S-3 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load or SI injection actuations.

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and NCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ICs FS1 or IC RS1

MNS Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. OP/1(2)/A/6100/SD-2 Cooldown to 400 Degrees F
3. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – NCS Activity

Initiating Condition: NCS activity greater than Technical Specification allowable limits

EAL:

SU4.1 Unusual Event

NCS activity > **any** of the following Technical Specification 3.4.16 limits:

- Dose Equivalent I-131 > 1.0 µCi/gm for > 48 hrs.
- Dose Equivalent I-131 > 60 µCi/gm
- Dose Equivalent Xe-133 > 280 µCi/gm

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

The specific iodine activity is limited to ≤ 1.0 µCi/gm Dose Equivalent I-131 for > 48 hrs. or ≤ 60 µCi/gm Dose Equivalent I-131 instantaneous. The specific Xe-133 activity is limited to ≤ 280 µCi/gm Dose Equivalent XE-133 (ref 1, 2).

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

MNS Basis Reference(s):

1. MNS Technical Specifications section 3.4.16 RCS Specific Activity
2. MNS Technical Specifications section 3.4.16 RCS Specific Activity Bases
3. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – NCS Leakage

Initiating Condition: NCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event
NCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. **OR** NCS identified leakage > 25 gpm for ≥ 15 min. **OR** Leakage from the NCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Identified leakage includes leakage such as that from pump seals or valve packing (except reactor coolant pump (NCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank, leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage; or NCS leakage through a steam generator to the secondary system (primary to secondary leakage) (ref. 1).

Unidentified leakage is all leakage (except NCP seal water injection or leakoff) that is not identified leakage (ref. 1).

Pressure Boundary leakage is leakage (except primary to secondary leakage) through a nonisolable fault in an NCS component body, pipe wall, or vessel wall (ref. 1)

NCS leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as NCS to the Component Cooling Water (KC), or systems that directly see NCS pressure outside containment such as Chemical & Volume Control System (NV), Nuclear Sampling system (NM) and Residual Heat Removal (ND) system (when in the shutdown cooling mode).

This IC addresses NCS leakage which may be a precursor to a more significant event. In this case, NCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the NCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an NCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage) or a location outside of containment.

The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the NCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

MNS Basis Reference(s):

1. MNS Technical Specifications Definitions section 1.1
2. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RPS setpoint is exceeded

AND

A subsequent automatic trip or manual trip action taken at the reactor control console (manual reactor trip switches or turbine manual trip) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability: 1

1- Power Operations

Definition(s):

None

Basis:

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) trip function. A reactor trip is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console (i.e., manual trip switches or turbine trip). Reactor shutdown achieved by use of other trip actions specified in EP/1(2)/A/5000/FR-S.1 Response to Nuclear Power Generation/ATWS (such as depressing manual pushbutton on turbine control panel, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

Following any automatic RPS trip signal, EP/1(2)/A/5000/E-0 (ref. 2) and EP/1(2)/A/5000/FR-S.1 (ref. 3) prescribe insertion of redundant manual trip signals to back up the automatic RPS trip function and ensure reactor shutdown is achieved. Even if the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the automatic trip, the lowest level of classification that must be declared is an Unusual Event (ref. 4).

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power below 5%, the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual trip following receipt of an automatic trip signal and there are no clear indications that the automatic trip failed (such as a time delay following indications that a trip setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic trip or manual actions. If a subsequent review of the trip actuation indications reveals that the automatic trip did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip using a different switch). Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip
- and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

MNS Basis Reference(s):

1. MNS Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. EP/1(2)/A/5000/E-0 Reactor Trip or Safety Injection
3. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees - Subcriticality
4. EP/1(2)/A/5000/FR-S.1 Response to Nuclear Power Generation/ATWS
5. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.2	**Unusual Event**

A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the reactor control console (manual reactor trip switches or turbine manual trip) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability: 1

- Power Operations

Definition(s):

None

Basis:

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor (reactor power < 5%). (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from a manual reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3 4).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console (i.e., manual trip switches or turbine trip). Reactor shutdown achieved by use of other trip actions specified in EP/1(2)/A/5000/FR-

S.1 Response to Nuclear Power Generation/ATWS (such as depressing manual pushbutton on turbine control panel, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

If both subsequent automatic and subsequent manual reactor trip actions in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 5%) following a failure of an initial manual trip, the event escalates to an Alert under EAL SA6.1

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip using a different switch). Depending upon several factors, the initial or subsequent effort to manually the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RTS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

MNS Basis Reference(s):

1. MNS Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. EP/1(2)/A/5000/E-0 Reactor Trip or Safety Injection
3. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees - Subcriticality
4. EP/1(2)/A/5000/FR-S.1 Response to Nuclear Power Generation/ATWS
5. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

EAL:

SA6.1	**Alert**
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%	
AND	
Manual trip actions taken at the reactor control console (manual reactor trip switches or turbine manual trip) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)	

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability: 1

- Power Operations

Definition(s):

None

Basis:

This EAL addresses any automatic or manual reactor trip signal that fails to shut down the reactor followed by a subsequent manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed.

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console (i.e., manual trip switches or turbine trip). Reactor shutdown achieved by use of other trip actions specified in EP/1(2)/A/5000/FR-S.1 Response to Nuclear Power Generation/ATWS (such as depressing manual pushbutton on turbine control panel, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent

subsequent core damage. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 1).

Escalation of this event to a Site Area Emergency would be under EAL SS6.1 or Emergency Coordinator judgment.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console (e.g., locally opening breakers). Actions taken at backpanels or other locations within the Control Room, or any location outside the Control Room, are not considered to be “at the reactor control console”.

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling or NCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

MNS Basis Reference(s):

1. MNS Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. EP/1(2)/A/5000/E-0 Reactor Trip or Safety Injection
3. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees - Subcriticality
4. EP/1(2)/A/5000/FR-S.1 Response to Nuclear Power Generation/ATWS
5. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or NCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%

AND EITHER:

- Core Cooling RED PATH conditions met
- Heat Sink RED PATH conditions met

Mode Applicability: 1

- Power Operations

Definition(s):

None

Basis:

This EAL addresses the following:

- Any automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and NCS barriers.

Reactor shutdown achieved by use of EP/1(2)/A/5000/FR-S.1 Response to Nuclear Power Generation/ATWS (such as depressing manual pushbutton on turbine control panel, emergency boration or manually driving control rods) are also credited as a successful manual

trip provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 4).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5% power (ref. 1, 4).

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met (ref. 2). Specifically, Core Cooling RED PATH conditions exist if either core exit T/Cs are reading greater than or equal to 1200°F or subcooling is 0°F AND no NC pumps are on AND core exit T/Cs are reading greater than or equal to 700°F AND Reactor Vessel Lower Range level less than or equal to 39% (ref. 2).

Indication of inability to adequately remove heat from the NCS is manifested by CSFST Heat Sink RED PATH conditions being met (ref. 2). Specifically, Heat Sink RED PATH conditions exist if narrow range level in at least on steam generator is not greater than or equal to 11% (32% ACC) and total feedwater flow to the intact steam generators is less than or equal to 450 gpm. (ref. 3).

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the NCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via IC RG1 or FG1.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees – Subcriticality
2. EP/1(2)/A/5000/F-0 Critical Safety Function Status Tress – Core Cooling
3. EP/1(2)/A/5000/F-0 Critical Safety Function Status Tress – Heat Sink
4. EP/1(2)/A/5000/FR-S.1 Response to Nuclear Power Generation/ATWS
5. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-4 onsite communication methods

OR

Loss of **all** Table S-4 ORO communication methods

OR

Loss of **all** Table S-4 NRC communication methods

Table S-4	**Communication Methods**		
System	**Onsite**	**ORO**	**NRC**
Public Address	X		
Internal Telephones	X		
Onsite Radios	X		
DEMNET		X	
Offsite Radio System		X	
Commercial Telephones		X	X
NRC Emergency Telephone System (ETS)			X

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Onsite/offsite communications include one or more of the systems listed in Table S-4 (ref. 1).

Public Address System

The McGuire Nuclear Station public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

Internal Telephone System

The McGuire Nuclear Station PBX telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, TSC and EOF.

DEMNET

DEMNET is the primary means of offsite communication. This circuit allows intercommunication among the EOF, TSC, control room, counties, and states. DEMNET operates as an internet based (VoIP) communications system with a satellite back-up. Should the internet transfer rate become slow or unavailable, the DEMNET will automatically transfer to satellite mode.

Offsite Radio System

A dedicated radio network can be used for communication with county and state warning points.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by Duke Energy. The local service provider provides primary and secondary power for their lines at the Central Office.

NRC Emergency Telephone System

The NRC uses a Duke Energy dedicated telephone line which allows direct telephone communications from the plant to NRC regional and national offices. The Duke Energy communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the McGuire Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Gaston, Catawba, Iredell, Lincoln, Cabarrus and Mecklenburg County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

MNS Basis Reference(s):

1. MNS Emergency Plan Section F Emergency Communications
2. MNS Emergency Plan Section B On-Site Emergency Organization.
3. NEI 99-01 CU5

Category: S – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

EAL:

SU8.1 Unusual Event

EITHER:

Any penetration is not isolated within 15 min. of a VALID containment isolation signal (Note 1)

OR

Containment pressure > 3 psig with **EITHER** a failure of both trains of NS **OR** failure of both trains of VX-CARF for ≥ 15 min. (Notes 1, 10)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 10: If the loss of containment cooling threshold is exceeded due to loss of both trains of VX-CARF, this EAL **only** applies if at least one train of VX-CARF is not operating, per design, after the 10 minute actuation delay for greater than or equal to 15 minutes.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

Basis:

The containment Phase B pressure setpoint (3 psig, ref. 1, 2) is the pressure at which the containment cooling systems should actuate and begin performing their function.

One full train of containment cooling operating per design is considered (ref. 1, 2):

- One train of Containment Air Return Fan System (VX-CARF), and
- One train of Containment Spray System (NS)

Once the Residual Heat Removal system is taking suction from the containment sump, with containment pressure greater than 3 psig and procedural guidance, one train of containment

spray is manually aligned to the containment sump. If unable to place one NS train in service or without an operating train of VX-CARF (the CARF with a 10-minute delay) within 15 minutes this EAL has been exceeded. At this point a significant portion of the ice in the ice condenser would have melted and the NS system would be needed for containment pressure control.
The Unusual Event threshold applies after automatic or manual alignment of the containment spray system has been attempted with containment pressure greater than 3 psig and less than one full train of NS is operating for greater than or equal to 15 minutes.

The Unusual Event threshold also applies if containment pressure is greater than 3 psig and at least one train of VX-CARF is not operating after a 10 minute delay for greater than or equal to 15 minutes. Without a single train of VX-CARF in service following actuation, the Unusual Event should be declared regardless of whether ECCS is in injection or sump recirculation mode after 15 minutes.

This EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For the first condition, the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

The second condition addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or ice condenser fans) are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or NCS fission product barriers.

MNS Basis Reference(s):

1. MNS Technical Specification 3.6.6
2. MNS Technical Specification 3.6.6 Bases
3. MNS Technical Specification 3.3.2
4. UFSAR Section 6.2 Containment Systems
5. NEI 99-01 SU7

Category: S – System Malfunction

Subcategory: 9 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

SA9.1 Alert

The occurrence of **any** Table S-5 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table S-5	Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post- event inspection to determine if the attributes of an explosion are present.

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

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).
- External flooding may be due to high lake level. MNS powerhouse yard elevation is 760 ft MSL. The administration building and yard are elevation 747 ft MSL. The maximum water level elevation at the site is 760.375 ft MFL (ref. 3, 4).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of 95 mph. (ref. 5).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area in the fire response procedure (ref. 5).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first condition addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance

should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or RS1.

MNS Basis Reference(s):

1. RP/0/A/5700/007 Earthquake
2. AP/0/A/5500/030 Plant Flooding
3. UFSAR Section 2.1 Site Location
4. UFSAR Section 3.4 Water Level (Flood) Design
5. UFSAR Section 3.3.1 Wind Loadings
6. AP/0/A/5500/45 Plant Fire
7. NEI 99-01 SA9

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

An independent spent fuel storage installation (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a cask/canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel.

An Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated.

The MNS ISFSI is contained wholly within the plant Protected Area. Therefore a security event related to the ISFSI would be applicable to EALs HU1.1, HA1.1and HS1.1

Minor surface damage that does not affect storage cask/canister boundary is excluded from the scope of these EALs.

ATTACHMENT 1
EAL Bases

Category: E - ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 dose limit

Table E-1 ISFSI Dose Limits

NAC Magnastor	NAC UMS	Transnuclear (TN-32)
• 190 mrem/hr (gamma) on the side of the cask (excludes air inlet/outlet ports) • 10 mrem/hr (neutron) on the side of the cask (excludes air inlet/outlet ports) • 900 mrem/hr (neutron + gamma) on the top of the cask (excludes air inlet/outlet ports)	• 100 mrem/hr (neutron + gamma) on the side of the cask • 100 mrem/hr (neutron + gamma) on the top of the cask • 200 mrem/hr (neutron + gamma) at air inlets and outlets	• 120 mrem/hr (gamma) or 20 mrem/hr (neutron) on top of the cask • 340 mrem/hr (gamma) or 40 mrem/hr (neutron) on the sides of the radial neutron shield • 560 mrem/hr (gamma) or 280 mrem/hr (neutron) on the side surfaces above the radial neutron shield region • 220 mrem/hr (gamma) or 400 mrem/hr (neutron) on the side surfaces below the radial neutron shield region

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the MNS ISFSI, Confinement Boundary is defined as the Transportable Storage Canister (TSC) for TN, UMS and MAGNASTOR storage systems.

Basis:

The MNS ISFSI utilizes three designs for dry spent fuel storage:

- The Transnuclear (TN) TN-32 dry spent fuel storage system
- The NAC-UMS dry spent fuel storage system
- The NAC-MAGNASTOR dry spent fuel storage system

All systems consist of a Transportable Storage Canister (TSC) and concrete Vertical Concrete Cask (VCC). The TSC is the CONFINEMENT BOUNDARY for all systems. The TSC is welded/bolted and designed to provide confinement of all radionuclides under normal, off-normal, and accident conditions (ref. 1, 2, 3).

Confinement boundary is defined as the barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. Therefore, damage to a confinement boundary must be a confirmed physical breach between the spent fuel and the environment for the TSC.

The values shown in Table E-1 represent 2 times the limits specified in the ISFSI Certificate of Compliance Technical Specification for radiation external to a loaded cask for each of the NAC-MAGNASTOR, NAC-UMS and TN designs. All Table E-1 ISFSI dose limits are based on surveys taken consistent with the locations specified in the associated Technical Specification (ref. 1, 2, 3).

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

MNS Basis Reference(s):

1. TN Generic Technical Specifications
2. NAC-UMS Certificate of Compliance
3. MAGNASTOR Technical Specifications and Design Features
4. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

EAL Group: Hot Conditions (NCS temperature > 200°F); EALs in this category are applicable only in one or more hot operating modes.

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

A. Fuel Clad (FC): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
B. Reactor Coolant System (NCS): The NCS Barrier includes the NCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
C. Containment (CMT): The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:

Any loss or any potential loss of either Fuel Clad or NCS Site

Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the NCS Barrier are weighted more heavily than the Containment Barrier.

- Unusual Event ICs associated with NCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.

- The fission product barrier thresholds specified within a scheme reflect plant-specific MNS design and operating characteristics.

- As used in this category, the term NCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of NCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the NCS due to the as-designed/expected operation of a relief valve is not considered to be NCS leakage.

- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and NCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and NCS fission product barriers were potentially lost, the Emergency Coordinator/EOF Director would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or NCS

EAL:

FA1.1	**Alert**
Any loss **OR any** potential loss of either Fuel Clad or NCS (Table F-1)	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, NCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and NCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or NCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or NCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

MNS Basis Reference(s):

1. NEI 99-01 FA1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss or potential loss of **any** two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss **OR** potential loss of **any** two barriers (Table F-1)	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, NCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and NCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and NCS potential loss thresholds existed, the Emergency Coordinator/EOF Director would have greater assurance that escalation to a General Emergency is less imminent.

MNS Basis Reference(s):

1. NEI 99-01 FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FG1.1 General Emergency
Loss of **any** two barriers
AND
Loss **OR** potential loss of third barrier (Table F-1)

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, NCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, NCS and Containment barriers
- Loss of Fuel Clad and NCS barriers with potential loss of Containment barrier
- Loss of NCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of NCS barrier

MNS Basis Reference(s):

1. NEI 99-01 FG1

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

Introduction

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. NCS or SG Tube Leakage

B. Inadequate Heat removal

C. CMT Radiation / NCS Activity

D. CMT Integrity or Bypass

E. Emergency Coordinator Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned "FC Loss A.1," the third Containment barrier Potential Loss in Category C would be assigned "CMT P-Loss C.3," etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three

fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and NCS barriers and a Potential Loss of the Containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the NCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,..., E.

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		**Reactor Coolant System (NCS) Barrier**		**Containment (CMT) Barrier**	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A NCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by EITHER: • UNISOLABLE NCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE NCS leakage • SG tube leakage 2. Integrity-RED PATH conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B Inadequate Heat Removal	1. Core Cooling-RED PATH conditions met	1. Core Cooling-ORANGE PATH conditions met 2. Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. Core Cooling-RED PATH conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C CMT Radiation / NCS Activity	1. EMF51A/B > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. EMF51A/B > Table F-2 column "NCS Loss"	None	None	1. EMF51A/B > Table F-2 column "CMT Potential Loss"
D CMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Coordinator/EOF Director judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of NCS leakage outside of containment	1. Containment-RED Path conditions met 2. Containment hydrogen concentration > 6% 3. Containment pressure > 3 psig with **EITHER** a failure of both trains of NS **OR** failure of both trains of VX-CARF for ≥ 15 min. (Notes 1, 10)
E EC Judgment	1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates potential loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates potential loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates loss of the containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates potential loss of the containment barrier

Barrier: Fuel Clad

Category: 1. NCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: 1. NCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

1. Core Cooling-RED PATH conditions met

Definition(s):

None

Basis:

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the SPDS display on the Operator Aid Computer (OAC) (ref. 1).

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees
2. EP/1(2)/A/5000/FR-C.1 Response to Inadequate Core Cooling
3. EP/1(2)/A/5000/FR-C.2 Response to Degraded Core Cooling
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-ORANGE Path conditions met

Definition(s):

None

Basis:

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path indicates indicates subcooling has been lost and that some fuel clad damage may potentially occur. The CSFSTs are normally monitored using the SPDS display on the Operator Aid Computer (OAC) (ref. 1).

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees
2. EP/1(2)/A/5000/FR-C.1 Response to Inadequate Core Cooling
3. EP/1(2)/A/5000/FR-C.2 Response to Degraded Core Cooling
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. Heat Sink-RED Path conditions met **AND** Heat sink is required

Definition(s):

None

Basis:

In combination with NCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The CSFSTs are normally monitored using the SPDS display on the Operator Aid Computer (OAC) (ref. 1).

The phrase "and heat sink required" precludes the need for classification for conditions in which NCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 2 tells the operator to determine if heat sink is required by checking that NCS pressure is greater than any non-faulted SG pressure and NCS T_{hot} is greater than 350ºF (347ºF ACC). If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect or place ND in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 2).

This condition indicates an extreme challenge to the ability to remove NCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees
2. EP/1(2)/A/5000/FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.B

Barrier: Fuel Clad

Category: C. CMT Radiation / NCS Activity

Degradation Threat: Loss

Threshold:

1.	EMF51A/B > Table F-2 column "FC Loss"

Table F-2 Containment Radiation – R/hr (EMF51A & B)			
Time After S/D (Hrs.)	NCS Loss	FC Loss	CMT Potential Loss
0-1	8.8	550	5500
1-2	8.4	400	4000
2-8	7.0	160	1600
>8	6.2	100	1000

Definition(s):

None

Basis:

The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the containment high range monitors, EMF51A & B. EMF51 & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale). Radiation Monitors EMF51A & B provide a diverse means of measuring the containment for high level gamma radiation. (ref. 1).

The Table F-2 values, column FC Loss represents, based on core damage assessment procedure, the expected containment high range radiation monitor (EMF51A & B) response based on a LOCA, for periods of 1, 2, 8 and >8 hours after shutdown, no sprays and NCS pressure < 1600 psig with ~2% fuel failure (ref. 1).

The value is derived as follows:

RP/0/A/5700/019 Figure 3 Containment Radiation Level vs. Time for 100% Clad Damage 1, 2, and 8 and >8 hours after shutdown without spray and NCS pressure < 1600 psig x 0.02 (rounded) (ref. 1).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for NCS Barrier Loss threshold C.1 since it indicates a loss of both the Fuel Clad Barrier and the NCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ECL to a Site Area Emergency.

MNS Basis Reference(s):

1. RP/0/A/5700/019 Core Damage Assessment

2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Fuel Clad

Category: C. CMT Radiation / NCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 µCi/gm

Definition(s):

None

Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold dose equivalent I-131 concentration is well above that expected for iodine spikes and corresponds to about 2% fuel clad damage. When reactor coolant activity reaches this level the Fuel Clad barrier is considered lost. (ref. 1).

This threshold indicates that NCS radioactivity concentration is greater than 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

There is no Potential Loss threshold associated with NCS Activity / Containment Radiation.

MNS Basis Reference(s):

1. RP/0/A/5700/019 Core Damage Assessment
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

Barrier: Fuel Clad

Category: C. CMT Radiation / NCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator/EOF Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the Emergency Coordinator in determining whether the Fuel Clad barrier is lost

MNS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates potential loss of the Fuel Clad barrier

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the Emergency Coordinator in determining whether the Fuel Clad barrier is potentially lost. The Emergency Coordinator should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

MNS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

Barrier: Reactor Coolant System

Category: A. NCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. An automatic or manual ECCS (SI) actuation required by **EITHER:**
   - UNISOLABLE NCS leakage
   - SG tube RUPTURE

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

RUPTURE - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

ECCS (SI) actuation is caused by (ref. 1):

- Pressurizer pressure < 1845 psig
- Containment pressure > 1.0 psig

This threshold is based on an UNISOLABLE NCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the NCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE NCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/E-0 Reactor Trip or Safety Injection
2. EP/1(2)/A/5000/E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

Barrier: Reactor Coolant System

Category: A. NCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. Operation of a standby charging pump is required by **EITHER:**
   - UNISOLABLE NCS leakage
   - SG tube leakage

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

The Chemical and Volume Control System (CVCS) includes two centrifugal charging pumps which take suction from the Volume Control Tank and return cooled, purified reactor coolant to the NCS. Normal charging flow is handled by one of the two charging pumps. Each charging pump is designed for a flow rate of 150 gpm. A second charging pump being required is indicative of a substantial NCS leak. (ref. 1).

This threshold is based on an UNISOLABLE NCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE NCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

MNS Basis Reference(s):

1. UFSAR Section 9.3.4 Chemical and Volume Control System

2. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

Barrier: Reactor Coolant System

Category: A. NCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. Integrity-RED path conditions met

Definition(s):

None

Basis:

The "Potential Loss" threshold is defined by the CSFST Reactor Coolant Integrity - RED path. CSFST NCS Integrity - Red Path plant conditions and associated PTS Limit A indicates an extreme challenge to the safety function when plant parameters are to the right of the limit curve following excessive NCS cooldown under pressure (ref. 1, 2).

This condition indicates an extreme challenge to the integrity of the NCS pressure boundary due to pressurized thermal shock – a transient that causes rapid NCS cooldown while the NCS is in Mode 3 or higher (i.e., hot and pressurized).

MNS Basis Reference(s):

1. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees
2. EP/1(2)/A/5000/FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Heat Sink-RED path conditions met **AND** Heat sink is required

Definition(s):

None

Basis:

In combination with FC Potential Loss B.2, meeting this threshold results in a Site Area Emergency.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The CSFSTs are normally monitored using the SPDS display on the Operator Aid Computer (OAC) (ref. 1).

The phrase "and heat sink required" precludes the need for classification for conditions in which NCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 2 tells the operator to determine if heat sink is required by checking that NCS pressure is greater than any non-faulted SG pressure and NCS T_{hot} is greater than 350°F. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect or place ND in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 1, 2)

This condition indicates an extreme challenge to the ability to remove NCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the NCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold B.2; both will be met. This condition warrants a Site Area Emergency declaration because inadequate NCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase NCS pressure to the point where mass will be lost from the system.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees
2. EP/1(2)/A/5000/FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal NCS Loss 2.B

Barrier: Reactor Coolant System

Category: C. CMT Radiation/ NCS Activity

Degradation Threat: Loss

Threshold:

1. EMF51A/B > Table F-2 column "NCS Loss"

Table F-2 Containment Radiation – R/hr (EMF51A & B)			
Time After S/D (Hrs.)	NCS Loss	FC Loss	CMT Potential Loss
0-1	8.8	550	5500
1-2	8.4	400	4000
2-8	7.0	160	1600
>8	6.2	100	1000

Definition(s):

N/A

Basis:

The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the containment high range monitors, EMF51A & B. EMF51A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale). Radiation Monitors EMF51A & B provide a diverse means of measuring the containment for high level gamma radiation. (ref. 1).

The value specified represents, based on core damage assessment procedure RP/0/A/5700/019 Figure 1, the expected containment high range radiation monitor (EMF51A & B) response based on a LOCA, for periods of 1, 2, 8 and >8 hours after shutdown with no fuel failure (ref. 1).

The value is derived as follows:

RP/0/A/5000/019 Figure 1 Containment Radiation Level vs. Time for NCS Release for periods of 1, 2, 8 and >8 hours after shutdown (rounded) (ref. 1).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical

Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold C.1 since it indicates a loss of the NCS Barrier only.

There is no Potential Loss threshold associated with NCS Activity / Containment Radiation.

MNS Basis Reference(s):

1. RP/0/A/5700/019 Core Damage Assessment
2. NEI 99-01 CMT Radiation / RCS Activity NCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. CMT Radiation/ NCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates loss of the NCS barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the NCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the NCS Barrier is lost.

MNS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment NCS Loss 6.A

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates potential loss of the NCS barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the NCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to the inability to reach final safety acceptance criteria before completing all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the NCS Barrier is potentially lost. The Emergency Coordinator should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

MNS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment NCS Potential Loss 6.A

Barrier: Containment

Category: A. NCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of containment

Definition(s):

FAULTED - The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

RUPTURED - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for NCS Barrier Potential Loss A.1 and Loss A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (part of the FAULTED definition) and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel clad barrier (i.e., NCS activity values) and IC SU5 for the NCS barrier (i.e., NCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive

steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

Affected SG is FAULTED Outside of Containment?

P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per SU5.1	Unusual Event per SU5.1
Requires operation of a standby charging (makeup) pump (*NCS Barrier Potential Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1
Requires an automatic or manual ECCS (SI) actuation (*NCS Barrier Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1

There is no Potential Loss threshold associated with NCS or SG Tube Leakage.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/E-0 Reactor Trip or Safety Injection
2. EP/1(2)/A/5000/E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Containment Loss 1.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: A. NCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-RED path conditions met

   AND

Restoration procedures **not** effective within 15 min. (Note 1)

Definition(s):

None

Basis:

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the SPDS display on the Operator Aid Computer (OAC) (ref. 1).

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing (ref. 1, 2, 3).

A direct correlation to status trees can be made if the effectiveness of the restoration procedures is also evaluated. If core exit thermocouple (TC) readings are greater than 1,200°F (ref. 1), Fuel Clad barrier is also lost.

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the NCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Coordinator should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

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

appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees
2. EP/1(2)/A/5000/FR-C.1 Response to Inadequate Core Cooling
3. EP/1(2)/A/5000/FR-C.2 Response to Degraded Core Cooling
4. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CMT Radiation/NCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CMT Radiation/NCS Activity

Degradation Threat: Potential Loss

Threshold:

1. EMF51A/B > Table F-2 column "CMT Potential Loss"

Table F-2 Containment Radiation – R/hr (EMF51A & B)			
Time After S/D (Hrs.)	NCS Loss	FC Loss	CMT Potential Loss
0-1	8.8	550	5500
1-2	8.4	400	4000
2-8	7.0	160	1600
>8	6.2	100	1000

Definition(s):

None

Basis:

The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the containment high range monitors, EMF51A & B. EMF51A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale). Radiation Monitors EMF51A & B provide a diverse means of measuring the containment for high level gamma radiation. (ref. 1).

The Table F-2 values, column CMT Potential Loss represents, based on core damage assessment procedure, the expected containment high range radiation monitor (EMF51A & B) response based on a LOCA, for periods of 1, 2, 8 and >8 hours after shutdown, no sprays and NCS pressure < 1600 psig with ~20% fuel failure (ref. 1).

The value is derived as follows:

RP/0/A/5700/019 Figure 3 Containment Radiation Level vs. Time for 100% Clad Damage 1, 2, 8 and >8 hours after shutdown without spray and NCS pressure < 1600 psig x 0.20 (rounded) (ref. 1).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of

fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and NCS Barrier Loss thresholds.

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the NCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ECL to a General Emergency.

MNS Basis Reference(s):

1. RP/0/A/5700/019 Core Damage Assessment
2. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

AND EITHER:

- Containment integrity has been lost based on EC judgment
- UNISOLABLE pathway from containment to the environment exists

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds.

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of NCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Coordinator will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the NCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of NCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 1. In this simplified example, leakage in an NCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then second threshold would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold to be met as well.

Following the leakage of NCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1.

MNS Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2.	Indications of NCS leakage outside of containment

Definition(s):

None

Basis:

ECA-1.2 LOCA Outside Containment (ref. 1) provides instructions to identify and isolate a LOCA outside of the containment. Potential NCS leak pathways outside containment include (ref. 1, 2):

- Residual Heat Removal (ND)
- Safety Injection (NI)
- Chemical & Volume Control (NV)
- NCP seals (NC)
- PZR/NCS Loop sample lines (NM)

Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the NCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of NCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if NCS mass is being lost outside of the containment.

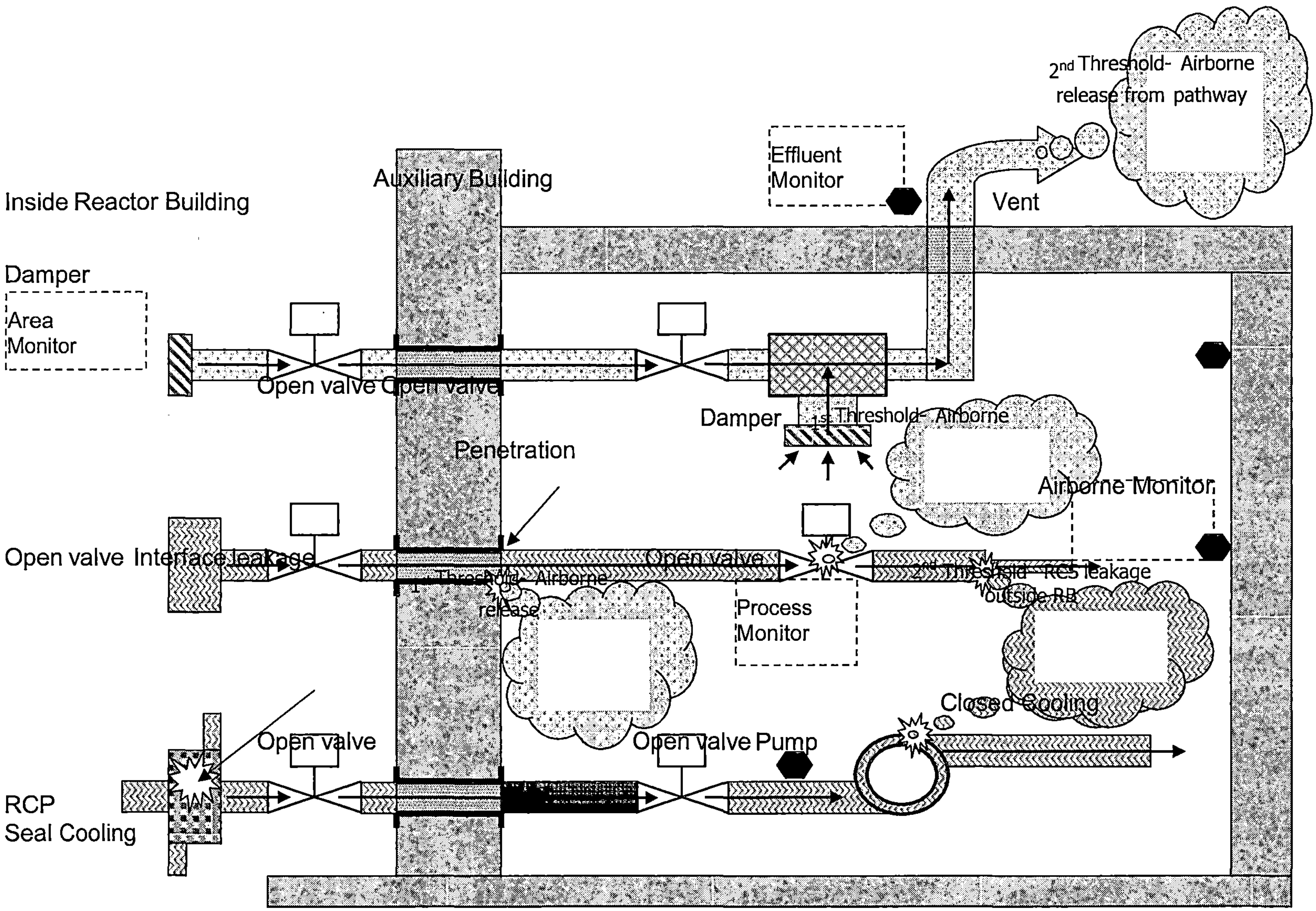
Refer to the middle piping run of Figure 1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold D.1 to be met as well.

To ensure proper escalation of the emergency classification, the NCS leakage outside of containment must be related to the mass loss that is causing the NCS Loss and/or Potential Loss threshold A.1 to be met.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/ECA-1.2 LOCA Outside Containment
2. EP/1(2)/A/5000/E-1 Loss of Reactor or Secondary Coolant
3. NEI 99-01 CMT Integrity or Bypass Containment Loss

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. Containment-RED Path conditions met

Definition(s):

None

Basis:

Critical Safety Function Status Tree (CSFST) Containment-RED path is entered if containment pressure is greater than or equal to 15 psig and represents an extreme challenge to safety function. (ref. 1).

15 psig is based on the containment design pressure (ref. 2).

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the NCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

MNS Basis Reference(s):

1. EP/1(2)/A/5000/F-0 Critical Safety Function Status Trees
2. UFSAR Section 6.2 Containment Systems
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

2.	Containment hydrogen concentration > 6%

Definition(s):

None

Basis:

Following a design basis accident, hydrogen gas may be generated inside the containment by reactions such as zirconium metal with water, corrosion of materials of construction and radiolysis of aqueous solution in the core and sump. (ref. 1).

The lower limit of deflagration of hydrogen in air is > 6% and is the maximum concentration at which hydrogen igniters can be placed in service (ref. 2).

To generate such levels of combustible gas, loss of the Fuel Clad and NCS barriers must have occurred. With the Potential Loss of the containment barrier, the threshold hydrogen concentration, therefore, will likely warrant declaration of a General Emergency.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a potential loss of the Containment Barrier.

MNS Basis Reference(s):

1. UFSAR Section 6.2 Containment Systems
2. EP/1(2)/A/5000/FR-Z.4 Response to High Containment Hydrogen Concentration
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3.	Containment pressure > 3 psig with **EITHER** a failure of both trains of NS **OR** failure of both trains of VX-CARF for ≥ 15 min. (Notes 1, 10)

Note 1: The Emergency Coordinator/EOF Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 10: If the loss of containment cooling threshold is exceeded due to loss of both trains of VX-CARF, this EAL **only** applies if at least one train of VX-CARF is not operating, per design, after the 10 minute actuation delay for greater than or equal to 15 minutes.

Definition(s):

None

Basis:

The containment Phase B pressure setpoint (3 psig, ref. 1, 2) is the pressure at which the containment cooling systems should actuate and begin performing their function.

One full train of containment cooling operating per design is considered (ref. 1, 2):

- One train of Containment Air Return Fan System (VX-CARF), and
- One train of Containment Spray System (NS)

Once the Residual Heat Removal system is taking suction from the containment sump, with containment pressure greater than 3 psig and procedural guidance, one train of containment spray is manually aligned to the containment sump. If unable to place one NS train in service or without an operating train of VX-CARF (the CARF with a 10-minute delay) within 15 minutes a potential loss of containment exists. At this point a significant portion of the ice in the ice condenser would have melted and the NS system would be needed for containment pressure control. The potential loss of containment applies after automatic or manual alignment of the containment spray system has been attempted with containment pressure greater than 3 psig and less than one full train of NS is operating for greater than or equal to 15 minutes.

The potential loss of containment also applies if containment pressure is greater than 3 psig and at least one train of VX-CARF is not operating after a 10 minute delay for greater than or equal to 15 minutes. Without a single train of VX-CARF in service following actuation, the potential loss should be credited regardless of whether ECCS is in injection or sump recirculation mode after 15 minutes.

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

MNS Basis Reference(s):

1. MNS Technical Specification 3.6.6
2. MNS Technical Specification 3.6.6 Bases
3. MNS Technical Specification 3.3.2
4. UFSAR Section 6.2 Containment Systems
5. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: F. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates loss of the Containment barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the Containment Barrier is lost.

MNS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Containment

Category: F. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator/EOF Director that indicates potential loss of the Containment barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the Containment Barrier is lost.

MNS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A

ATTACHMENT 3

Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

Background

NEI 99-01 Revision 6 ICs AA3 and HA5 prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes for AA3 and HA5 states:

The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.

The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

Further, as specified in IC HA5:

The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.

ATTACHMENT 3

Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

MNS Table R-2 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

MNS Procedure and Step	Step Action	Building/Elevation/Room	Mode	If action not performed does this prevent cooldown/ shutdown?
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.8.4.1	Perform OP/1/A/6100/SD 1 (Prepare For Cooldown).	N/A	N/A	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.8.6	Perform NC System degas per OP/1/A/6100/SD-10 (NC System, PRT and NCDT Degas).	N/A	N/A	No
OP/1&2/A/6100/003, Enclosure 4.2, Steps 3.8.8.1 & 3.8.9.1	Open breakers on Transformer Cooling Groups.	Transfer Yard	1	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.8.14	Perform Main Steam Safety Valve testing.	Main Steam Doghouses	1	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.8.17.1	Check transfer of Aux Steam from C htr Bleed to Main Steam (Close 1SP-1 (Main Steam to 1A CF Pump Turb Isol) and 1SP-2 (Main Steam to 1B CF Pump Turb Isol).1AS-11).	Turbine Bldg. Basement (739') North Wall	1	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.8.21	Stop G HDT Pumps per OP/1/B/6250/004 (Feedwater Heater Vents, Drains, and Bleed System).	Turbine Bldg. Basement (739') West Wall	1	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.8.23	Stop C HDT Pumps per OP/1/B/6250/004 (Feedwater Heater Vents, Drains, and Bleed System).	Turbine Bldg. Basement (739') HP Heater Panel	1	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.8.34	Transfer of Aux Steam to Unit 2 or Aux Electric Boilers per OP/1/B/6250/007 B (Auxiliary Electric Boilers).	Service Bldg. (739') or Auxiliary Boiler Room	1	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.12.7	Close 1SP-1 (Main Steam to 1A CF Pump Turb Isol) and 1SP-2 (Main Steam to 1B CF Pump Turb Isol).	Turbine Bldg. Mezz (760') at CF Pumps	1	No

ATTACHMENT 3

Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

MNS Procedure and Step	Step Action	Building/Elevation/Room	Mode	If action not performed does this prevent cooldown/ shutdown?
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.13.16.6	Shutdown MG Sets per OP/1/A/6150/008 (Rod Control), Enclosure 4.5 (M/G Shutdown).	MG Set Room (767')	3	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.14.4	Secondary System Wet Layup Chemical addition (Chemistry).	Secondary Chemistry Lab and TB Basement (739')	3	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.14.6.2	Begin performance of stroke time testing of Pzr PORVs.	Inside Containment	3	No
OP/1&2/A/6100/SD-1, Step 3.5	When RP allows access to Lower Containment, begin Enclosure 4.2 (Pre-Cooldown Containment Entry). This enclosure performs a Containment Inspection with RP, Engineering and Operations involvement.	Inside Containment	3	No
OP/1&2/A/6100/SD-1, Step 3.3.4	After required amount of boron is added for SDM requirements for blocking P-11, Primary Chemistry samples NC System.	Aux. Bldg. (NM Lab 716') Counting Room (767')	3	No
OP/1&2/A/6100/SD-1, Step 3.4.9	After required amount of boron is added for SDM Shutdown Boron Concentration, Primary Chemistry samples NC System.	Aux. Bldg. (NM Lab 716') Counting Room (767')	3	No
OP/1&2/A/6100/SD-1, Step 3.5.7	After required amount of boron is added for Crud Burst Boron Concentration, Primary Chemistry samples NC System.	Aux. Bldg. (NM Lab 716') Counting Room (767')	3	No
OP/1&2/A/6100/SD-1, Step 3.6.7	After required amount of boron is added for Refueling Boron Concentration, Primary Chemistry samples NC System.	Aux. Bldg. (NM Lab 716') Counting Room (767')	3	No
OP/1&2/A/6100/SD-	Have Radwaste align	Aux. Bldg. (716')	1-3	No

ATTACHMENT 3

Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

MNS Procedure and Step	Step Action	Building/Elevation/Room	Mode	If action not performed does this prevent cooldown/ shutdown?
10, Step 3.5.1.1	Nitrogen for NCDT Degas per OP/1/A/6200/600 (WG Support of Unit 1 Shutdown).	Radwaste Area		
OP/1&2/A/6100/SD-10, Step 3.6.2	Radwaste performs Phase 1 PRT Degas per OP/0/A/6200/518 (Waste Gas Operation).	Aux. Bldg. (716') Radwaste Area	1-3	No
OP/1&2/A/6100/SD-10, Step 3.7.1	Radwaste performs NC System Degas per OP/1&2/A/6200/600 (WG Support Of Unit 1/2 Shutdown).	Aux. Bldg. (716') Radwaste Area	1-3	No
OP/1&2/A/6100/SD-10, Step 3.8.3	Radwaste performs NCDT Degas per OP/0/A/6200/518 (Waste Gas Operation).	Aux. Bldg. (716') Radwaste Area	1-3	No
OP/1&2/A/6100/SD-10, Step 3.9.1 & 3.9.2	Radwaste performs Phase 2 PRT Degas per OP/1&2/A/6200/600 (WG Support Of Unit 1/2 Shutdown) and OP/0/A/6200/518 (Waste Gas Operation).	Aux. Bldg. (716') Radwaste Area	1-3	No
OP/1&2/A/6100/SD-2, Step 3.2.3	Radwaste crossties BATs.	Aux. Bldg. (733') BAT Area	3	No
OP/1&2/A/6100/SD-2, Step 3.5	When less than 1000 psig, IAE places NC System Narrow Range Pressure Transmitters in service.	Aux. Bldg. (733') Electrical Pene (733' and 750')	3	Yes
OP/1&2/A/6100/SD-2, Enclosure 4.2, Step 3.2.3.2	To maximize charging flow, adjust NC Pump Seal Water Injection Throttles to 8-10 gpm.	Aux. Bldg. (733') Ledge Outside VCT Room	3	No
OP/1&2/A/6100/SD-4, Step 3.2.3	Radwaste crossties BATs.	Aux. Bldg. (733') BAT Area	3	No
OP/1&2/A/6100/SD-4, Steps 3.11.1, 3.11.2, & 3.12.3	Perform plant shutdown tagging.	ETA, ETB, Aux. Bldg. (733' and 750') South End	3	Yes
OP/1&2/A/6100/SD-4, Enclosure 4.2, Step 3.10.2	To maximize charging flow, adjust NC Pump Seal Water Injection Throttles to 8-10 gpm.	Aux. Bldg. (733') Ledge Outside VCT Room	3	No

ATTACHMENT 3

Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

MNS Procedure and Step	Step Action	Building/Elevation/Room	Mode	If action not performed does this prevent cooldown/ shutdown?
OP/1&2/A/6100/SD-4, Step 3.15	Secondary Chemistry to check S/G sulfate meets chemistry criteria for continued cooldown.	Chemistry Lab and Turbine Bldg. (739') North Wall	4	No
OP/1&2/A/6100/SO-10, Step 3.8	Stroke time testing of PORVs per PT/1/A/4151/005 (NC Valve Stroke Timing Test Using Air). Enclosures 13.4, 13.5, 13.6	Inside Containment	4	No
OP/1&2/A/6100/SO-10, Step 3.11.1, 3.11.2, 3.11.3	Rack out and tag one NV and both NI Pumps per OP/0/A/6350/008 (Operation of Station Breakers).	ETA (750'), ETB (733')	4	Yes
OP/1&2/A/6100/SO-10, Step 3.11.6	Tag out PD Pump at 1MXK-F2C (Reciprocating Charging Pump No 1).	Aux. Bldg. (750') North End	4	Yes
OP/1&2/A/6100/SO-10, Step 3.12	If LTOP vent requirements are to be satisfied by securing 1NC-36B (Pzr PORV) open, Maintenance gags 1NC-36B (Pzr PORV).	Inside Containment	4	No
OP/1&2/A/6100/SD-6A(B), Step 3.3	Unlock and close 1/2ND-119 (1/2 A ND ECCS Sump Suction Relief Inlet Isol #2).	P/C, RHole, near 1NI-185, Outside CAD 212 (716') ABPC thru CAD Door, FF59 (716')	4	Yes
OP/1&2/A/6100/SD-6A(B), Enclosure 4.1, Step 3.6	Perform PT/1/A/4206/030 (Draining ECCS Sump Piping Drain Reservoir Train A), Enclosure 13.1 (Draining ECCS Sump Piping Drain Reservoir Train A in Modes 1 4) and PT/1/A/4206/031 (Draining ECCS Sump Piping Drain Reservoir Train B), Enclosure 13.1 (Draining ECCS Sump Piping Drain Reservoir Train B in Modes 1 4).	Aux Building Pipechase (716')	4	No
OP/1&2/A/6100/SD-6A(B), Encl. 4..1, Step 3.8	Monitor and shift NC System Filters on high DP.	Aux Building (716'/733') at NC Filters Room	4	No

ATTACHMENT 3

Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

MNS Procedure and Step	Step Action	Building/Elevation/Room	Mode	If action not performed does this prevent cooldown/ shutdown?
OP/1&2/A/6100/SD-6A(B), Encl. 4.2 Step 3.7	De-energize 1/2ND-68A (A ND Pump & A Hx Miniflow) in the open position at 1/2EMXA-F12B (1/2ND-68A).	ETA Aux. Bldg. (750')	4	Yes
OP/1&2/A/6100/SD-6A(B), Step 3.31.1	Adjust flow through Cation Bed Demineralizer per OP/1/A/6200/001 D (Chemical and Volume Control System Demineralizers).	Aux Building (750') over Demin Pits	4	No
OP/1&2/A/6100/S0-6, Step 3.4.3	De-energize 1/2ND-67B (B ND Pump & B Hx Miniflow) in the open position at 1/2EMXB1-2C (1/2ND-67B).	ETB Aux. Bldg. (733')	4	Yes

Table R-2 & H-2 Results

Table R-2/H-2	Safe Operation & Shutdown Rooms/Areas		
Bldg. Elevation	**Unit 1 Room/Area**	**Unit 2 Room/Area**	**Modes**
Auxiliary 716'	P/C,RHole, near 1NI-185, Outside CAD 212	ABPC thru CAD Door, FF59	4
Auxiliary 750'	800 (1EMXA)	820 (2EMXA)	3, 4
	803 (1ETA)	805 (2ETA)	3, 4
Auxiliary 733'	702 (Elec. Pene.)	713 (Elec. Pene.)	3
	722 (1EMXB-1)	724 (2EMXB-1)	3, 4
	705 (1ETB)	716 (2ETB)	3, 4

Plant Operating Procedures Reviewed

1. OP/1&2/A/6100/003
2. OP/1&2/A/6100/SD-1
3. OP/1&2/A/6100/SD-2
4. OP/1&2/A/6100/SD-4
5. OP/1&2/A/6100/SD-10
6. OP/1&2/A/6100/SD-6A(B)
7. OP/1&2/A/6100/SO-6
8. OP/1&2/A/6100/SO-10

ATTACHMENT 3
Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

G. Public Education and Information

Information will be made available to the public on a yearly basis concerning notification of a nuclear plant emergency and the response that will be required from the public sector.

G.1/G.2 Public Education and Information Program

On an annual basis, the licensee will update and distribute to residents within the plume exposure pathway emergency-planning zone, emergency-planning information concerning McGuire Nuclear Station. It will provide educational information on radiation, emergency planning contacts, protective actions, primary emergency alert system radio stations, evacuation routes, pick-up points for school children, reception centers, and information for residents with special needs. Local telephone numbers to call with questions will also be listed.

Public information for the transient population includes lake-access signs and emergency planning information. Transient locations will be identified by the Emergency Planning Manager/designee, the site Nuclear Communications staff, and state and county emergency management officials. These locations may include but are not limited to motels, hotels, marinas, and lake access (signs).

The list of transient locations will be reviewed quarterly and updated as needed. Locations will be contacted periodically to ensure adequate copies of materials are available.

G.3.a Communications - Location and Contacts

During a drill or emergency, public information at McGuire Nuclear Station will be coordinated and disseminated through the on-site media center located on Hagers Ferry Rd., Huntersville, N.C. or the Joint Information Center (JIC) located in the Energy Center at 526 South Church Street, Charlotte, N. C. During initial stages of an emergency situation, response to media questions relative to plant status will be provided at the on-site media center. The Charlotte media center, also located in the Energy Center, will be activated as needed. The news release will indicate the location of the primary media center. A company spokesperson will be designated as the primary contact for the news media.

If the Emergency Operations Facility (EOF) is <u>not</u> activated, the normal Duke Energy news release process is followed. If the EOF is activated, then the Joint Information Center (JIC) implements procedures for gathering and disseminating information..

G.3.b Communications - Media Center

In a nuclear plant emergency, the licensee relies on the news media to provide prompt, accurate information to local residents and the public. To provide ready access to current information on plant status, a media center is promptly established. An on-site media center will provide space for a limited number of media. A larger media center, located in the Energy Center at 526 South Church Street, Charlotte, NC (near the EOF) can be activated as needed to support additional media.

G.4.a Company Spokesperson

A company spokesperson will provide plant status and company information during scheduled news conferences and media briefings at a designated media center. Designated public spokespersons are the president of Duke Energy nuclear, the chief nuclear officer and his direct reports, and their designees.

G.4.b Spokesperson Information Exchange

State, county and licensee spokespersons/public information officers are co-located in the Joint Information Center (JIC) to promote a timely exchange and coordination of emergency information. If the JIC is not activated, or if the state or counties do not send public information officers to the JIC, information will be shared via a Joint Information System (JIS) utilizing email, common shared platforms such as WebEOC, faxes or other communication vehicles agreed upon by the affected agencies.

G.4.c Rumor Control

A licensee liaison will work with state, county, and federal public information officers in the JIC or via the JIS to acknowledge rumors and determine the origin. A coordinated response will be made to deal with rumors or correct misinformation.

Customer inquiries are handled by our Customer Contact Centers. Employees are updated via the company intranet/portal. Elected officials and regulatory agencies are updated through our Corporate Communications and governmental affairs departments. Industry groups would assist in disseminating information to other industry groups.

G.5 News Media Training Sessions

The licensee will annually provide the news media with information about emergency planning, radiation, and points of contact for release of public information in an emergency.

H. Emergency Facilities and Equipment

H.1 Technical Support Center (TSC)/Operations Support Center (OSC)

H.1.a Control Room. The Control Room is utilized for evaluation and control of the initial phase of an emergency, including corrective actions and notification and activation of McGuire, Duke Energy, state and local emergency response organizations. The Control Room has redundant (telephone and alternate) two-way communications with emergency centers and off-site agencies. See Figure F-1 for communication scheme.

H.1.b Technical Support Center. (Figure H-1) The Technical Support Center (TSC) is utilized for evaluation of plant status by knowledgeable plant, vendor, NRC and other support groups during an emergency. This center will also be utilized to direct the on-site and initial off-site aspects of an emergency. Anticipated occupants are defined in Emergency Planning Group Manual Section 1.1, On-site Emergency Organization. The TSC has the following capabilities:

1. Redundant two-way communications with the Control Room, the OSC, the Emergency Operations Facility and the Nuclear Regulatory Commission Operations Center. See Figure F-2 for communication scheme.

2. Monitoring for direct radiation and airborne radioactive materials with local readout of radiation level and alarms if levels are exceeded.

3. Display, printout or trend record of comprehensive data necessary to monitor reactor system status and to evaluate plant system abnormalities, in-plant and off-site radiological parameters and meteorological parameters are available. This capability is provided via the operator aid computer. Capabilities to access and display parameters, individually or in groups is provided.

4. Ready access to as-built plant drawings such as general arrangements, flow diagrams, electrical one-lines, instrument details, etc.

5. Radiological habitability during postulated radiological accidents to the same degree as the Control Room.

6. Provisions for staffing by the Station Manager (Emergency Coordinator), advisors and representatives from the site as necessary. Room is also provided for NRC personnel. Space for up to 35 persons plus instrumentation displays are provided.

   The TSC is located near the Control Room, on elevation 767, in the Service Building. The TSC is within one (1) minute walking distance from the Control Room. This is a permanent facility.

H.1.c Operations Support Center. (Figure H-2) The Operations Support Center (OSC) is that place designated for Operations, Radiation Protection, Chemistry, Maintenance, IAE, and others as necessary, to report to in an emergency condition. This center will be used to brief and prepare site personnel for work assignments in support of the emergency condition. The OSC is located on the Auxiliary Building roof office, elevation 784'. Workspace and resources are shared with the Outage Control Center (OCC). The OSC shall have priority over the OCC if any emergency is declared during an outage. The OSC has adequate capacity and supplies including provisions for respiratory protection, protective clothing, portable lighting, portable radiation monitoring equipment and communications equipment.

H.1.d Alternate Facilities. (Figures H-9 and H-10) Alternate TSC and OSC facilities have been established in the McGuire Admin Building as a contingency. Communications equipment similar to that provided in the designated TSC and OSC facilities is available but not all regulatory required equipment/capability is provided.

H.2 Emergency Operations Facility (EOF)

The Emergency Operations Facility (EOF) is utilized for direction and control of all emergency and recovery activities with emphasis on the coordination of off-site activities such as communications with local, state and federal agencies, and coordination of corporate and other outside support. Anticipated occupants are the EOF organization and appropriate state and federal agency representatives.

The EOF has the following capabilities:

a. The capability for obtaining and displaying plant data and radiological information for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves.

b. The capability to analyze plant technical information and provide technical briefings on event conditions and prognosis to licensee and offsite response organizations for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves.

c. The capability to support response to events occurring simultaneously at more than one nuclear power reactor site if the emergency operations facility serves more than one site.

The Common EOF in Charlotte serves as an alternate facility that would be accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and having the following characteristics required collectively of the alternate facilities for use when onsite emergency facilities cannot be safely accessed during hostile action:

- The capability for communication with the emergency operations facility, control room, and plant security.
- The capability to perform offsite notifications.
- The capability for engineering assessment activities, including damage control team planning and preparation.

The EOF has redundant two-way communications with the Technical Support Center and appropriate off-site support agencies. (See Section F).

The EOF is located at 526 South Church Street, Charlotte, NC in the Energy Center Phase II, third floor (Rooms 0300, 0330, 0331, 0332, 0333, 0334, 0335, 0336, 0337, 0337-A, 0340, 0341, 0342, 0343, 0343-A, 0343-B, 0344 and 0345). The EOF layout and location are shown on Figures H-3 thru H-5.

The Joint Information Center and Media Center are utilized for the origination of news briefings and interviews. Anticipated staffing includes the News Group personnel, industry and government representatives and support personnel. News media personnel can be accommodated for press conferences, etc., in the Media Center. (See Figure H-6 and H-7.)

The Joint Information Center has two-way communications with the Emergency Operations Facility and corporate headquarters.

The Joint Information Center (JIC) is located in Duke's Energy Center, 526 South Church Street, Charlotte, N.C. The JIC is located on the first floor, room ECI-0111.

The facilities and resources in the JIC include:

- Work space
- Telephones
- Facsimile machines
- Copy machines
- Podium and PA system
- Tone alert radio
- TV monitor and VCR for real time viewing of the press conferences and taped review of news broadcasts from all three major networks
- Status board
- Wall charts dealing with nuclear site systems and evacuation zones
- Name tags
- Limited clerical support as needed
- Meals during long term activation
- Security escort to other JIC facilities as needed

The media center is located in Duke's Energy Center, 526 South Church Street, Charlotte, N.C. The center is located on the first floor in the O.J. Miller Auditorium.

The facilities and resources in the Media Center include:

- PA system and direct access to recording
- 18 telephones for news media
- Court recorders for prompt press conference transcripts
- Charts dealing with nuclear site systems and evacuation zones
- Modem/computer connections for the news media
- Overhead projector
- Slide projector
- Screen
- Press kits
- News releases
- Technical resources
- Security, registration and badging

H.3 State and Local Government Emergency Operations Centers

See County and State Plans.

H.4 Activation and Staffing

McGuire emergency response facilities (TSC, OC, EOF) are activated as required by the appropriate Emergency Response Procedure. Activation of the TSC, OSC, and EOF is required for Alert and higher emergency conditions. Timely activation and staffing of the Emergency Operations Facility is important to allow the Nuclear Station staff the ability to correct the situation with minimal interference from outside organizations. The Emergency Coordinator will perform the role and function of the EOF Director until activation of the EOF has taken place. The EOF Organization will be alerted and activated for Alert and higher emergency classifications.

H.5 Assessment Actions

Onsite monitoring systems used to initiate emergency measures are defined in Section I. Those used for conducting assessment evaluations during any emergency condition are listed below:

H.5.a Meteorological. A description of the primary meteorological measurement facility is found in Appendix 2. These basic meteorological parameters are displayed in the Control Room, see Figure H-8, Generalized Meteorological System.

1. During periods of primary system unavailability, an alternate source of meteorological data is established as the NWS (NATIONAL WEATHER SERVICE) office. Wind direction and speed are from standard NWS instrumentation at conventional heights.

   Wind direction from the NWS can replace the tower (60 m) wind direction. Wind speed from the NWS can replace the lower tower (10 m) wind speed for

dose calculation purposes; it can also replace the tower (60 m) wind speed for transport speed considerations.

A monthly telephone contact, initiated by plant personnel, with the NWS office will be established to insure that this basic meteorological information can be accessed. See PT/0/A/4600/089.

2. The following field checks will be performed each week by plant personnel:

<u>Wind Direction</u>

(a) Recorder Time Accuracy
(b) Recorder Zero
(c) Translator Zero
(d) Translator Full Scale

<u>Wind Speed</u>

(a) Recorder Time Accuracy
(b) Recorder Zero
(c) Translator Zero
(d) Translator Full Scale

<u>Delta - Temperature</u>

(a) Recorder Time Accuracy

3. Onsite meteorological instruments will be calibrated at a frequency specified by Selective Licensee Commitments. During calibration periods, basic meteorological data, characteristic of site conditions, will be accessible from the NWS. These instruments will be calibrated in accordance with approved procedures.

<u>Hydrologic</u>

A hydrological description of the McGuire Nuclear Site is located in the MNS FSAR, Section 2.4.

<u>Seismic</u>

A description of the seismic monitoring instrumentation and area seismology studies are found in McGuire FSAR, Sections 3.7 and 2.5 respectively.

H.5.b Radiological Monitors

Radiological monitors including process monitors, area monitors, post-accident monitoring equipment, effluent monitors, personnel monitoring devices, portable monitors and sampling equipment are described in various Radiation Protection procedures, the McGuire FSAR, Emergency Plan Implementing Procedures and Safety Evaluation Report.

H.5.c. Plant Parameters

Equipment and instrumentation to monitor plant parameters such as reactor coolant pressure, temperature, levels, containment pressure, temperature, humidity, sump levels, hydrogen concentrations, system flow rates, status, line-ups, are included in operating and emergency procedures. Examples of specific instruments used for accident evaluation are given in Section I.

H.5.d Fire Detection

Fire detection devices of the ionization-chamber and thermal type are located throughout the site.

H.6 Data, Monitoring Equipment and Analysis Facilities

Provisions have been made and exist to obtain data from off-site agencies or monitoring equipment and analysis facilities. The provisions are described below:

a. Meteorological information is available from the National Weather Service as described in Section H.5.a. Monitoring of the Catawba River for hydrologic data is conducted within the Duke System of dams and hydro-electric facilities. Seismic data is available from the U.S. Geological Survey Office as provided for in the McGuire Procedure RP/0/A/5700/007 (Earthquake).

b. Radiological monitors for emergency environmental monitoring are provided in emergency kits. The established environmental monitoring network and sampling equipment in the surrounding area are also available to provide emergency assessment data. Environmental Radiological Monitoring equipment includes radioiodine and particulate continuous air samplers and thermoluminescent dosimeters. The thermoluminescent dosimeters are posted and collected in accordance with Table 1, Branch Technical Position, Rev. 1 of November, 1979. A procedure lists locations of posted thermoluminescent dosimeters and air samplers.

c. See Section C.3.

H.7 Offsite Radiological Monitoring

As described in H.6.b above.

H.8 Meteorology Instrumentation and Procedures

See Section H.5.a.

H.9 Operations Support Center

See Section H.1.c.

H.10 Emergency Equipment/Instrumentation Inspection, Inventory, Operational Check, Calibration

McGuire Procedure PT/0/A/4600/088, Functional Check of Emergency Vehicle and Equipment, defines the inspection, inventory and operational checks required of emergency equipment. Various Radiation Protection procedures define the criteria for calibration of all monitoring equipment located in the emergency kits.

H.11 Emergency Kits

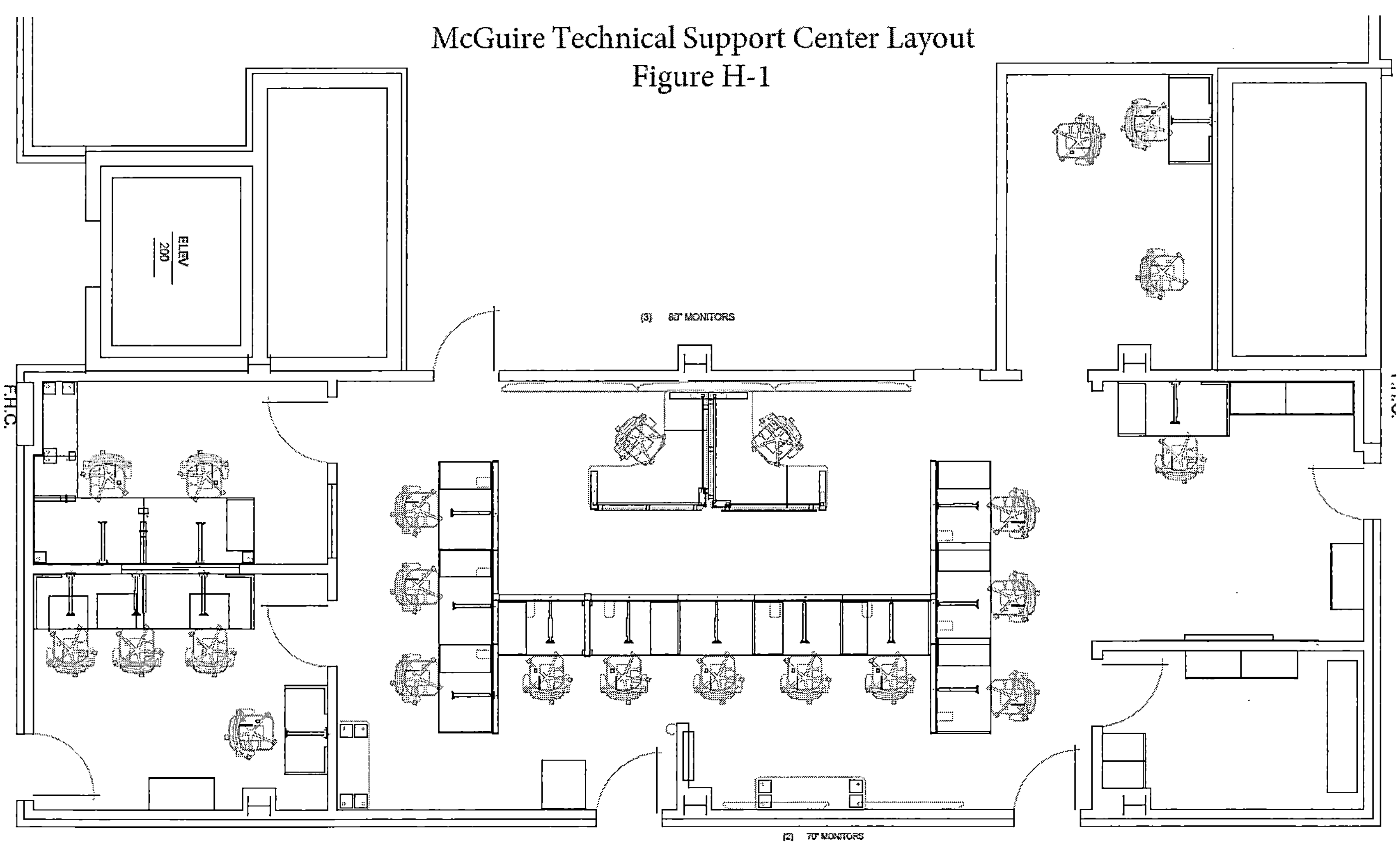
Radiological Emergency kits are described in PT/0/A/4600/088, Functional Check of Emergency Vehicle and Equipment.

H.12 Receipt and Analysis of Field Monitoring Data

Duke Energy's Emergency Operations Facility (Radiological Assessment Manager) will be the central point for the receipt of off-site monitoring data results and sample media analysis results collected by Duke personnel. Resources exist within the organization to evaluate the information and make recommendations based upon the evaluations. The Radiological Assessment Manager's group will perform these evaluations and make recommendations to the EOF Director for protective actions. The EOF Director is the individual responsible for making protective action recommendations to off-site agencies after activation of the EOF.

McGuire Technical Support Center Layout

Figure H-1



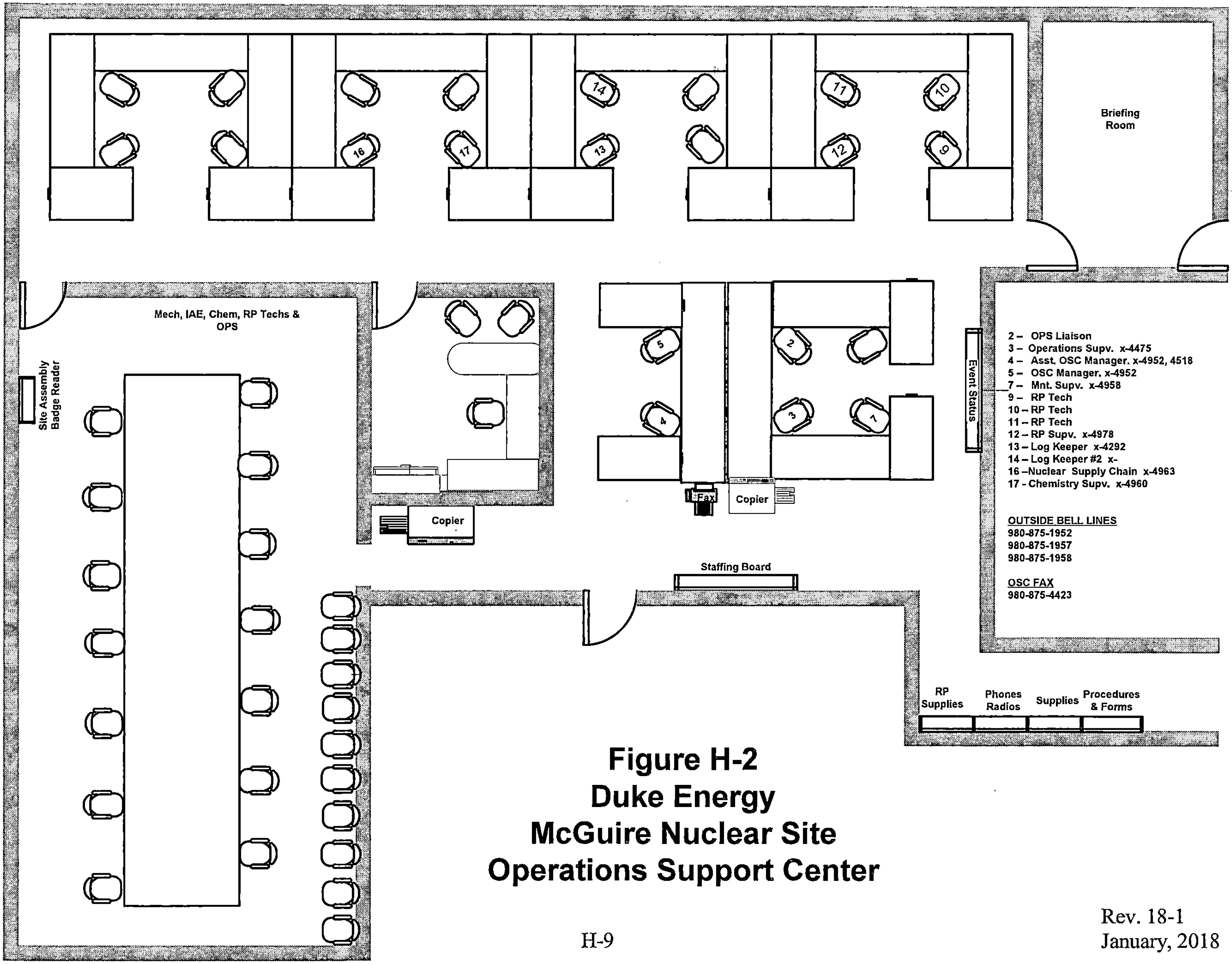
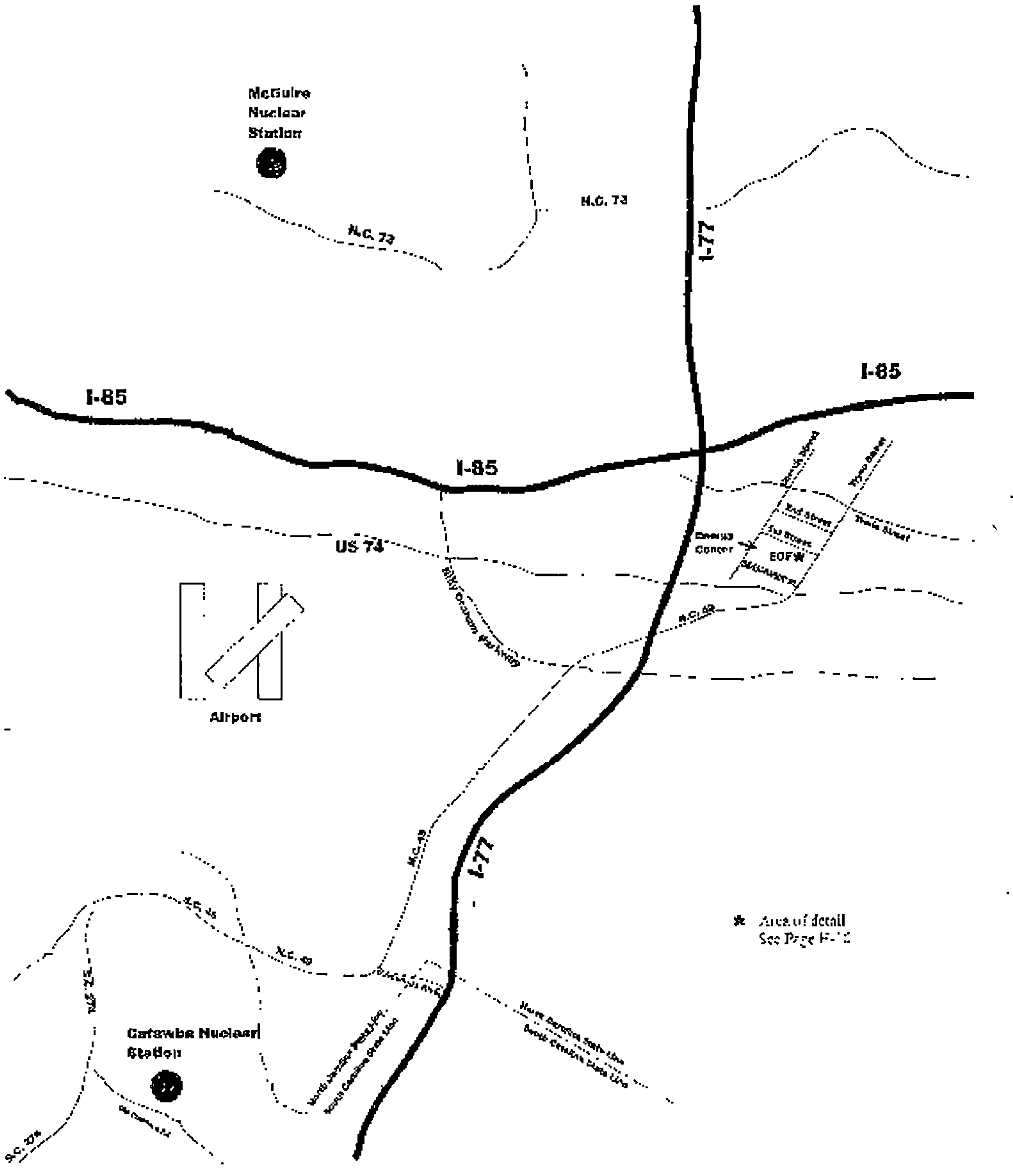


Figure H-2
Duke Energy
McGuire Nuclear Site
Operations Support Center

FIGURE H-3
DUKE ENERGY
EMERGENCY RESPONSE
MNS/CNS/ONS EOF
GENERAL LOCATION



The Media Center and Joint Information Center are in the Energy Center Phase I on the 1st floor. The EOF is in the Energy Center Phase II on the 3rd floor.

Figure H-4
DUKE ENERGY
GENERAL OFFICE RESPONSE
FACILITY

GENERAL OFFICE BUILDING LAYOUT - CHARLOTTE, NC

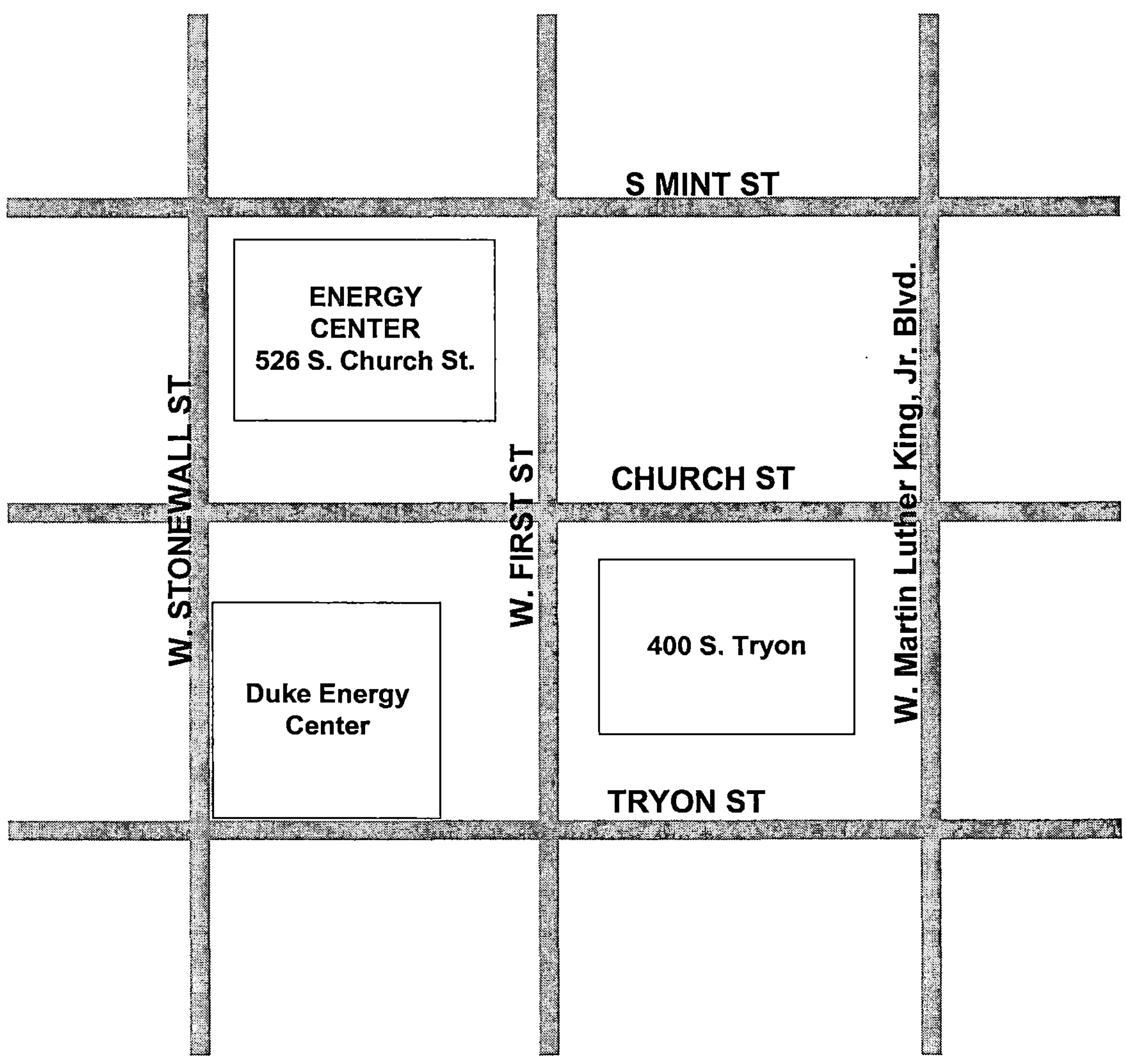


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FIGURE H-5
Emergency Operations Facility
EOF GENERAL ARRANGEMENT

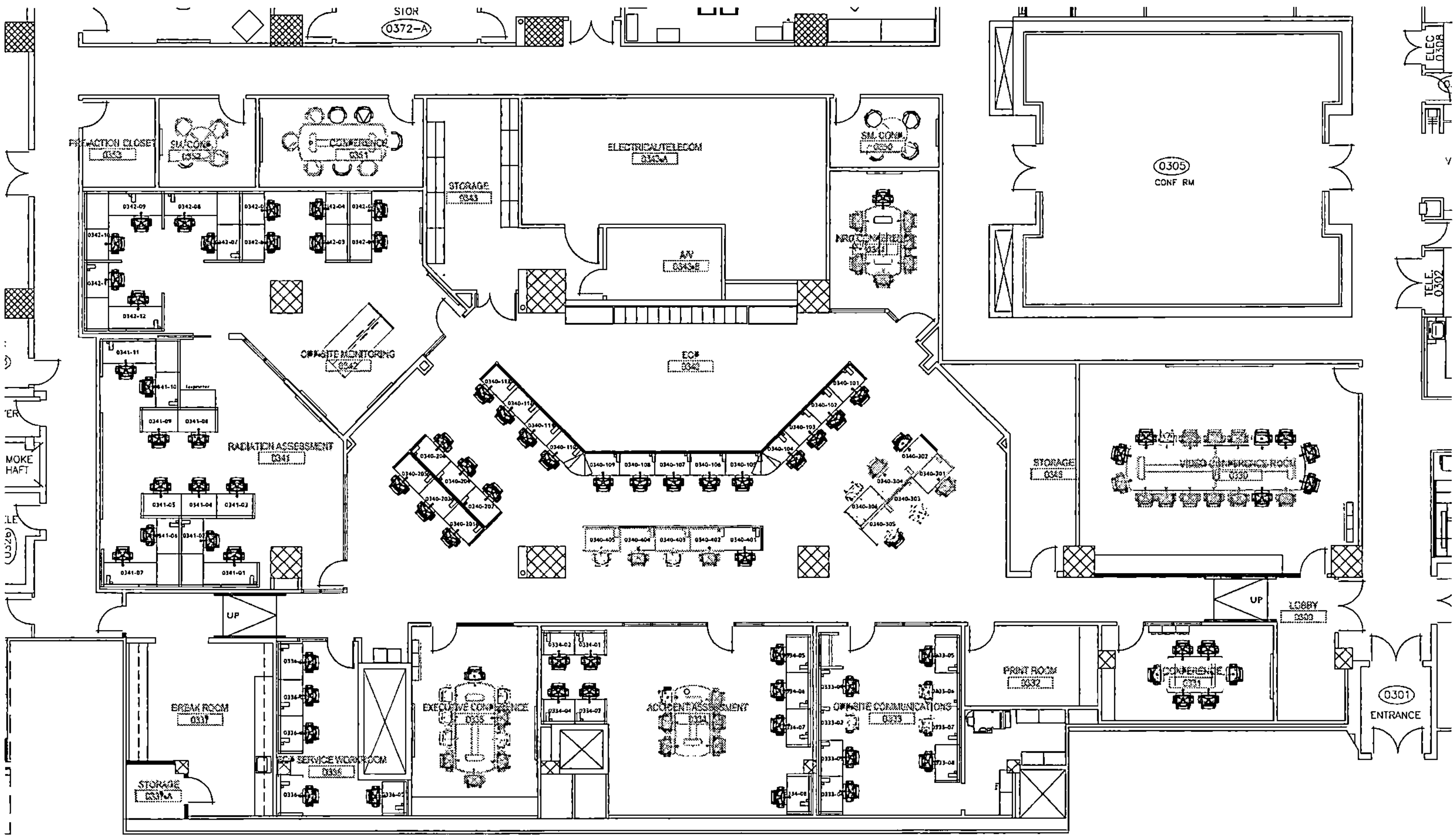

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Figure H-6
Duke Energy
Media Center

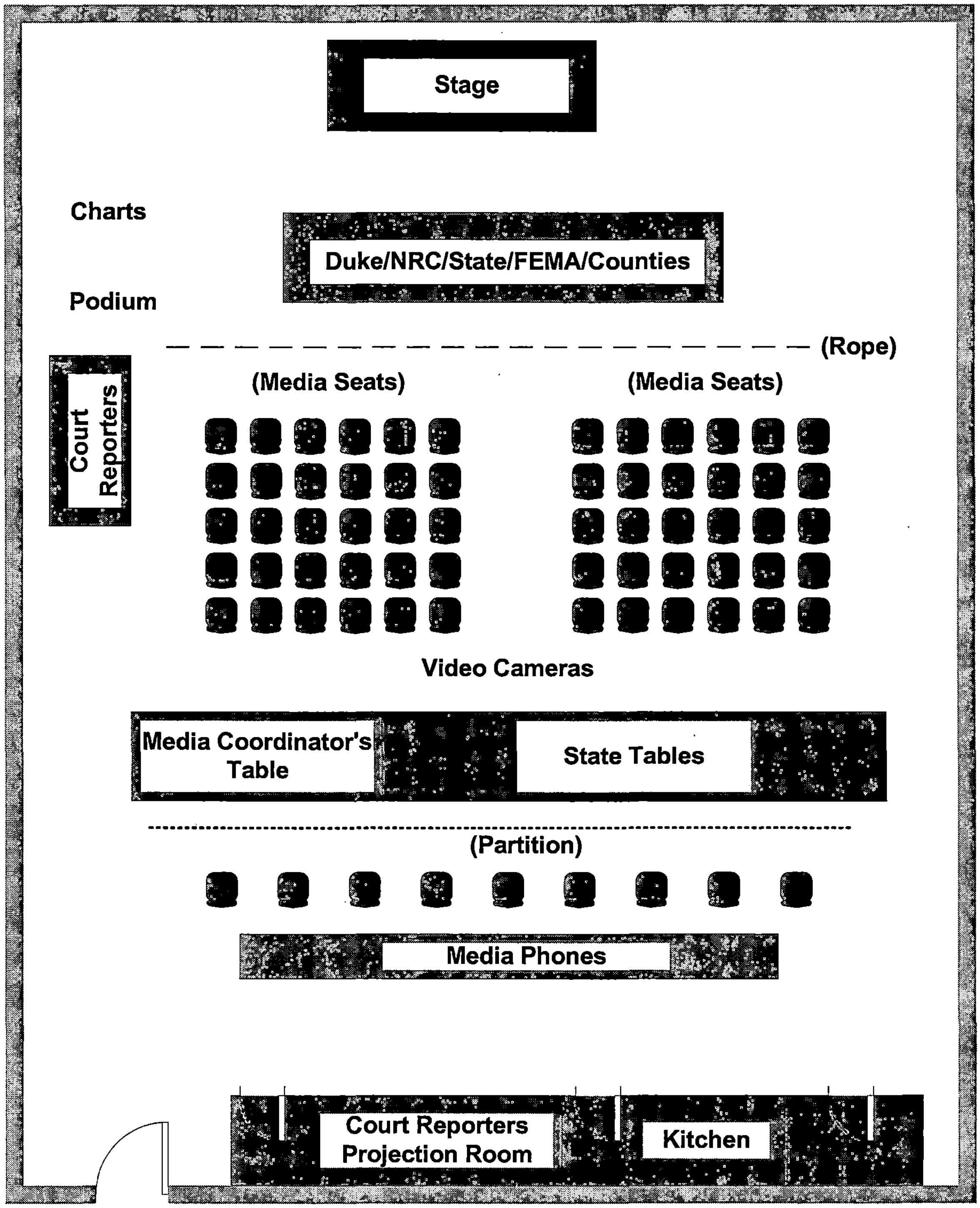

Figure H-7
Duke Energy
Joint Information Center

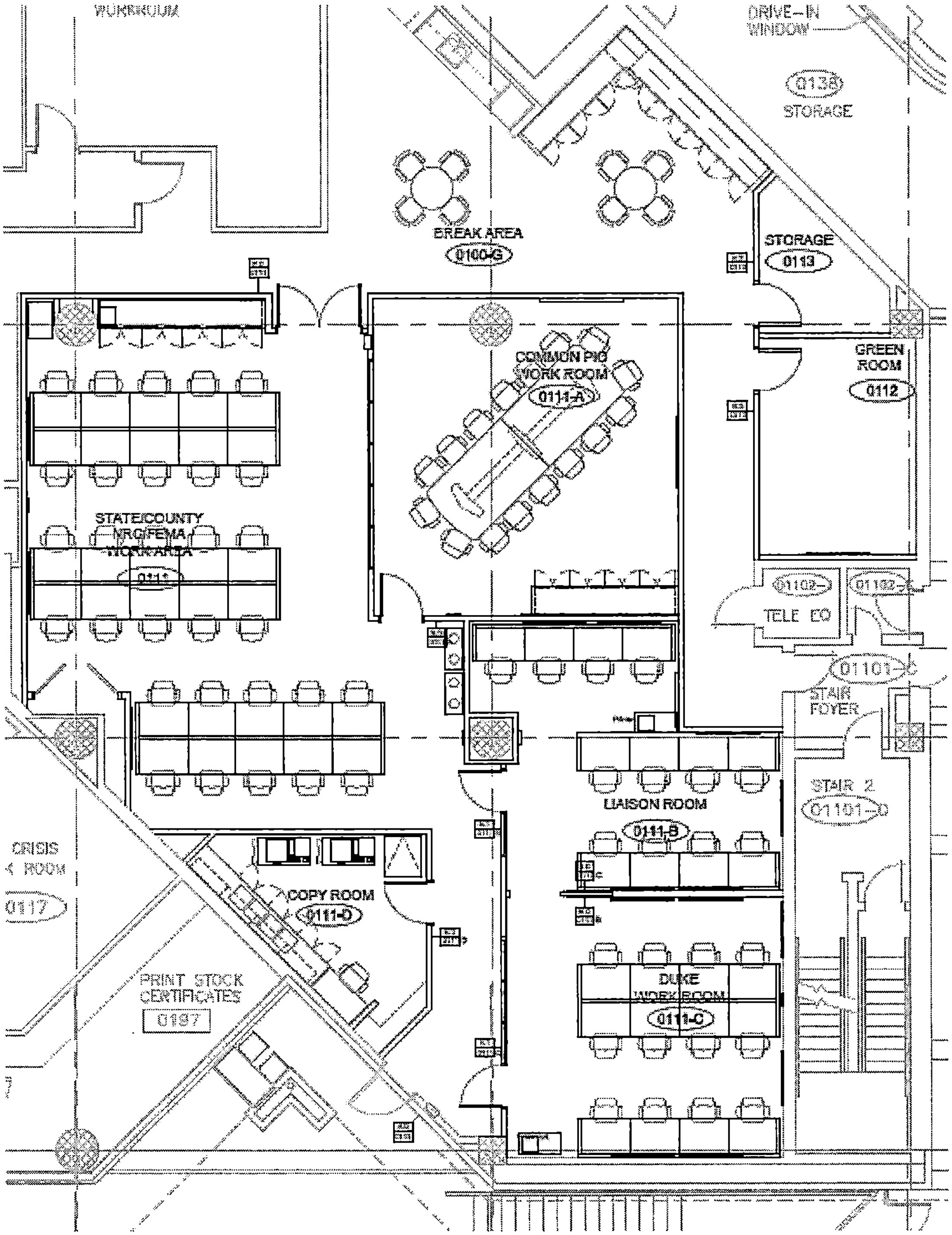

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FIGURE H-8
McGuire and Catawba Nuclear Sites
Generalized Met System

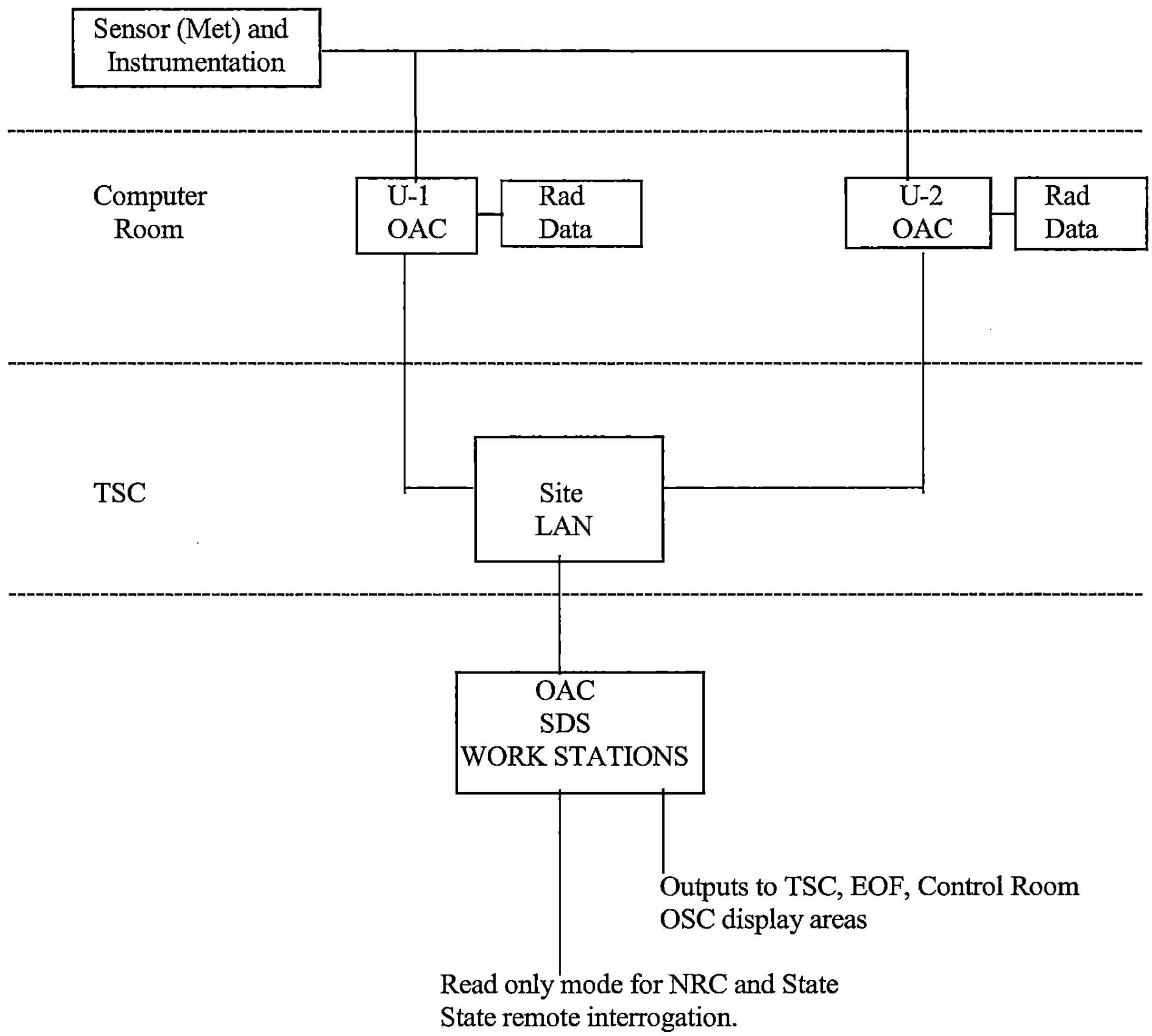
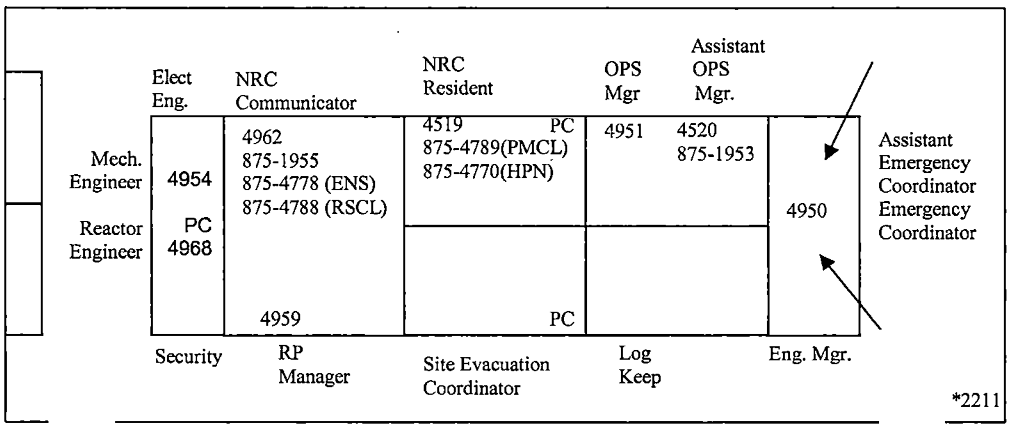


FIGURE H-9

MCGUIRE NUCLEAR SITE
ALTERNATE TECHNICAL SUPPORT CENTER
(EXECUTIVE BOARD ROOM, ROOM 111, ADMIN. BUILDING)



Other TSC Position Locations

- Site Evacuation Coordinator (EP Room 114) - *4458, *4977, *875-1951.
- Offsite Communicator (EP Room 117 -- *4970, DEMNET, *Radio, *875-1951.
- IT Support (CBX Equipment Room 112) -- *4248.
- OAC Support (CBX Equipment Room 112) -- *4999.
- Dose Assessor (SCR Room 100D) -- *4405.
- Company Spokesperson (Rooms 118 and 141) -- *4400, *4419, *4233.
- NRC (NRC Office, Room 126) -- *875-1681.
- Other, use Jaguar Room as needed (Room 144) -- *4826.

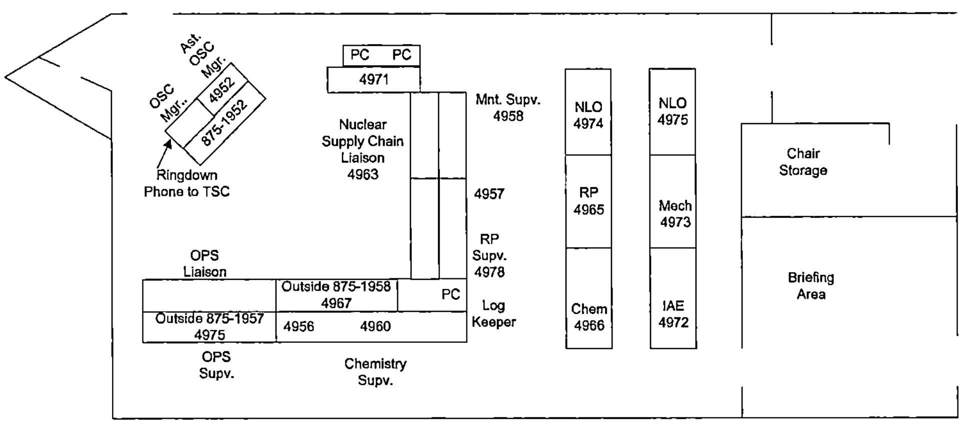
Office Equipment

- FAX (Mail Room, Room 116) -- *875-4506.
- FAX (EP Room 114) -- *875-4382.
- Copier (Mail Room, Room 116).
- Copier (SA Room 170).
- CBX (CBX Office in Admin. Building Lobby).

* Indicates existing phones. All others are to be plugged in when the Alternate TSC is activated.

FIGURE H-10

MCGUIRE NUCLEAR SITE
ALTERNATE OPERATIONS SUPPORT CENTER
(TRAINING ROOM TR155, ADMIN. BUILDING)



<u>Office Equipment</u>

- FAX, Mail Room, Room 116 -- *875-4506.
- FAX, EP, Room 114 -- *875-4382.
- Copier, Mail Room, Room 116.
- Copier, SA, Room 170.
- CBX, CBX Office in Lobby.

* Indicates existing phone. All others are to be plugged in when the Alternate OSC is activated.

I. ACCIDENT ASSESSMENT

To assure the adequacy of methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition.

I.1 Emergency Action Level Procedures

Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 , MNS conducted an EAL implementation upgrade project that produced the EALs discussed in Section D.

The EAL Technical Basis Document (Section D) and EAL Wallboards, will identify the system parameter and effluent parameter values which can be used to determine the emergency condition.

I.2 Onsite Capability and Resources to Provide Initial Values and Continuing Assessment

I.2.a. Post Accident Sampling

The requirement to have reactor coolant and containment atmosphere post accident sample panels (PASS) has been deleted per NRC License Amendment 199/180 by letter dated September 17, 2001. As a result, Emergency Plan Implementing Procedures HP/1/B/1009/015, Unit 1 Nuclear Post-Accident Containment Air Sampling System Operating Procedure and HP/2/B/1009/015, Unit 2 Nuclear Post-Accident Containment Air Sampling System Operating Procedure have been deleted. These EPIPs were replaced by Radiation Protection group procedure HP/0/B/1009/032, Sampling Containment Atmosphere Under Accident Conditions. **HP/0/B/1009/032 is <u>not</u> an EPIP or a part of the EPLAN. It is listed in this paragraph for <u>reference purposes only</u>.** HP/0/B/1009/032 provides contingency methods for containment atmosphere sampling under accident conditions.

Also as a result of NRC License Amendment 199/180, OP/0/B/6200/090, PALSS Operation for Accident Sampling has been deleted from the EPLAN as an Emergency Plan Implementing Procedure. However, OP/0/B/6200/090 will be retained as a Chemistry group procedure to provide contingency methods for reactor coolant and containment sump sampling under accident conditions. **OP/0/B/6200/090 is <u>not</u> an EPIP or a part of the EPLAN. It is listed in this paragraph for <u>reference purposes only</u>.**

I.2.b. Radiation and Effluent Monitors

Radiological monitoring capabilities include process and effluent monitoring systems (FSAR 11.4); area monitoring system (FSAR 12.1.4); plus station portable monitoring instruments, laboratory counters and analyzers (FSAR 12.3.2.4), including emergency high-range instruments and air samplers.

In addition, there are two (2) high range containment monitors, two (2) high range unit vent monitors, four (4) steam line monitors per unit and four (4) N-16 steam line monitors per unit.

I.2.c In-plant Iodine Instrumentation

Radioiodine sampling cartridges are used for sampling containments and unit vents. Radiation Protection personnel are knowledgeable in the appropriate site procedures required and are trained in the equipment required to determine airborne iodine concentrations in the plant under all conditions. Procedures to determine airborne iodine concentrations will cover analyses to be done if counting room capabilities are not available.

I.3.a/ I.3.b Method For Determining Release Source Term

Procedures HP/0/B/1009/006, HP/0/B/1009/010 and AD-EP-ALL-0202 are used on shift, in the TSC and/or EOF for the calculation of potential off-site doses based on a Design Basis Accident, release of primary coolant, or release of GAP activity situation scaled to actual containment monitor readings. Provisions for use of actual source terms exist in the procedures.

The magnitude of the release is based on actual effluent monitoring readings, plant system parameters (containment pressure), area meteorology and the duration of the release.

I.4 Effluent Monitor Readings Vs Onsite/Offsite Exposure

The procedures referenced in I.3.a/I.3.b establish the relationship between effluent monitor readings and on-site/off-site exposures and contamination for various meteorological conditions.

I.5 Meteorological Information Availability

Meteorological information will be available to the Emergency Operations Facility, the Technical Support Center, the Control Room through use of the Station Operator Aid Computer (OAC) and by direct telephone communication. Meteorological information will be available to the NRC through the Emergency Response Data System (ERDS), Health Physics Network (HPN) or by direct telephone communications with the individual responsible for making off-site dose assessments either at the Technical Support Center or the Emergency Operations Facility.

Meteorological information will also be given to both the county Emergency Operations Centers and the State of North Carolina during the follow-up notification ENF.

I.6 Release Rates/Projected Dose For Offscale Instrumentation

If instrumentation used for dose assessment is offscale or inoperable, dose rates within the Reactor Building will be determined using procedure HP/0/B/1009/002, Alternative Method for Determining Dose Rate Within the Reactor Building, or HP/0/B/1009/006, Procedure for Quantifying High Level Radioactivity Release During Accident Conditions.

I.7/ I.8 Field Monitoring Within E.P.Z.

Field monitoring within the McGuire Emergency Planning Zone will be performed in accordance with fleet and site specific procedures.

Two off-site field monitoring teams are comprised from site personnel and are under the direction of the Field Monitoring Coordinator. On-site monitoring is performed by Radiation Protection personnel under the direction of the OSC Radiation Protection Supervisor. Procedures describe how to obtain the vehicles to be used, routes to be used, sampling and monitoring equipment to be used, locations of TLD's and directions for taking KI tablets.

An emergency radio system is available for the field monitoring teams to use to relay information to the TSC/EOF. The state will be able to monitor the results of the field monitoring teams and relay results to the counties.

I.9 Detect and Measure Radioiodine Concentration in the EPZ

Appropriate instrumentation to measure radioactivity in counts per minute (cpm) and determine dose rate in mrem/hr shall be used for detection and measurement of radioiodine concentration. The air sample will be taken with a Portable Air Sampler equipped with a Silver Zeolite or equivalent cartridge and particulate filter. Air sampling results will be obtained through the use of a portable single channel Analyzer and appropriate gamma sensitive detector OR a count rate meter utilizing direct corrected count rate (ccpm) of Silver Zeolite or equivalent cartridge cross referenced against an estimated Iodine 131 μCi/cc (microcuries per cubic centimeter) concentration attachment.

Interference from the presence of noble gas and background radiation shall not decrease the minimum detectable activity of 1E-7 μCi/cc (microcuries per cubic centimeter) under field conditions.

These samples taken by the offsite monitoring teams will be evaluated further by one of the available laboratory facilities described in Section C.3. A multi-channel analyzer will be used to perform this evaluation.

I.10 Relationship Between Contamination Levels and Integrated Dose/Dose Rates

Provisions for relating contamination levels, water, and air to dose rates for key isotopes is found in HP/0/B/1009/021.

I.11 Plume Tracking

The state of North Carolina has arrangements to locate and track an airborne plume of radioactive materials. Duke Energy will have monitoring teams in the field, fixed TLD sites and the capability for airborne monitoring to assist in plume tracking.

J. PROTECTIVE RESPONSE

To assure that a range of protective actions is available for the plume exposure pathway for emergency workers and the public. Guidelines for protective actions during an emergency, consistent with Federal guidance, are developed and in place and protective actions for the ingestion exposure pathway appropriate to the locale have been developed.

To protect onsite personnel during hostile action and ensure the continued ability to safely shutdown the reactor and perform the functions of the emergency plan a range of protective actions are in place.

J.1. Onsite Alerting and Notification

The means and time required to warn, alert and/or notify employees not having emergency assignments (non-essential), visitors, contractor and construction personnel and other individuals who may be on or passing through the owner-controlled area are described in RP/0/A/5700/011, Conducting a Site Assembly, Site Evacuation or Containment Evacuation.

Methods to notify and alert onsite personnel (essential and non-essential) during hostile action activities are describe in AP/0/A/5500/047, Security Events and RP/0/A/5700/011, Conducting A Site Assembly, Site Evacuation or Containment Evacuation.

J.2 Evacuation Routes and Transportation

The Operations Shift Manager/Emergency Coordinator or designee uses site and local area maps, information available from meteorological tower instrument readouts and current radiological data for determining the evacuation route. Evacuation routes for onsite individuals to suitable offsite locations, including alternatives for weather or radiological conditions is provided in RP/0/A/5700/011, Conducting a Site Assembly, Site Evacuation or Containment Evacuation.

J.3 Personnel Monitoring

Radiation Protection emergency personnel survey teams equipped with portable monitoring instruments will monitor employees, visitors, contract workers and vehicles for contamination at the Relocation Sites. Monitoring will be performed in accordance with Radiation Protection procedure HP/0/B/1009/024, Personnel Monitoring for Emergency Conditions.

J.4 Site Evacuation Procedures - Decontamination/Non-Essential Personnel Criteria

Non-essential personnel may be evacuated from the plant site in the event of a Site Area Emergency and will be evacuated in the event of a General Emergency. Provisions are made for the decontamination of vehicles and personnel at an off-site location if the situation should warrant.

All members of the general public who are on-site must be evacuated if there is a possibility they may be exposed to dose rates in excess of any of the following:

External Radiation Level = 2 mrems/hr
Airborne Radioactivity = 1 times DAC for an unrestricted area

During hostile threat conditions that do not require Site Sheltering, Site Relocation of non-essential personnel to locations outside of the protected area are performed in accordance with AP/0/A/5500/047, Security Events and RP/0/A/5700/011, Conducting A Site Assembly, Site Evacuation or Containment Evacuation.

J.5 Personnel Accountability

Within thirty minutes of a Site Assembly, all persons within the Protected Area at the McGuire Nuclear Site can be accounted for and any person(s) determined to be missing, will be identified by name. RP/0/A/5700/011, Conducting a Site Assembly, Site Evacuation or Containment Evacuation, provides for the accounting of personnel (on site) continuously thereafter.

When hostile threat conditions permit, personnel accountability is performed in accordance with RP/0/A/5700/011, Conducting A Site Assembly, Site Evacuation or Containment Evacuation.

J.6 Protective Equipment - Breathing Apparatus, Protective Clothes, KI

Protective equipment and supplies will be distributed to respiratory qualified personnel remaining on site or arriving on site during the emergency to minimize the effects of radiological exposures or contamination. Protective measures will be utilized as follows:

Individual Respiratory Protection - Respiratory protective equipment will be used when airborne radioactivity levels exceed the appropriate limits specified in 10CFR20, Appendix B.

Self-contained breathing apparatus will also be used in areas that are deficient in oxygen or when fighting fires. Respiratory protective equipment will be issued by Radiation Protection or Safety and Health Services. Self-contained breathing apparatus are available with other fire fighting equipment for use by the site fire brigade.

Individual Thyroid Protection - All efforts should be made to utilize respiratory protective equipment to minimize ingestion and/or inhalation of radionuclides and to maintain internal

exposure below the limits specified in 10CFR20, Appendix B. However, if an unplanned incident involves the accidental or potential ingestion or inhalation of radioactive iodine, Potassium Iodide Tablets (KI) are available to distribution by AD-EP-ALL-0204, Distribution of Potassium Iodide Tablets in the Event of a Radioiodine Release.

Use of Protective Clothing - Protective clothing will be issued when contamination levels exceed 1000 dpm/100 cm^2 beta-gamma and 20 dpm/100 cm^2 alpha of smearable contamination. Protective clothing items are located in the Change Rooms inside the Radiation Control Area, available for emergency use. Special fire-fighting protective clothing and equipment is available in designated site supply storage areas for use by fire brigade personnel.

J.7 Protective Actions Recommendations

The Emergency Coordinator (Operations Shift Manager or Station Manager) or the EOF Director shall be responsible for contacting the state and/or local governments to give prompt notification for implementing protective measures within the plume exposure pathway, and beyond it if necessary.

Protective Action Guides are adopted from EPA 400-R-92-001 and in the State Plan guidance on the use of KI and are shown in Figure J-1. A flowchart to aid the Emergency Coordinator/EOF Director in making Protective Action Recommendations is also shown in Figure J-1. {PIP-G-03-606}

As described in section B.4, the Emergency Coordinator and the EOF Director are responsible for making protective action recommendations. Prior to activation of the EOF, the Emergency Coordinator will be responsible for making these recommendations. After activation of the EOF, the EOF Director assumes this responsibility. Protective action recommendations will be provided to the off-site authorities (states and counties) who are responsible for implementing public protective actions. The pre-established warning message format (Figure E-1) will be used in transmitting the recommendations.

The mechanism for making dose projections upon EOF activation is as follows:

The Radiological Assessment Manager is responsible for making dose projections on a periodic basis. Calculations are made using a computer based dose projection model to calculate projected dose to the population-at-risk for either potential or actual release conditions. For conditions in which a release has not occurred but fuel damage has taken place and radiation levels in the containment building atmosphere are significant, a scoping analysis will be performed to determine what recommendations would be made if containment integrity were lost at that time. The analysis will be based upon a design leak rate and upon a projected penetration failure indicated by a hole size of certain diameter. This analysis will include the use of actual containment pressure, realistic meteorology, and actual source term. A Total Effective Dose Equivalent and Committed Dose Equivalent thyroid dose will be calculated at various distances from the plant (site boundary, 2 miles, 5 miles, 10 miles and beyond if needed). These dose projections are compared to the Protective Action Guides in Figure J-1, which are derived from the "Manual of Protective Action Guides and Protective Actions for

Nuclear Incidents" (EPA-400-R-92-001) and in the State Plan guidance on use of KI. Based on these comparisons, protective action recommendations are developed by the Radiological Assessment Manager. The Radiological Assessment Manager informs the EOF Director of the situation and recommendations for protective actions. {PIP-G-03-606}

If dose projections show that PAGs have been exceeded at 10 miles, the dose assessment code and in-field measurements, when available, shall be used to calculate doses at various distances down wind to determine how far from the site PAG levels are exceeded. The Radiological Assessment Manager shall forward the results to the EOF Director who will communicate this information to the offsite authorities.

J.8 Evacuation Time Estimates

An Analysis of Evacuation Time Estimates is available at the site and a summary of the Time Estimates is included in Figure J-3 and Appendix 4.

Under normal weather and for the critical time period (weekday during school hours), the maximum evacuation time for the McGuire EPZ is 4 hours 35 minutes. The critical component in the evacuation is the permanent resident population, all other segments of the population can be evacuated in less than the maximum time.

Under adverse weather conditions (winter storm), the evacuation time for the McGuire EPZ is 5 hours 40 minutes. This evacuation time assumes evacuation of the entire EPZ. Figure J-3 provides more detailed information including evacuation times for individual zones. Appendix 4 discusses the ETE used by the site, state and local planners.

A description of the methods and assumptions used in developing the analysis of evacuation time estimates is included in the current Evacuation Time Estimate study for McGuire Nuclear Site. (MNS-ETE-12132012, Rev. 000; MNS EVACUATION TIME ESTIMATES (ETE) DATED December 2012). The Evacuation Time Estimates will be considered in evaluating protective action recommendations from the Technical Support Center or the Emergency Operations Facility. A copy of the most recent study is available in the MNS Master File under MNS-ETE-12132012.000 or EP Office area.

An updated ETE analysis will be submitted to the NRC under §50.4 no later than 365 days after MNS determination that the criteria for updating the ETE have been met and at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.

The criteria for determination that an updated ETE analysis have been met:

a) The availability of the most recent decennial census data from the U.S. Census Bureau:

   OR

b) If at any time during the decennial period, the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ to increase by 25 percent or 30 minutes, whichever is less, from the currently NRC approved or updated ETE.

During the years between decennial period censuses MNS will estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. MNS will maintain these estimates so that they are available for NRC inspection during the period between decennial censuses and shall submit these estimates to the NRC with any updated ETE analysis.

MNS ETE analysis, using the 2010 decennial census data from the U.S. Census Bureau, was submitted to the NRC via §50.4 on December 13, 2012.

J.9 Implementing Protective Measures

If protective actions for any off-site location are deemed necessary, the emergency management agency of the affected County, in conjunction with the appropriate State agency (NC-Department of Public Safety) has the legal authority and responsibility for initiating protective measures for the general public in the plume exposure pathway EPZ including evacuation of these areas. Use of sheltering as an alternative to evacuation for impediments to evacuation and special populations is a decision that will be made by the offsite officials. Sheltering in lieu of evacuation should also be considered during a short term release. A short term release is any release that can be accurately projected to be less than the affected protective action zone's evacuation time. An example would be a "puff release". In addition, sheltering may be appropriate (when available) for areas not designated for immediate evacuation because: 1) it positions the public to receive additional instructions; and 2) it may provide protection equal to or greater than evacuation. {PIP G-04-337} Public notification of the emergency, the resources used to determine if an evacuation is necessary, the evacuation routes, and the methods used for evacuating persons in the plume exposure pathway EPZ are outlined in the appropriate County and State emergency plans.

See County and State Plans for more detailed information.

For hostile action events, a range of protective actions for onsite workers including site relocation of essential personnel from potential target buildings, timely evacuation of non-essential site personnel, dispersal of critical personnel to safe locations, on site sheltering of

personnel and accountability of personnel after the attack are provided in emergency plan implementing procedures AP/0/A/5500/047, Security Events and RP/0/A/5700/011, Conducting A Site Assembly, Site Evacuation or Containment Evacuation.

J.10 Implementation of Protective Measures for Plume Exposure Pathway

J.10.a EPZ Maps

Figures J-1 and 2 describe the EPZ's, government jurisdictions and evacuation zones for McGuire Nuclear Site. Evacuation routes are displayed in Figure J-5.

J.10.b EPZ - Population Distribution Map

Figure J-6 describes the population distribution by Emergency Planning subzone. The FSAR describes the population distribution by sector.

J.10.c EPZ - Population Alerting and Notification

As described in Appendix 3 of this plan, a system exists for alerting and notifying the population (resident and transient) within the EPZ areas. This system is activated by the county and state organization and includes the use of large fixed-site sirens and the Emergency Alert System (EAS). A back-up means of alerting and notification is described in the State and County Emergency Plans.

J.10.d EPZ - Protecting Immobile Persons

See County and State Plans.

J.10.e Use of Radioprotective Drugs For Persons in EPZ

See County and State Plans.

J.10.f Conditions For Use of Radioprotective Drugs

See County and State Plans.

J.10.g State/County Relocation Plans

See County and State Plans.

J.10.h Reception Center Locations

See County and State Plans.

J.10.i Evacuation Route - Traffic Capacities

See County and State Plans.

J.10.j Evacuated Area Access Control

See County and State Plans.

J.10.k Planning For Contingencies in Evacuation

See County and State Plans.

J.10.l State/County Evacuation Time Estimates

The estimates shown in Appendix 4 are references in the County and State Plans.

J.10.m Bases For Protective Action Recommendations

Figure J-1 describes the considerations used by Duke management in developing protective action recommendations.

J.11 Ingestion Pathway Planning

See County and State Plans.

J.12 Reception Center - Registering & Monitoring

See County and State Plans

Figure J-1

Guidance for Offsite Protective Actions

INITIAL PAR Determination

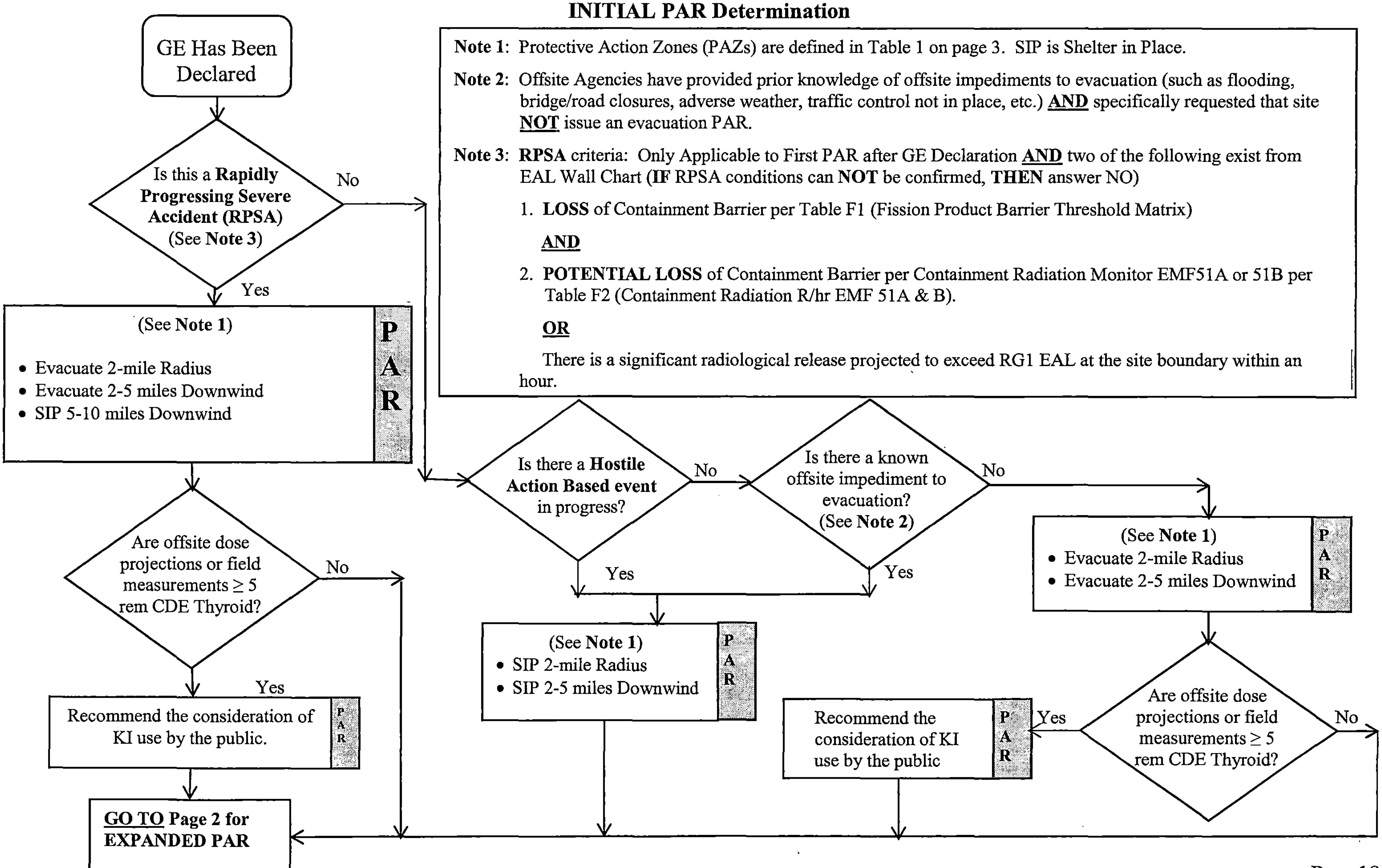


Figure J-1

EXPANDED PAR Determination

Note 1: Protective Action Zones (PAZs) are defined in Table 1 on page 3. SIP is Shelter in Place.

(IF a PAZ has been accurately selected for evacuation, it shall remain selected.)

Note 2: Offsite Agencies have provided prior knowledge of offsite impediments to evacuation (such as flooding, bridge/road closures, adverse weather, traffic control not in place, etc.) **AND** specifically requested that site **NOT** issue an evacuation PAR.

Note 3: A short-term release is one that can be accurately projected to be < three hours and controlled by the licensee. This consideration would typically apply to controlled venting of containment.

Note 4: Plant conditions exist which would require the classification of a General Emergency per the EALs. This does NOT include consideration of offsite dose-based EALs.

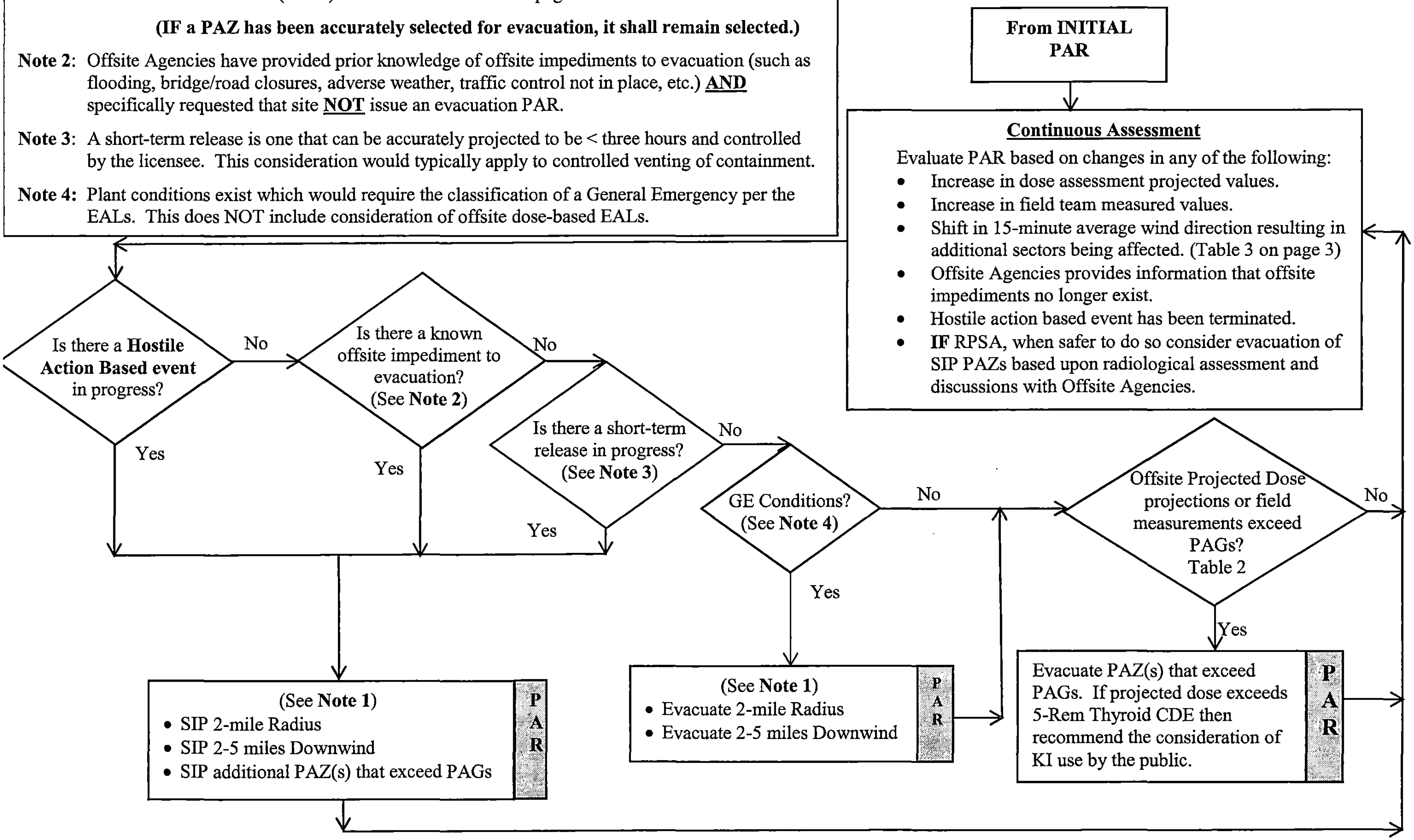


FIGURE J-1
3 OF 3
GUIDANCE FOR OFFSITE PROTECTIVE ACTIONS

Table 1			
Protective Action Zones			
Wind Direction	2 Mile Radius	2-5 Miles Downwind	5-10 Miles Downwind
0.1 - 22.5	B,C,L,M	D,O,R	E,F,S
22.6 – 45.0	B,C,L,M	D,O,R	E,Q,S
45.1 – 67.5	B,C,L,M	D,N,O,R	E,P,Q,S
67.6 – 90.0	B,C,L,M	D,N,O,R	P,Q,S
90.1 – 112.5	B,C,L,M	N,O,R	K,P,Q,S
112.6 – 135.0	B,C,L,M	A,N,O,R	I,K,P,Q,S
135.1 – 157.5	B,C,L,M	A,N,O	I,K,P,Q
157.6 – 180.0	B,C,L,M	A,N	H,I,J,K,P
180.1 – 202.5	B,C,L,M	A,N	G,H,I,J,K,P
202.6 – 225.0	B,C,L,M	A,D,N	G,H,I,J,K,P
225.1 – 247.5	B,C,L,M	A,D	F,G,H,I,J
247.6 – 270.0	B,C,L,M	A,D	F,G,H,I,J
270.1 – 292.5	B,C,L,M	A,D	E,F,G,H,J
292.6 – 315.0	B,C,L,M	A,D,R	E,F,G
315.1 – 337.5	B,C,L,M	D,R	E,F,G,S
337.6 – 360.0	B,C,L,M	D,R,O	E,F,S

Table 2	
PROTECTIVE ACTION GUIDES (PAGs) (Projected Dose or Field Measurements)	
Total Effective Dose Equivalent (TEDE)	Committed Dose Equivalent (CDE) Thyroid
$\geq$ 1 Rem	$\geq$ 5 Rem

Table 3	
WIND SPEED/DIRECTION	
ENF Radiation Protection Manager	Line 9
McGuire SDS	Group Display **ERORD5**
DPC Meteorological Lab	704-382-0139 704-373-7896
National Weather Service Greer, S.C	864-879-1085 800-268-7785

Figure J-2
Description of Evacuation Regions

Region	Description	Degrees From North:	Sub-Zone																		
			A	**B**	**C**	**D**	**E**	**F**	**G**	**H**	**I**	**J**	**K**	**L**	**M**	**N**	**O**	**P**	**Q**	**R**	**S**
R01	2-Mile Ring	N/A		X	X									X	X						
R02	5-Mile Ring	N/A	X	X	X	X								X	X	X	X			X	
R03	Full EPZ	N/A	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
R04	Catawba County	N/A											X								
R05	Gaston County	N/A																		X	X
R06	Iredell County	N/A	X								X	X									
R07	Lincoln County	N/A												X	X	X	X	X	X		
R08	Mecklenburg County	N/A	X	X	X	X	X	X	X	X											
Evacuate 2-Mile Radius and Downwind to 5 Miles																					
Region	**Wind Direction From:**	**Degrees From North:**	**Sub-Zone**																		
			A	**B**	**C**	**D**	**E**	**F**	**G**	**H**	**I**	**J**	**K**	**L**	**M**	**N**	**O**	**P**	**Q**	**R**	**S**
R09	N, NNE	0.1 - 45.0		X	X	X								X	X					X	
R10	NE, ENE	45.1 - 90.0		X	X									X	X		X			X	
R11	E,ESE, SE	90.1 - 157.6		X	X									X	X	X	X				
R12	SSE	157.5 - 180.0		X	X									X	X	X					
R13	S	180.1 - 202.5	X	X	X									X	X	X					
R14	SSW, SW	202.6 - 247.5	X	X	X									X	X						
R15	WSW, W	247.6 - 292.5	X	X	X	X								X	X						
R16	WNW, NW, NNW	292.6 - 360.0		X	X	X								X	X						
Evacuate 5-Mile Radius and Downwind to the EPZ Boundary																					
Region	**Wind Direction From:**	**Degrees From North:**	**Sub-Zone**																		
			A	**B**	**C**	**D**	**E**	**F**	**G**	**H**	**I**	**J**	**K**	**L**	**M**	**N**	**O**	**P**	**Q**	**R**	**S**
R17	N	0.1 - 22.5	X	X	X	X	X	X						X	X	X	X			X	X
R18	NNE	22.6 - 45.0	X	X	X	X	X							X	X	X	X			X	X
R19	NE	45.1 - 67.5	X	X	X	X	X							X	X	X	X		X	X	X
R20	ENE	67.6 - 90.0	X	X	X	X								X	X	X	X		X	X	X
R21	E	90.1 - 112.5	X	X	X	X								X	X	X	X	X	X	X	X
R22	ESE	112.6 -135.0	X	X	X	X								X	X	X	X	X	X	X	
R23	SE	135.1 - 157.6	X	X	X	X							X	X	X	X	X	X	X	X	
R24	SSE	157.5 - 180.0	X	X	X	X					X		X	X	X	X	X	X		X	
R25	S	180.1 - 202.5	X	X	X	X					X	X	X	X	X	X	X	X		X	
R26	SSW	202.6 - 225.0	X	X	X	X				X	X	X	X	X	X	X	X	X		X	
R27	SW	225.1 - 247.5	X	X	X	X			X	X	X	X		X	X	X	X			X	
R28	WSW	247.6 - 270.0	X	X	X	X		X	X	X		X		X	X	X	X			X	
R29	W	270.1 - 292.5	X	X	X	X		X	X	X				X	X	X	X			X	

Figure J-2

Region	Description	Degrees From North:	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S
R30	WNW	292.6 - 315.5	X	X	X	X		X	X					X	X	X	X			X	
R31	NW	315.1 - 337.5	X	X	X	X	X	X	X					X	X	X	X			X	
R32	NNW	337.6 - 360.0	X	X	X	X	X	X						X	X	X	X			X	
Staged Evacuation - 2-Mile Radius Evacuates, then Evacuate Downwind to 5 Miles																					
Region	**Wind Direction From:**	**Degrees From North:**	**A**	**B**	**C**	**D**	**E**	**F**	**G**	**H**	**I**	**J**	**K**	**L**	**M**	**N**	**O**	**P**	**Q**	**R**	**S**
R33	N, NNE	0.1 - 45.0		X	X	X								X	X					X	
R34	NE, ENE	45.1 - 90.0		X	X									X	X		X			X	
R35	E,ESE, SE	90.1 - 157.6		X	X									X	X	X	X				
R36	SSE	157.5 - 180.0		X	X									X	X	X					
R37	S	180.1 - 202.5	X	X	X									X	X	X					
R38	SSW, SW	202.6 - 247.5	X	X	X									X	X						
R39	WSW, W	247.6 - 292.5	X	X	X	X								X	X						
R40	WNW, NW, NNW	292.6 - 360.0		X	X	X								X	X						
R41	5-Mile Ring	N/A	X	X	X	X								X	X	X	X			X	

Column header "Sub-Zone" spans columns A–S.

Sub-Zone(s) Shelter-in-Place until 90% ETE for R01 then Evacuate	Sub-Zone(s) Shelter-in-Place	Sub-Zone(s) Evacuate

Figure J-3

MNS ETE Based on 2010 Census

Time to Clear the Indicated Area of 90 Percent of the Affected Population														
	Summer		Summer		Summer	Winter			Winter			Winter	Winter	Summer
	Midweek		Weekend		Midweek Weekend	Midweek			Weekend			Midweek Weekend	Weekend	Midweek
Scenario:	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)
	Midday		Midday		Evening	Midday			Midday			Evening	Midday	Midday
	Good Weather	Rain	Good Weather	Rain	Good Weather	Good Weather	Rain	Ice	Good Weather	Rain	Ice	Good Weather	Special Event	Roadway Impact
Entire 2-Mile Region, 5-Mile Region, EPZ and each County														
R01 (B,C,L,M)	2:35	2:50	1:50	1:50	1:50	2:45	2:55	3:10	1:50	1:50	1:55	1:50	1:50	2:40
R02 (A,B,C,D,L,M,N,O,R)	2:50	3:10	2:25	2:45	2:20	2:45	3:10	3:35	2:25	2:40	3:00	2:20	2:25	3:30
R03 (All Sub-Zones)	4:35	5:00	3:30	3:55	3:10	4:35	5:00	5:40	3:30	3:50	4:10	3:10	3:40	4:50
R04 (K)	2:25	2:25	2:00	2:00	2:00	2:25	2:25	2:25	2:00	2:00	2:00	2:00	2:00	2:25
R05 (R,S)	2:20	2:30	2:10	2:15	2:10	2:20	2:30	2:35	2:05	2:15	2:25	2:05	2:10	2:20
R06 (A,I,J)	2:40	2:50	2:25	2:35	2:20	2:40	2:55	3:15	2:25	2:35	2:55	2:20	2:25	3:25
R07 (L,M,N,O,P,Q)	3:30	3:40	3:05	3:20	2:50	3:20	3:30	3:55	2:50	3:05	3:25	2:40	3:05	3:30
R08 (A,B,C,D,E,F,G,H)	4:35	5:00	3:30	3:55	3:10	4:40	5:05	5:45	3:30	3:50	4:10	3:10	3:35	4:50
2-Mile Region and Keyhole to 5 Miles														
R09 (B,C,D,L,M,R)	2:35	2:50	2:20	2:35	2:30	2:35	2:50	3:10	2:20	2:30	2:55	2:30	2:25	3:35
R10 (B,C,L,M,O)	2:30	2:40	1:55	1:55	1:55	2:35	2:45	2:55	1:55	1:55	2:00	1:55	1:55	2:30
R11 (B,C,L,M,N,O)	2:30	2:35	1:55	1:55	1:55	2:30	2:35	2:50	1:55	1:55	2:00	1:55	1:55	2:30
R12 (B,C,L,M,N)	2:30	2:40	1:55	1:55	1:55	2:35	2:40	2:55	1:55	1:55	1:55	1:55	1:55	2:30
R13 (A,B,C,L,M,N)	2:20	2:40	2:10	2:20	2:00	2:20	2:40	3:00	2:10	2:20	2:40	2:00	2:10	3:45
R14 (A,B,C,L,M)	2:25	2:40	2:10	2:20	2:00	2:25	2:45	3:00	2:10	2:20	2:35	2:00	2:10	3:40

Figure J-3

MNS ETE Based on 2010 Census

	Summer		Summer		Summer	Winter			Winter			Winter	Winter	Summer
	Midweek		Weekend		Midweek Weekend	Midweek			Weekend			Midweek Weekend	Weekend	Midweek
Scenario:	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)
	Midday		Midday		Evening	Midday			Midday			Evening	Midday	Midday
	Good Weather	Rain	Good Weather	Rain	Good Weather	Good Weather	Rain	Ice	Good Weather	Rain	Ice	Good Weather	Special Event	Roadway Impact
2-Mile Region, and Keyhole to 5 Miles														
R15 (A,B,C,D,L,M)	2:50	3:15	2:30	2:45	2:25	2:50	3:10	3:35	2:25	2:40	3:05	2:20	2:30	3:35
R16 (B,C,D,L,M)	2:35	2:50	2:25	2:35	2:30	2:35	2:50	3:10	2:20	2:30	2:55	2:30	2:25	3:40
5-Mile Region and Keyhole to EPZ Boundary														
R17 (A,B,C,D,E,F,L,M,N, O,R,S)	4:10	4:40	3:15	3:35	3:00	4:20	4:40	5:15	3:10	3:30	3:50	2:55	3:10	4:25
R18 (A,B,C,D,E,L,M,N, O,R,S)	2:45	3:05	2:30	2:45	2:30	2:50	3:05	3:25	2:30	2:40	3:00	2:25	2:30	3:15
R19 (A,B,C,D,E,L,M,N, O,Q,R,S)	2:55	3:15	2:35	2:50	2:35	2:55	3:15	3:35	2:35	2:45	3:05	2:35	2:35	3:20
R20 (A,B,C,D,L,M,N, O,Q,R,S)	2:45	3:10	2:25	2:40	2:20	2:45	3:00	3:30	2:20	2:35	2:55	2:20	2:25	3:25
R21 (A,B,C,D,L,M,N, O,P,Q,R,S)	3:00	3:15	2:35	2:55	2:30	2:55	3:15	3:40	2:30	2:45	3:10	2:30	2:35	3:30
R22 (A,B,C,D,L,M,N, O,P,Q,R,S)	3:00	3:20	2:35	2:55	2:35	3:00	3:15	3:40	2:30	2:45	3:10	2:30	2:40	3:30
R23 (A,B,C,D,K,L,M,N, O,P,Q,R)	3:05	3:20	2:40	2:55	2:35	3:00	3:15	3:45	2:30	2:45	3:10	2:30	2:40	3:35
R24 (A,B,C,D,I,K,L,M,N, O,P,R)	3:05	3:25	2:45	3:00	2:45	3:05	3:20	3:45	2:35	2:50	3:15	2:40	2:45	3:30

Figure J-3

MNS ETE Based on 2010 Census

	Summer		Summer		Summer	Winter			Winter			Winter	Winter	Summer
	Midweek		Weekend		Midweek Weekend	Midweek			Weekend			Midweek Weekend	Weekend	Midweek
Scenario:	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)
	Midday		Midday		Evening	Midday			Midday			Evening	Midday	Midday
	Good Weather	Rain	Good Weather	Rain	Good Weather	Good Weather	Rain	Ice	Good Weather	Rain	Ice	Good Weather	Special Event	Roadway Impact
5-Mile Region and Keyhole to EPZ Boundary														
R25 (A,B,C,D,I,J,K,L,M, N,O,P,R)	3:00	3:20	2:45	3:00	2:40	3:05	3:20	3:45	2:40	2:55	3:15	2:35	2:45	3:30
R26 (A,B,C,D,H,I,J,K,L, M,N,O,P,R)	3:15	3:35	2:50	3:05	2:40	3:15	3:30	4:00	2:45	3:00	3:25	2:35	2:50	3:45
R27 (A,B,C,D,G,H,I,J, L,M,N,O,R)	3:45	4:05	3:10	3:30	2:55	3:40	4:05	4:35	3:10	3:25	3:50	2:50	3:15	4:35
R28 (A,B,C,D,F,G,H,J, L,M,N,O,R)	4:30	5:00	3:30	3:50	3:05	4:35	5:05	5:35	3:25	3:45	4:10	3:00	3:35	4:50
R29 (A,B,C,D,F,G,H,L, M,N,O,R)	4:30	4:55	3:25	3:45	3:00	4:30	4:55	5:35	3:20	3:35	4:05	3:00	3:25	4:40
R30 (A,B,C,D,F,G,L,M, N,O,R)	4:25	4:45	3:15	3:40	3:00	4:25	4:50	5:20	3:15	3:35	4:00	3:00	3:25	4:35
R31 (A,B,C,D,E,F,G,L, M,N,O,R)	4:35	4:55	3:35	3:45	3:05	4:30	5:05	5:30	3:25	3:45	4:05	3:05	3:35	4:40
R32 (A,B,C,D,E,F,L,M, N,O,R)	4:10	4:35	3:10	3:30	2:55	4:20	4:40	5:10	3:15	3:30	3:50	2:55	3:15	4:20

Figure J-3

	Summer				Summer	Winter			Winter			Winter	Winter	Winter
	Midweek		Weekend		Midweek Weekend	Midweek			Weekend			Midweek Weekend	Weekend	Midweek
Scenario:	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)
	Midday		Midday		Evening	Midday			Midday			Evening	Midday	Midday
	Good Weather	Rain	Good Weather	Rain	Good Weather	Good Weather	Rain	Ice	Good Weather	Rain	Ice	Good Weather	Special Event	Roadway Impact
Staged Evacuation - 2 Mile Region and Keyhole to 5 miles														
R33 (B,C,D,L,M,R)	3:10	3:20	3:10	3:20	3:30	3:10	3:20	3:30	3:15	3:20	3:30	3:30	3:15	4:00
R34 (B,C,L,M,O)	2:30	2:40	2:25	2:25	2:25	2:35	2:45	2:55	2:25	2:25	2:25	2:25	2:25	2:30
R35 (B,C,L,M,N,O)	2:40	2:45	2:35	2:40	2:35	2:40	2:50	2:55	2:35	2:40	2:45	2:35	2:35	2:40
R36 (B,C,L,M,N)	2:40	2:45	2:35	2:35	2:35	2:40	2:45	2:55	2:35	2:35	2:40	2:35	2:35	2:40
R37 (A,B,C,L,M,N)	2:45	2:55	2:45	2:50	2:50	2:45	2:55	3:05	2:45	2:55	3:00	2:55	2:45	3:35
R38 (A,B,C,L,M)	2:50	2:55	2:50	2:55	2:55	2:50	2:55	3:10	2:50	2:55	3:05	2:55	2:45	3:45
R39 (A,B,C,D,L,M)	3:15	3:25	3:15	3:20	3:25	3:15	3:30	3:45	3:15	3:20	3:35	3:25	3:15	4:00
R40 (B,C,D,L,M)	3:15	3:20	3:15	3:20	3:30	3:15	3:20	3:35	3:15	3:20	3:30	3:30	3:15	4:00
R41 (A,B,C,D,L,M, N,O,R)	3:15	3:25	3:10	3:15	3:20	3:15	3:20	3:40	3:10	3:20	3:30	3:20	3:10	3:55

Figure J-4

MNS ETE Based on 2010 Census

Time to Clear the Indicated Area of 100 Percent of the Affected Population														
	Summer		Summer		Summer	Winter			Winter			Winter	Winter	Summer
	Midweek		Weekend		Midweek Weekend	Midweek			Weekend			Midweek Weekend	Weekend	Midweek
Scenario:	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)
	Midday		Midday		Evening	Midday			Midday			Evening	Midday	Midday
	Good Weather	Rain	Good Weather	Rain	Good Weather	Good Weather	Rain	Ice	Good Weather	Rain	Ice	Good Weather	Special Event	Roadway Impact
Entire 2-Mile Region, 5-Mile Region, EPZ and each County														
R01 (B,C,L,M)	4:30	4:30	4:30	4:30	4:30	4:30	4:35	4:35	4:30	4:30	4:30	4:30	4:30	4:30
R02 (A,B,C,D,L,M,N,O,R)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	5:00	4:35	4:35	4:35	4:35	4:35	4:45
R03 (All Sub-Zones)	6:10	6:50	5:00	5:25	5:05	6:20	7:15	7:55	5:05	5:15	5:40	5:05	5:00	7:10
R04 (K)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35
R05 (R,S)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35
R06 (A,I,J)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	5:20
R07 (L,M,N,O,P,Q)	4:35	4:50	4:35	4:35	4:35	4:35	4:35	5:05	4:35	4:35	4:35	4:35	4:35	4:35
R08 (A,B,C,D,E,F,G,H)	6:05	6:50	4:45	5:10	4:40	6:20	7:00	7:45	4:40	5:15	5:40	4:40	4:50	7:05
2-Mile Region and Keyhole to 5 Miles														
R09 (B,C,D,L,M,R)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	5:15
R10 (B,C,L,M,O)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35
R11 (B,C,L,M,N,O)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35
R12 (B,C,L,M,N)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35

Figure J-4

	Summer		Summer		Summer	Winter			Winter			Winter	Winter	Summer
	Midweek		Weekend		Midweek Weekend	Midweek			Weekend			Midweek Weekend	Weekend	Midweek
Scenario:	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)
	Midday		Midday		Evening	Midday			Midday			Evening	Midday	Midday
	Good Weather	Rain	Good Weather	Rain	Good Weather	Good Weather	Rain	Ice	Good Weather	Rain	Ice	Good Weather	Special Event	Roadway Impact
Entire 2-Mile Region, and Keyhole to 5 miles														
R13 (A,B,C,L,M,N)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	5:30
R14 (A,B,C,L,M)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	5:30
R15 (A,B,C,D,L,M)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	5:45
R16 (B,C,D,L,M)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:40	4:35	4:35	4:35	4:35	4:35	5:20
5-Mile Region and Keyhole to EPZ Boundary														
R17 (A,B,C,D,E,F,L,M,N, O,R,S)	5:55	6:10	5:00	5:10	5:00	6:15	6:30	7:20	5:05	5:10	5:35	5:00	5:05	6:40
R18 (A,B,C,D,E,L,M,N, O,R,S)	4:40	4:40	4:40	4:40	4:40	4:40	4:40	5:10	4:40	4:40	4:40	4:40	4:40	5:40
R19 (A,B,C,D,E,L,M,N, O,Q,R,S)	5:15	6:00	4:55	5:10	4:55	5:35	6:05	6:20	4:55	5:05	5:30	4:55	5:00	5:40
R20 (A,B,C,D,L,M,N, O,Q,R,S)	4:40	4:40	4:40	4:40	4:40	4:40	4:40	5:05	4:40	4:40	4:40	4:40	4:40	5:45
R21 (A,B,C,D,L,M,N, O,P,Q,R,S)	4:40	5:05	4:40	4:40	4:40	4:40	5:05	5:15	4:40	4:40	4:40	4:40	4:40	5:45
R22 (A,B,C,D,L,M,N, O,P,Q,R)	4:40	4:45	4:40	4:40	4:40	4:40	4:40	5:10	4:40	4:40	4:40	4:40	4:40	5:40

Figure J-4

	Summer		Summer		Summer	Winter			Winter			Winter	Winter	Summer
	Midweek		Weekend		Midweek Weekend	Midweek			Weekend			Midweek Weekend	Weekend	Midweek
Scenario:	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)
	Midday		Midday		Evening	Midday			Midday			Evening	Midday	Midday
	Good Weather	Rain	Good Weather	Rain	Good Weather	Good Weather	Rain	Ice	Good Weather	Rain	Ice	Good Weather	Special Event	Roadway Impact
5-Mile Region and Keyhole to EPZ Boundary														
R23 (A,B,C,D,K,L,M,N, O,P,Q,R)	4:40	5:10	4:40	4:40	4:40	4:40	4:50	5:25	4:40	4:40	4:45	4:40	4:40	5:45
R24 (A,B,C,D,I,K,L,M,N, O,P,R)	4:40	5:05	4:40	4:40	4:40	4:40	4:50	5:20	4:40	4:40	4:40	4:40	4:40	5:40
R25 (A,B,C,D,I,J,K,L,M, N,O,P,R)	4:40	5:05	4:40	4:40	4:40	4:40	4:45	5:20	4:40	4:40	4:40	4:40	4:40	5:40
R26 (A,B,C,D,H,I,J,K,L, M,N,O,P,R)	4:40	5:05	4:40	4:40	4:40	4:40	4:45	5:20	4:40	4:40	4:40	4:40	4:40	6:05
R27 (A,B,C,D,G,H,I,J, L,M,N,O,R)	5:05	5:25	4:40	4:40	4:40	4:55	5:30	6:05	4:40	4:40	4:40	4:40	4:40	6:40
R28 (A,B,C,D,F,G,H,J, L,M,N,O,R)	6:00	6:30	4:40	4:55	4:40	6:15	7:05	7:30	4:40	4:45	5:30	4:40	4:55	7:10
R29 (A,B,C,D,F,G,H,L, M,N,O,R)	6:10	6:25	4:40	5:00	4:40	6:20	7:00	7:30	4:40	4:45	5:25	4:40	4:40	7:05
R30 (A,B,C,D,F,G,L,M, N,O,R)	6:10	6:35	4:40	4:45	4:40	6:15	6:55	7:20	4:40	4:45	5:20	4:40	4:40	7:00
R31 (A,B,C,D,E,F,G,L, M,N,O,R)	6:05	6:45	4:45	5:10	4:40	6:15	7:15	7:50	4:50	5:05	5:35	4:40	4:50	6:55
R32 (A,B,C,D,E,F,L,M, N,O,R)	6:00	6:30	4:40	4:50	4:40	6:05	6:35	7:00	4:55	5:00	5:20	4:40	4:55	6:30

Figure J-4

	Summer		Summer		Summer	Winter			Winter			Winter	Winter	Winter
	Midweek		Weekend		Midweek Weekend	Midweek			Weekend			Midweek Weekend	Weekend	Midweek
Scenario:	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)
	Midday		Midday		Evening	Midday			Midday			Evening	Midday	Midday
	Good Weather	Rain	Good Weather	Rain	Good Weather	Good Weather	Rain	Ice	Good Weather	Rain	Ice	Good Weather	Special Event	Roadway Impact
Staged Evacuation - 2 mile Region and Keyhole to 5 Miles														
R33 (B,C,D,L,M,R)	4:35	4:55	4:35	4:40	4:35	4:35	4:50	5:10	4:35	4:45	5:10	4:35	4:35	5:25
R34 (B,C,L,M,O)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35
R35 (B,C,L,M,N,O)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35
R36 (B,C,L,M,N)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35
R37 (A,B,C,L,M,N)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	5:30
R38 (A,B,C,L,M)	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	4:35	5:30
R39 (A,B,C,D,L,M)	4:40	4:55	4:35	4:45	4:35	4:40	5:05	5:15	4:35	4:45	5:10	4:35	4:35	5:55
R40 (B,C,D,L,M)	4:35	4:55	4:35	4:45	4:35	4:50	4:50	5:10	4:35	4:45	5:10	4:35	4:35	5:25
R41 (A,B,C,D,L,M, N,O,R)	4:55	5:00	4:35	4:45	4:35	5:00	5:10	5:20	4:35	4:50	5:10	4:35	4:35	5:55

Figure J-5

Evacuation Route map for MNS

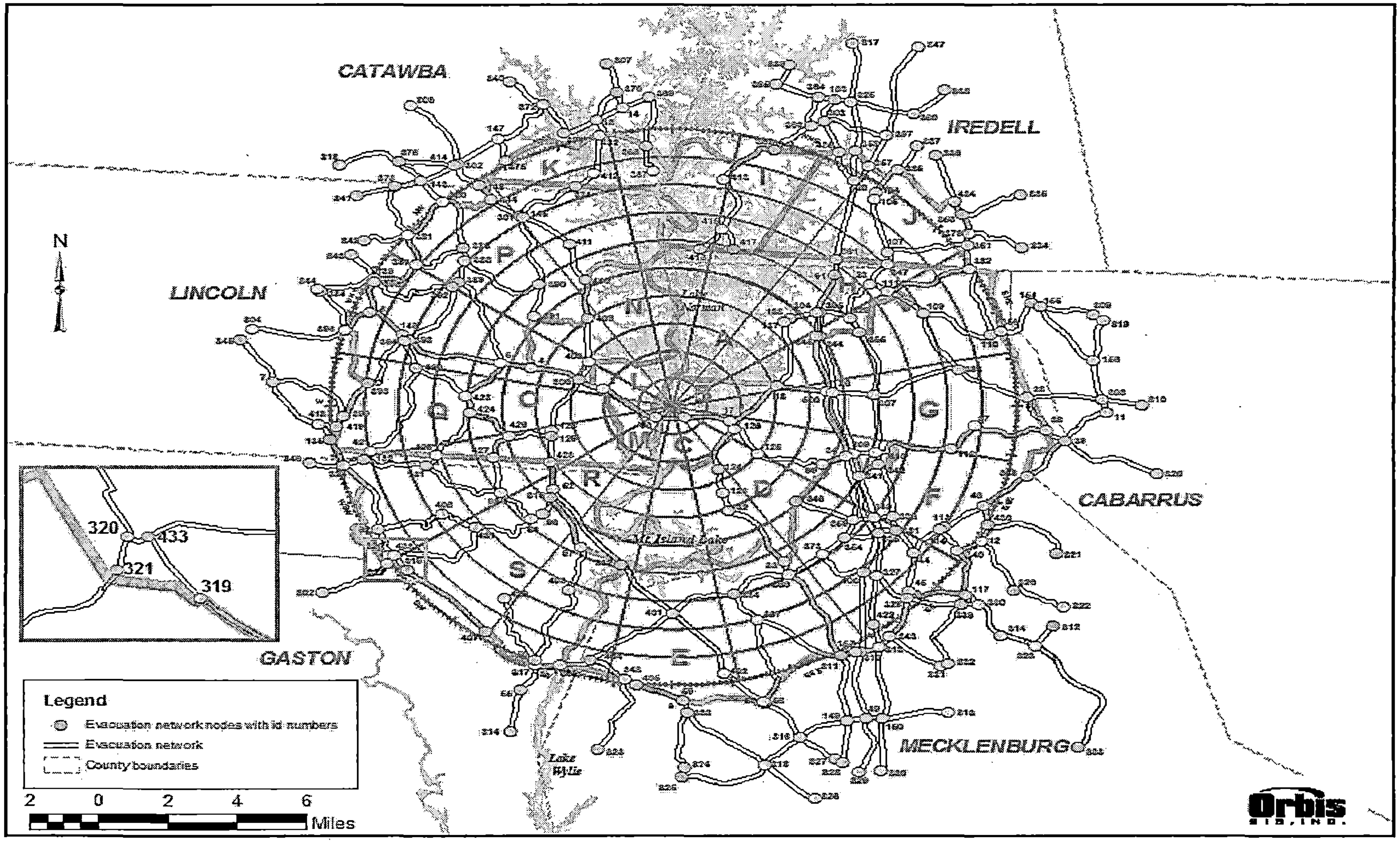


Figure J-6

Summary of Population and Demand

Sub-Zone	Residents	Transit-Dependent	Transients	Employees	Special Facilities	Schools	Shadow Population	External Traffic	Total
A	18433	335	1478	3632	0	338			24,216
B	931	17	0	2231	0	896			4,075
C	1484	27	0	0	0	0			1,511
D	22994	418	2065	1163	102	4900			31,642
E	37228	674	472	2354	0	4134			44,862
F	30364	552	7850	14154	740	6591			60,251
G	25408	462	1150	3311	149	5356			35,836
H	9665	176	694	2604	0	4826			17,965
I	8053	146	19	121	0	0			8,339
J	7447	135	128	1398	107	1471			10,686
K	2272	41	111	0	0	0			2,424
L	1247	23	13	0	0	0			1,283
M	238	4	0	0	0	0			242
N	5381	98	134	1079	0	509			7,201
O	3705	67	0	501	0	0			4,273
P	10377	189	5232	1048	112	3408			20,366
Q	3394	62	13	56	0	0			3,525
R	1667	30	0	43	0	0			1,740
S	14970	272	0	756	236	1652			17,886
Shadow Region						9085	61906		70,991
Total	205,258	3,728	19,359	34,451	1,446	43,166	61,906		369,314

NOTE: Shadow population has been reduced to 20%.
NOTE: Shadow Facilities include both medical facilities and correctional facilities.

K. RADIOLOGICAL EXPOSURE CONTROL

To assure that means for controlling radiological exposures in an emergency are established for emergency workers.

K.1 Onsite Exposure Guidelines

Onsite exposure guidelines consistent with EPA 400-R-92-001, Table 2-2, "Guidance on Dose Limits for Workers Performing Emergency Services" are shown in Figure K-1.

Members of outside emergency services responding to a call from the site are considered emergency workers and must also be protected from excessive radiation doses. Their doses are not to exceed guidelines as established in Figure K-1.

K.2 Doses in Excess of 10CFR Part 20

The Emergency Coordinator is responsible for authorizing emergency workers to receive doses in excess of 10CFR20 limits. An on-site radiation protection program shall be implemented during emergencies which shall be consistent with ALARA conditions. The site will be responsible for providing medical treatment and rescue efforts for life-saving missions. Site procedures are in place for expeditious decision-making with reasonable consideration of the relative risks involved in a lifesaving mission involving radiation exposure.

K.3 Emergency Personnel Exposure and Records

K.3.a Distribution of Dosimetry

Provisions have been made for maintaining records of emergency personnel during a radiological emergency on a 24-hour per day basis. The Operations Support Center will provide a means for keeping track of exposure to personnel involved in a radiological accident. Distribution of dosimeters (self-reading and TLD badges) will be provided for all personnel.

The issuance of High Range and/or Multiple Dosimetry will be in accordance with Radiation Protection procedures.

Should any offsite agency respond to an emergency at the site during a nuclear emergency, dosimeters will be provided for their use to determine any exposure.

K.3.b Dose Records

The Operations Support Center through the Radiation Protection section shall have the responsibility of keeping records of the doses received by emergency personnel involved in any radiological accident. Normal operating procedures shall be followed for the use of dosimeters and the TLD badges. Distribution of the dosimeters and badges shall be through Radiation Protection.

K.4 <u>State/Local Plan for Authorizing Doses Exceeding PAG's</u>

See County and State Plans.

K.5 <u>Decontamination</u>

K.5.a Action Levels For Determining the Need For Decontamination

Guidelines as established in the System Radiation Protection Procedures will be used to determine action levels for decontamination. Pre-planning efforts have been established by the Radiation Protection Section.

K.5.b Radiological Decontamination

AD-RP-ALL-2009, Personnel Contamination Monitoring and Reporting, defines the specific action levels for determining the need for decontamination of personnel. PT/0/A/4600/088, Functional Check of Emergency Vehicle and Equipment, defines the means for availability of supplies, instruments and equipment. Radiation Protection procedures provides direction for waste disposal. HP/0/B/1009/024, Personnel Monitoring for Emergency Conditions promotes means for decontamination of emergency personnel. Handling of contaminated injured personnel is provided in HP/0/B/1009/022, Accident and Emergency Response. Transportation of contaminated injured personnel is described in site procedure RP/0/A/5700/005, Care and Transportation of Injured Individual(s) From Site to Offsite Medical Facility.

K.6 <u>Contamination Control Measures</u>

K.6.a <u>Area Access Control</u> - The site will be evacuated when site management declares a Site Evacuation and a potential threat exists for safety of non-essential personnel. Once the site has been evacuated, access to the site will be limited by the Highway Patrol on the public highway and then Site Security will limit access to the site except through established access procedures.

K.6.b <u>Drinking Water and Food Supplies</u> - Drinking water and food supplies can be brought in by private vendor if necessary. Arrangements will be made by the Commodities and Facilities Manager/Designee.

K.6.c Recovery efforts will be determined by the Emergency Operations Facility Organization (see Section M).

K.7 Decontamination of Personnel at Relocation Assembly Area

Should non-essential plant personnel be evacuated from site to a relocation area, provisions for extra clothing and decontaminants suitable for any type of contamination have been made. Extra clothing and supplies have been placed at the relocation site to take care of site personnel.

Relocation assembly areas have been determined so that site personnel can be relocated to a safe site quickly and can be decontaminated (if necessary), monitored and released. Records will be made of the exposure of all personnel released from the relocation site. (Site procedures provide for emergency supplies to be provided at the off-site relocation assembly area.)

FIGURE K-1

Emergency Worker Exposure Guidelines (a)			
Dose Limits			
Activity	Total Effective Dose Equivalent (TEDE)	Lens of Eye	Other Organs (b)
All	5 rem	15 rem	50 rem
Protecting Valuable Property	10 rem	30 rem	100 rem
Lifesaving or Protection of Large Populations	25 rem	75 rem	250 rem
Lifesaving or Protection of Large Populations (c)	> 25 rem	> 75 rem	> 250 rem

(a) Excludes declared pregnant women

(b) Includes skin and body extremities

(c) Only on a volunteer basis to persons fully aware of the risks involved

Based on EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

L. MEDICAL AND PUBLIC HEALTH SUPPORT

L.1 Hospital and Medical Support

Hospitals -

Carolinas Medical Center; Charlotte, NC - (Agreement #1 App. 5)

Medical Support -

Local - Carolinas Medical Center (Agreement #1 App. 5)
Backup - Oak Ridge National Lab Hospital, Oak Ridge, Tenn. (Agreement #13, App. 5)

Ambulance Service

Mecklenburg Emergency Medical Services Agency (MEDIC), Charlotte, NC (Agreement #6 App. 5)

L.2 On-site First-Aid Capability

The on-site medical facilities include two First Aid areas and a bioassay facility. One First Aid facility, located outside of the protected area southeast of the Administration Building, is used for the treatment of persons injured, outside the Protected Area, in accidents or emergencies not involving radioactive contamination. This facility is equipped with a sink, a treatment chair, lavatory, a wheeled stretcher, a resuscitator, and medical/first aid supplies. The second First Aid area, located in the Auxiliary Building near the Radiation Protection office area, is used for treatment of persons injured in accidents or emergencies involving radiological contamination. This facility has a decontamination area with a shower, a treatment table and medical/first aid supplies.

The bioassay facility, located in the Administration Building, is used in emergencies to determine if personnel have inhaled or ingested radioactive materials, or if such materials have entered wounds or been absorbed through the skin. The bioassay facility is equipped with a shielded body-burden analyzer and a thyroid-burden analyzer; liquid scintillation counting capabilities for tritium analyses are available in the Radiation Protection area and laboratory in the Radiation Control Area.

L.3 Public, Private, Military Hospitals; Emergency Medical Facilities

See State of North Carolina FNF Plans.

L.4 Transport of Accident Victims

McGuire Nuclear Site has agreements with the Carolinas Medical Center, AND MEDIC to provide transportation for any medical emergency patient (may or may not be contaminated).

If contaminated, efforts will be made to decontaminate the victim before transportation as long as the decontamination does not obstruct the medical attention given the victim or cause an unnecessary delay in transporting. During transportation Radiation Protection personnel will accompany the victim and prevent the further spread of contamination using procedure RP/0/A/5700/005, "Care and Transportation of Contaminated Injured Individuals".

Any item(s) found to be contaminated after the treatment of a contaminated patient at the Carolinas Medical Center or any other medical facility will be decontaminated or replaced by Duke Energy.

M. RECOVERY AND REENTRY PLANNING AND POST ACCIDENT OPERATIONS

M.1 Recovery/Reentry Plans and Procedures

In any site emergency involving radioactive contamination or other emergency condition, the immediate action is directed to limiting the consequences of the incident in a manner that will afford maximum protection to the public. Once the immediate protective actions have established an effective control over the incident, the emergency actions will shift into the recovery phase. AD-EP-ALL-0110, Recovery, provides guidance in establishing the recovery/reentry organization and actions. The EOF Director will inform members of the response organization that a recovery operation is to be initiated and inform them of any changes in the organization that may occur. Implementation of Recovery Operations would occur as follows:

1. Termination of General Emergency or Site Area Emergency
2. Deescalation to Non-Emergency Condition
3. Activation of Recovery Organization

The emergency is not considered to be over until Duke Energy, NRC and the states agree that the public is afforded comparable safety assurance to that which exists during periods of normal station operation. Specifically:

1. Radiation levels in site areas are stable or decreasing with time.

2. Releases of radioactive materials to the environment from the site are under control or have ceased.

3. Any fire, flooding or similar emergency conditions are controlled or have ceased.

Public officials are kept informed of recovery plans so that they can properly carry out their responsibilities to the public.

Periodic briefings of media representatives are held to inform the public of recovery plans and progress made.

Periodic status reports are given to company employees at other locations and to government and industry representatives.

M.1.a Outline of Site Recovery Plans

A Recovery Plan will be developed, based on event, to provide a plan for recovery from and return to an operational status following a notification of Unusual Event, Alert, Site Area Emergency or a General Emergency.

The plans and procedures for site reentry will consider existing as well as potential conditions inside containment. Prior to reentry, the following shall be addressed:

1. Review all available radiation survey data.

2. Determine site areas potentially affected by radiological hazards.

3. Review radiation exposure history of all personnel scheduled to participate in recovery operations. Determine the need for additional personnel.

4. Review the adequacy of radiation survey equipment available. Determine the need for additional equipment and a source of procurement.

5. Pre-plan team activities, including areas to be surveyed, anticipated radiation levels, survey equipment required, protective clothing requirements, access control procedures, exposure control procedures and communication capabilities.

6. Conduct comprehensive radiation survey of site facilities and define all radiological problem areas.

7. Isolate and post with appropriate warning signs all "high radiation areas" and areas of contamination.

8. Perform visual inspection of site areas and equipment.

9. All radiological conditions discovered and existing in the facility as determined by the re-entry survey will be evaluated by site management.

10. Upon evaluation of the radiological condition, site management will determine what procedures are required to restore the site to a normal status.

11. Personnel radiation exposure will be closely controlled and documented.

12. Recovery coordinators will take appropriate actions to ensure that emergency personnel and equipment leaving the Radiation Control Area are not contaminated, that radiological conditions at the scene of the emergency are properly defined, barricaded and posted with appropriate signs.

M.1.b <u>Outline of Recovery Plans</u>

Recovery from an emergency is guided by the following principles:

a. The protection of the public health and safety is the foremost consideration in formulating recovery plans.

b. Public officials would be kept informed of recovery plans so that they can properly carry out their responsibilities to the public.

c. Periodic information would be provided to the news media so that they can provide information to the public regarding recovery plans and progress made.

d. Periodic status reports would be given to company employees at other locations and to government and industry representatives.

e. The radiation doses to employees and other radiation workers would be kept as low as reasonably achievable.

M.2 Recovery Organization

Before entering the Recovery phase, the EOF Director/Emergency Coordinator should establish a Recovery organization that is appropriate for the existing on-site and off-site conditions. Figure M-1 describes a suggested organization structure. It may be modified or supplemented as necessary to fit the particular circumstances. In some situations (such as no core damage), the normal organization may be adequate and a Recovery Organization may not be needed.

The recovery activities would be managed much like a normal outage, except that certain activities unique to the post accident situation may be managed by the Recovery organization. The organization would function as a matrix management organization to coordinate activities with the normal company organization. This organization may be located at the Emergency Operations Facility or the site, as appropriate.

The primary positions in the Recovery Organization are described below:

Recovery Manager - Overall management of recovery activities.
Coordination with Federal, State, and local government agencies

Onsite Recovery Director - Directs the recovery activities onsite to restore the plant to pre-incident conditions.

Offsite Recovery Director - Directs interface with Federal, State and local agencies during the recovery process.

Radiological Assessment Manager (if needed) - Coordinates radiological and environmental assessment with Federal and State agencies. Coordinates offsite radwaste management and decontamination activities.

Company Spokesperson - Manages communications of recovery activities. Informs the news media, employees, etc.

Other Support - Individuals from other company or outside groups may be assigned to specific recovery positions or to perform specific tasks based on recovery needs.

EOF Services Manager or Nuclear supply Chain Manager - Coordinates activities such as purchasing, finance, insurance, human resources, transportation, etc.

Other site management and supervisory personnel will interface with recovery operations as necessary and as warranted.

M.3 Information to Members of Recovery Organization

The EOF Director will take the following steps to inform members of the Emergency Operations Facility, Site Organization, and off-site agencies that Recovery Operations are being initiated and that activities associated with bringing the plant to a safe shutdown condition are completed:

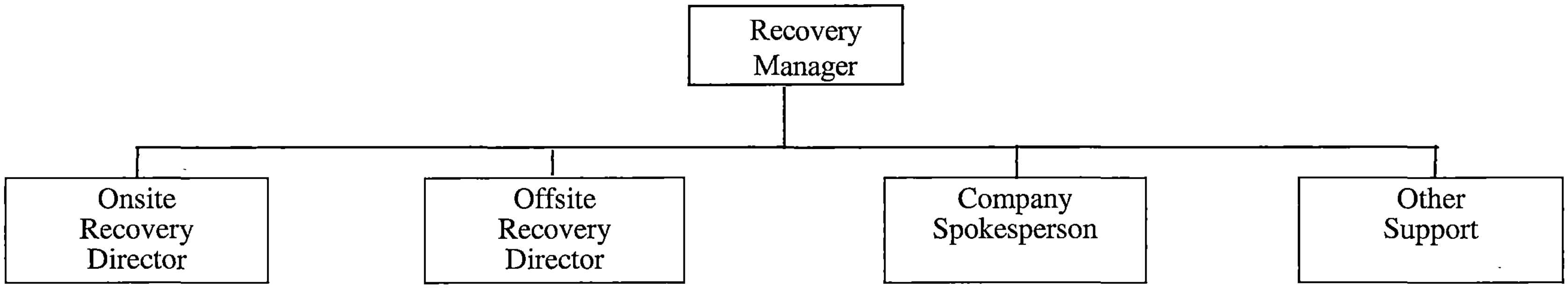
1. Develop a brief message as to the time and date of Recovery Operations initiation as well as any necessary organizational realignments.

2. Distribute the message to EOF Managers, News Manager, Emergency Coordinator, state and local officials, NRC and other representatives. Ask that each person inform those under his/her direction.

M.4 Total Population Exposure Estimates

The Radiological Assessment Manager (or successor in Recovery/ Reentry Operations) will periodically update the estimate of total population exposure. See Section I.10.

FIGURE M-1
RECOVERY ORGANIZATION



Directs Recovery activities to restore the plant to pre-incident conditions or permanent safe shutdown

Directs the interface between Duke Energy and Federal, State and Local Agencies during the Recovery Phase.

Directs the Public Information Program during the Recovery Phase.

Other individuals or groups assigned specific tasks to support activities during the Recovery Phase, such as Legal, Financial, Engineering, etc.

P. RESPONSIBILITY FOR THE PLANNING EFFORT

To assure that responsibilities for plan development, review and distribution of emergency plans are established and that the Emergency Planning staff are properly trained.

P.1 Emergency Planning Staff Training

Emergency Planning Group personnel will attend training/workshops, information exchange meetings with other licensees, and conferences held by industry and government agencies, as available, to maintain current knowledge of the overall planning effort. The Emergency Preparedness Manager is required to attend offsite training on an annual basis. This training will be documented in site Emergency Planning files or the Training group database (i.e. People Soft, etc.).

P.2 Emergency Response Planning

The Site Vice President has the overall authority and responsibility for the Site Emergency Plan. This planning effort is delegated to the Emergency Preparedness Manager.

P.3 Site Emergency Preparedness Manager

The Emergency Preparedness Manager has the overall authority and responsibility for site emergency response planning as well as the responsibility for the development and updating of the site Emergency Plan and coordination of this plan with other response organizations.

P.4 Review of Emergency Plan

Review and updating of the site Emergency Plan and Emergency Plan Implementing Procedures shall be certified to be current on an annual basis. Any changes identified by drills and exercises shall be incorporated into the Site Emergency Plan.

On an annual basis, the Emergency Preparedness Manager will provide each state and local organization responsible for off-site activation and protective action decision-making, a copy of the nuclear site procedures appropriate for their area on emergency classification and notification. A response will be requested by letter within 30 days that a review has been completed with concurrence with the EAL's used for event classification and for protective action recommendations. If problem areas are identified, the Emergency Preparedness Manager will ensure resolution.

P.5 Distribution of Revised Plans

The Emergency Plan and approved changes shall be forwarded to individuals and organizations listed in App. 6. Revised pages shall be dated and marked to show where changes have been made.

P.6 Supporting Plans

Figure P-1 gives a detailed listing of supporting plans to the McGuire Nuclear Site Emergency Plan.

P.7 Implementing Procedures

Written procedures will be established, implemented, and maintained covering the activities associated with emergency plan implementation. Each procedure, and changes thereto, shall be reviewed and approved by the responsible implementing manager prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

McGuire Emergency Plan Implementing Procedures are listed in Figure P-2 with a reference to the section of Emergency Plan implemented by each procedure. Figure P-3 contains the distribution list for McGuire Emergency Plan Implementing Procedures.

P.8 Table of Contents

The McGuire Nuclear Site Emergency Plan contains a specific table of contents. The McGuire Nuclear Site Emergency Plan has been cross referenced to the applicable sections of NUREG-0654 Rev. 1.

P.9 Audit of Emergency Plan

The Nuclear Oversight (NOS) Audit Manager will arrange for an independent review of McGuire Nuclear Station's Emergency Preparedness Program at a frequency specified in 10 CFR 50.54(t). NOS will audit the Plan, Emergency Plan Implementing Procedures, Training, Drills and Exercise, facilities and equipment for conformance with 10 CFR 50.47, 10 CFR 50.54, and 10 CFR 50 Appendix E. The independent review will include the following plans, procedures, training programs, drills/exercises, equipment, and State/local government interfaces:

1. McGuire Nuclear Site Emergency Plan and Implementing Procedures
2. State/Local Support Agency Training Program
3. Emergency Response Training Program
4. Public and Media Training/Awareness
5. Equipment - Communications, Monitoring, Meteorological, Public Alerting
6. State/Local Plan Interface

The review findings will be submitted to the appropriate corporate and nuclear site management. Appropriate portions of the review findings will be reported to the involved federal, state, and local organizations. The corporate or nuclear site management, as appropriate, will evaluate the findings affecting their area of responsibility and ensure effective corrective actions are taken. The result of the review, along with recommendations for improvements, will be documented and retained for a period of five years.

P.10 Telephone Number Updates

Telephone numbers reflected in the online organization charts will be updated quarterly in accordance with PT/0/A/4600/091, Periodic Test of Technical Support Center Communications and Supplies.

DUKE ENERGY
MCGUIRE NUCLEAR SITE
FIGURE P-1

SUPPORTING PLANS

1. North Carolina Emergency Response Plan in support of McGuire Nuclear Site

2. South Carolina Operational Radiological Emergency Response Plan in support of Fixed Nuclear Facilities (McGuire Nuclear Site)

3. Iredell County, N.C., Radiological Emergency Response Plan in Support of the McGuire Nuclear Site

4. Mecklenburg County, N.C., Radiological Emergency Response Plan in Support of the McGuire Nuclear Site

5. Gaston County, N.C., Radiological Emergency Response Plan in Support of the McGuire Nuclear Site

6. Lincoln County, N.C., Radiological Emergency Response Plan in Support of the McGuire Nuclear Site

7. Catawba County, N.C., Radiological Emergency Response Plan in Support of the McGuire Nuclear Site

8. Cabarrus County, N.C., Radiological Emergency Response Plan in Support of the McGuire Nuclear Site

9. Emergency Response Plan, Water Reactors Division, Westinghouse Electric Corporation

10. N.R.C. Region II Incident Response Plan

11. Interagency Radiological Assistance Plan - Region 3 - U.S. Department of Energy

12. INPO Emergency Response Plan

MCGUIRE

FIGURE P-2

PAGE 1 OF 4

EMERGENCY PLAN IMPLEMENTING PROCEDURES

Procedure #	Title	Emergency Plan Section Implemented
AP/0/A/5500/047	Security Events (Proprietary Info)	Section J
RP/0/A/5700/000	Classification of Emergency	Section D, E, I
RP/0/A/5700/001	Notification of Unusual Event	Section D, E, I.1, J.7
RP/0/A/5700/002	Alert	Section D, E, I.1, J.7
RP/0/A/5700/003	Site Area Emergency	Section D, E, I.1, J.7, M.1
RP/0/A/5700/004	General Emergency	Section D, E, I.1, J.7, M.1
RP/0/A/5700/006	Natural Disasters	Section D
RP/0/A/5700/007	Earthquake	Section D, H.6
RP/0/A/5700/008	Release of Toxic or Flammable Gases	Section D
RP/0/A/5700/09	Collisions/Explosions	Section D
RP/0/A/5700/010	NRC Immediate Notification	Section D
RP/0/A/5700/011	Conducting a Site Assembly, Site Evacuation or Containment Evacuation	Section E.2, J, K.7

MCGUIRE FIGURE P-2 PAGE 2 OF 4 EMERGENCY PLAN IMPLEMENTING PROCEDURES		
Procedure #	Title	Emergency Plan Section Implemented
RP/0/A/5700/019	Core Damage Assessment	Section I
RP/0/A/5700/022	Spill/Incident Response Procedure	Appendices 7, 8, 9
RP/0/A/5700/026	Operations/Engineering Required Actions in the Technical Support Center (TSC)	
RP/0/B/5700/029	Notifications to Offsite Agencies from the Control Room	Section E
HP/0/B/1009/002	Alternative Methods for Determining Dose Rate within the Reactor Building	Section D, I.6
HP/0/B/1009/003	Recovery Plan	Section M
HP/0/B/1009/006	Procedure for Quantifying High Level Gaseous Radioactivity Release During Accident Conditions	Section D, I.3
HP/0/B/1009/010	Releases of Liquid Radioactive Materials Exceeding Selected Licensee Commitments	Section D, I.3
HP/0/B/1009/021	Estimating Food Chain Doses Under Post-Accident Conditions	I.10
HP/0/B/1009/022	Accident and Emergency Response	Section I, Section E
HP/0/B1009/024	Personnel Monitoring for Emergency Conditions	J.3, K.7

MCGUIRE
FIGURE P-2
PAGE 3 OF 4
EMERGENCY PLAN IMPLEMENTING PROCEDURES

Procedure #	Title	Emergency Plan Section Implemented
AD-EP-ALL-0101	Emergency Classification	Section D, E, I
AD-EP-ALL-0103	Activation and Operation of the EOF	Section B, H, M.1
AD-EP-ALL-0104	ERO Common Guidelines and Forms	Section B, E, F, G, I, J, K, M
AD-EP-ALL-0105	Activation and Operation of the TSC	Section B, H, M.1
AD-EP-MNS-0105	MNS Site Specific TSC Support	Section H
AD-EP-ALL-0106	Activation and Operation of the OSC	Section H
AD-EP-MNS-0106	MNS Site Specific OSC Support	Section H
AD-EP-ALL-0108	Joint Information System Support	Section G
AD-EP-ALL-0109	Protective Action Recommendations	J.7
AD-EP-ALL-0110	Recovery	Section M
AD-EP-ALL-0202	Emergency Response Offsite Dose Assessment	Section I
AD-EP-ALL-0203	Field Monitoring During Declared Emergency	Section E, I.7, I.8, I.9
AD-EP-MNS-0203	MNS Site Specific Field Monitoring	Section E, I.7, I.8, I.9
AD-EP-ALL-0204	Distribution of Potassium Iodide Tablets in the Event of a Radioiodine Release	J.6
AD-EP-ALL-0205	Emergency Exposure Controls	Section K.2
AD-EP-ALL-0304	State and County Notifications	Section E

MCGUIRE
FIGURE P-2
PAGE 4 OF 4
EMERGENCY PLAN IMPLEMENTING PROCEDURES

Procedure #	Title	Emergency Plan Section Implemented
EP Manual Section 1.1	Emergency Organization	Sections B, E, H
PT/0/A/4600/088	Functional Check of Emergency Vehicle and Equipment	Section H.11

FIGURE P-3
McGUIRE NUCLEAR SITE
EMERGENCY PLAN IMPLEMENTING PROCEDURES DISTRIBUTION

Control No.

2. Radiation Protection Manager
3. Emergency Preparedness Manager, Oconee
4. McGuire Nuclear Training
5. Operations Staff Manager
6. Site Emergency Planner, MG01EP
7. NRC Site Representative, McGuire Nuclear Site (forwarded by McGuire Emergency Planning)
8. Operator Training Director
13. Emergency Preparedness Manager, CNS
14. Director, Division of Radiation Protection
16. NCEM REP Program Manager
17. Tina Kuhr, Fleet Emergency Planning
19. Emergency Operations Facility, EOF Director's Area (MNS Emergency Planning, custodians)
20. McGuire Nuclear Site, Document Control
21. NCEM Western Branch Office Manager
22. NRC Document Control Desk, Washington D.C. (forwarded 1 copy by McGuire Emergency Planning)

FIGURE P-3
McGUIRE NUCLEAR SITE
EMERGENCY PLAN IMPLEMENTING PROCEDURES DISTRIBUTION

Control No.

23. NRC, Regional Administrator, Atlanta, GA (forwarded 1 copy by McGuire Emergency Planning)
24. NRC, Regional Administrator, Atlanta, GA (forwarded 1 copy by McGuire Emergency Planning)
25. NRC Office of Nuclear Materials Safety and Safeguards

APPENDIX 1

1.0 DEFINITIONS

AFFECTED PERSONS
Persons who have received radiation exposure or have been physically injured as a result of an accident to a degree requiring special attention as individuals, e.g., decontamination, first aid or medical services.

ANNUAL
For periodic emergency planning requirements, annual is defined as twelve months, with a maximum interval of 456 days.

ASSESSMENT ACTION
Those actions taken during or after an accident to obtain and process information that is necessary to make decisions to implement specific emergency measures.

BIENNIAL
For periodic emergency planning requirements, biennial is defined as at least once every two years, with a maximum interval of 912 days. (Note that this does not apply to the scheduling of biennial exercises. An exercise can occur at any time during the second calendar year after the previous exercise.)

CORRECTIVE ACTIONS
Emergency measures taken to ameliorate or terminate an emergency situation at or near the source of the problem to prevent an uncontrolled release of radioactive material or to reduce the magnitude of the release, e.g., shutting down equipment, fire-fighting, repair and damage control.

DRILL
A drill is a supervised instruction period aimed at testing, developing, and maintaining skills in a particular operation.

EMERGENCY ACTION LEVELS (EAL's)
A pre-determined, site-specific, observable threshold for a plant Initiating Condition that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (onsite or offsite); a discrete, or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

EMERGENCY OPERATIONS FACILITY (EOF)
The Emergency Operations Facility is the facility utilized for direction and control of all emergency and recovery activities with emphasis on the coordination of off-site activities such as dispatching mobile emergency monitoring teams, communications with local, state and federal agencies, and coordination of corporate and other outside support.

EMERGENCY PLANNING ZONE (EPZ)
The area for which planning is needed to assure that prompt and effective actions can be taken to protect the public in the event of an accident. The plume exposure EPZ is about 10 miles in radius and the ingestion exposure EPZ is about 50 miles in radius.

EXCLUSION AREA
The nuclear site property out to a radius of 2500 feet, that meets the 10CFR100 definition.

EXERCISE
An exercise is an event that tests the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations.

INGESTION EXPOSURE PATHWAY
The principle exposure from this pathway would be from ingestion of contaminated water or foods such as milk or fresh vegetables. The time of potential exposure could range in length from hours to months.

MONTHLY
For periodic emergency planning requirements, monthly is defined as once each month, with a maximum interval of 38 days.

OPERATIONAL SUPPORT CENTER (OSC)
In the event of an emergency, shift support personnel (e.g., auxiliary operators and technicians) other than those required and allowed in the control room shall report to this center for further orders and assignment.

PLUME EXPOSURE PATHWAY
The principle exposure sources from this pathway are (a) external exposure to gamma radiation from the plume and from deposited material and (b) inhalation exposure from the passing radioactive plume. The time of potential exposure could range from hours to days.

POPULATION-AT-RISK
Those persons for whom protective actions are being or would be taken.

PROTECTED AREA
An area encompassed by physical barriers and to which access is controlled.

PROTECTIVE ACTIONS
Those emergency measures taken after an uncontrolled release of radioactive materials has occurred, for the purpose of preventing or minimizing radiological exposures to persons that would be likely to occur if the actions were not taken.

PROTECTIVE ACTION GUIDES (PAG)
Projected radiological dose or dose-commitment values to individuals in the general population that warrant protective action following a release of radioactive material. Protective actions would be warranted provided the reduction in individual dose expected to be achieved by carrying out the preventive action is not offset by excessive risks to individual safety in taking the protective action. The PAG does not include the dose that has unavoidably occurred prior to the assessment.

QUARTERLY
For periodic emergency planning requirements, quarterly is defined as once every three months, with a maximum interval of 112 days.

RECOVERY ACTIONS
Those actions taken after the emergency to restore affected property as nearly as practicable to its pre-emergency condition.

SEMI-ANNUAL
For periodic emergency planning requirements, semi-annual is defined as once every 6 months, with a maximum interval of 228 days.

SITE
That part of the nuclear site property consisting of the Reactor, Auxiliary, Turbine, Service Buildings and grounds, contained within the outer security area fence.

TECHNICAL SUPPORT CENTER (TSC)
This on-site center is for use by plant management, technical and engineering support personnel. In an emergency, this center shall be used for assessment of plant status and potential off-site impact in support of the control room command and control function.

TRIENNIAL
For periodic emergency planning requirements, triennial is defined as at least once every three years, with maximum interval of 1369 days.

VITAL AREA
Areas within the Protected Area that house equipment important for nuclear safety. Access to a Vital Area is allowed only if an individual has been authorized to be in that area per the Security plan, therefore Vital Area is a Security term.

WEEKLY
For periodic emergency planning requirements, weekly is defined as once every 7 days, with a maximum interval of 9 days.