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EMERGENCY PLAN SECTION	PAGE NUMBER	REVISION NUMBER	DATE
Emergency Plan Approval Cover Sheet		16-1	March 2016
Table of Contents			
	1	16-1	March 2016
	2	16-1	March 2016
	3	16-1	March 2016
	4	16-1	March 2016
	5	16-1	March 2016
	6	16-1	March 2016
List of Figures, Tables and Attachments			
	7	16-1	March 2016
	8	16-1	March 2016
	9	16-1	March 2016
Introduction			
	i-1	16-1	March 2016
	i-2	16-1	March 2016
	i-3	16-1	March 2016
	i-4	16-1	March 2016
	i-5	16-1	March 2016
	i-6	16-1	March 2016
Section A			
	A-1	16-1	March 2016
	A-2	16-1	March 2016
	A-3	16-1	March 2016
	A-4	149	March 2017
	A-5	16-1	March 2016
Section B			
	B-1	16-1	March 2016
	B-2	16-1	March 2016
	B-3	16-1	March 2016
	B-4	16-1	March 2016
	B-5	16-1	March 2016
	B-6	16-1	March 2016
	B-7	16-1	March 2016
	B-8	16-1	March 2016
	B-9	16-1	March 2016
	B-10	16-1	March 2016
	B-11	16-1	March 2016
	B-12	16-1	March 2016
	B-13	16-1	March 2016
Section C			
	C-1	16-1	March 2016
	C-2	16-1	March 2016
Section D			
	1-258 (All pages)	147	March 2017
Section E			
	E-1	146	May 2015
	E-2	146	May 2015
	E-3	146	May 2015
	E-4	146	May 2015
	E-5	146	May 2015

LIST OF EFFECTIVE PAGES (LOEP)

EMERGENCY PLAN SECTION	PAGE NUMBER	REVISION NUMBER	DATE
	E-6	146	May 2015
	E-7	146	May 2015
	E-8	146	May 2015
	E-9	146	May 2015
	E-10	146	May 2015
	E-11	146	May 2015
Section F			
	F-1	16-1	March 2016
	F-2	16-1	March 2016
	F-3	16-1	March 2016
	F-4	16-1	March 2016
	F-5	16-1	March 2016
	F-6	16-1	March 2016
	F-7	16-1	March 2016
Section G			
	G-1	14-3	June, 2014
	G-2	14-3	June, 2014
	G-3	14-3	June, 2014
Section H			
	H-1	16-1	March 2016
	H-2	16-1	March 2016
	H-3	16-1	March 2016
	H-4	16-1	March 2016
	H-5	16-1	March 2016
	H-6	16-1	March 2016
	H-7	16-1	March 2016
	H-8	16-1	March 2016
	H-9	16-1	March 2016
	H-10	16-1	March 2016
	H-11	16-1	March 2016
	H-12	16-1	March 2016
	H-13	16-1	March 2016
	H-14	16-1	March 2016
Section I			
	I-1	146	May 2015
	I-2	146	May 2015
	I-3	146	May 2015
	I-4	146	May 2015
Section J			
	J-1	146	May 2015
	J-2	146	May 2015
	J-3	146	May 2015
	J-4	146	May 2015
	J-5	146	May 2015
	J-6	146	May 2015
	J-7	146	May 2015
	J-8	146	May 2015
	J-9	146	May 2015

LIST OF EFFECTIVE PAGES (LOEP)

EMERGENCY PLAN SECTION	PAGE NUMBER	REVISION NUMBER	DATE
Section K			
	K-1	14-3	June, 2014
	K-2	14-3	June, 2014
	K-3	14-3	June, 2014
	K-4	14-3	June, 2014
Section L			
	L-1	11-1	May 2011
	L-2	11-1	May 2011
Section M			
	M-1	13-1	August, 2013
	M-2	13-1	August, 2013
	M-3	13-1	August, 2013
	M-4	13-1	August, 2013
	M-5	13-1	August, 2013
Section N			
	N-1	16-1	March 2016
	N-2	16-1	March 2016
	N-3	16-1	March 2016
Section O			
	O-1	14-3	June, 2014
	O-2	14-3	June, 2014
Section P			
	P-1	146	May 2015
	P-2	146	May 2015
	P-3	146	May 2015
	P-4	146	May 2015
	P-5	146	May 2015
	P-6	146	May 2015
	P-7	146	May 2015
Section Q			
	Q-1	16-1	March 2016
Appendix 1			
	Q-1.1	16-1	March 2016
	Q-1.2	16-1	March 2016
	Q-1.3	16-1	March 2016
	Q-1.4	16-1	March 2016
	Q-1.5	16-1	March 2016
	Q-1.6	16-1	March 2016
	Q-1.7	16-1	March 2016
	Q-1.8	16-1	March 2016
Section Q, Appendix 2			
	Q-2.1	16-1	March 2016
	Q-2.2	16-1	March 2016
	Q-2.3	16-1	March 2016
	Q-2.4	16-1	March 2016
Appendix 3			
	Q-3.1	16-1	March 2016
	Q-3.2	16-1	March 2016
	Q-3.3	16-1	March 2016
	Q-3.4	16-1	March 2016

LIST OF EFFECTIVE PAGES (LOEP)

EMERGENCY PLAN SECTION	PAGE NUMBER	REVISION NUMBER	DATE
	Q-3.5	16-1	March 2016
Appendix 4			
	Q-4.1	16-1	March 2016
Appendix 5			
	Q-5.1	16-1	March 2016
	Q-5.2	16-1	March 2016
	Q-5.3	149	March 2017

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A. Assignment of Responsibility

Planning Objective

To assure that State, Local, Federal, private sector, Duke Energy Corporate and Catawba Nuclear Station organizations that are part of the overall response organization within the Catawba Emergency Planning Zone are identified.

A.1.a Organization

The principal organizations that are part of the overall response organization within the Catawba Emergency Planning Zone are listed below:

Federal

NRC (Nuclear Regulatory Commission)
FEMA (Federal Emergency Management Agency)
DOE (Department of Energy)

NOTE: NRC, FEMA, and DOE will coordinate response of other Federal Agencies per the Federal Radiological Emergency Response Plan (FRERP).

South Carolina State

S.C. Emergency Management Division of the S.C. Adjutant General's Office (Note 1)
S.C. Department of Health and Environmental Control, Bureau of Radiological Health

North Carolina State

N.C. Department of Crime Control and Public Safety, Division of Emergency Management (Note 1)
N.C. Department of Environment, Health and Natural Resources, Division of Radiation Protection

Local Government

The county governments and municipal governments (within the counties) to include the emergency service departments and other agencies interrelated to these local governments within the 10-mile EPZ (plume exposure pathway) of Catawba Nuclear Station are:
York
Gaston
Mecklenburg

The county governments (and municipal governments within the counties) to include the emergency service departments and other agencies interrelated to these local governments within a 50-mile EPZ (ingestion exposure pathway) of Catawba Nuclear Station are:

South Carolina (Note 2)

Cherokee	Lancaster
Chester	Newberry
Chesterfield	Spartanburg
Fairfield	Union
Kershaw	York

North Carolina (Note 2)

Anson	Cleveland	Mecklenburg	Union
Burke	Gaston	Rowan	
Cabarrus	Iredell	Rutherford	
Catawba	Lincoln	Stanley	

Note 1: This agency has the principal state responsibility for emergency response.

Note 2: Agreement letters with these agencies are not a part of the Catawba Nuclear Station Emergency Plan unless specifically noted in A-3.

Private Sector

The principal organizations in the private sector that are part of the overall response organization for the EPZ are:

Westinghouse
AT & T
The Independent Telephone Companies
Radio and Television Stations
Bethel Volunteer Fire Department
Various vendors such as GTS and Bartlett
Carolinas Medical Center
Center for Emergency Medicine (Rock Hill, S.C.)
Member's Southeastern Electric Exchange
The Salvation Army
The American Red Cross
Piedmont Medical Center (Rock Hill, SC)

Non-Government Organizations

INPO (Institute of Nuclear Power Operations), risk management companies and the ANI (American Nuclear Insurers)

A.1.b Concept of Operations

All emergencies or accident situations at the station are handled initially by the Shift Manager. When an abnormal situation occurs, the Shift Manager is able, utilizing station operating and emergency procedures and from background, training and experience, to determine if the abnormal situation is an emergency condition. During the course of the emergency condition and as response personnel are notified, and emergency centers are staffed (OSC, TSC, EOF), the Shift Manager is the person in charge, and assumes the functions of the Emergency Coordinator until the arrival of the Station Manager/designee. When the Station Manager/designee arrives and relieves the Shift Manager of the Emergency Coordinator function, he/she becomes the person in charge or the decision-maker. When the Emergency Operations Facility (EOF) is activated and operational, the EOF Director at the EOF is responsible for company emergency response.

The Control Room at the station is the initial center for coordination of emergency response for all emergency conditions. For emergencies classified as Alert, Site Area Emergency and General Emergency, the Emergency Coordinator shall activate the Emergency Response Organization.

The TSC acts in support of the command and control function of the Control Room and provides an area for other station personnel who have expertise in all areas of plant operation to support the emergency response. This facility is equipped with communication equipment, Operator Aid Computer (OAC) terminals, line printers, off-site and on-site computer access, plant drawings, procedures and other materials and equipment to support its function. Personnel in the TSC will be able to assess the accident condition and make responsible recommendations to the Control Room, the EOF and off-site agencies as necessary to provide for the safety of plant personnel and members of the general public. After the EOF is operational and activated, it will assume many of the functions of the TSC and will rely on the TSC as a vital link to the station. The TSC will provide the EOF with up-to-date plant parameters, which will allow this facility to perform its assigned tasks.

The responsibility of the Control Room, TSC and EOF for the various emergency response functions is further detailed in Figure A-1.

A.1.c Block Diagram of Organization Interrelationships

See Figures B-1a and B1b, B-2, B-3, B-4, and B-5.

A.1.d Key Decision-Making

During the course of any emergency condition at Catawba, several persons have the potential to be "in charge" or to be the "Key Decision Maker". Prior to TSC activation and arrival of the Station Manager/designee, the Shift Manager assumes the functions of the Emergency Coordinator at the Station and is in charge. When the Station Manager/designee arrives on-site and assumes the Emergency Coordinator function, he/she becomes the person in charge of emergency response and becomes the key decision-maker. After EOF is operational and activated, the EOF Director is responsible for company emergency response.

A.1.e 24 Hour Emergency Response

The Catawba Station emergency response organization beginning with the Control Room through the TSC is capable of responding to an emergency 24 hours per day, 7 days per week. Section E.2 describes the notification scheme within the station emergency response organization.

A.2.a Responsibility For and Functions of State and Local Government Emergency Response Organization

(See State and County Plans)

A.2.b Legal Basis For Authority

(See State and County Plans)

A.3 Agreement Letters For Emergency Response Support from Off-site Agencies

Section Q, Appendix 5 contains letters of agreement with the following organizations:

Piedmont Medical Center
Carolinas Medical Center
York County Emergency Management
Bethel Volunteer Fire Department
Charlotte-Mecklenburg Emergency Management Office
Gaston County Emergency Management
Center for Emergency Medicine (Rock Hill, SC)
North Carolina Division of Emergency Management
South Carolina Emergency Management Division
Radiation Emergency Assistance Center/Training Site (REAC/TS)
DOE - Savannah River
INPO - Fixed Nuclear Facility Voluntary Assistance Agreement
JIC - Joint Information Center
York County Sheriff

1. Duke Energy has established numerous support agreements and contracts with organizations that may be required to provide assistance in the event of an emergency.
2. All agreements or contracts are reviewed annually to assure each contributes the desired support to the Emergency Preparedness Program.
3. Letters of Agreement and Contracts, including the review frequency, will be documented according to the site's protocol.

A.4 Individual Responsible for Continuity of Resources

The emergency response organization is capable of continuous (24 hours/day) operation for an extended period of time. The EOF Director is the individual responsible for assuring continuity of resources within the emergency response organization.

FIGURE A-1

RESPONSIBILITY FOR EMERGENCY RESPONSE FUNCTIONS

	Emergency Class			
Emergency Response Functions	Unusual Event	Alert	Site Area Emergency	General Emergency
Supervision of reactor operations and manipulation of controls	CR	CR	CR	CR
Management of plant operations	CR (TSC)	TSC	TSC	TSC
Technical support to reactor operations	CR (TSC)	TSC	TSC	TSC
Management of corporate emergency response resources	CR (TSC) (EOF)	EOF	EOF	EOF
Radiological effluent and environs monitoring, assessment and dose projection	CR (TSC) (EOF)	EOF	EOF	EOF
Inform state and local emergency response organizations and make recommendations for public protective actions	CR (TSC) (EOF)	EOF	EOF	EOF
Management of recovery operations	CR (TSC) (EOF)	TSC/EOF	TSC/EOF	TSC/EOF
Technical support of recovery operations	CR (TSC) (EOF)	TSC/EOF	TSC/EOF	TSC/EOF

NOTE: (TSC) (EOF) indicates that activation of these facilities or the performance of this function is optional for the indicated emergency class.

EMERGENCY PLAN SECTION D EMERGENCY CLASSIFICATION SYSTEM

Revision 147

TABLE OF CONTENTS

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Catawba Nuclear Station (CNS). It should be used to facilitate review of the CNS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of RP/0/A/5000/001 Classification of Emergency, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Coordinator 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 CNS 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), CNS 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 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 Emergency Classification Level (ECL) from Alert to a Site Area Emergency or a General Emergency

2.4 Fission Product Barrier Classification Criteria

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

<u>*Alert:*</u>

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

<u>*Site Area Emergency:*</u>

Loss or potential loss of any two barriers

<u>*General Emergency:*</u>

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

2.4 EAL Organization

The CNS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
    - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
    - EALs applicable only under 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 Defueled 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 CNS EAL categories are aligned to and represent the NEI 99-01"Recognition Categories." Subcategories are used in the CNS scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The CNS EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal Rad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – Hazards 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 – System 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 – Fission Product Barrier Degradation	None
Cold Conditions:	
C – Cold Shutdown / Refueling System Malfunction	1 – NCS Level 2 – Loss of Essential AC Power 3 – NCS Temperature 4 – Loss of Vital DC Power 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 Operation, 2 - Startup, 3 – Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, D - Defueled, 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 CNS-relevant information concerning the EALas well as a description of the rationale for the EAL as provided in NEI 99-01 Rev. 6.

CNS 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

3 Hot Shutdown

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

4 Cold Shutdown

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

5 Refueling

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

D Defueled

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 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 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 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 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 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, 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 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 or condition 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 CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents

4.1.7 CNS UFSAR Figure 1-20 Plot Plan

4.1.8 Technical Specifications Table 1.1-1 Modes

4.1.9 OP/0/A/6100/014 Penetration Control for Modes 5, 6 and NO Mode - Enclosure 4.7 Setting, Maintaining and Securing from Containment Penetration Control

4.1.10 PRO-NGGC-0201 NGG Procedure Writers Guide

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

4.1.12 CNS ISFSI Certificates of Compliance

4.1.13 CNS Emergency Plan

4.2 Implementing

4.2.1 RP/0/A/5000/001 Classification of Emergency

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

4.2.3 CNS 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 CNS ISFSI, Confinement Boundary is defined as the Transportable Storage Canister (TSC) for both NAC-UMS and MAGNASTAR 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 CNS, Containment Closure is established when the requirements of OP/0/A/6100/014 Penetration Control for Modes 5, 6 and NO Mode - Enclosure 4.7 Setting, Maintaining and Securing from Containment Penetration Control are met (ref. 4.1.9).

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

Environmental 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 CNS 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.

Fission Product Barrier Threshold

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

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.

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

Intrusion

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

Independent Spent Fuel Storage Installation (ISFSI)

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

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.

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. Access to this area is generally restricted to those entering on official business. (ref. 4.1.13).

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 CNS UFSAR Figure 1-20 Plot Plan (ref. 4.1.7).

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 CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents (ref. 4.1.6).

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
CA .... Auxiliary Feedwater
CNS .... Catawba 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
ECCS .... Emergency Core Cooling System
EC .... Emergency Coordinator
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...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
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 .... Safe Shutdown Facility
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
WOG .... Westinghouse Owners Group

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

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

CNS	NEI 99-01 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

CNS	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, 2 3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1
HU4.2	HU4	2

CNS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1, 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
SA3.1	SA2	1
SA6.1	SA5	1
SA9.1	SA9	1

CNS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
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

7.3 Attachment 3, Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases

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 > 2 times the SLC/TS 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 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	1/2EMF36L	----	----	4.18E+6 cpm	5.75E+3 cpm
Gaseous	Unit Vent Noble Gas High	1/2EMF36H	2.21E+4 cpm	2.22E+3 cpm	2.42E+2 cpm	----
Liquid	Liquid Waste Effluent Line	0EMF49L	----	----	----	4.50E+6 cpm
Liquid	Monitor Tank Discharge	0EMF57L	----	----	----	4.97E+5 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 and Technical Specification release rate and concentration limits associated with the specified monitors (ref. 2, 3, 4, 7).

Gaseous Releases

Instrumentation that may be used to assess this EAL is listed below (ref. 1, 5):

- Unit Vent Noble Gas Low Range – 1/2EMF36L has a range of $10^1 - 10^7$ cpm

Liquid Releases

Instrumentation that may be used to assess this EAL is listed below (ref. 1, 6):

- Liquid Waste Effluent Line Monitor – 0EMF49L (batch release) has a range of $10^1 - 10^7$ cpm
- Monitor Tank Discharge Monitor – 0EMF57L has a range of $10^1 - 10^7$ cpm

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.

CNS Basis Reference(s):

1. CNS ODCM Section 3.0 Setpoint Calculations
2. CNS-SLC 16.11-1 Liquid Effluents
3. CNS-SLC 16.11-6 Gaseous Effluents
4. EP-EALCALC-CNS-1401 CNS Radiological Effluent EAL Values, Rev. 0
5. UFSAR Table 11-20 Airborne Process Radiation Monitoring Equipment
6. UFSAR Table 11-19 Liquid Process Radiation Monitoring Equipment
7. Technical Specifications Section 5.5.5 Radioactive Effluent Controls Program
8. 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/TC 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/TC limits for ≥ 60 min. (Notes 1, 2)	

Note 1: The Emergency Coordinator 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.

CNS Basis Reference(s):

1. CNS Offsite Dose Calculation Manual
2. CNS-SLC 16.11-1 Liquid Effluents
3. CNS-SLC 16.11-6 Gaseous Effluents
4. Technical Specifications Section 5.5.5 Radioactive Effluent Controls Program
5. AD-RP-ALL-2003 Investigation of Unusual Radiological Occurences
6. 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 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	1/2EMF36L	----	----	4.18E+6 cpm	5.75E+3 cpm
Gaseous	Unit Vent Noble Gas High	1/2EMF36H	2.21E+4 cpm	2.22E+3 cpm	2.42E+2 cpm	----
Liquid	Liquid Waste Effluent Line	0EMF49L	----	----	----	4.50E+6 cpm
Liquid	Monitor Tank Discharge	0EMF57L	----	----	----	4.97E+5 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. 3, 4).

Instrumentation that may be used to assess this EAL is listed below (ref. 1, 2):

- Unit Vent Noble Gas High Range – EMF36H has a range of $10^1 - 10^6$ cpm

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.

CNS Basis Reference(s):

1. CNS ODCM Section 3.0 Setpoint Calculations
2. UFSAR Table 11-20 Airborne Process Radiation Monitoring Equipment
3. EP-EALCALC-CNS-1401 CNS Radiological Effluent EAL Values, Rev. 0
4. SDQA-70400-COM Unified RASCAL Interface (URI)
5. 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 CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents.

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.

CNS Basis Reference(s):

1. HP/0/B/1009/026 On-Shift Offsite Dose Assessment
2. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
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:

RA1.3	**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 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 CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents.

Basis:

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

This EAL addresses a release of 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.

CNS Basis Reference(s):

1. CNS 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 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 CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents.

Basis:

HP/0/B/1009/004, Environmental Monitoring for Emergency Conditions Within the Ten Mile Radius of CNS provides 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.

CNS Basis Reference(s):

1. HP/0/B/1009/004 Environmental Monitoring for Emergency Conditions Within the Ten Mile Radius of CNS

2. 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 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	1/2EMF36L	----	----	4.18E+6 cpm	5.75E+3 cpm
Gaseous	Unit Vent Noble Gas High	1/2EMF36H	2.21E+4 cpm	2.22E+3 cpm	2.42E+2 cpm	----
Liquid	Liquid Waste Effluent Line	0EMF49L	----	----	----	4.50E+6 cpm
Liquid	Monitor Tank Discharge	0EMF57L	----	----	----	4.97E+5 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, 2).

Instrumentation that may be used to assess this EAL is Unit Vent Noble Gas High Range Monitor – EMF36H and has a range of $10^1 - 10^6$ cpm (ref 3, 4).

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.

CNS Basis Reference(s):

1. EP-EALCALC-CNS-1401 CNS Radiological Effluent EAL Values, Rev. 0
2. SDQA-70400-COM Unified RASCAL Interface (URI)
3. CNS ODCM Section 3.0 Setpoint Calculations
4. UFSAR Table 11-20 Airborne Process Radiation Monitoring Equipment
5. 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 CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents.

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.

CNS Basis Reference(s):

1. SDQA-70400-COM Unified RASCAL Interface (URI)
2. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
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:

RS1.3	**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 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 CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents.

Basis:

HP/0/B/1009/004, Environmental Monitoring for Emergency Conditions Within the Ten Mile Radius of CNS provides 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.

CNS Basis Reference(s):

1. HP/0/B/1009/004 Environmental Monitoring for Emergency Conditions Within the Ten Mile Radius of CNS
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 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 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	1/2EMF36L	----	----	4.18E+6 cpm	5.75E+3 cpm
Gaseous	Unit Vent Noble Gas High	1/2EMF36H	2.21E+4 cpm	2.22E+3 cpm	2.42E+2 cpm	----
Liquid	Liquid Waste Effluent Line	0EMF49L	----	----	----	4.50E+6 cpm
Liquid	Monitor Tank Discharge	0EMF57L	----	----	----	4.97E+5 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, 2).

Instrumentation that may be used to assess this EAL is Unit Vent Noble Gas High Range Monitor – EMF36H and has a range of $10^1 – 10^6$ cpm (ref 3, 4).

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.

CNS Basis Reference(s):

1. EP-EALCALC-CNS-1401 CNS Radiological Effluent EAL Values, Rev. 0
2. SDQA-70400-COM Unified RASCAL Interface (URI)
3. CNS ODCM Section 3.0 Setpoint Calculations
4. UFSAR Table 11-20 Airborne Process Radiation Monitoring Equipment
5. 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 CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents.

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.

CNS Basis Reference(s):

1. SDQA-70400-COM Unified RASCAL Interface (URI)
2. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
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:

RG1.3 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 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 CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents.

Basis:

HP/0/B/1009/004, Environmental Monitoring for Emergency Conditions Within the Ten Mile Radius of CNS provides 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.

CNS Basis Reference(s):

1. HP/0/B/1009/004 Environmental Monitoring for Emergency Conditions Within the Ten Mile Radius of CNS
2. 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 **any** of the following radiation monitors:

- 1EMF15 (2EMF4) Spent Fuel Building Refueling Bridge
- 1EMF17 (2EMF2) Reactor Building Refueling Bridge

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 actuated by 1(2)KFPS5120 at a setpoint of 39' (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 CAVITY level are not classifiable under this EAL.

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.

CNS Basis Reference(s):

1. OP/1(2)/B/6100/010N E/2 Spent Fuel Pool Level Hi/Lo
2. AP/1(2)/A/5500/026 Loss of Refueling Canal Level
3. AP/1(2)/A/5500/041 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.

CNS Basis Reference(s):

1. AP/1(2)/A/5500/026 Loss of Refueling Canal 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:

- 1EMF15 (2EMF4) Spent Fuel Building Refueling Bridge
- 1EMF17 (2EMF2) Reactor Building Refueling Bridge
- 1EMF42 (2EMF42) Spent Fuel Pool Ventilation
- 1EMF39L (2EMF39L) Containment Noble Gas

Mode Applicability:

All

Definition(s):

None

Basis:

The specified radiation monitors are those expected to see increase area radiation levels as a result of damage to irradiated fuel (ref. 1).

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.

CNS Basis Reference(s):

1. AP/1(2)/A/5500/025 Damaged Spent Fuel
2. HP/0/B/1000/010 Determination of Radiation Monitor Setpoints
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**
Lowering of spent fuel pool level to 24.5 ft. (Level 2) on 1(2)KFP5780 or 1(2)NVP8790	

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).

SFP level indicators 1(2)KFP5780 (radar) or 1(2)NVP8790 (pressure) located on the back of 1(2)MC7 provide continuous wide range SFP level indication to the top of the spent fuel racks (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.

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.

CNS Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. EC109413
3 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**
Lowering of spent fuel pool level to 14.5 ft. (Level 3) on 1(2)KFP5780 or 1(2)NVP8790	

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).

SFP level indicators 1(2)KFP5780 (radar) or 1(2)NVP8790 (pressure) located on the back of 1(2)MC7 provide continuous wide range SFP level indication to the top of the spent fuel racks (ref. 2).

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 AG1 or RG2.

CNS Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. EC109413
3. 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 at least 14.5 ft. (Level 3) on 1(2)KFP5780 or 1(2)NVP8790 for ≥ 60 min. (Note 1)	

Note 1: The Emergency Coordinator 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).

SFP level indicators 1(2)KFP5780 (radar) or 1(2)NVP8790 (pressure) located on the back of 1(2)MC7 provide continuous wide range SFP level indication to the top of the spent fuel racks (ref. 2).

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.

CNS Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. EC109413
3. NEI 99-01 AG2

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 (EMF12)

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). EMF 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.

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

CNS Basis Reference(s):

1. OP/1(2)/B/6100/010Z C/2 Control Room
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**	**Mode**
Auxiliary 577'	Rm 478 (1EMXA)	Rm 469 (2EMXA)	4
	Rm 496 (1ETA)	Rm 486 (2ETA)	4
	Rm 496 (1EMXS)	Rm 486 (2EMXS)	4
	AB-577', JJ-57 (1MXK)	AB-577', JJ-57 (2MXK)	4
Auxiliary 560'	Rm 330 (1EMXJ)	Rm 320 (2EMXJ)	4
	Rm 372 (1ETB)	Rm 362 (2ETB)	4
	Rm 372 (1EMXD)	Rm 362 (2EMXD)	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.

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

CNS Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/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, D – Defueled).

The events of this category pertain to the following subcategories:

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 emergency 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 NCS inventory results in NCS water level less than a required lower limit for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator 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 3.9.6 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.

CNS Basis Reference(s):

1. EP/1(2)/A/5000/FR-I.2 Response to Low Pressurizer Level
2. CNS Technical Specifications Section 3.9.6 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

EAL:

CU1.2	**Unusual Event**

NCS water level **cannot** be monitored

AND EITHER

- UNPLANNED increase in Containment Floor & Equipment Sump or Incore Sump (alarm) due to a loss of NCS inventory
- Visual observation of UNISOLABLE NCS leakage

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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:

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 (ref. 1). 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.

The Incore Sump level cannot be monitored in the CR but alarms on high level.

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.

CNS Basis Reference(s):

1. AP/1(2)/A/5500/010 Reactor Coolant Leak
2. 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**
UNPLANNED loss of NCS inventory as indicated by NCS water level < 6.5% (wide range)	

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:

6.5% wide range NCS level indication is the lowest level to assure adequate net positive suction head and prevent ND pump cavitation 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 6.5% 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.

CNS Basis Reference(s):

1. OP/1(2)/A/6150/006 Draining the Reactor Coolant System
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 Containment Floor & Equipment Sump or Incore Sump (alarm) due to a loss of NCS inventory
- Visual observation of UNISOLABLE NCS leakage

Note 1: The Emergency Coordinator 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):

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:

In Cold Shutdown mode, the NCS will normally be intact and standard NCS monitoring means are available. In the Refuel mode, the NCS is not intact and NCS 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 (ref. 1). 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.

The Incore Sump level cannot be monitored in the CR but alarms on high level.

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.

CNS Basis Reference(s):

1. AP/1(2)/A/5500/010 Reactor Coolant Leak
2. 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 Containment Floor & Equipment Sump or Incore Sump (alarm) due to a loss of NCS inventory
- Visual observation of UNISOLABLE NCS leakage
- Reactor Building Refueling Bridge Monitor 1EMF17 (2EMF2) reading > 9,000 mR/hr
- Erratic Source Range or Gamma Metric Monitor indication

Note 1: The Emergency Coordinator 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):

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 (ref. 1). 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. 2).

The Incore Sump level cannot be monitored in the CR but alarms on high level.

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. 1EMF17 (2EMF2), 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.

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

CNS Basis Reference(s):

1. OP/1(2)/A/6150/006 Draining the Reactor Coolant System
2. AP/1(2)/A/5500/010 Reactor Coolant Leak
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 Containment Floor & Equipment Sump or Incore Sump (alarm) due to a loss of NCS inventory
- Visual observation of UNISOLABLE NCS leakage
- Reactor Building Refueling Bridge Monitor 1EMF17 (2EMF2) reading > 9,000 mR/hr
- Erratic Source Range or Gamma Metric Monitor indication

AND

Any Containment Challenge indication, Table C-1

Note 1: The Emergency Coordinator 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-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 CNS, Containment Closure is established when the requirements of OP/0/A/6100/014 Penetration Control for Modes 5, 6 and NO Mode - Enclosure 4.7 Setting, Maintaining and Securing from Containment Penetration Control 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 (ref. 1). 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. 2).

The Incore Sump level cannot be monitored in the CR but alarms on high level.

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. 1EMF17 (2EMF2), 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.

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% by volume in the presence of oxygen (>5%).
- 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*.

CNS Basis Reference(s):

1. OP/1(2)/A/6150/006 Draining the Reactor Coolant System
2. AP/1(2)/A/5500/010 Reactor Coolant Leak
3. OP/0/A/6100/014 Penetration Control for Modes 5, 6 and NO Mode. Enclosure 4.7 Setting, Maintaining and Securing from Containment Penetration Control
4. 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 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) (if already aligned)
• ATD (Train B)
• SATB (Train B) (if already aligned)
Onsite:
• D/G A (Train A)
• D/G B (Train B)

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling, D - Defueled

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 Normal Auxiliary Transformers (ATC & ATD). 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). However, alignment of SATA or SATB to an essential bus takes longer than 15 minutes and therefore should only be credited if already aligned.

Each essential bus has a dedicated diesel generator (D/G A & D/G 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).

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 defueled 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 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.

CNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/007 Loss of Normal Power
3. 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 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, D - Defueled

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 Normal Auxiliary Transformers (ATC & ATD). 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). However, alignment of SATA or SATB to an essential bus takes longer than 15 minutes and therefore should only be credited if already aligned.

Each essential bus has a dedicated diesel generator (D/G A & D/G 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).

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 defueled 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.

CNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/007 Loss of Normal Power
3. ECA-0.0 EP/1(2)/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.

CNS Basis Reference(s):

1. CNS Technical Specifications Table 1.1-1
2. CNS UFSAR Section 7.0 Instrumentation and Controls
3. AP/1(2)/A/5500/019 Loss of Residual Heat Removal 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 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.

CNS Basis Reference(s):

1. CNS Technical Specifications Table 1.1-1
2. CNS UFSAR Section 7.0 Instrumentation and Controls
3. AP/1(2)/A/5500/019 Loss of Residual Heat Removal System
4. OP/1(2)/A/6150/006 Draining the Reactor Coolant 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 > 10 psig due to a loss of NCS cooling (this does **not** apply during water-solid plant conditions)

Note 1: The Emergency Coordinator 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 CNS, Containment Closure is established when the requirements of OP/0/A/6100/014 Penetration Control for Modes 5, 6 and NO Mode - Enclosure 4.7 Setting, Maintaining and Securing from Containment Penetration Control 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 10 psig RPV pressure increase can be read on various instruments such as NCPT5141 C-Loop N/R, 0 – 800 psig and Point #4 on SMCR5810 (CR chart recorder, 0 – 600 psi). (ref. 4, 5).

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.

RCS reduced inventory condition exists when NCS level is ≤ 16% (ref. 7).

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.

CNS Basis Reference(s):

1. CNS Technical Specifications Table 1.1-1
2. CNS UFSAR Section 7.0 Instrumentation and Controls
3. AP/1(2)/A/5500/019 Loss of Residual Heat Removal System
4. IP/1(2)/B/3121/011A
5. IP/1(2)/A/3122/055A
6. OP/0/A/6100/014 Penetration Control for Modes 5, 6 and NO Mode. Enclosure 4.7 Setting, Maintaining and Securing from Containment Penetration Control
7. OP/1(2)/A/6150/006 Draining the Reactor Coolant System
8. 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 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:

Four 125 VDC distribution centers are provided for the 125VDC Vital Instrumentation and Control Power System. Four distribution centers (EDA, EDC, EDB and EDD), one per load group, supply the four independent channels of vital instrumentation and control, and are each powered directly from an independent 125 volt battery and battery charger. Each of the four distribution centers supplies one DC panel board and one 125VDC-120VAC static inverter (ref. 1).

The Class 1E DC loads have an operating voltage range of 105 to 135 volts. 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.

CNS Basis Reference(s):

1. CNS UFSAR Section 8.0 Electrical Power
2. AP/1(2)/A/5500/029 Loss of Vital or Aux Control Power
3. 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	
Commercial Telephones		X	X
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System (ETS)		X	X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D – Defueled

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 Catawba Plant 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 Catawba Site 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.

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.

Satellite Phones

Portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. These portable systems can be powered by internal batteries, external DC sources as well as external AC sources.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

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 Catawba 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, York, Gaston and Mecklenburg County EOCs

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

CNS Basis Reference(s):

1. CNS Emergency Plan Section F Emergency Communications
2. 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:

6 - 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. CNS plant yard elevation is 593.5 ft MSL. The minimum external access elevation for the Auxiliary, Turbine and Service Buildings is 594.0 ft MSL (ref. 1, 3).

- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 95 mph. (ref. 4).
- 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.

CNS Basis Reference(s):

1. RP/0/A/5000/007 Natural Disaster and Earthquake
2. AP/0/A/5500/030 Plant Flooding
3. UFSAR Section 3.4 Water Level (Flood) Design
4. Updated FSAR Section 3.3.1 Wind Loadings
5. AP/0/A/5500/045 Plant Fire
6. 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 the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Coordinator 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 CNS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

This EAL is based on the Duke Energy Physical Security Plan for CNS (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 CNS (ref. 1).

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

CNS Basis Reference(s):

1. Duke Energy Physical Security Plan for CNS
2. AP/0/A/5500/046 Hostile Aircraft Activity
3. RP/0/B/5000/026 Site Response to a Security Threat
4. AP/0/A/5500/048 Extensive Damage Mitigation
5. 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:

This EAL is based on the Duke Energy Physical Security Plan for CNS (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 CNS 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 CNS (ref. 1).

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

CNS Basis Reference(s):

1. Duke Energy Physical Security Plan for CNS
2. AP/0/A/5500/046 Hostile Aircraft Activity
3. RP/0/B/5000/026 Site Response to a Security Threat
4. AP/0/A/5500/048 Extensive Damage Mitigation
5. 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:

This EAL is based on the Duke Energy Physical Security Plan for CNS (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 CNS 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 CNS (ref. 1).

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

CNS Basis Reference(s):

1. Duke Energy Physical Security Plan for CNS
2. AP/0/A/5500/046 Hostile Aircraft Activity
3. RP/0/B/5000/026 Site Response to a Security Threat
4. AP/0/A/5500/048 Extensive Damage Mitigation
5. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

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

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 CNS (ref. 1).

CNS Basis Reference(s):

1. Duke Energy Physical Security Plan for CNS
2. AP/0/A/5500/046 Hostile Aircraft Activity
3. RP/0/B/5000/026 Site Response to a Security Threat
4. AP/0/A/5500/048 Extensive Damage Mitigation
5. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

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

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

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.

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 CNS (ref. 1).

CNS Basis Reference(s):

1. Duke Energy Physical Security Plan for CNS
2. AP/0/A/5500/046 Hostile Aircraft Activity
3. RP/0/B/5000/026 Site Response to a Security Threat
4. AP/0/A/5500/048 Extensive Damage Mitigation
5. 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 CNS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (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 CNS UFSAR Figure 1-20 Plot Plan.

Basis:

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 CNS (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 CNS (ref. 1).

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

CNS Basis Reference(s):

1. Duke Energy Physical Security Plan for CNS
2. RP/0/B/5000/026 Site Response to a Security Threat
3. AP/0/A/5500/048 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 Supervisor

AND EITHER of the following has occurred:

Any of the following safety functions cannot be controlled or maintained

- Reactivity
- 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 CNS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (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 CNS UFSAR Figure 1-20 Plot Plan.

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 CNS (ref. 1).

CNS Basis Reference(s):

1. Duke Energy Physical Security Plan for CNS
2. RP/0/B/5000/026 Site Response to a Security Threat
3. AP/0/A/5500/048 Extensive Damage Mitigation
4. AP/1(2)/A/5500/017 Loss of Control Room
5. NEI 99-01 HG1

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-4, B/8	

Mode Applicability:

All

Definition(s):

None

Basis:

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

Five strong motion accelerographs are installed within Unit 1 structures. 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/5000/007 Natural Disaster and Earthquake provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions. (ref. 3)

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 CNS. Provide the analyst with the following CNS coordinates: **35° 03' 04" north latitude, 81° 04' 10" west longitude** (ref. 4). Alternatively, near real-time seismic activity can be accessed via the NEIC website:

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

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.

CNS Basis Reference(s):

1. Updated FSAR Section 3.1 Conformance with General Design Criteria
2. Updated FSAR Section 3.7.4.2 Location and Description of Instrumentation
3. RP/0/A/5000/007 Natural Disaster and Earthquake
4. Updated FSAR section 2.1.1.1 Specification of Location (Unit 1)
5. 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 CNS UFSAR Figure 1-20 Plot Plan.

Basis:

Response actions associated with a tornado onsite is provided in RP/0/A/5000/007 Enclosure 4.2 Tornado Warning Issued for York County or Tornado On-Site (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.

CNS Basis Reference(s):

1. RP/0/A/5000/007 Natural Disaster and Earthquake
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/Service Buildings and Auxiliary/Diesel Buildings 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.

CNS Basis Reference(s):

1. AP/0/A/5500/030 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 CNS UFSAR Figure 1-20 Plot Plan.

Basis:

As used here, the term "offsite" is meant to be areas external to the CNS 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.

CNS 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 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.

CNS 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 should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Reactor Building (Containment) • Auxiliary Building • Diesel Generator Rooms • RN Pump House • 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.

Table H-1 Fire Areas are based on CNS-1465.00-00-0006 Design Basis Specification for the Plant Fire Protection and AP/0/A/5500/045 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.

CNS Basis Reference(s):

1. CNS-1465.00-00-0006 Design Basis Specification for the Plant Fire Protection
2. AP/0/A/5500/045 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 should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Reactor Building (Containment) • Auxiliary Building • Diesel Generator Rooms • RN Pump House • 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. 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 CNS-1465.00-00-0006 Design Basis Specification for the Plant Fire Protection and AP/0/A/5500/045 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.

CNS Basis Reference(s):

1. CNS-1465.00-00-0006 Design Basis Specification for the Plant Fire Protection
2. AP/0/A/5500/045 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 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 CNS UFSAR Figure 1-20 Plot Plan.

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.

CNS 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 CNS UFSAR Figure 1-20 Plot Plan.

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.

CNS 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	Mode
Auxiliary 577'	Rm 478 (1EMXA)	Rm 469 (2EMXA)	4
	Rm 496 (1ETA)	Rm 486 (2ETA)	4
	Rm 496 (1EMXS)	Rm 486 (2EMXS)	4
	AB-577', JJ-57 (1MXK)	AB-577', JJ-57 (2MXK)	4
Auxiliary 560'	Rm 330 (1EMXJ)	Rm 320 (2EMXJ)	4
	Rm 372 (1ETB)	Rm 362 (2ETB)	4
	Rm 372 (1EMXD)	Rm 362 (2EMXD)	4

Mode Applicability:

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 do 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.*

CNS Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/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	

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.

CNS Basis Reference(s):

1. AP/1(2)/A/5500/017 Loss of Control Room
2. OP/1(2)/A/6100/004 Shutdown Outside the Control Room From Hot Standby to Cold Shutdown.
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

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 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

CNS Basis Reference(s):

1. AP/1(2)/A/5500/017 Loss of Control Room
2. OP/1(2)/A/6100/004 Shutdown Outside the Control Room From Hot Standby to Cold Shutdown.
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 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 is the designated onsite individual having the responsibility and authority for implementing the CNS 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 to fall under the emergency classification level description for an Unusual Event.

CNS Basis Reference(s):

1. CNS Emergency Plan section 3.0 Site Emergency Organization Section B.2 Emergency Coordinator
2. NEI 99-01 HU7

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 warrant declaration of an Alert

EAL:

HA7.1 Alert

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

Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the CNS 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 to fall under the emergency classification level description for an Alert.

CNS Basis Reference(s):

1. CNS Emergency Plan section 3.0 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 warrant declaration of a Site Area Emergency

EAL:

HS7.1 Site Area Emergency
Other conditions exist which in the judgment of the Emergency Coordinator 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 CNS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

SITE BOUNDARY - Area as depicted in CNS-SLC-16.11-16 Figure 16.11-16-1 Unrestricted Area and Site Boundary for Radioactive Effluents.

Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the CNS 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 to fall under the emergency classification level description for a Site Area Emergency.

CNS Basis Reference(s):

1. CNS Emergency Plan section 3.0 Site Emergency Organization Section B.2 Emergency Coordinator
2. NEI 99-01 HS7

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 warrant declaration of a General Emergency

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the Emergency Coordinator 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 CNS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (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 is the designated onsite individual having the responsibility and authority for implementing the CNS 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 to fall under the emergency classification level description for a General Emergency.

CNS Basis Reference(s):

1. CNS Emergency Plan section 3.0 Site Emergency Organization Section B.2 Emergency Coordinator
2. NEI 99-01 HG7

Category S – System Malfunction

EAL Group: Hot Conditions (NCS temperature > 200°F); 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 V 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 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) (if already aligned)
• ATD (Train B)
• SATB (Train B) (if already aligned)
Onsite:
• D/G A (Train A)
• D/G B (Train B)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

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 Normal Auxiliary Transformers (ATC & ATD). 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). However, alignment of SATA or SATB to an essential bus takes longer than 15 minutes and therefore should only be credited if already aligned.

Each essential bus has a dedicated diesel generator (D/G A & D/G 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).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

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.

Escalation of the emergency classification level would be via IC SA1.

CNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/007 Loss of Normal Power
3. NEI 99-01 SU1

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 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) (if already aligned) • ATD (Train B) • SATB (Train B) (if already aligned) **Onsite:** • D/G A (Train A) • D/G B (Train B)

Mode Applicability:

1 - Power Operation, 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 Normal Auxiliary Transformers (ATC & ATD). 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). However, alignment of SATA or SATB to an essential bus takes longer than 15 minutes and therefore should only be credited if already aligned.

Each essential bus has a dedicated diesel generator (D/G A & D/G 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).

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.

CNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/007 Loss of Normal Power
3. 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 capabilityto essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1)	

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

Mode Applicability:

1 - Power Operation, 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 S-1) 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 Normal Auxiliary Transformers (ATC & ATD). 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). However, alignment of SATA or SATB to an essential bus takes longer than 15 minutes and therefore should only be credited if already aligned.

Each essential bus has a dedicated diesel generator (D/G A & D/G 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).

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.

CNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/007 Loss of Normal Power
3. ECA-0.0 EP/1(2)/5000/ECA-0.0 Loss of All AC Power
4. NEI 99-01 SS1

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

SSF fails to supply NC pump seal injection **OR** CA supply to SGs

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 should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 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 CNS Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 1) or that has resulted in indications of an actual loss of adequate core cooling.

The SSF is capable of providing the necessary functions (reactor coolant pump seal injection and auxiliary feedwater supply to the steam generators) to maintain a hot shutdown condition for up to 72 hours. No fission product barrier degradation would be expected if the SSF is functioning as intended.

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 Normal Auxiliary Transformers (ATC & ATD). 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 A & D/G 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. 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.

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.

CNS Basis Reference(s):

1. UFSAR Section 8.4.2 Station Blackout Duration
2. EP/1/A/5000/F-0 Critical Safety Function Status Tress – Core Cooling
3. UFSAR Section 8.0 Electric Power
4. AP/1(2)/A/5500/007 Loss of Normal Power
5. ECA-0.0 EP/1(2)/5000/ECA-0.0 Loss of All AC Power
6. NEI 99-01 SG1

Category: S –System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **all** 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 **all** vital DC buses EDA, EDD, EDB and EDC for ≥ 15 min.

(Note 1)

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

Mode Applicability:

1 - Power Operation, 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). However, alignment of SATA or SATB to an essential bus takes longer than 15 minutes and therefore should only be credited if already aligned.

Each essential bus has a dedicated diesel generator (D/G A & D/G 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 Safe 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).

Four 125 VDC distribution centers are provided for the 125VDC Vital Instrumentation and Control Power System. Four distribution centers (EDA, EDD, EDB and EDC), one per load group, supply the four independent channels of vital instrumentation and control, and are each powered directly from an independent 125 volt battery and battery charger. Each of the four distribution centers supplies one DC panel board and one 125VDC-120VAC static inverter (ref. 1, 3).

The Class 1E DC loads have an operating voltage range of 105 to 135 volts. 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.

CNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. AP/1(2)/A/5500/007 Loss of Normal Power
3 AP/1(2)/A/5500/029 Loss of Vital or Aux Control Power
4. ECA-0.0 EP/1(2)/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 **all** vital DC buses EDA, EDC, EDB, EDD and for ≥ 15 min. (Note 1)	

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Four 125 VDC distribution centers are provided for the 125VDC Vital Instrumentation and Control Power System. Four distribution centers (EDA, EDC, EDB and EDD), one per load group, supply the four independent channels of vital instrumentation and control, and are each powered directly from an independent 125 volt battery and battery charger. Each of the four distribution centers supplies one DC panel board and one 125VDC-120VAC static inverter (ref. 1, 2).

The Class 1E DC loads have an operating voltage range of 105 to 135 volts. 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.

CNS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2 AP/1(2)/A/5500/029 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 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 • In-core T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow in at least one S/G

Mode Applicability:

1 - Power Operation, 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-1 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.

CNS Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. OP/1(2)/A/6700/003 Operation With the Operator Aid Computer Out of Service
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 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 • In-core T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow in at least one S/G

Table S-3 Significant Transients
• Reactor trip • Runback > 25% thermal power • Electrical load rejection > 25% electrical load • Safety injection actuation

Mode Applicability:

1 - Power Operation, 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-1 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-2 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.

CNS Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. OP/1(2)/A/6700/003 Operation with the Operator Aid Computer Out of Service
3. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – NCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
NCS activity > Technical Specification 3.4.16 limits	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Technical Specification Section 3.4.16, as modified in the Facility Operating License, limits NC System Dose Equivalent I-131 to ≤ 0.46 µCi/gm. Technical Specification Section 3.4.16 also limits NC System Dose Equivalent Xe-133 to ≤ 280 µCi/gm. (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.

CNS Basis Reference(s):

1. CNS Technical Specifications section 3.4.16 RCS Specific Activity
2. Facility Operating License Attachment B
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 should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 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 (RCP) 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 (ref. 1).

Unidentified leakage is all leakage (except RCP seal water injection or leakoff) that is not identified leakage (ref. 1).

Pressure Boundary leakage is leakage (except SG leakage) through an unisolable 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) (ref. 2)

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

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.

CNS Basis Reference(s):

1. CNS Technical Specifications Definitions section 1.1
2. UFSAR Section 5.2.5.2.1 Intersystem Leakage
3. 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 success 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 Operation

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.

CNS Basis Reference(s):

1. CNS 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 success 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 Operation

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 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.

CNS Basis Reference(s):

1. CNS 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 Operation

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 RTS 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 RTS.

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.

CNS Basis Reference(s):

1. CNS 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 $\geq$ 5%

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power $\geq$ 5%

AND EITHER:

- Core Cooling RED PATH conditions met
- Heat Sink RED PATH conditions met

Mode Applicability:

1 - Power Operation

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).

Indication of inability to adequately remove heat from the NCS is manifested by CSFST Heat Sink RED PATH conditions being met (ref. 3).

This IC addresses a failure of the RTS 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.

CNS 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	
Commercial Telephones		X	X
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System (ETS)		X	X

Mode Applicability:

1 - Power Operation, 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 Catawba Plant 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 Catawba Site 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.

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.

Satellite Phones

A portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. These portable systems can be powered by internal batteries, external DC sources as well as external AC sources.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

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 Catawba 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, York, Gaston and Mecklenburg County EOCs

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

CNS Basis Reference(s):

1. CNS Emergency Plan Section F Emergency Communications
2. NEI 99-01 SU6

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 < one full train of containment cooling operating per design for > 15 min. (Notes 1, 10)

Note 1: The Emergency Coordinator 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 Operation, 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.

CNS Basis Reference(s):

1. CNS Technical Specification 3.6.6
2. CNS Technical Specification 3.6.6 Bases
3. CNS 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 Operation, 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. CNS plant yard elevation is 593.5 ft MSL. The minimum external access elevation for the Auxiliary, Turbine and Service Buildings is 594.0 ft MSL (ref. 1, 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 95 mph. (ref. 4).
- 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.

CNS Basis Reference(s):

1. RP/0/A/5000/007 Natural Disaster and Earthquake
2. AP/0/A/5500/030 Plant Flooding
3. UFSAR Section 3.4 Water Level (Flood) Design
4. Updated FSAR Section 3.3.1 Wind Loadings
5. AP/0/A/5500/045 Plant Fire
6. NEI 99-01 SA9

Category E – <u>Independent Spent Fuel Storage Installation (ISFSI)</u>

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 CNS 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.

Category: E - ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Notification of Unusual Event

Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded Vertical Storage Cask (VSC) > **any** of the following:

- 100 mrem/hr (neutron + gamma) on the side of the VSC
- 100 mrem/hr (neutron + gamma) on the top of the VSC
- 200 mrem/hr (neutron + gamma) at the air inlets or outlets of the VSC

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 CNS ISFSI, Confinement Boundary is defined as the Transportable Storage Canister (TSC) for both NAC-UMS and MAGNASTOR storage systems.

Basis:

The CNS ISFSI utilizes two designs for dry spent fuel storage:

- The NAC-UMS dry spent fuel storage system
- The MAGNASTOR dry spent fuel storage system

Both systems consist of a Transportable Storage Canister (TSC) and concrete Vertical Storage Cask (VSC). The TSC is the CONFINEMENT BOUNDARY for both systems. The TSC is welded and designed to provide confinement of all radionuclides under normal, off-normal, and accident conditions (ref. 1, 2).

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 represent 2 times the limits specified in the ISFSI Certificate of Compliance Technical Specification for radiation external to a loaded VSC for a NAC-UMS canister (ref. 1).

The specified ISFSI dose limits are based on surveys taken consistent with the locations specified in the associated Technical Specification (ref. 1, 2).

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.

CNS Basis Reference(s):

1. NAC-UMS Certificate of Compliance #1015 Technical Specifications
2. MAGNASTOR Certificate of Compliance #1031 Technical Specifications
3. 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 CNS 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 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 Operation, 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

CNS 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 Operation, 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 would have greater assurance that escalation to a General Emergency is less imminent.

CNS 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 Operation, 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

CNS Basis Reference(s):

1. NEI 99-01 FG1

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 B would be assigned "FC Loss B.1," the third Containment barrier Potential Loss in Category D would be assigned "CMT P-Loss D.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. CSFST Integrity-RED Path conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. CSFST Core Cooling-RED Path conditions met	1. CSFST Core Cooling-ORANGE path conditions met 2. CSFST Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED Path conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C **CMT Radiation / NCS Activity**	1. EMF53A/B > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. EMF53A/B > Table F-2 column "NCS Loss"	None	None	1. EMF53A/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 judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of NCS leakage outside of containment	1. CSFST Containment-RED Path conditions met 2. Containment hydrogen concentration > 6% 3. Containment pressure > 3 psig with < one full train of containment cooling operating per design for > 15 min. (Note 1)
E **EC Judgment**	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator 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. CSFST 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.

CNS Basis Reference(s):

1. EP/1(2)/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. CSFST Core Cooling-ORANGE Path conditions met

Definition(s):

None

Basis:

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path 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.

CNS Basis Reference(s):

1. EP/1(2)/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:

2. CSFST 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 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. 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.

CNS Basis Reference(s):

1. EP/1(2)/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. EMF53A/B > Table F-2 column "FC Loss"

Table F-2 Containment Radiation – R/hr (EMF53A/B)			
Time After S/D (Hrs.)	**FC Loss**	**NCS Loss**	**CMT Potential Loss**
0-1	550	8.8	5500
1-2	400	8.4	4000
2-8	160	7.0	1600
>8	100	6.2	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, EMF53A & B. EMF53A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale). Radiation Monitors EMF53A & 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 (EMF53A & B) response based on a LOCA, for periods of 1, 2, 8 and 16 hours after shutdown, no sprays and NCS pressure < 1600 psig with ~2% fuel failure (ref. 2).

The value is derived as follows:

RP/0/A/5000/015 Figure 3 Containment Radiation Level vs. Time for 100% Clad Damage 1, 2, 8 and 16 hours after shutdown without spray and NCS pressure < 1600 psig x 0.02 (rounded) (ref. 2).

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.

CNS Basis Reference(s):

1. IP/0/3314/004 Radiation Monitoring System RP-2C High Range Process Channel Calibration
2. RP/0/A/5000/015 Core Damage Assessment
3. 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.

CNS Basis Reference(s):

1. RP/0/A/5000/015 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 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 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.

CNS Basis Reference(s):

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 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.

CNS 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 > 1.2 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.

CNS 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. CSFST 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 Curve A indicates an extreme challenge to the safety function when plant parameters are to the left 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).

CNS 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

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST 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.

CNS Basis Reference(s):

1. EP/1(2)/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. EMF53A/B > Table F-2 column "NCS Loss"

Table F-2 Containment Radiation – R/hr (EMF53A/B)			
Time After S/D (Hrs.)	**FC Loss**	**NCS Loss**	**CMT Potential Loss**
0-1	550	8.8	5500
1-2	400	8.4	4000
2-8	160	7.0	1600
>8	100	6.2	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, EMF53A & B. EMF53A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale). Radiation Monitors EMF53A & 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/5000/015 Figure 1, the expected containment high range radiation monitor (EMF53A & B) response based on a LOCA, for periods of 1, 2, 8 and 16 hours after shutdown with no fuel failure (ref. 2).

The value is derived as follows:

RP/0/A/5000/015 Figure 1 Containment Radiation Level vs. Time for RCS Release for periods of 1, 2, 8 and 16 hours after shutdown (rounded) (ref. 2).

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.

CNS Basis Reference(s):

1. IP/0/3314/004 Radiation Monitoring System RP-2C High Range Process Channel Calibration
2. RP/0/A/5000/015 Core Damage Assessment
3. 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 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.

CNS 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 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 Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

CNS 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.

CNS 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

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. CSFST 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.

CNS Basis Reference(s):

1. EP/1(2)/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. EMF53A/B > Table F-2 column "CMT Potential Loss"

Table F-2 Containment Radiation – R/hr (EMF53A/B)			
Time After S/D (Hrs.)	**FC Loss**	**NCS Loss**	**CMT Potential Loss**
0-1	550	8.8	5500
1-2	400	8.4	4000
2-8	160	7.0	1600
>8	100	6.2	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, EMF53A & B. EMF53A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale). Radiation Monitors EMF53A & 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 (EMF53A & B) response based on a LOCA, for periods of 1, 2, 8 and 16 hours after shutdown, no sprays and NCS pressure < 1600 psig with ~20% fuel failure (ref. 2).

The value is derived as follows:

RP/0/A/5000/015 Figure 3 Containment Radiation Level vs. Time for 100% Clad Damage 1, 2, 8 and 16 hours after shutdown with no spray and NCS pressure < 1600 psig x 0.20 (rounded) (ref. 2).

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.

CNS Basis Reference(s):

1. IP/0/3314/004 Radiation Monitoring System RP-2C High Range Process Channel Calibration
2. RP/0/A/5000/015 Core Damage Assessment
3. 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 RCP 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.

CNS 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)
- RCP 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.

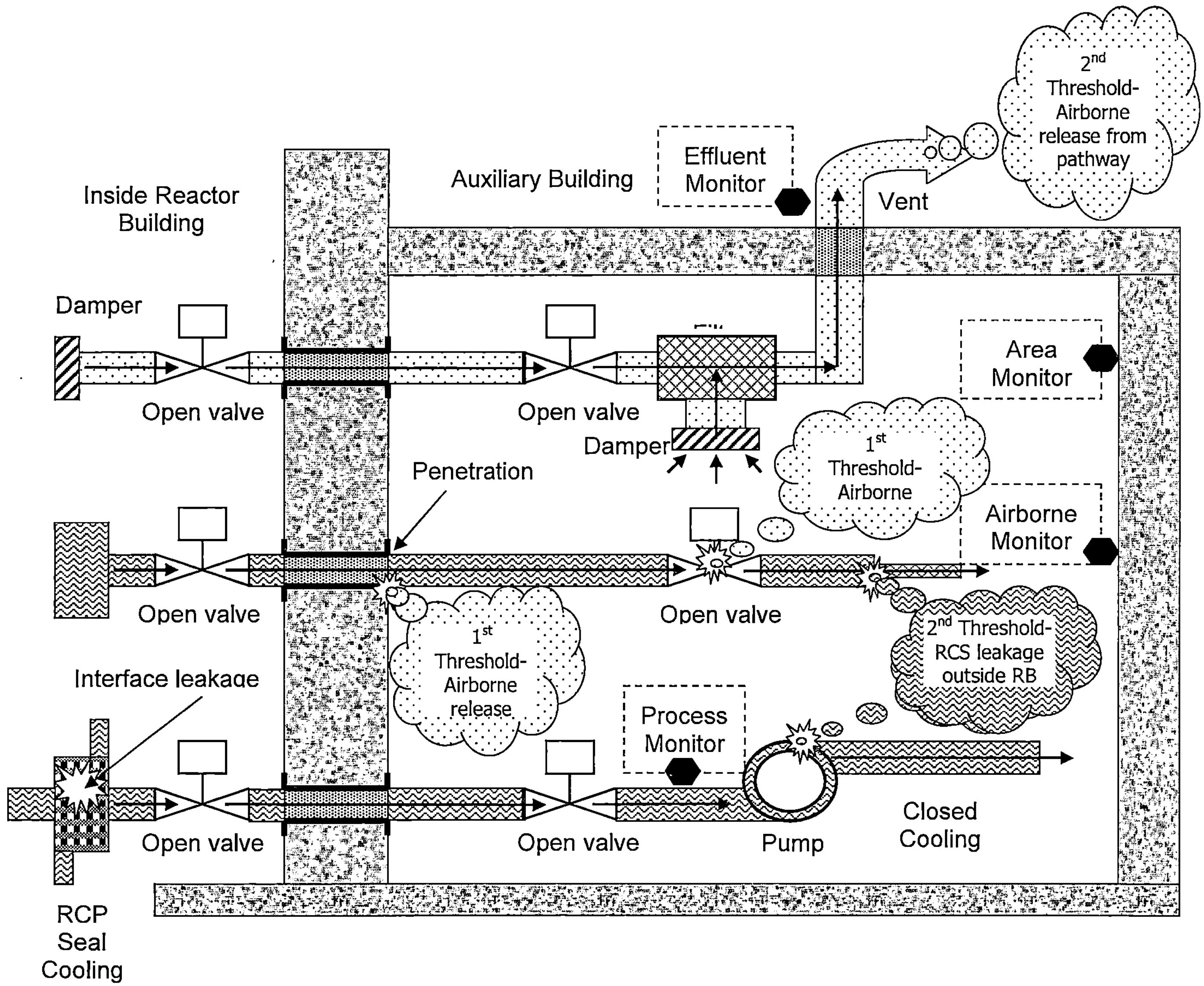
Refer to the middle piping run of Figure 1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the 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.

CNS 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.	CSFST 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. The CSFSTs are normally monitored using the SPDS display on the Operator Aid Computer (OAC) (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.

CNS 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

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 Containment Hydrogen Purge and Sample System (VY) is used to monitor the hydrogen concentration inside containment after a severe accident involving core damage. Samples of Containment air are obtained via the containment hydrogen/oxygen sample lines to the Post Accident Containment Sample (PACS) panel located in the auxiliary building. Additionally, the containment hydrogen analyzer system continuously monitors the hydrogen concentration inside containment (ref. 1).

The lower limit of deflagration of hydrogen in air is approximately 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.

CNS 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 < one full train of containment cooling operating per design for > 15 min. (Notes 1, 10)

Note 1: The Emergency 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.

CNS Basis Reference(s):

1. CNS Technical Specification 3.6.6
2. CNS Technical Specification 3.6.6 Bases
3. CNS 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 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.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency 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.

CNS 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 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.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency 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.

CNS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A

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.

The review at CNS was completed using the following Controlling Procedures:

- OP/1(2)/A/6100/003 (Controlling Procedure For Unit Operation)
- OP/1(2)/A/6100/002 (Controlling Procedure For Unit Shutdown)

CNS 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:

CNS Procedure and Step	Step Action	Building/Elevation/Room	Mode	If action not performed does this prevent cooldown/ shutdown?
OP/1/A/6100/003 Enclosure 4.3 Step 3.19: U2 Encl. 4.3 Step 3.20	Coordinate with Chemistry and Radwaste while performing NC System Degas. Radwaste continues Degas OPS thru shutdown & cooldown	Auxiliary Building (Various Locations)	1	No
OP/1/A/6100/003, Enclosure 4.3, Steps 3.26, 3.27 and 3.28; U2 Encl. 4.3, Steps 3.28, 3.29 and 3.30	Align heater vent orifices per OP/1(2)/B/6250/004 (Feedwater Heaters, Vents, Drains and Bleed Systems) ,Align VI and SP valves associated with CFPTs	Turbine Building (Various Locations)	1	No
OP/1/A/6100/003, Enclosure 4.2, Step 3.11: U2 Encl. 4.2 Step 3.10	Align Auxiliary Steam to CFPTs.	Turbine Building (Various Locations)	1	No
OP/1&2/A/6100/003, Enclosure 4.2, Steps 3.13; U2 Encl. 4.2 Step 3.12	Align "C" Htr Drain Pump per OP/1(2)/B/6250/004 (Feedwater Heaters, Vents, Drains and Bleed Systems) for removal from service	Turbine Building (568')	1	No
OP/1/A/6100/003, Enclosure 4.2, Step 3.14; U2 Encl. 4.2 Step 3.13	Plant activities to ensure Main Turbine Sealing Steam system responds as required.	Turbine Building (594')	1	No
OP/1&2/A/6100/003, Enclosure 4.2, Step 3.18: U2 Encl. 4.2 Step 3.17	Ensure Moisture Separator Reheater low load valve operation per OP/1(2)/B/6250/013 (Moisture Separator Reheater Operation)	Turbine Building (619')	1	No
OP/1/A/6100/003 Enclosure 4.2 Step 3.19: U2 Encl. 4.2 Step 3.18	Secure one Main CFPT per OP/1(2)/A/6250/001 (Condensate and Feedwater System)	Turbine Building (Mainly 594')	1	No

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases

CNS Procedure and Step	Step Action	Building/Elevation/Room	Mode	If action not perform does this prevent cooldown/ shutdown?
OP/1/A/6100/003, Enclosure 4.2, Step 3.20; U2 Encl. 4.2 Step 3.19	Secure half Main Transformer Cooling Fans and oil pumps	Outside in Main Transformer yard	1	No
OP/1(2)/A/6100/003 Enclosure 4.2 Step 3.21:	Shutdown the Main Turbine per OP/1(2)/B/6300/001 (Turbine generator)	Turbine Building and Transformer Yard	1	No
OP/1(2)/A/6100/003, Enclosure 4.2, Step 3.28 &3.29	Bypass "F" LP heaters	Turbine Building (594" LP Htr Panel).	1	No
OP/1(2)/A/6100/003, Enclosure 4.2, Step 3.34	Transfer of Aux Steam to on line Unit per OP/0/B/6250/007 A (Auxiliary Steam System Alignment) or place Aux Electric Boiler in service per OP/1/B/6250/007 B (Auxiliary Electric Boilers)	Service Bldg. (568')	1	No
OP/1(2)/A/6100/003, Enclosure 4.2, Step 3.37	Isolate Unit Related Steam supply to Aux Steam Header	Turbine Building. (594')	1	No
OP/1(2)/A/6100/002, Enclosure 4.1, Step 3.4	Initiate action to reduce VCT pressure per OP/1(2)/6200/001 (Chemical and Volume Control System)	Auxiliary Building (577' Mechanical Pent. Room)	1, 2, 3	No
OP/1(2)/A/6100/002, Enclosure 4.1, Step 3.9	Align S/G reverse purge.	Both Doghouses	1	No
OP/1(2)/A/6100/002, Enclosure 4.1, Step 3.52	Align CM system flow for Low Pressure cleanup thru Upper Surge Tank.	Turbine Building (619')	3	No
OP/1(2)/A/6100/002 Enclosure 4.2 or 4.7, Step 3.3	Shutdown Rod Control System per OP/1(2)/6150/008 (Rod Control)	Auxiliary Building (594' Electrical Pent Room)	3	No
OP/1(2)/A/6100/002 Enclosure 4.2 or 4.7 Step 3.10	Chemistry obtains samples to ensure boron concentration good to allow NCS cooldown to begin	Auxiliary Building (543' Sample Lab)	3	No

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases

CNS Procedure and Step	Step Action	Building/Elevation/Room	Mode	If action not performed does this prevent cooldown/ shutdown?
OP/1(2)/A/6100/002 Enclosure 4.2 or 4.7, Step 3.22	Perform PZR PORVs stroke testing per PT/1(2)/A/4200/023 A (NC valve Inservice Test)	Auxiliary. Building (577', IF performing IWVR Containment 635' as well)	3	No
OP/1(2)/A/6100/002 Enclosure 4.2 or 4.7 Step 3.23	Support placing N2 Cover gas on NCDT per OP/1(2)/A/6500/014 (Operations Controlled Liquid Waste Systems)	Auxiliary Building (Various Locations on 577' & 560')	3	No
OP/1(2)/A/6100/002, Enclosure 4.2 or 4.7 Step 3.31	Removing CLAs from service per OP/1(2)/A/6200/009 (Cold Leg Accumulator Operation).	Auxiliary Building (577' & 560' Ess. MCC Bkrs)	3	No
OP/1&2/A/6100/002 Enclosure 4.2 or 4.7, Step 3.45.1	Remove CAPT and one Motor Driven CA Pump from service per OP/1(2)/A/6250/002 (Auxiliary Feedwater System).	Auxiliary Building. (577' & 560 Ess MCC Bkrs)	4	No
OP/1(2)/A/6100/002 Enclosure 4.2 or 4.7, Step 3.46.3	Open NCS Loop Suction Vlvs for train of ND to be placed in service per OP/1(2)/A/6200/004 (Residual Heat removal System).	Auxiliary. Building. (577' & 560' Ess. MCC Bkrs)	4	Yes
OP/1(2)/A/6100/002 Enclosure 4.2 or 4.7, Step 3.48.2	Rack out appropriate NI and NV Pump Motor Bkrs per OP/0/A/6350/010 (Operation of Station Breakers and Disconnects)	Auxiliary Building (577' & 560 Electrical Pent Rooms)	4	Yes
OP/1(2)/A/6100/002 Enclosure 4.2 or 4.7 Step 3.52.2	Support placing first train of ND in service per OP/1(2)/A/6200/004 (Residual Heat removal System)	Auxiliary Building (577' or 560' Ess MCC Bkr s)	4	Yes

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases

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**	**Mode**
Auxiliary 577'	Rm 478 (1EMXA)	Rm 469 (2EMXA)	4
	Rm 496 (1ETA)	Rm 486 (2ETA)	4
	Rm 496 (1EMXS)	Rm 486 (2EMXS)	4
	AB-577', JJ-57 (1MXK)	AB-577', JJ-57 (2MXK)	4
Auxiliary 560'	Rm 330 (1EMXJ)	Rm 320 (2EMXJ)	4
	Rm 372 (1ETB)	Rm 362 (2ETB)	4
	Rm 372 (1EMXD)	Rm 362 (2EMXD)	4

Plant Operating Procedures Reviewed

1. OP/1(2)/A/6100/003 (Controlling Procedure for Unit Operation)
2. OP/1(2)/A/6100/002 (Controlling Procedure for Unit Shutdown)
3. OP/1(2)/B/6250/004 (Feedwater Heaters, Vents, Drains and Bleed Systems)
4. OP/1(2)/B/6250/013 (Moisture Separator Reheater Operation)
5. OP/1(2)/A/6250/001 (Condensate and Feedwater System)
6. OP/1(2)/B/6300/001 (Turbine generator)
7. OP/0/B/6250/007 A (Auxiliary Steam System Alignment)
8. OP/1/B/6250/007 B (Auxiliary Electric Boilers)
9. OP/1(2)/6200/001 (Chemical and Volume Control System)
10. OP/1(2)/6150/008 (Rod Control)
11. PT/1(2)/A/4200/023 A (NC valve Inservice Test)
12. OP/1(2)/A/6500/014 (Operations Controlled Liquid Waste Systems)
13. OP/1(2)/A/6200/009 (Cold Leg Accumulator Operation)
14. OP/1(2)/A/6250/002 (Auxiliary Feedwater System)
15. OP/1(2)/A/6200/004 (Residual Heat removal System)
16. OP/0/A/6350/010 (Operation of Station Breakers and Disconnects)

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DUKE ENERGY
CATAWBA NUCLEAR STATION

APPENDIX INDEX

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.

ALERT
Events are in process 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 protection action guideline exposure levels.

ALL (As relates to Operating Mode Applicability)
Modes 1,2,3,4,5,6 and No Mode (Defueled)

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.)

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

CARF
Containment Air Return Fan

CIVIL DISTURBANCE
A group of ten (10) or more people violently protesting station operations or activities at the site. A civil disturbance is considered to be violent when force has been used in an attempt to injure site personnel or damage plant property.

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.

CREDIBLE THREAT
A threat should be considered credible when:
- Physical evidence supporting the threat exists.
- Information independent (law enforcement) from the actual threat message exists that supports the threat.
- A specific group or organization claims responsibility for the threat.

DEGRADING
Plant conditions involve at least one of the following:
- Plant parameters (ex. temperature, pressure, level, voltage, frequency) are trending unfavorably away from expected or desired values **AND** plant conditions could result in a higher classification or Protective Action Recommendation (PAR) before the next follow-up notification.
- Environmental site conditions (ex., wind, ice/snow, ground tremors, hazardous/toxic/radioactive material leak, fire) impacting plant operations or personnel safety are worsening **AND** plant conditions could result in a higher classification or Protective Action Recommendation (PAR) before the next follow-up notification.

DRILL
A drill is a supervised instruction period aimed at testing, developing, and maintaining skills in a particular operation.

EMERGENCY ACTION LEVELS (EALs)
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, observable event; results of analyses; entry into specific emergency operating procedures; 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.

EMERGENCY RELEASE
Any unplanned, quantifiable radiological release to the environment during an emergency event. The release does not have to be related to a declared emergency.

EPA PAG
Environmental Protection Agency Protective Action Guidelines for exposure to a release of radioactive material.

EXCLUSION AREA
The nuclear station property, including the site, out to a radius of 2500 feet that meets the 10CFR100 definition.

EXPLOSION
A rapid, violent unconfined combustion or a catastrophic failure of pressurized equipment (e.g., a steamline or feedwater line break) that imparts energy sufficient to potentially damage or creates shrapnel to actually damage permanent structures, systems or components. An electrical breaker flash that creates shrapnel and results in damage to other components beyond scorching should also be considered.

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.

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

FIRE
Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flames is preferred but is NOT required if large quantities of smoke and heat are observed. An electrical breaker flash that creates high temperatures for a short duration and merely localized scorching to that breaker and its compartment should not be considered a fire.

FRESHLY OFF-LOADED REACTOR CORE
The complete removal and relocation of all fuel assemblies from the reactor core and placed in the spent fuel pool. (Typical of a "No Mode" operation during a refuel outage that allows safety system maintenance to occur and results in maximum decay heat load in the spent fuel pool system.)

FUNCTIONAL
A component is fully capable of meeting its design function. It would be declared INOPERABLE if unable to meet Technical Specifications.

GENERAL EMERGENCY
Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity 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.

HOSTAGE
A person or object held as leverage against the site to ensure demands will be met by the site.

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

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

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

IMPROVING
Plant conditions involve at least one of the following:

- Plant parameters (ex., temperature, pressure, level, voltage, frequency) are trending favorably toward expected or desire values **AND** plant conditions could result in a lower classification or emergency termination before the next follow-up notification.
- Environmental site conditions (ex., wind, ice/snow, ground tremors, hazardous/toxic/radioactive material leak, fire) have become less of a threat to plant operations or personnel safety **AND** plant conditions could result in a lower classification or emergency termination before the next follow-up notification.

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.

INOPERABLE
A component does not meet Technical Specifications. The component may be functional, capable of meeting its design.

INABILITY TO DIRECTLY MONITOR
Operational Aid Computer data points are unavailable or gauges/panel indications are not readily available to the operator.

INTRUSION
A person(s) present in a specified area without authorization. Discovery of a **BOMB** in a specified area is indication of **INTRUSION** into that area by a **HOSTILE FORCE**.

ISFSI
Independent Spent Fuel Storage Installation - Includes the components approved for loading and storage of spent fuel assemblies.

LOSS
A component is INOPERABLE and not FUNCTIONAL.

MONTHLY
For periodic emergency planning requirements, monthly is defined as once each month, with a maximum interval of 38 days.

NO MODE
Defueled.

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.

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

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.

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

PROLONGED
A duration beyond normal limits, defined as "greater than 15 minutes" or as determined by the judgment of the Emergency Coordinator.

PROTECTED AREA
Typically, the site specific area which normally encompasses all controlled areas within the security **PROTECTED AREA** fence.

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.

REACTOR COOLANT SYSTEM (RCS/NCS) LEAKAGE
RCS Operational Leakage as defined in the Technical Specification Basis B 3.4.13.

RECOVERY ACTIONS
Those actions taken after the emergency to restore affected property as nearly as practicable to its pre-emergency condition.

RUPTURED (As relates to Steam Generator)
Existence of primary to secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

SABOTAGE
Deliberate damage, misalignment or misoperation of plant equipment with the intent to render the equipment unavailable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of **SABOTAGE** until this determination is made by security supervision.

SECURITY CONDITION

Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel or a potential degradation to the level of safety of the plant. A **SECURITY CONDITION** does not involve a **HOSTILE ACTION**.

<u>SEMI-ANNUAL</u>
For periodic emergency planning requirements, semi-annual is defined as once every 6 months, with a maximum interval of 228 days.

<u>SIGNIFICANT PLANT TRANSIENT</u>
An unplanned event involving one or more of the following: (1) Automatic turbine runback >25% thermal reactor power, (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip, (4) Safety Injection, (5) Thermal power oscillations >10%.

<u>SITE</u>
That part of the nuclear station property consisting of the Reactor, Auxiliary, Turbine, Service Buildings and grounds, contained within the outer security area fence.

<u>SITE AREA EMERGENCY</u>
Events are in process 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 the 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.

<u>SITE BOUNDARY</u>
That area, including the protected area, in which Duke Energy has the authority to control all activities, including exclusion or removal of personnel and property.

<u>SLC</u>
Selected Licensee Commitments.

<u>STABLE</u>
Plant conditions are neither degrading nor improving.

<u>SUSTAINED</u>
A duration of time long enough to confirm that the CSF is valid (not momentary).

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.

TERMINATION
Exiting the emergency condition.

TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE)
The sum of external dose exposure to radioactive plume, to radionuclides deposited on the ground by the plume, and the internal exposure inhaled radionuclides deposited in the body.

TOXIC GAS
A gas that is dangerous to life or health by reason of inhalation or skin contact (e.g. chlorine).

UNCONTROLLED
Event is not the result of planned actions by the plant staff.

UNPLANNED
An event or action is UNPLANNED if it is not the expected result of normal operations, testing or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

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

VIOLENT
Force has been used in an attempt to injure site personnel or damage plant property.

VISIBLE DAMAGE
Damage to equipment or structure that is readily observable without measurements, testing or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected structure, system, or component. Example damage: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering.

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 plant. 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.

APPENDIX 2
CATAWBA NUCLEAR STATION
METEOROLOGICAL PROGRAM

INTRODUCTION

In response to guidance provided by NUREG-0654, Revision 1 and supporting documents, Regulatory Guide 1.23, Proposed Revision 1, Regulatory Guide 1.111, Revision 1, and Regulatory Guide 1.109, Duke has reviewed the existing meteorological system at Catawba Nuclear Station and, based on that review, has developed a plan for upgrading the meteorology system.

The meteorological measurement program at Catawba Nuclear Station was originally designed to best describe the meteorological conditions on-site by taking into account source characteristics, terrain features and modeling needs. Duke has changed the meteorological system by upgrading the instrumentation and modifying the data transfer and access methodologies. The modifications include:

1) The meteorological microprocessor has been replaced with a digital data link connecting the instrumentation and the station.

2) The analog chart recorders have been replaced with digital chart recorders.

3) The data is scanned and averaged by the station process monitoring computer and transferred to databases accessible by the ERO.

EFFLUENT DISPERSION MODEL

The Class A model has calculation capability that can produce initial transport and diffusion estimates for the plume exposure emergency planning zone within fifteen minutes following classification of an incident. The Class B model is a numerical model that represents actual spatial (space) and temporal (time) variations affecting plume distribution; it can provide estimates of deposition and relative concentration of radioactivity within the plume exposure and ingestion planning zones for the duration of the release. More detailed description can be found in INPO 86-008 Dose Assessment Manual.

The effluent dispersion model at Catawba uses a variable trajectory, puff advection dispersion model to simulate atmospheric transport and diffusion of radioisotopes from Catawba Nuclear Station. Plume trajectories are calculated using meteorological data obtained directly from the site meteorological tower. Puffs are transported by the horizontal wind field which varies with time. The diffusion (or spread) of each puff is based on a Gaussian distribution model. The dimensions of individual puffs, which compose the plume, are determined as a function of travel distance and atmospheric stability. Further, the initial dimensions of puffs are adjusted to account for building wake effects. Plume growth during changing atmospheric stability conditions is determined using a virtual source concept. Each puff is released at a rate which is based on current fifteen minute forecasted meteorology. The puff advection model is used for both the real-time and the forecast operating modes. In the real time mode, the model uses actual Operator Aid Computer (OAC) fifteen minute averaged data as it becomes available. For a forecast, the user is required to enter one time set of meteorological data representative of the entire period.

Radioisotopes released to atmosphere are assumed to be distributed in a Gaussian manner, subject to reflection in the vertical direction between the surface boundary and mixing layer lid (i.e., mixing height) above. The diffusion of release materials is expressed in terms of a normalized concentration χ/Q. Normalized concentrations are multiplied by a source strength Q to provide an estimate of cloud concentration $\chi(Ci/m^3)$. Puff depletion that takes in consideration the removal of iodines and particulate from the plume as a result of dry and wet deposition; which is also calculated. Deposition fluxes are provided to assist in the identification of areas where relative high levels of surface contamination might be expected to occur. Diffusion and deposition for each puff are determined after each advection step. Modeled release from Catawba Nuclear Station is assumed to be at or below the containment structure; therefore, all releases are modeled as being emitted from ground-level sources. The model uses modified σ_y and σ_z diffusion parameters to account for building downwash effects on ground level releases. The model dispersion routines include the concept of a mixing height which recognizes that the atmosphere is heated from below as the earth absorbs the sun's ultraviolet radiation. The height above ground for this boundary, between lower unstable and upper stable air is known as the mixing height. The value for mixing height used in the model is based on seasonal afternoon mean at the site. Atmospheric stability is determined from the vertical temperature gradient (delta-temperature) for stability classification. At the end of each advection step, total dry and wet deposition from all puffs are calculated and accumulated at each model receptor location.

INSTRUMENTATION

Figure 2-1 shows the type and number of parameters measured at Catawba Nuclear Station. The meteorological conditions present at Catawba Nuclear Station warrant the use of the basic described meteorological variables. These include wind speed and wind direction measured at high and low levels, and delta-temperature. Ambient air temperature, dew point temperature and precipitation instrumentation are also provided but are not required as input for off-site dose assessment calculations.

DATA HANDLING

Meteorological data used for dose calculations are 15 minute running averages of the variables. The 15 minute running averages are determined by the Operator Aid Computer (OAC) which scans the variables each minute. The data is stored on databases that are accessed by the personnel performing the dose calculations. As a backup, the variables are also recorded each five seconds on digital chart recorders located in the Control Room. These systems meet the accuracy and other specifications suggested in Regulatory Guide 1.23, Proposed Revision 1.

DOSE ASSESSMENT METHODOLOGY

The first radiological indication of a problem in a reactor building is through increased control room monitor readings from containment particulate and noble gas (EMF) skid package. It is assumed that the first monitor to indicate increase of containment activity is the noble gas monitor because it is a non-integrating, near instantaneous response to increased noble gas radioactivity in containment. Leak rate from containment to the annulus or bypass to the environment may be based on containment design basis leakage, or leakage may be a function of containment pressure and hole size. Unit vent release may be from several ventilation source intakes including annulus and Auxiliary Building ventilation systems. It is possible both Unit 1 and 2 vents could contribute to an off-site release because of shared ventilation. Each unit vent is monitored with particulate and noble gas (EMF) skid package with indication and detection as previously stated. There are four main steam lines per unit (A,B,C,D) with coded Safety Relief valves; Power Operated Relief Valve (PORV), atmospheric steam dump valves and each unit has an auxiliary feedwater pump turbine valve release path. Steamlines have monitors (EMFs) installed, including N^{16} detectors that may provide first indication of primary to secondary leakage. Steam generator tube leakage is monitored through the affected unit Condensate Steam Air Ejector Monitor. Steam Release (MSR) accumulator program on the Operator Aid Computer scans these valves and calculate pounds mass (lbm) released based on valves being read closed or not closed.

The model can be used to calculate Source Term release through up to five release pathways and has capability of maintaining an inventory of up to twenty-four radioisotopes for each selected accident type(s). The model assumes a release to include noble gases, iodines, and particulates unless release path grab sample is obtained and analyzed, and model direct entry of nuclides is selected for Source Term calculation. Dose calculation methods attempt to predict dose concentration at specific receptor locations downwind from the release point. The model provides dose calculations from plume exposure, inhalation and material deposited on the ground consistent with methods of the EPA-400-R-92-001 document, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*. Using dose rate conversion factors, the model calculates a combined dose from external exposure from the plume with plume inhalation and four day external exposure from material deposited on the ground (the sum of which is referred to as the Total Effective Dose Equivalent [TEDE]), as well as the Committed Dose Equivalent to the Thyroid from inhalation of radioiodines (referred to as CDE). For the forecast period (expected release duration using a default of four hours), the TEDE and its separate components, and CDE Thyroid dose is calculated and then used to determine Protective Action Recommendations (PAR) consistent with Protective Action Guides (PAGs) given in EPA 400.

DETAILED DESCRIPTION OF SUBSYSTEMS

Sensors to Operator Aid Computer

Lightning protection is provided for all sensors and signal conditioning equipment; wind sensors are outfitted with heating jackets, when necessary, for protection against icing conditions. Signal conditioners are housed in an environmentally controlled building at the base of the microwave tower. Signals to the plant are converted from analog to digital and transmitted via a data link. For each variable, one channel transmits data to the OAC and another transmits to the chart recorders.

Operator Aid Computer (OAC) to Plant Databases

The Operator Aid Computer systems use process monitoring equipment. Meteorological data is received at the station, converted from digital to analog, and scanned each minute by the Unit 2 OAC. Each minute, the fifteen minute running average of each parameter is calculated and passed to the Unit 1 OAC. Each OAC transmits data to two databases, one hosted on the site VAX system and the other hosted on a site PC server. ERO personnel can access the data on either database using PCs located in each emergency facility. Alternatively, the current data may be accessed directly on either OAC using terminals located in the Technical Support Center.

Digital Chart Recorders

Meteorological data is also received at the station, converted from digital to analog, and scanned every five seconds by digital chart recorders. These are located in the Control Room. The recorders accumulate the average of the samples for each hour and print this information on the charts.

QUALITY ASSURANCE

Meteorological components have been designed, procured and installed as a non-safety related system. New equipment has been purchased from suppliers who have provided high quality, reliable products in the past. Surveillance during construction was provided as for any other non-safety system.

Maintenance, calibration and repair procedures are available at the site for inspection. Inventories of meteorological system spare parts, sensors and components are maintained in company files.

FIGURE 2-1

CATAWBA NUCLEAR STATION
METEOROLOGICAL PARAMETERS OF THE UPGRADED SYSTEM

Measurement System	60 m (upper level)	Upper wind speed and direction Upper RTD
	10 m (lower level)	Lower wind speed and direction Lower RTD

NOTE 1: ΔT is obtained by subtracting the lower RTD from the upper RTD.

NOTE 2: Ambient dry bulb temperature, dew point and precipitation parameters are provided but not required as input for off-site dose assessment calculations.

APPENDIX 3

DUKE ENERGY
CATAWBA NUCLEAR STATION
ALERT AND NOTIFICATION SYSTEM DESCRIPTION

GENERAL DESCRIPTION

The Alert and Notification System for Catawba Nuclear Station consists of an acoustic alerting signal and notification of the public by commercial broadcast (EAS - Emergency Alert System). The system is designed to meet the acceptance criteria of Section B of Appendix 3, NUREG-0654, FEMA-REP-1, Rev. 1.

An engineering study of the Catawba Nuclear Station Alerting System was prepared by Duke-Energy and was submitted February, 1983. This is an annotated version of the study.

The Emergency plans of Duke Energy, the States of North Carolina and South Carolina, and the counties of Mecklenburg, Gaston, and York include the organizations and individuals, by title, who will be responsible for decision-making as regards the alert and notification system. The county locations from which the sirens would be activated and, potentially, the request for an EAS message would come are manned 24 hours per day. Each organization's plan describes provisions for use of public communications media or other emergency instructions to members of the public. The plans of both states include a description of the information that would be communicated to the public under given circumstances.

A. Concept of Operations

A system of 89 fixed sirens is installed and operational in the 10 mile EPZ area around Catawba Nuclear Station. A backup means of alerting and notification is described in the State and County Plans. This backup method includes reverse 911 and area-wide emergency service vehicles traversing the area giving both an alerting signal and notification message.

Each county will control the activation of the sirens within its boundaries.

B. Criteria for Acceptance

The alert and notification system for the Catawba Nuclear Station provides an alerting signal and an informational or instructional message to the population (via the EAS) on an area-wide basis throughout the 10 mile EPZ within 15 minutes from the time the cognizant off-site agencies have determined the need for such alerting exists. The emergency plans of each state include evidence of EAS preparation for emergency situations and the means for activating the system.

C. Physical Implementation

1. The activation of this alert and notification system requires procedures and relationships between both Duke Energy and the off-site agencies that support Duke and Catawba Nuclear Station.

   When an incident is determined to have reached the level requiring public protective actions, Duke contacts the cognizant off-site agency via the Duke Emergency Management Network (DEMNET) and provides its recommendations. This system is available for use 24 hours per day and links the Control Room, TSC, EOF, SERT headquarters, the county warning points/EOCs, and the state Warning Point/EOCs.

2. The alert and notification system has multipurpose use built into it. The sirens are capable of producing a three minute steady signal for the nuclear plant emergency, natural disasters or nuclear attack. Procedures exist at the counties to allow activation of the sirens.

The expected performance of the sirens used in this system is described in Figure 3-1. These sirens complement existing alerting systems. The ambient background sound level in the Catawba area is taken to be 50 db for areas of "less than 2000 persons/per square mile" and 60 db for areas above this density. On this basis, the siren coverages are designed to provide a signal 10db above the average daytime ambient background.

Furthermore, the sirens have been located to assure that the maximum sound levels received by any member of the public should be lower than 126 db.

The basis for our selection of the 60 db(c) and 70 db(c) criteria is documented as follows:

Location of heavy industry - There is limited "heavy industry" in the Catawba 10 mile EPZ as described in Chapter 2 of the Catawba Nuclear Station UFSAR.

Attenuation factors with distance - 10 db loss per distance doubled (See Figure 3-1)

Siren output db(c) at 100 ft. vs. assumed range and acoustic frequency spectra -
2001AC: 127 $\pm$ 1.0db at 100 feet

<u>Assumed ranges per</u> Figure 3-1, 10 db loss column

<u>Frequency Spectra</u>:

2001AC: top frequency 705Hz

Map showing siren location - See Figure 3-2

Mounting height of sirens - 50 feet (approximate)

Special weather condition considerations (such as expected heavy snow) - None

The siren will produce a 3-minute steady signal and is capable of repetition.

Test or Maintenance	Required frequency	Duke frequency
Silent Test performed by County	Every two weeks	Weekly
Silent Test performed by Telcomm	Every two weeks	Weekly and following Corrective and Preventative Maintenance
Growl Test	Quarterly and when Preventive Maintenance is performed. A Growl Test is performed following Preventive Maintenance	Full Cycle Test is performed in lieu of the Quarterly Growl Test.
Full Cycle Test	Annually	Full Cycle/Quarterly
Preventive Maintenance	At least Annually	Annually

Note: Full Cycle Test may substitute for a growl test.

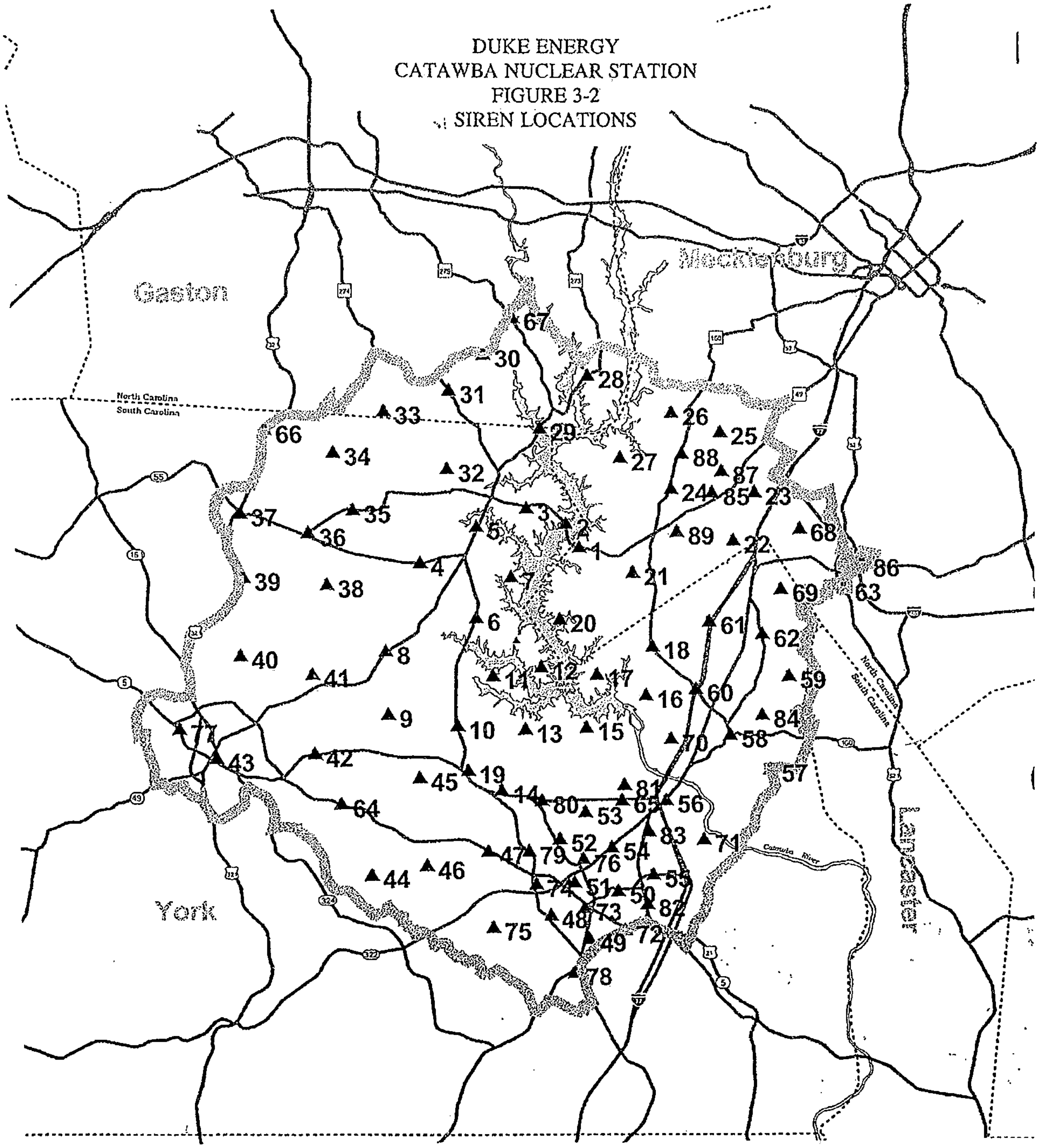
FIGURE 3-1

SIREN RANGE IN FEET

FIGURED AT 12 and 10 dB LOSS PER DISTANCE DOUBLED

Minimum Level Coverage in dB	**2001 AC 126dB(C) Siren**	
	12	**10**
85	1125	1830
80	1500	2600
75	2000	3680
73	2260	4210
70	2700	5200
68	3000	6000
65	3600	7400
60	4800	10400

DUKE ENERGY
CATAWBA NUCLEAR STATION
FIGURE 3-2
SIREN LOCATIONS



EMERGENCY PLANNING ZONE (EPZ)
FOR THE CATAWBA NUCLEAR STATION

APPENDIX 4

DUKE ENERGY
CATAWBA NUCLEAR STATION
EVACUATION TIME ESTIMATES

The Evacuation Time Estimates (ETEs) for the Catawba Nuclear Station described in part J of this plan, dated December 2012, KLD Engineering, P.C. Report KLD TR-510, Catawba Nuclear Station, Development of Evacuation Time Estimates, Revision 1, was submitted under separate cover and is considered to be incorporated as part of this document by reference.

See the following:

- CNS-ETE-12132012, Rev. 000 (Part 1 of 2): PART 1 OF 2 - EVACUATION TIME ESTIMATES (ETE) REPORTS DATED 12/13/2012, REVISION 000 FOR CATAWBA NUCLEAR STATION.
- CNS-ETE-12132012, Rev. 000 (Part 2 of 2): PART 2 OF 2 - EVACUATION TIME ESTIMATES (ETE) REPORTS DATED 12/13/2012, REVISION 000 FOR CATAWBA NUCLEAR STATION.

The studies have been submitted for regulatory review and have been made available to site, state, and local planners for their use.

The evacuation study is available in the CNS Emergency Planning office for study and review.

APPENDIX 5

AGREEMENT LETTERS

This Appendix contains a list of written agreements between Duke Energy and other organizations that may be required to provide support to the Catawba Nuclear Station in the event of an onsite radiological emergency. The actual agreements are maintained on file by CNS Emergency Preparedness.

1. Piedmont Medical Center - Describes the arrangements between Piedmont Medical Center and Duke Energy Corporation relative to the medical care and treatment and to also have injured personnel that may also have radioactive contamination.

2. Carolinas Medical Center - Describes the arrangements between Carolinas Medical Center and Duke Energy Corporation relative to the medical care and treatment and to also have injured personnel that may also have radioactive contamination.

3. Bethel Volunteer Fire Department Describes the type of assistance which the Bethel Volunteer Fire Department will provide to the Catawba Nuclear Station in the event of an emergency such as a radioactive release, hostile action, large scale fire, natural disaster (i.e. hurricane, tornado, earthquake, or flooding), or hazardous material issue.

4. Memorandum of Understanding between Duke Energy Carolinas, LLC and York County, South Carolina - Describes both emergency and non-emergency assistance by York County to support the Catawba Nuclear Station Emergency Plan.

5. Memorandum of Understanding between Duke Energy Carolinas, LLC and Mecklenburg County, North Carolina - Describes both emergency and non-emergency assistance by Mecklenburg County to support the Catawba Nuclear Station Emergency Plan.

6. Memorandum of Understanding between Duke Energy Carolinas, LLC and Gaston County, North Carolina - Describes both emergency and non-emergency assistance by Gaston County to support the Catawba Nuclear Station Emergency Plan.

7. Memorandum of Understanding among the State of North Carolina Department of Public Safety, North Carolina Emergency Management (NCEM), and Duke Energy Carolinas, LLC - Describes both emergency and non-emergency assistance by the State of North Carolina Department of Public Safety, North Carolina Emergency Management (NCEM), and the State of North Carolina Division of Health Service Regulation, Radiation Protection Section (RPS) to support the Catawba Nuclear Station Emergency Plan.

8. Memorandum of Understanding among the South Carolina Emergency Management Division, the South Carolina Department of Health and Environmental Control, and Duke Energy Carolinas, LLC - Describes both emergency and non-emergency assistance by the South Carolina Emergency Management Division, the Carolina Department of Health and Environmental Control to support the Catawba Nuclear Station Emergency Plan.

9. Center for Emergency Medicine - Describes the arrangements Center of Emergency Medicine and Duke Energy Corporation relative to the medical care and treatment and to also have injured personnel that may also have radioactive contamination.

10. Deleted

11. REACTS - Describes the arrangement for the US Department of Energy (DOE) REAC/TS facilities and team to be available to provide back-up capability and assistance to Duke Energy Carolinas, LLC, and Duke Energy Progress, Inc. in the event of a radiological emergency.

12. DOE - Savannah River - DOE - Savannah River--Describes the arrangements between the US Department of Energy, National Nuclear Safety Administration to support the Emergency Plans of the Duke Energy Carolinas and Duke Energy Progress nuclear sites DOE/NNSA assistance will be advice, detection and identification of radioactive materials, and/or monitoring and assessment actions essential for the control of the immediate hazards to health and safety.

13. INPO - Certifies that INPO will assist the Catawba Nuclear Station in acquiring of other organizations in the nuclear industry as described in Section 1 of the Emergency Resources Manual, INPO 03-001 and the United States Industry Response Framework.

14. Deleted

15. Joint Information Center - Establishes an agreement regarding, and provides reference to , the operating guidelines, processes, and procedures governing the use of Joint Information System (JIS) and Joint Information Centers (JIC) by providing a holistic approach for a communications response to a declared emergency or significant event at the Catawba Nuclear Station.

16. Memorandum of Understanding between CNS EP, Work Control, Operations, Site Services and Information Technology on Use of OSC/OCC Area - Establishes that the OSC/OCC/WCC is a multi-purpose facility with the OSC in a state of readiness at all times for compliance with the station's Emergency Plan.

17. Alternate Site Agreement - Describes the terms and conditions of the agreement between the Catawba Nuclear Station and the McGuire Nuclear Station for using either facilities existing business unit space; in this case the Technical Support Center or Alternate Technical Support Center as an alternate site Emergency Operations Facility in the event of a service disruption and/or a disaster rendering the primary Emergency Operations Facility unavailable and relocation of the primary Emergency Operations Facility is necessary.

18. Carolinas Delivery Operations Departmental Interface Agreement - Describes the use of the Emergency Operations Facility by Carolinas Delivery Operations for emergency situations.

19. Memorandum of Understanding between Nuclear Generation Department and the Distribution Maintenance and Construction-West Department Concerning Use of the York Operations Center as Catawba Nuclear Station's Evacuation/ Assembly/Staging Site - Provides an off-site location where personnel released from Catawba Nuclear Station can assemble, be monitored for radiation and, if necessary decontaminated.

20. Memorandum of Understanding between Safe Industries and Catawba, McGuire and Oconee Nuclear Sites - Describes the agreement to the request by Duke Energy regarding assistance with technical support after hours and in emergency situation. In the event a Duke Energy site is in need of emergency technical support , trouble shooting, or assistance with the equipment or operation of Hale pumps

21 Operating Agreement between Duke Energy's Lincoln Combustion Turbine Facility and McGuire, Catawba and Oconee Nuclear Stations Nuclear Supply Chain - Documents the contingency plan between Duke Energy's Lincoln Combustion Turbine Facility and Duke Energy's McGuire, Catawba, and Oconee Nuclear Stations concerning the Lincoln Combustion Turbine Facility providing the emergency supply of diesel fuel during a disruption of normal diesel fuel supply.

22. York County Sherriff's Office to Support the Emergency Plan of the Catawba Nuclear Station - Provides for assistance to support the Catawba Nuclear Station's Emergency Plan, including assistance expected to be provided in the event of an emergency.

These agreements are verified current through annual recertification of the Catawba Emergency Plan. A copy of the annual recertification (including the agreements) is maintained on file by CNS Emergency Preparedness.

Catawba Nuclear Station
Emergency Plan Revision 17-1
Attachment 1
10CFR 50.54(q) Evaluations

Duke Energy

ACTION REQUEST - 02107407

Action Request Cross References

Ref Type	Ref Nbr	Ref Sub	Ref Nbr Type	Status	Limit AR Cls	Description
AR	02099449		DRR	COMPLETE	N	EPA A rev 149

Action Request Assignment Details

ASSIGNMENT NUMBER : 01 **SUB** :

Type	: EP01	**Due Date**	: 03/16/2017	**Pri Resp Fac**	:
Status	: COMPLETE	**Reschedule**	:	**Pri Resp Group**	:
Assigned To	: STACI N FISCHER			**Sec Resp Fac**	:
Subject	: 50.54(Q) SCREEN			**Sec Resp Group**	:
Aff Facility	: CN	**Unit**	:	**System**	:
UCR	:	**Schedule Ref**	:		
Organization	:	**Department**	: 13650	**Discipline**	:
Est Manhrs	:	**Ext Comp Date**	:		

Description

COMPLETE 50.54(Q) SCREEN IN ACCORDANCE WITH AD-EP-ALL-0602.

Action Request Assignment Completion Notes

Action Request Assignment Status History

Updated Date	Updated By	Assgn Status	Assgn Due Date
03/13/2017	I44004	INPROG	
03/13/2017	I44004		03/16/2017
03/13/2017	I44004	ACC/ASG	
03/13/2017	I44004	AWAIT/C	
03/14/2017	TAA7322	COMPLETE	

Duke Energy

ACTION REQUEST - 02107407

Action Request Assignment Attributes

Action Request Assignment Routing/Return Comments

Routing Comments from the X601 Panel	Updated On	Updated By
*** No Routing Comments Found ***		

Routing Comments from the X602 Panel	Updated On	Updated By
*** No Return Comments Found ***		

Action Request Assignment Completion Approval

Route List : 001 Route List Initiator : I44004

Passport	Fac	Group /	Type	Send Date	Send Time	Action Taken	Action Date /	Time	Last Name
I80034			A	03/13/2017	15:40	APPROVED	03/13/2017	15:44	WHITE
TAA7322			A	03/13/2017	15:44	APPROVED	03/14/2017	09:18	ARLOW

Action Request Assignment Cause/Action

Action Request Assignment Reference Documents

Facility	Doc Type	Sub Type	Document	Sheet	Rev	Minor Rev	Title

Action Request Assignment Reference Equipment

Facility	Unit	System	Equip Type	Equip Number	Equip Tag	Equip Status	Rev	Rev Status

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Screening and Evaluation Number	Applicable Sites	
EREG #: 2107407	BNP	☐
	CNS	●
	CR3	☐
	HNP	☐
5AD #: 2104030	MNS	☐
	ONS	☐
	RNP	☐
	GO	☐

Document and Revision
Emergency Plan Section A, Revision 149

Part I. Description of Activity Being Reviewed (event or action, or series of actions that may result in a change to the emergency plan or affect the implementation of the emergency plan):

A.3 Agreement Letters for Emergency Response Support from Off-site Agencies
Changed:

"These Letters of Agreement shall be updated as necessary and at least once every three (3) years."

to

"1. Duke Energy has established numerous support agreements and contracts with organizations that may be required to provide assistance in the event of an emergency.
2. All agreements or contracts are reviewed annually to assure each contributes the desired support to the Emergency Preparedness Program.
3. Letters of Agreement and Contracts, including the review frequency, will be documented according to the site's protocol."

Part II. Activity Previously Reviewed?	Yes ☐	No ●
Is this activity Fully bounded by an NRC approved 10 CFR 50.90 submittal or Alert and Notification System Design Report? If yes, identify bounding source document number or approval reference and ensure the basis for concluding the source document fully bounds the proposed change is documented below: Justification:	10 CFR 50.54(q) Effectiveness Evaluation is not required. Enter justification below and complete Attachment 4, Part V.	Continue to Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part III
Bounding document attached (optional)		☐

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Part III. Editorial Change	Yes	☐	No	●
Is this activity an editorial or typographical change only, such as formatting, paragraph numbering, spelling, or punctuation that does not change intent? Justification:	10 CFR 50.54(q) Effectiveness Evaluation is not required. Enter justification and complete Attachment 4, Part V & VI.		Continue to Attachment 4, Part IV and address non editorial changes	

Part IV. Emergency Planning Element and Function Screen (Reference Attachment 1, Considerations for Addressing Screening Criteria) Does this activity involve any of the following, including program elements from NUREG-0654/FEMA REP-1 Section II? If answer is yes, then check box.		
1	10 CFR 50.47(b)(1) Assignment of Responsibility (Organization Control)	
1a	Responsibility for emergency response is assigned.	●
1b	The response organization has the staff to respond and to augment staff on a continuing basis (24-7 staffing) in accordance with the emergency plan.	☐
2	10 CFR 50.47(b)(2) Onsite Emergency Organization	
2a	Process ensures that onshift emergency response responsibilities are staffed and assigned	☐
2b	The process for timely augmentation of onshift staff is established and maintained.	☐
3	10 CFR 50.47(b)(3) Emergency Response Support and Resources	
3a	Arrangements for requesting and using off site assistance have been made.	☐
3b	State and local staff can be accommodated at the EOF in accordance with the emergency plan. (NA for CR3)	☐
4	10 CFR 50.47(b)(4) Emergency Classification System	
4a	A standard scheme of emergency classification and action levels is in use. (Requires final approval of Screen and Evaluation by EP CFAM.)	☐
5	10 CFR 50.47(b)(5) Notification Methods and Procedures	
5a	Procedures for notification of State and local governmental agencies are capable of initiating notification of the declared emergency within 15 minutes (60 minutes for CR3) after declaration of an emergency and providing follow-up notification.	☐
5b	Administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway. (NA for CR3)	☐
5c	The public ANS meets the design requirements of FEMA-REP-10, Guide for Evaluation of Alert and Notification Systems for Nuclear Power Plants, or complies with the licensee's FEMA-approved ANS design report and supporting FEMA approval letter. (NA for CR3)	☐

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Part IV. Emergency Planning Element and Function Screen (cont.)		
6	10 CFR 50.47(b)(6) Emergency Communications	
6a	Systems are established for prompt communication among principal emergency response organizations.	☐
6b	Systems are established for prompt communication to emergency response personnel.	☐
7	10 CFR 50.47(b)(7) Public Education and Information	
7a	Emergency preparedness information is made available to the public on a periodic basis within the plume exposure pathway emergency planning zone (EPZ). (NA for CR3)	☐
7b	Coordinated dissemination of public information during emergencies is established.	☐
8	10 CFR 50.47(b)(8) Emergency Facilities and Equipment	
8a	Adequate facilities are maintained to support emergency response.	☐
8b	Adequate equipment is maintained to support emergency response.	☐
9	10 CFR 50.47(b)(9) Accident Assessment	
9a	Methods, systems, and equipment for assessment of radioactive releases are in use.	☐
10	10 CFR 50.47(b)(10) Protective Response	
10a	A range of public PARs is available for implementation during emergencies. (NA for CR3)	☐
10b	Evacuation time estimates for the population located in the plume exposure pathway EPZ are available to support the formulation of PARs and have been provided to State and local governmental authorities. (NA for CR3)	☐
10c	A range of protective actions is available for plant emergency workers during emergencies, including those for hostile action events.	☐
10d	KI is available for implementation as a protective action recommendation in those jurisdictions that chose to provide KI to the public.	☐
11	10 CFR 50.47(b)(11) Radiological Exposure Control	
11a	The resources for controlling radiological exposures for emergency workers are established.	☐
12	10 CFR 50.47(b)(12) Medical and Public Health Support	
12a	Arrangements are made for medical services for contaminated, injured individuals.	☐
13	10 CFR 50.47(b)(13) Recovery Planning and Post-accident Operations	
13a	Plans for recovery and reentry are developed.	☐
14	10 CFR 50.47(b)(14) Drills and Exercises	
14a	A drill and exercise program (including radiological, medical, health physics and other program areas) is established.	☐
14b	Drills, exercises, and training evolutions that provide performance opportunities to develop, maintain, and demonstrate key skills are assessed via a formal critique process in order to identify weaknesses.	☐
14c	Identified weaknesses are corrected.	☐
15	10 CFR 50.47(b)(15) Emergency Response Training	
15a	Training is provided to emergency responders.	☐

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Part IV. Emergency Planning Element and Function Screen (cont.)		
16	10 CFR 50.47(b)(16) Emergency Plan Maintenance	
16a	Responsibility for emergency plan development and review is established.	☐
16b	Planners responsible for emergency plan development and maintenance are properly trained.	☐

PART IV. Conclusion If no Part IV criteria are checked, a 10 CFR 50.54(q) Effectiveness Evaluation is not required, then complete Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part V. Go to Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part VI for instructions describing the NRC required 30 day submittal.	☐
If any Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part IV criteria are checked, then complete Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part V and perform a 10 CFR 50.54(q) Effectiveness Evaluation. Shaded block requires final approval of Screen and Evaluation by EP CFAM.	●

Part V. Signatures:		
Preparer Name (Print): Staci Fischer	Preparer Signature: See CAS	Date: See CAS
Reviewer Name (Print): Jeffery White	Reviewer Signature: See CAS	Date: See CAS
Approver (EP Manager Name (Print): Tom Arlow	Approver Signature: See CAS	Date: See CAS
Approver (CFAM, as required) Name (Print) N/A	Approver Signature: N/A	Date: N/A

Part VI. NRC Emergency Plan and Implementing Procedure Submittal Actions

Create two EREG General Assignments.	
• One for EP to provide the 10 CFR 50.54(q) summary of the analysis, or the completed 10 CFR 50.54(q), to Licensing.	●
• One for Licensing to submit the 10 CFR 50.54(q) information to the NRC within 30 days after the change is put in effect.	●

QA RECORD

DUKE ENERGY.

Duke Energy

ACTION REQUEST - 02107407

Action Request Assignment Details

ASSIGNMENT NUMBER : 02 SUB :

Type	: EP02	Due Date	: 03/16/2017	Pri Resp Fac	:
Status	: COMPLETE	Reschedule	:	Pri Resp Group	:
Assigned To	: STACI N FISCHER			Sec Resp Fac	:
Subject	: 50.54(Q) EVALUATION			Sec Resp Group	:
Aff Facility	: CN	Unit	:	System	:
UCR	:	Schedule Ref	:		
Organization	:	Department	: 13650	Discipline	:
Est Manhrs	:	Ext Comp Date	:		

Description

COMPLETE 50.54(Q) EVALUATION IN ACCORDANCE WITH AD-EP-ALL- 0602.

Action Request Assignment Completion Notes

Action Request Assignment Status History

Updated Date	Updated By	Assgn Status	Assgn Due Date
03/13/2017	I44004	INPROG	
03/13/2017	I44004		03/16/2017
03/13/2017	I44004	ACC/ASG	
03/13/2017	I44004	AWAIT/C	
03/14/2017	TAA7322	COMPLETE	

Action Request Assignment Attributes

Action Request Assignment Routing/Return Comments

Routing Comments from the X601 Panel	Updated On	Updated By
*** No Routing Comments Found ***		

Duke Energy

ACTION REQUEST - 02107407

Routing Comments from the X602 Panel	Updated On	Updated By
*** No Return Comments Found ***		

Action Request Assignment Completion Approval

Route List : 001

Route List Initiator : I44004

Passport	Fac	Group /	Type	Send Date	Send Time	Action Taken	Action Date /	Time	Last Name
I80034			A	03/13/2017	15:41	APPROVED	03/13/2017	15:46	WHITE
TAA7322			A	03/13/2017	15:46	APPROVED	03/14/2017	09:19	ARLOW

Action Request Assignment Cause/Action

Action Request Assignment Reference Documents

Facility	Doc Type	Sub Type	Document	Sheet	Rev	Minor Rev	Title

Action Request Assignment Reference Equipment

Facility	Unit	System	Equip Type	Equip Number	Equip Tag	Equip Status	Rev	Rev Status

Action Request Assignment Cross References

Ref Type	Ref Nbr	Ref Sub	Ref Nbr Type	Status	Limit AS Cls	Description

Action Request Assignment Appendices

APPENDIX 1

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Screening and Evaluation Number	Applicable Sites	
EREG #: 2107407	BNP	☐
	CNS	●
	CR3	☐
	HNP	☐
5AD #: 2104030	MNS	☐
	ONS	☐
	RNP	☐
	GO	☐

Document and Revision
Emergency Plan Section A, Revision 149

Part I. Description of Proposed Change:

A.3 Agreement Letters for Emergency Response Support from Off-site Agencies

Changed:

"These Letters of Agreement shall be updated as necessary and at least once every three (3) years."

to

"1. Duke Energy has established numerous support agreements and contracts with organizations that may be required to provide assistance in the event of an emergency.

2. All agreements or contracts are reviewed annually to assure each contributes the desired support to the Emergency Preparedness Program.

3. Letters of Agreement and Contracts, including the review frequency, will be documented according to the site's protocol."

Attachment 6, 10 CFR 50.54(q) Initiating Condition (IC) and Emergency Action Level (EAL) and EAL Bases Validation and Verification (V&V) Form , is attached (required for IC or EAL change)	Yes ☐ No ●

Part II. Description and Review of Licensing Basis Affected by the Proposed Change:

Three licensing basis documents were reviewed for applicability, 1) the Catawba Emergency plan, Revision 2, as the original plan approved by the NRC, 2) the current Catawba Emergency plan, revision 148, and 3) "Duke Power Company Response to Supplement 1 to NUREG-0737, Emergency Response Capability for Catawba Nuclear Station, Volume 1."

Applicable sections of the Emergency Plan titled, **"Catawba Nuclear Station Emergency Plan, revision 2 - January 1983:**

Section A, Assignment of Responsibility

A.3 Agreement Letter For Emergency Response Support

Appendix 5 contains letters of agreement with the following organizations:

- York General Hospital and Ambulance Service
- Charlotte Memorial Hospital and Medical Center
- Municipal-County Emergency Preparedness Agency of York County
- Bethel Volunteer Fire Department
- Department of Emergency Management, Mecklenburg County

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

(Charlotte, NC)
Department of Emergency Management Gaston County
North Carolina Department of Crime Control and Public Safety
South Carolina Department of Health and Environmental Control

NOTE: Agreements with other agencies are found in the Crisis Management Plan Appendix 5.

[Preparer's Comment - there is not a requirement to update the Agreement letters at least once every three years in the latest NRC-approved Catawba Emergency Plan]

The current revision of the **Catawba Emergency Plan** is revision 148. The following sections are affected by the change described in this evaluation:

Section A - Assignment of Responsibility
A.3 Agreement Letters for Emergency Response Support from Off-site Agencies
Section Q, Appendix 5, contains letters of agreement with the following organizations:

Piedmont Medical Center
Carolinas Medical Center
York County Emergency Management
Bethel Volunteer Fire Department
Charlotte-Mecklenburg Emergency Management Office
Gaston County Emergency Management
Center for Emergency Medicine (Rock Hill, SC)
North Carolina Division of Emergency Management
South Carolina Emergency Management Division
Radiation Emergency Assistance Center/Training Site (REAC/TS)
DOE - Savannah River
INPO - Fixed Nuclear Facility Voluntary Assistance Agreement
JIC - Joint Information Center
York County Sheriff

These Letters of Agreement shall be updated as necessary and at least once every three (3) years.

Duke Power Company Response to Supplement 1 to NUREG-0737, Emergency Response Capability for Catawba Nuclear Station, Volume 1 does not contain applicable sections related to assignment of responsibility or letters of agreement.

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Part III. Description of How the Proposed Change Complies with Regulation and Commitments.

If the emergency plan, modified as proposed, no longer complies with planning standards in 10 CFR 50.47(b) and the requirements in Appendix E to 10 CFR Part 50, then ensure the change is rejected, modified, or processed as an exemption request under 10 CFR 50.12, Specific Exemptions, rather than under 10 CFR 50.54(q):

10 CFR 50.47(b) (1).

"Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis."

10 CFR 50, Appendix E, section IV. A.6

"A description of the local offsite services to be provided in support of the licensee's emergency organization."

NUREG-0654 Section II.A.3

"Each plan shall include written agreements referring to the concept of operations developed between Federal, State, and local agencies and other support organizations having an emergency response role within the Emergency Planning Zones. The agreements shall identify the emergency measures to be provided and the mutually acceptable criteria for their implementation, and specify the arrangements for exchange of information. These agreements may be provided in an appendix to the plan or the plan itself may contain descriptions of these matters and a signature page in the plan may serve to verify the agreements. The signature page format is appropriate for organizations where response functions are covered by laws, regulations or executive orders where separate written agreements are not necessary."

Conclusion

The change continues to comply with Regulations and Commitments because the emergency responsibilities of the various supporting organizations remain specifically established. The agreements continue to describe the local offsite services that may be provided in support of the site Emergency Response Organization, and the plans remain in a written form with the identification of emergency measures to be provided. The regulations do not contain a periodicity for review, update, or re-issuing the agreement letters.

The latest NRC-approved Catawba Emergency Plan (Revision 2) does not contain a required timeframe to update or re-issue the agreement letters. The three-year requirement was added to Revision 4 in April 1984. The reason behind the addition cannot be determined at this time. With the proposed change, the agreement letters will continue to be reviewed every year as part of the annual review of the Catawba Emergency Plan. The re-issuing of these agreement letters will be done on an as- needed basis instead of a three-year periodic basis.

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Part IV. Description of Emergency Plan Planning Standards, Functions and Program Elements Affected by the Proposed Change (Address each function identified in Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part IV of associated Screen):

PLANNING STANDARDS

10 CFR 50.47(b) (1).

"Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the *emergency responsibilities of the various supporting organizations have been specifically established*, and each principal response organization has staff to respond and to augment its initial response on a continuous basis."

FUNCTIONS

Two emergency planning functions have been defined for 10 CFR 50.47(b)(1):

(1) Responsibility for emergency response is assigned.

(2) The response organization has the staff to respond and to augment staff on a continuing basis (i.e., 24/7 support) in accordance with the emergency plan.

10 CFR 50, Appendix E, section IV. A.6 provides supporting requirements:

"A description of the local offsite services to be provided in support of the licensee's emergency organization."

PROGRAM ELEMENTS

NUREG-0654 Section II.A.3

"Each plan shall include written agreements referring to the concept of operations developed between Federal, State, and local agencies and other support organizations having an emergency response role within the Emergency Planning Zones. The agreements shall identify the emergency measures to be provided and the mutually acceptable criteria for their implementation, and specify the arrangements for exchange of information. These agreements may be provided in an appendix to the plan or the plan itself may contain descriptions of these matters and a signature page in the plan may serve to verify the agreements. The signature page format is appropriate for organizations where response functions are covered by laws, regulations or executive orders where separate written agreements are not necessary."

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Part V. Description of Impact of the Proposed Change on the Effectiveness of Emergency Plan Functions:

Two emergency planning functions have been defined for 10 CFR 50.47(b)(1):

(1) Responsibility for emergency response is assigned.

(2) The response organization has the staff to respond and to augment staff on a continuing basis (i.e., 24/7 support) in accordance with the emergency plan.

Conclusion

The change continues to comply with Emergency Plan Functions because the emergency responsibilities of the various supporting organizations remain specifically established. The agreements continue to describe the local offsite services that may be provided in support of the site Emergency Response Organization, and the plans remain in a written form with the identification of emergency measures to be provided.

The regulations do not contain a periodicity for review, update, or re-issuing the agreement letters.

The Emergency Plan Planning Standards, Functions and Program Elements do not contain a required timeframe to update or re-issue the agreement letters. With the proposed change, the agreement letters will continue to be reviewed every year as part of the annual review of the Catawba Emergency Plan. The re-issuing of these agreement letters will be done on an as- needed basis instead of a three-year periodic basis.

Part VI. Evaluation Conclusion.

Answer the following questions about the proposed change.

1	Does the proposed change comply with 10 CFR 50.47(b) and 10 CFR 50 Appendix E?	Yes ●	No ☐
2	Does the proposed change maintain the effectiveness of the emergency plan (i.e., no reduction in effectiveness)?	Yes ●	No ☐
3	Does the proposed change maintain the current Emergency Action Level (EAL) scheme?	Yes ●	No ☐
4	Choose one of the following conclusions:		
a	The activity does continue to comply with the requirements of 10 CFR 50.47(b) and 10 CFR 50, Appendix E, and the activity does not constitute a reduction in effectiveness or change in the current Emergency Action Level (EAL) scheme. Therefore, the activity can be implemented without prior NRC approval.	●	
b	The activity does not continue to comply with the requirements of 10 CFR 50.47(b) or 10 CFR 50 Appendix E or the activity does constitute a reduction in effectiveness or EAL scheme change. Therefore, the activity cannot be implemented without prior NRC approval.	☐	

Part VII. Disposition of Proposed Change Requiring Prior NRC Approval

Will the proposed change determined to require prior NRC approval be either revised or rejected?	Yes ☐	No ☐

If No, then initiate a License Amendment Request in accordance 10 CFR 50.90 and AD-LS-ALL-0002, Regulatory Correspondence, and include the tracking number:______________________.

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Part VIII. Signatures: EP CFAM Final Approval is required for changes affecting risk significant planning standard 10 CFR 50.47(b)(4).		
Preparer Name (Print): Staci Fischer	Preparer Signature: See CAS	Date: See CAS
Reviewer Name (Print): Jeffery White	Reviewer Signature: See CAS	Date: See CAS
Approver (EP Manager) Name (Print): Tom Arlow	Approver Signature: See CAS	Date: See CAS
Approver (CFAM, as required) Name (Print): N/A	Approver Signature: N/A	Date: N/A

If the proposed activity is a change to the E-Plan or implementing procedures, then create two EREG General Assignments.	
• One for EP to provide the 10 CFR 50.54(q) summary of the analysis, or the completed 10 CFR 50.54(q), to Licensing.	•
• One for Licensing to submit the 10 CFR 50.54(q) information to the NRC within 30 days after the change is put in effect.	•

QA RECORD

Duke Energy

ACTION REQUEST - 02107784

Action Request Cross References

Ref Type	Ref Nbr	Ref Sub	Ref Nbr Type	Status	Limit AR Cls	Description
AR	02107483		DRR	COMPLETE	N	EPA Q rev 149

Action Request Assignment Details

ASSIGNMENT NUMBER : 01 **SUB** :

Type	: EP01	**Due Date**	: 03/16/2017	**Pri Resp Fac**	:
Status	: COMPLETE	**Reschedule**	:	**Pri Resp Group**	:
Assigned To	: STACI N FISCHER			**Sec Resp Fac**	:
Subject	: 50.54(Q) SCREEN			**Sec Resp Group**	:
Aff Facility	: CN	**Unit**	:	**System**	:
UCR	:	**Schedule Ref**	:		
Organization	:	**Department**	: 13650	**Discipline**	:
Est Manhrs	:	**Ext Comp Date**	:		

Description

COMPLETE 50.54(Q) SCREEN IN ACCORDANCE WITH AD-EP-ALL-0602.

Action Request Assignment Completion Notes

Action Request Assignment Status History

Updated Date	Updated By	Assgn Status	Assgn Due Date
03/14/2017	I44004	INPROG	
03/14/2017	I44004		03/16/2017
03/14/2017	I44004	ACC/ASG	
03/14/2017	I44004	AWAIT/C	
03/14/2017	TAA7322	COMPLETE	

Duke Energy

ACTION REQUEST - 02107784

Action Request Assignment Attributes

Action Request Assignment Routing/Return Comments

Routing Comments from the X601 Panel	Updated On	Updated By
*** No Routing Comments Found ***		

Routing Comments from the X602 Panel	Updated On	Updated By
*** No Return Comments Found ***		

Action Request Assignment Completion Approval

Route List : 001   Route List Initiator : I44004

Passport	Fac	Group /	Type	Send Date	Send Time	Action Taken	Action Date /	Time	Last Name
I80034			A	03/14/2017	08:41	APPROVED	03/14/2017	08:48	WHITE
TAA7322			A	03/14/2017	08:48	APPROVED	03/14/2017	09:16	ARLOW

Action Request Assignment Cause/Action

Action Request Assignment Reference Documents

Facility	Doc Type	Sub Type	Document	Sheet	Rev	Minor Rev	Title

Action Request Assignment Reference Equipment

Facility	Unit	System	Equip Type	Equip Number	Equip Tag	Equip Status	Rev	Rev Status

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Screening and Evaluation Number	Applicable Sites	
EREG #: 2107784	BNP	☐
	CNS	●
	CR3	☐
	HNP	☐
5AD #: 2107785	MNS	☐
	ONS	☐
	RNP	☐
	GO	☐

Document and Revision
Emergency Plan Section Q, Revision 149

Part I. Description of Activity Being Reviewed (event or action, or series of actions that may result in a change to the emergency plan or affect the implementation of the emergency plan):

Appendix 5, last statements changed from:
"These agreements are verified current through annual recertification of the Catawba Emergency Plan.
A copy of the annual recertification (including the agreements) is maintained on file by CNS Emergency Preparedness.
The actual agreements are re-confirmed every 3 years and maintained on file by CNS Emergency Preparedness."

to

"These agreements are verified current through annual recertification of the Catawba Emergency Plan.
A copy of the annual recertification (including the agreements) is maintained on file by CNS Emergency Preparedness."

Part II. Activity Previously Reviewed?	Yes ☐	No ●
Is this activity Fully bounded by an NRC approved 10 CFR 50.90 submittal or Alert and Notification System Design Report? If yes, identify bounding source document number or approval reference and ensure the basis for concluding the source document fully bounds the proposed change is documented below: Justification:	10 CFR 50.54(q) Effectiveness Evaluation is not required. Enter justification below and complete Attachment 4, Part V.	Continue to Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part III
Bounding document attached (optional)		☐

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Part III. Editorial Change	Yes ☐	No ●
Is this activity an editorial or typographical change only, such as formatting, paragraph numbering, spelling, or punctuation that does not change intent? Justification:	10 CFR 50.54(q) Effectiveness Evaluation is not required. Enter justification and complete Attachment 4, Part V & VI.	Continue to Attachment 4, Part IV and address non editorial changes

Part IV. Emergency Planning Element and Function Screen (Reference Attachment 1, Considerations for Addressing Screening Criteria)

Does this activity involve any of the following, including program elements from NUREG-0654/FEMA REP-1 Section II? If answer is yes, then check box.

1	10 CFR 50.47(b)(1) Assignment of Responsibility (Organization Control)	
1a	Responsibility for emergency response is assigned.	●
1b	The response organization has the staff to respond and to augment staff on a continuing basis (24-7 staffing) in accordance with the emergency plan.	☐
2	10 CFR 50.47(b)(2) Onsite Emergency Organization	
2a	Process ensures that onshift emergency response responsibilities are staffed and assigned	☐
2b	The process for timely augmentation of onshift staff is established and maintained.	☐
3	10 CFR 50.47(b)(3) Emergency Response Support and Resources	
3a	Arrangements for requesting and using off site assistance have been made.	☐
3b	State and local staff can be accommodated at the EOF in accordance with the emergency plan. (NA for CR3)	☐
4	10 CFR 50.47(b)(4) Emergency Classification System	
4a	A standard scheme of emergency classification and action levels is in use. (Requires final approval of Screen and Evaluation by EP CFAM.)	☐
5	10 CFR 50.47(b)(5) Notification Methods and Procedures	
5a	Procedures for notification of State and local governmental agencies are capable of initiating notification of the declared emergency within 15 minutes (60 minutes for CR3) after declaration of an emergency and providing follow-up notification.	☐
5b	Administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway. (NA for CR3)	☐
5c	The public ANS meets the design requirements of FEMA-REP-10, Guide for Evaluation of Alert and Notification Systems for Nuclear Power Plants, or complies with the licensee's FEMA-approved ANS design report and supporting FEMA approval letter. (NA for CR3)	☐

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Part IV. Emergency Planning Element and Function Screen (cont.)		
6	10 CFR 50.47(b)(6) Emergency Communications	
6a	Systems are established for prompt communication among principal emergency response organizations.	☐
6b	Systems are established for prompt communication to emergency response personnel.	☐
7	10 CFR 50.47(b)(7) Public Education and Information	
7a	Emergency preparedness information is made available to the public on a periodic basis within the plume exposure pathway emergency planning zone (EPZ). (NA for CR3)	☐
7b	Coordinated dissemination of public information during emergencies is established.	☐
8	10 CFR 50.47(b)(8) Emergency Facilities and Equipment	
8a	Adequate facilities are maintained to support emergency response.	☐
8b	Adequate equipment is maintained to support emergency response.	☐
9	10 CFR 50.47(b)(9) Accident Assessment	
9a	Methods, systems, and equipment for assessment of radioactive releases are in use.	☐
10	10 CFR 50.47(b)(10) Protective Response	
10a	A range of public PARs is available for implementation during emergencies. (NA for CR3)	☐
10b	Evacuation time estimates for the population located in the plume exposure pathway EPZ are available to support the formulation of PARs and have been provided to State and local governmental authorities. (NA for CR3)	☐
10c	A range of protective actions is available for plant emergency workers during emergencies, including those for hostile action events.	☐
10d	KI is available for implementation as a protective action recommendation in those jurisdictions that chose to provide KI to the public.	☐
11	10 CFR 50.47(b)(11) Radiological Exposure Control	
11a	The resources for controlling radiological exposures for emergency workers are established.	☐
12	10 CFR 50.47(b)(12) Medical and Public Health Support	
12a	Arrangements are made for medical services for contaminated, injured individuals.	☐
13	10 CFR 50.47(b)(13) Recovery Planning and Post-accident Operations	
13a	Plans for recovery and reentry are developed.	☐
14	10 CFR 50.47(b)(14) Drills and Exercises	
14a	A drill and exercise program (including radiological, medical, health physics and other program areas) is established.	☐
14b	Drills, exercises, and training evolutions that provide performance opportunities to develop, maintain, and demonstrate key skills are assessed via a formal critique process in order to identify weaknesses.	☐
14c	Identified weaknesses are corrected.	☐
15	10 CFR 50.47(b)(15) Emergency Response Training	
15a	Training is provided to emergency responders.	☐

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Part IV. Emergency Planning Element and Function Screen (cont.)		
16	10 CFR 50.47(b)(16) Emergency Plan Maintenance	
16a	Responsibility for emergency plan development and review is established.	☐
16b	Planners responsible for emergency plan development and maintenance are properly trained.	☐

PART IV. Conclusion If no Part IV criteria are checked, a 10 CFR 50.54(q) Effectiveness Evaluation is not required, then complete Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part V. Go to Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part VI for instructions describing the NRC required 30 day submittal.	☐
If any Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part IV criteria are checked, then complete Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part V and perform a 10 CFR 50.54(q) Effectiveness Evaluation. Shaded block requires final approval of Screen and Evaluation by EP CFAM.	●

Part V. Signatures:		
Preparer Name (Print): Staci Fischer	Preparer Signature: See CAS	Date: See CAS
Reviewer Name (Print): Jeffery White	Reviewer Signature: See CAS	Date: See CAS
Approver (EP Manager Name (Print): Tom Arlow	Approver Signature: See CAS	Date: See CAS
Approver (CFAM, as required) Name (Print) N/A	Approver Signature: N/A	Date: N/A

Part VI. NRC Emergency Plan and Implementing Procedure Submittal Actions

Create two EREG General Assignments.	
• One for EP to provide the 10 CFR 50.54(q) summary of the analysis, or the completed 10 CFR 50.54(q), to Licensing.	●
• One for Licensing to submit the 10 CFR 50.54(q) information to the NRC within 30 days after the change is put in effect.	●

QA RECORD

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Duke Energy

ACTION REQUEST - 02107784

Action Request Assignment Details

ASSIGNMENT NUMBER : 02 **SUB** :

Type	: EP02	**Due Date**	: 03/16/2017	**Pri Resp Fac**	:
Status	: COMPLETE	**Reschedule**	:	**Pri Resp Group**	:
Assigned To	: STACI N FISCHER			**Sec Resp Fac**	:
Subject	: 50.54(Q) EVALUATION			**Sec Resp Group**	:
Aff Facility	: CN	**Unit**	:	**System**	:
UCR	:	**Schedule Ref**	:		
Organization	:	**Department**	: 13650	**Discipline**	:
Est Manhrs	:	**Ext Comp Date**	:		

Description

COMPLETE 50.54(Q) EVALUATION IN ACCORDANCE WITH AD-EP-ALL- 0602.

Action Request Assignment Completion Notes

Action Request Assignment Status History

Updated Date	Updated By	Assgn Status	Assgn Due Date
03/14/2017	I44004	INPROG	
03/14/2017	I44004		03/16/2017
03/14/2017	I44004	ACC/ASG	
03/14/2017	I44004	AWAIT/C	
03/14/2017	TAA7322	COMPLETE	

Action Request Assignment Attributes

Action Request Assignment Routing/Return Comments

Routing Comments from the X601 Panel	Updated On	Updated By
*** No Routing Comments Found ***		

Duke Energy

ACTION REQUEST - 02107784

Routing Comments from the X602 Panel	Updated On	Updated By
*** No Return Comments Found ***		

Action Request Assignment Completion Approval

Route List : 001 | Route List Initiator : I44004

Passport	Fac	Group /	Type	Send Date	Send Time	Action Taken	Action Date /	Time	Last Name
I80034			A	03/14/2017	08:41	APPROVED	03/14/2017	08:50	WHITE
TAA7322			A	03/14/2017	08:50	APPROVED	03/14/2017	09:16	ARLOW

Action Request Assignment Cause/Action

Action Request Assignment Reference Documents

Facility	Doc Type	Sub Type	Document	Sheet	Rev	Minor Rev	Title

Action Request Assignment Reference Equipment

Facility	Unit	System	Equip Type	Equip Number	Equip Tag	Equip Status	Rev	Rev Status

Action Request Assignment Cross References

Ref Type	Ref Nbr	Ref Sub	Ref Nbr Type	Status	Limit AS Cls	Description

Action Request Assignment Appendices

APPENDIX 1

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Screening and Evaluation Number	Applicable Sites	
EREG #: 2107784	BNP	☐
	CNS	●
	CR3	☐
	HNP	☐
5AD #: 2107785	MNS	☐
	ONS	☐
	RNP	☐
	GO	☐

Document and Revision
Emergency Plan Section Q, Revision 149

Part I. Description of Proposed Change:

Appendix 5, last statements changed from:

"These agreements are verified current through annual recertification of the Catawba Emergency Plan. A copy of the annual recertification (including the agreements) is maintained on file by CNS Emergency Preparedness. The actual agreements are re-confirmed every 3 years and maintained on file by CNS Emergency Preparedness."

to

"These agreements are verified current through annual recertification of the Catawba Emergency Plan. A copy of the annual recertification (including the agreements) is maintained on file by CNS Emergency Preparedness."

Attachment 6, 10 CFR 50.54(q) Initiating Condition (IC) and Emergency Action Level (EAL) and EAL Bases Validation and Verification (V&V) Form , is attached (required for IC or EAL change)	Yes ☐ No ●

Part II. Description and Review of Licensing Basis Affected by the Proposed Change:

Three licensing basis documents were reviewed for applicability, 1) the Catawba Emergency plan, Revision 2, as the original plan approved by the NRC, 2) the current Catawba Emergency plan, revision 148, and 3) "Duke Power Company Response to Supplement 1 to NUREG-0737, Emergency Response Capability for Catawba Nuclear Station, Volume 1."

Applicable sections of the Emergency Plan titled, **"Catawba Nuclear Station Emergency Plan, revision 2 - January 1983:**

Section A, Assignment of Responsibility

A.3 Agreement Letter For Emergency Response Support

Appendix 5 contains letters of agreement with the following organizations:

- York General Hospital and Ambulance Service
- Charlotte Memorial Hospital and Medical Center
- Municipal-County Emergency Preparedness Agency of York County
- Bethel Volunteer Fire Department
- Department of Emergency Management, Mecklenburg County (Charlotte, NC)

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Department of Emergency Management Gaston County
North Carolina Department of Crime Control and Public Safety
South Carolina Department of Health and Environmental Control

NOTE: Agreements with other agencies are found in the Crisis Management Plan Appendix 5.

[Preparer's Comment - there is not a requirement to update the Agreement letters at least once every three years in the latest NRC-approved Catawba Emergency Plan]

The current revision of the **Catawba Emergency Plan** is revision 148. The following sections are affected by the change described in this evaluation:

Section A - Assignment of Responsibility
A.3 Agreement Letters for Emergency Response Support from Off-site Agencies
Section Q, Appendix 5, contains letters of agreement with the following organizations:

Piedmont Medical Center
Carolinas Medical Center
York County Emergency Management
Bethel Volunteer Fire Department
Charlotte-Mecklenburg Emergency Management Office
Gaston County Emergency Management
Center for Emergency Medicine (Rock Hill, SC)
North Carolina Division of Emergency Management
South Carolina Emergency Management Division
Radiation Emergency Assistance Center/Training Site (REAC/TS)
DOE - Savannah River
INPO - Fixed Nuclear Facility Voluntary Assistance Agreement
JIC - Joint Information Center
York County Sheriff

These Letters of Agreement shall be updated as necessary and at least once every three (3) years.

Duke Power Company Response to Supplement 1 to NUREG-0737, Emergency Response Capability for Catawba Nuclear Station, Volume 1 does not contain applicable sections related to assignment of responsibility or letters of agreement.

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Part III. Description of How the Proposed Change Complies with Regulation and Commitments.

If the emergency plan, modified as proposed, no longer complies with planning standards in 10 CFR 50.47(b) and the requirements in Appendix E to 10 CFR Part 50, then ensure the change is rejected, modified, or processed as an exemption request under 10 CFR 50.12, Specific Exemptions, rather than under 10 CFR 50.54(q):

10 CFR 50.47(b) (1)

> “Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.”

10 CFR 50, Appendix E, section IV. A.6

> "A description of the local offsite services to be provided in support of the licensee's emergency organization."

Conclusion

The change continues to comply with Regulations and Commitments because the emergency responsibilities of the various supporting organizations remain specifically established. The agreements continue to describe the local offsite services that may be provided in support of the site Emergency Response Organization, and the plans remain in a written form with the identification of emergency measures to be provided. The regulations do not contain a periodicity for review, update, or re-issuing the agreement letters.

The Catawba Emergency Plan (Revision 2) does not contain a required timeframe to update or re-issue the agreement letters. The three-year requirement was added to Revision 4 in April 1984. The reason behind the addition cannot be determined at this time. With the proposed change, the agreement letters will continue to be reviewed every year as part of the annual review of the Catawba Emergency Plan. The re-issuing of these agreement letters will be done on an as- needed basis instead of a three-year periodic basis.

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Part IV. Description of Emergency Plan Planning Standards, Functions and Program Elements Affected by the Proposed Change (Address each function identified in Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part IV of associated Screen):

PLANNING STANDARDS

10 CFR 50.47(b) (1).

"Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the *emergency responsibilities of the various supporting organizations have been specifically established*, and each principal response organization has staff to respond and to augment its initial response on a continuous basis."

FUNCTIONS

Two emergency planning functions have been defined for 10 CFR 50.47(b)(1):

(1) Responsibility for emergency response is assigned.

(2) The response organization has the staff to respond and to augment staff on a continuing basis (i.e., 24/7 support) in accordance with the emergency plan.

10 CFR 50, Appendix E, section IV. A.6 provides supporting requirements:

"A description of the local offsite services to be provided in support of the licensee's emergency organization."

PROGRAM ELEMENTS

NUREG-0654 Section II.A.3

"Each plan shall include written agreements referring to the concept of operations developed between Federal, State, and local agencies and other support organizations having an emergency response role within the Emergency Planning Zones. The agreements shall identify the emergency measures to be provided and the mutually acceptable criteria for their implementation, and specify the arrangements for exchange of information. These agreements may be provided in an appendix to the plan or the plan itself may contain descriptions of these matters and a signature page in the plan may serve to verify the agreements. The signature page format is appropriate for organizations where response functions are covered by laws, regulations or executive orders where separate written agreements are not necessary."

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Part V. Description of Impact of the Proposed Change on the Effectiveness of Emergency Plan Functions:

Two emergency planning functions have been defined for 10 CFR 50.47(b)(1):

(1) Responsibility for emergency response is assigned.

(2) The response organization has the staff to respond and to augment staff on a continuing basis (i.e., 24/7 support) in accordance with the emergency plan.

Conclusion

The change continues to comply with Emergency Plan Functions because the emergency responsibilities of the various supporting organizations remain specifically established. The agreements continue to describe the local offsite services that may be provided in support of the site Emergency Response Organization, and the plans remain in a written form with the identification of emergency measures to be provided.

The regulations do not contain a periodicity for review, update, or re-issuing the agreement letters.

The Emergency Plan Planning Standards, Functions and Program Elements do not contain a required timeframe to update or re-issue the agreement letters. With the proposed change, the agreement letters will continue to be reviewed every year as part of the annual review of the Catawba Emergency Plan. The re-issuing of these agreement letters will be done on an as- needed basis instead of a three-year periodic basis.

Part VI. Evaluation Conclusion.

Answer the following questions about the proposed change.

1	Does the proposed change comply with 10 CFR 50.47(b) and 10 CFR 50 Appendix E?	Yes ●	No ☐
2	Does the proposed change maintain the effectiveness of the emergency plan (i.e., no reduction in effectiveness)?	Yes ●	No ☐
3	Does the proposed change maintain the current Emergency Action Level (EAL) scheme?	Yes ●	No ☐
4	Choose one of the following conclusions:		
a	The activity does continue to comply with the requirements of 10 CFR 50.47(b) and 10 CFR 50, Appendix E, and the activity does not constitute a reduction in effectiveness or change in the current Emergency Action Level (EAL) scheme. Therefore, the activity can be implemented without prior NRC approval.	●	
b	The activity does not continue to comply with the requirements of 10 CFR 50.47(b) or 10 CFR 50 Appendix E or the activity does constitute a reduction in effectiveness or EAL scheme change. Therefore, the activity cannot be implemented without prior NRC approval.	☐	

Part VII. Disposition of Proposed Change Requiring Prior NRC Approval

Will the proposed change determined to require prior NRC approval be either revised or rejected?	Yes ☐	No ☐

If No, then initiate a License Amendment Request in accordance 10 CFR 50.90 and AD-LS-ALL-0002, Regulatory Correspondence, and include the tracking number:______________________________.

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

Part VIII. Signatures: EP CFAM Final Approval is required for changes affecting risk significant planning standard 10 CFR 50.47(b)(4).		
Preparer Name (Print): Staci Fischer	Preparer Signature: See CAS	Date: See CAS
Reviewer Name (Print): Jeffery White	Reviewer Signature: See CAS	Date: See CAS
Approver (EP Manager) Name (Print): Tom Arlow	Approver Signature: See CAS	Date: See CAS
Approver (CFAM, as required) Name (Print): N/A	Approver Signature: N/A	Date: N/A

If the proposed activity is a change to the E-Plan or implementing procedures, then create two EREG General Assignments. • One for EP to provide the 10 CFR 50.54(q) summary of the analysis, or the completed 10 CFR 50.54(q), to Licensing.	•
• One for Licensing to submit the 10 CFR 50.54(q) information to the NRC within 30 days after the change is put in effect.	•

QA RECORD

Catawba Nuclear Station
Emergency Plan Revision 17-1
Attachment 2
Plan Update Instructions

Replace Revision 16-2 Coversheet with Revision 17-1 Coversheet

List of Effective Pages (LOEP)
Replace all pages of this section

Tab A - Assignment of Responsibility
Replace all pages of this section

Tab D – Emergency Classification System
Replace all pages of this section

Tab Q – Appendix Index
Replace all pages of this section

Catawba Nuclear Station
Emergency Plan Revision 17-1
Attachment 3
Emergency Plan Revision 17-1